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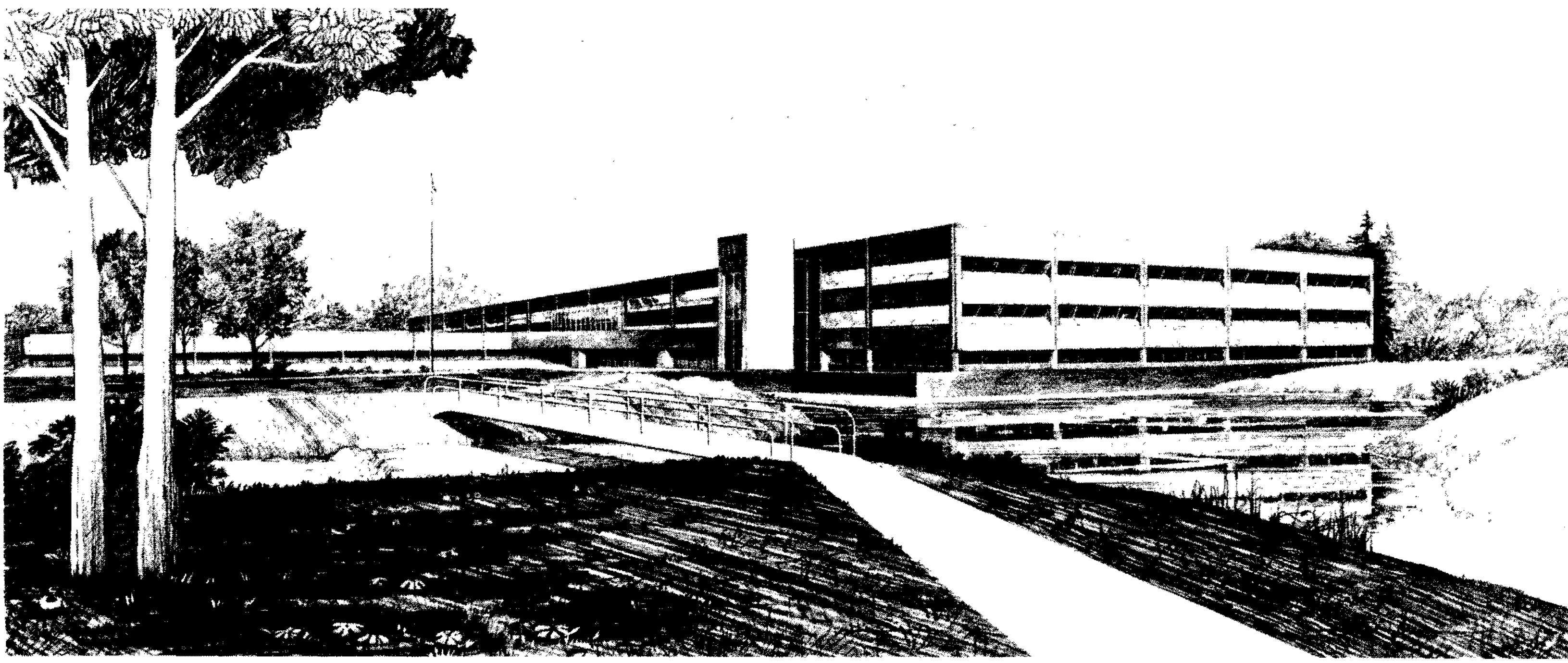
TMI-2 CORE EXAMINATION PLAN

Johan O. Carlson, Editor

J. C. HAIRE 11-6-84

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. DEPARTMENT OF ENERGY
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EG&G Idaho

EGG-TMI-6169
Revision 1

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Johan O. Carlson, Editor

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Idaho Falls, Idaho 83415

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ABSTRACT

The role of the Three Mile Island Unit 2 (TMI-2) core examination in the resolution of major nuclear safety issues is delineated in this plan. Relevant data needs are discussed, and approaches for recovering data from the TMI-2 plant are identified. Specific recommendations and justifications are provided for in situ documentation and off-site artifact examination activities. The research and development program is being managed by EG&G Idaho, Inc.

SUMMARY

Examination of the Three Mile Island Unit 2 (TMI-2) core will provide data to address the following major nuclear safety issues facing the light water reactor industry:

1. Fission product release, transport, and deposition (analyzing fission product retention in the core, on reactor vessel internals, and in the primary system)

2. Core coolability (understanding the damage state and processes of the core and reactor vessel internals)

3. Containment integrity (evaluating hydrogen generation)

4. Recriticality (assessing the segregation of fuel and control materials)

5. 10 CFR 50.46 issues (determining fuel and cladding behavior during a loss-of-coolant accident).

The TMI-2 core examination plan is divided into four categories. The first category includes in situ examinations at TMI-2 and is intended to provide on-site documentation of the post-accident condition of the core. This is to be done primarily by additional closed-circuit television camera inspections of the core and lower vessel, and ultrasonic mapping of the core cavity.

The second category includes characterization of surface deposits on reactor coolant system (RCS) artifacts from locations other than the core region. It includes examination of the following reactor coolant system components and structures: control rod leadscrews, leadscrew support tube, plenum cover debris, resistance thermal detectors (RTDs)/thermowells, steam generator handhole covers, and makeup and letdown system filters.

The third category includes characterization of the condition of the core by examination of samples taken from the reactor core and lower vessel. Samples included in this category are core debris grab samples from the near-surface rubble bed, fueled rod segments, core stratification samples, distinct fuel assembly and control rod cluster components (e.g., cladding, control rods, spiders, spacer grids, end fittings, hold down springs, etc.), in-core instrumentation, and debris from the lower vessel.

The fourth category includes examination of miscellaneous components and items not specifically included in Categories 1, 2, and 3. Examinations in this category include those of reactor building basement solids and the reactor coolant drain tank.

ACKNOWLEDGMENTS

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ACRONYMS

APSRA	Axial power shaping rod assembly
CCTV	Closed-circuit television
CFR	Code of Federal Regulations
CR	Control rod
CRA	Control rod assembly
CRDM	Control rod drive mechanism
DOE	U.S. Department of Energy
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
ESCA	Electron spectroscopy for chemical analysis
GPU Nuclear	General Public Utilities Nuclear Corporation
IDCOR	Industry Degraded Core Rulemaking
INEL	Idaho National Engineering Laboratory
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MOLE	Molecular optical laser examiner
NRC	U.S. Nuclear Regulatory Commission
OTSG	Once-through steam generators
PH	Precipitation hardened
PORV	Pilot-operated relief valve
RCS	Reactor coolant system
R&D	Research and development
RTD	Resistance thermal detector
RV	Reactor vessel
SEM	Scanning electron microscopy
SG	Steam generator
SIMS	Secondary ion mass spectrometry
SPND	Self-powered neutron detector
STEM	Scanning transmission electron microscopy
TEG	Technical evaluation group
TMI-2	Three Mile Island Unit 2
WDX	Wave length dispersive X-ray

TMI-2 CORE EXAMINATION PLAN

1. INTRODUCTION AND HISTORICAL BACKGROUND

On March 28, 1979, the Unit 2 pressurized water reactor at Three Mile Island (TMI-2) underwent an accident resulting in severe damage to the core of the reactor. As a consequence of the TMI-2 accident, numerous aspects of light water reactor safety have been questioned. 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 Three Mile Island activities program to develop technology for recovery from a serious reactor accident and conduct relevant research and development that will substantially enhance nuclear power plant safety.

This document is intended to provide recommendations for work to be conducted under the research and development part of the TMI activities program of DOE. It also is intended to outline how these recommendations were developed and discuss their supporting technical basis. The recommendations made are intended to provide guidance in executing the research and development part of the TMI-2 activities program of DOE (hereinafter referred to as the "TMI-2 core examination plan") by

- Identifying types of data that should be obtained through the program

- Suggesting approaches suitable for obtaining the data sought.

As a top-tier document, the scope outlined herein stops short of detailed specification of the technical data acquisition activities.

A broad spectrum of needs for data has arisen to support successful resolution of the severe accident safety questions that have arisen since the TMI-2 accident. The nuclear community generally has acknowledged the importance of examining TMI-2 to provide information satisfying some of these needs. This is reflected by the fact that 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. The organizations, commonly referred to as the GEND Group--General Public Utilities Nuclear Corporation, Electric Power Research Institute (EPRI), Nuclear Regulatory Commission, and Department of Energy--currently are involved actively in reactor recovery and accident research. The areas to which each of the individual GEND organizations are committing available resources have been defined and coordinated to minimize overlap. DOE is providing a portion of the funds for reactor recovery (in those areas where accident recovery knowledge generally will benefit the U.S. light water reactor industry). Additionally, DOE is providing most of the funds for acquiring severe accident technical data where such data are needed but would not otherwise be available from the cleanup effort.

Limitations on DOE resources available for technical data acquisition have dictated that the TMI-2 core examination be planned, executed, and designed to meet specific technical objectives rather than be an open-ended program of scientific inquiry. This, in turn, requires not only that technical activities planned for this program are designed to satisfy specific data needs, but also that these activities are selected carefully to maximize useful data obtained from available resources. With this in mind, criteria used to develop recommendations presented herein are summarized as follows:

- Each recommended activity should be designed to satisfy identified and specific data needs.

- Each recommended activity should be practical in light of

- The as-built and current damage state of the TMI-2 plant

- Constraints imposed by available engineering and scientific methods used for obtaining desired data

- Constraints imposed by the plant recovery plans, objectives, and schedule.

o Each recommended activity should be consistent with the objective of maximizing useful data obtained using the available resources. At the same time, special attention must be paid to the uniqueness or cost-effectiveness of TMI-2 as a source of requisite data.

Section 2 of this plan describes objectives of DOE regarding the TMI-2 core examination plan. Section 3 provides the approach taken to identify the types of samples and examinations to be conducted. Section 4 identifies specific data needs that could be satisfied by data obtained from TMI-2. Section 5 presents some of the technical background that led to the identification of practical approaches for obtaining the desired data. Finally, Section 6 summarizes recommendations that were developed based on consideration of data needs that would be satisfied through each recommended activity.

2. OBJECTIVES

DOE has stated its objectives in participating in the TMI-2 recovery and data acquisition programs. The Secretary of DOE, in his annual report to Congress, stated the following:

> Our objective is to obtain data that can be used by the Nuclear Regulatory Commission, utilities, and the nuclear manufacturing industry to enhance the safety of commercial reactors generally and to ensure adequate preparedness and protection for the public should a significant accident of any type occur.[1]

This statement establishes that DOE has a broad charter in its TMI-2 programs to acquire data useful to all parts of the nuclear industry. Such a charter, however, must be carried out within DOE's expressed intent to limit their TMI-2-related expenditures to an established dollar amount. Hence, the primary objective of this document is to recommend data acquisition activities that maximize the useful data obtainable and are consistent with the available resources.

This document is intended to provide guidance for EG&G Idaho in developing the TMI-2 core examination plan, which will satisfy DOE objectives and outline for other interested parties those considerations that shaped the recommendations presented herein. The scope of this document, therefore, includes the following:

1. Identification of the types of data retrievable from TMI-2 that can be used by the various organizations cited in the Secretary's statement (Refernce 1)

2. Identification of suitable approaches for obtaining the requisite data

3. Development of a list of recommended sampling and examination activities, considering funding constraints and the value and cost of TMI-2 data acquisition activities.

While not specifying the details of the TMI-2 core examination plan, it is the objective of this document to provide recommendations that are sufficient to provide overall guidance for the program. This is achieved by (a) identifying the types of data that should be obtained and (b) providing guidance for *in situ* measurements, artifact recovery, and subsequent off-site analyses to obtain the requisite data.

3. APPROACH

EG&G Idaho, Inc. was selected by DOE as the lead laboratory to organize and manage the research and development (R&D) activities for the damaged TMI-2 reactor. The Core Activities Program and the Radiation and Environment (often referred to as "Source Term") Program were organized to assist in the management of the R&D effort. Each of these EG&G Idaho programs operated somewhat independently and contracted the consulting and advisory assistance of two separate technical evaluation groups, the TMI-2 Core Damage Assessment and the Radiation and Environment TEGs. The TEGs consist of specialists from the entire nuclear industry, both commercial and governmental.

During the past year, EG&G Idaho has been in process of combining the two programs to coordinate and centralize the investigative efforts. The two separate TEGs were combined to form the current TMI-2 Core Damage Assessment/Fission Product Behavior TEG. Following the objectives outlined herein, and using financial restraints from DOE and knowledge obtained to date regarding the condition of the damaged reactor, EG&G Idaho, with the assistance of Battelle Columbus Laboratories, combined the two previous program plans into a draft combined plan. The combined TEG used this draft to formulate the recommended plan outlined in Section 6 (Table 9) of this document.

The evolutionary process for the recommendations presented herein involved several successive iterations of the following steps. The first step involved careful consideration of data needs together with an evaluation of the types of data that could reasonably be obtained from TMI-2 examination. The approach used to identify data needs was to identify major safety issues and assess types of data needed to address them. Recommendations for the data that should be obtained were developed through consideration of the following:

1. The as-built plant

2. The current damage state as inferred from

a. The accident event sequence

b. Available accident analysis results

c. Available experimental data

d. Assessment of how the post-accident state may have changed since March 1979

3. The requirement for minimal interference with the ongoing recovery program

4. Practical difficulties involved

5. Estimated costs involved to obtain the data.

These considerations ensured that the recommendations that evolved and are presented herein are consistent with criteria identified in Section 1. Cost estimates were developed as the recommendations were made. In general, the preliminary recommendations were based on (a) subjective engineering evaluations of the usefulness of the recoverable data and (b) difficulties involved in obtaining the data. As the list of recommendations evolved and matured, the engineering difficulties and costs associated with execution of each recommendation were examined in greater depth, so informed choices could be made in light of the value of the data and the cost associated with obtaining it.

The second step involved presentation of draft recommendations developed by working groups within the TEG to the TEG membership for their review and concurrence. Since the TEG membership includes a broad representation of nuclear power industry organizations, this step was intended to ensure that the recommendations will meet needs of most of the organizations and thereby satisfy the intent of the Secretary's statement to Congress (see Reference 1).

The recommendations contained in this document are therefore the result of an evolutionary process. In general terms, that process involved iterative execution of the two steps identified above at successively greater levels of detail, first within the individual TEG groups and then within the combined TEG. The iterative process was cost-effective in that it allowed the list of recommendations to be framed as it evolved, without necessitating expensive studies of engineering feasibility and cost for items not included in the final recommendations. For the recommendations included in the final plan, the process has been carried out in sufficient detail to give reasonable assurance of the following:

- The recommended TMI-2 core examination plan will produce data that meet the intent of the Secretary's statement to Congress in a cost-effective manner

- The plan can be executed within cost guidelines established by DOE

- The plan represents the general-consensus recommendations of interested organizations within the nuclear power industry

- The plan will have minimal interference with the TMI-2 recovery effort and, in fact, will supply data supportive of that effort

- The plan is based on a reasonable and carefully executed thought process using criteria that are consistent with DOE objectives.

4. DATA NEEDS AND TYPES OF DATA RECOVERABLE FROM TMI-2

The first step in developing recommendations for the TMI-2 core examination plan was identifying the set of needed and reasonably-recoverable data. Comprehensive identification of "data that can be used by" the various organizations cited in the Secretary's statement to Congress would be a formidable task. Rather than attempt to survey these organizations, this document assumes that the set of data needed by these organizations is similar to that needed to resolve the major nuclear safety issues facing the light water reactor industry. The validity of this assumption is supported by the fact that the combined TEG membership, representing as it does a broad cross section of the organizations involved, endorses the contents of this document. The remainder of this section, therefore, focuses on the data needs derived from consideration of the following five major nuclear safety issues (shown in the approximate order of their importance):

1. Fission product release, transport, and deposition (analyzing fission product retention in the core, on reactor vessel internals, and in the primary system)

2. Core coolability (understanding the damage processes of the core and reactor vessel internals)

3. Containment integrity (evaluating hydrogen generation)

4. Recriticality (assessing the segregation of fuel and control materials)

5. 10 CFR 50.46 issues [determining fuel and cladding behavior during a loss-of-coolant accident (LOCA)].

Before discussing these issues individually, it is important to state that the reader must take a very broad view of these issues. Currently debated issues such as emergency response and safety equipment survivability are not being ignored; rather, they are part of the larger issues listed above. Appropriate emergency response, for example, is a matter of the following:

1. Diagnosing the status of an accident (by recognizing clues such as temperature, pressure, flow, etc., that knowledge of core damage processes reveals)

2. Knowing what level of release of fission products from the fuel, if any, to expect from the accident

3. Knowing whether fission product containment and retention systems can fail or be bypassed by the accident

4. Determining what the environmental source term will be and, therefore, the appropriate on-site and off-site emergency responses.

The reader should also recognize that data needed for resolution of these nuclear safety issues also are being addressed by a number of projects conducted in test reactors and laboratories. However, studies of the TMI-2 plant will provide unique information on full-scale, reactor-wide variations in fuel rod damage timing and mechanisms, rubble bed formation and coolability, and fission product release and deposition. What follows is an attempt to identify the set of data retrievable from TMI-2 that will have the most impact on the major nuclear safety issues and will complement and support other ongoing safety research programs while minimizing duplication of effort.

4.1 Fission Product Release, Transport, and Deposition

Fission product release, transport, and deposition is clearly the most fundamental issue and underlies all other safety issues. This issue is basic because the biological effects of ionizing radiation from fission product radionuclides are the fundamental hazard of nuclear power. Since the biological effects and the dispersal and transport of radionuclides in the environment are relatively well-understood phenomena, the technical issue of fission product release and transport reduces to (a) the behavior of fission products within the reactor system and (b) the source term to the environment. Simply restated, the issue is this: For a given accident type or category, what radionuclides (chemical form, physical form, and quantity) remain in the fuel, are retained in the primary system, are released to the containment, and are released to the environment?

Because there are many postulated accident sequences and reactor designs, the industry will have to rely heavily on the results of computer code analyses for resolution of this issue (and the other issues cited previously). Hence, data needed to resolve this issue are determined largely by data needed to support the development and validation of fission product release and transport codes. The following section is intended to provide some insight into these data needs so TMI-2 data can be seen in the context of the overall data needs. Fission product release, transport, and deposition is the most fundamental of the issues identified here. The primary focus of the discussion in Section 4.1.1 is on computer codes used for source term analyses. However, most of the discussion is sufficiently general and applies to data needs of codes that can be used to resolve the other four major nuclear safety issues identified above.

4.1.1 Overview of Computer Code Data Needs

The development of computer codes to analyze source terms for core damage accidents in nuclear power plants must proceed through the following three sequential steps:

1. Identification of the dominant physical phenomena and processes

2. Development of a model to represent these processes

3. Validation of the model.

Execution of each of these steps can generate a wide variety of data needs, each of which would, ideally, be satisfied before proceeding to the next step in the development process. Since such an overall sequential approach would be protracted and expensive in practice, analysts frequently make assumptions where

1. Existing knowledge is not adequate to uniquely define the phenomena or processes of interest.

   Before the TMI-2 accident, the key processes or phenomena governing the release, transport, and deposition of fission products in light water reactor plants under severe accident conditions could not be identified by direct physical observation, at least on the scale of a modern, commercial-size power reactor. At that time, an alternative approach was necessary to define the phenomenology that would serve as a basis for model development. This alternative approach, which is the basis for many codes now used in risk and safety analysis studies, is based on plausible assumptions supported by informed judgment and insight, supplemented whenever possible by laboratory experimental information. Examination of TMI-2 provides a unique opportunity to obtain data on key phenomena involved in a core damage accident in a full-scale light water reactor (LWR) plant. It therefore represents an opportunity to check the basic assumptions and the extrapolation of laboratory data incorporated in the codes.

2. The complete analytical description of the model system is not tractable or is inconsistent with design objectives for the model code.

   When a computer code is constructed to analyze a system, the system is represented as a mathematical model. In constructing models, the analyst often is forced to make simplifying assumptions (e.g., the system consists of large, well-mixed volumes). Because of the complexity of the severe accident phenomena that govern fission product release and transport, many of the codes currently used in risk and safety analyses incorporate simplifying assumptions. These simplifying assumptions can be necessitated either in the development of suitable descriptions for individual processes or phenomena, or in the development of a suitably coupled description for the overall system.

These phenomenological and simplifying assumptions introduce uncertainties about the accuracy of code predictions. Therefore, it is important to check the accuracy and completeness of the assumptions involved and/or check their adequacy, through suitable validation of the individual process models and overall system model.

If sufficient data could be obtained to conduct a meaningful integral validation, and code predictions agreed sufficiently with the validation data, costs associated with data acquisition supporting the source term codes could be minimized. However, even if good agreement was found between code predictions and TMI-2 data, this would only benchmark the codes involved. These codes are intended to be applicable to a broad spectrum of severe core damage accidents in LWR plants, of which the TMI-2 accident is only one specific example. If the code predictions were to disagree significantly with the data, it would be necessary to identify the reasons for the disagreements. The reasons could be associated with (a) omissions of key phenomena, (b) omissions of models that include important interactions between key phenomena, or (c) inadequate descriptions for individual processes included in the code because of the simplifying assumptions made. It is, therefore, important to collect data that could support such model development or improvement.

The code data needs that should be considered as a basis for specifying the TMI-2 core examination plan can be categorized as follows:

1. Data needed to identify the key phenomena and interactions between those phenomena that govern fission product release, transport, and deposition

2. Data needed to test the adequacy of the specific assumptions (both phenomenological and simplifying) implicit in the use of individual process and overall system models

3. Data needed to support development or improvement of individual process models and/or the overall system description

4. Data needed to support validation of mature individual process models and/or the overall system description

5. Data needed to satisfy input requirements of the codes involved.

It should be recognized that as development of the individual codes progresses and matures, the focus of the data needs also generally progresses through the five categories listed above.

4.1.2 Data Needs to Assess the Source Term for a Core Damage Accident

Rather than attempt to discuss individual source term codes and their data needs, this document will attempt to provide a rational basis for those needs by discussing, in fairly general terms, what analysis of core damage accident source terms entails. In general, analysis of source terms for these accidents requires that the accident progression, plant thermal-hydraulic response, and associated radionuclide release and transport be investigated. In the past, such analyses have, for the most part, involved the use of one set of "physical process codes" to analyze the accident progression and plant thermal-hydraulic response. These codes have, in turn, been used to provide input to the "radionuclide transport analysis codes" as outlined in Figure 1.

The radionuclide release and transport behavior codes have been designed to calculate release of radionuclides from overheated and damaged fuel and describe subsequent "airborne" transport of these materials along escape pathways in the primary system and containment. In fact, the release of radionuclides from the fuel and airborne transport along escape pathways defined by specific accident sequences are often analyzed by separate codes.

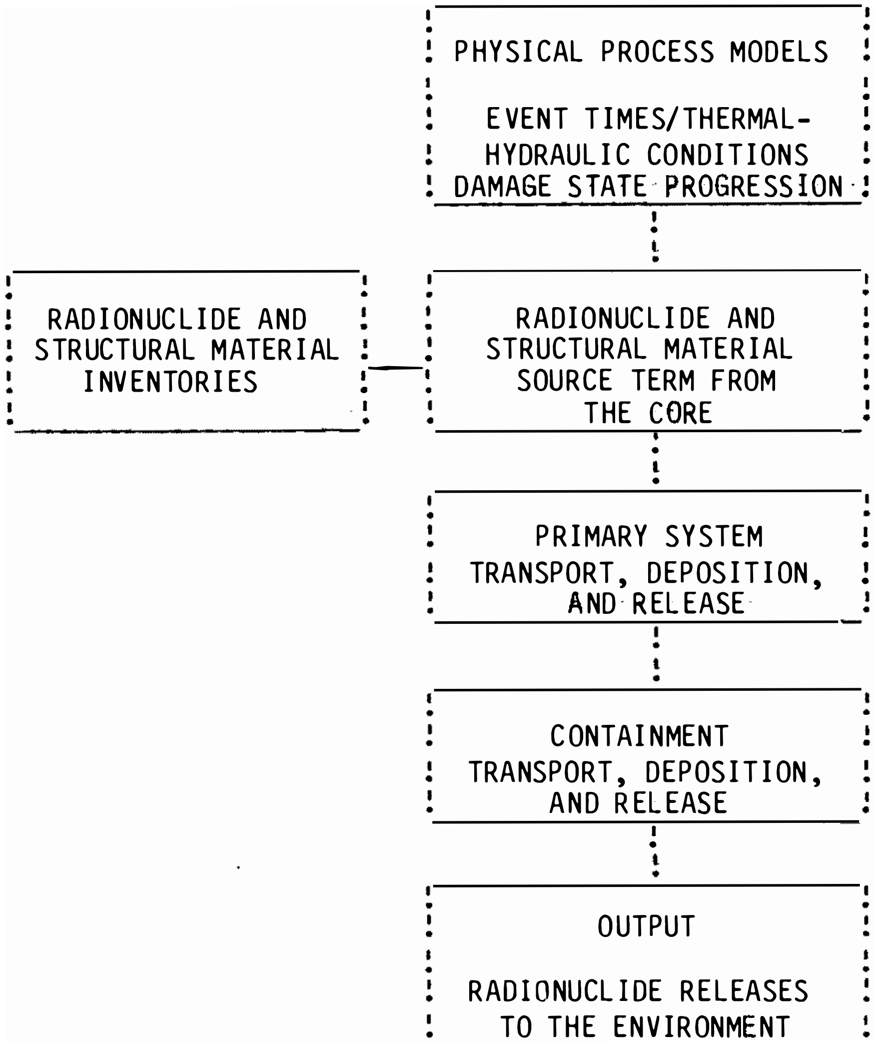

Figure 1. Key steps involved in analyzing the release and transport of radionuclides in light water reactors.

Some of the newer codes (e.g., SCDAP, MELPROG, CONTAIN, and MELCOR) are still under development. Attempts are being made to incorporate at least some important aspects of the coupling between the thermal-hydraulic and radionuclide transport behavior under severe accident conditions into these codes. Although there are some assumptions implicit in the separation of the analysis of severe accident phenomena as shown in Figure 1, this overview of the approach is adequate for this discussion of data needs.

Together with suitable thermal-hydraulic input from the physical process analysis codes, the radionuclide behavior codes are designed primarily to calculate environmental source terms for severe accident sequences. This implies the codes must provide a basis for estimating the amount of release of each of the radiologically significant radionuclides as well as the chemical and physical forms in which they are released to the environment. Not all of the 600+ radionuclides in an LWR core are important contributors to the environmental source term.[2] Reactor accident consequences arise from the amounts of individual radionuclides that escape to the environment. The codes currently used to analyze radionuclide release, transport, and deposition generally reduce the problem to analyzing the behavior of the most important fission products. The codes assume that isotopic inventories can be calculated from elemental inventories at any location from the following expression:

$$M_{ij}(t) = M_i(t)\ F_j(t)$$

where

$M_{ij}(t)$ = the inventory of any isotope j of element i at any location at time t

$M_i(t)$ = the elemental inventory of element i at this location at the same time t

$F_j(t)$ = the overall fractional contribution of isotope j to the elemental inventory of element i at time t (as would, for example, be obtained from an ORIGEN[a] calculation).

In addition, the fission product elements are further combined into groups based on judgments of their similarities in properties that govern release, transport, and deposition behavior. [See, for example, the WASH-1400 classification.] Data needed to test the adequacy of these two simplifying assumptions are

1. Data on isotopic compositions of deposits along the transport pathway at TMI-2
2. Data on elemental ratios for deposits along the transport pathway.

The data needed to support the physical process models are discussed at some length in the following subsections. A summary is presented in Table 1 of the types of data needed to support codes which analyze (a) release from the core, (b) primary system transport and deposition, and (c) containment building transport. The importance of the data needs identified herein is emphasized by calculations made before the TMI-2 accident, of fission product release from the reactor primary system during a severe accident. Those calculations appear to be overestimated. Such calculations did not fully account for the significant retention of fission products within both the primary system and the containment. It appears from TMI-2 data that such retention can significantly lower the calculated source term and thereby reduce the estimated public health consequences. As summarized in Table 1, the data needed by the industry include quantity and timing of releases from the fuel, chemical reactions of fission products in the accident environment, physical form of the released radionuclides (gases, volatile compounds, aerosols, etc.), and the role of primary system and containment retention phenomena (chemisorption, physisorption, condensation, agglomeration, etc.).

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a. A. G. Groff, ORIGEN-2--A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621, July 1980.

TABLE 1. SUMMARY OF THE TYPES OF TMI-2 DATA THAT COULD SATISFY THE IDENTIFIED DATA NEEDS FOR THE REGULATORY ANALYSIS CODES

Type of Data Needed	Core Source Term Codes	Primary System Transport Codes	Containment Building Transport Codes
1. Data needed to identify key phenomena	a. Fission gas bubble concentration and distribution in fuel	a. Data on vapor/surface and aerosol/surface interactions	a. Data on vapor/surface and aerosol/surface interactions for the reactor building coolers and piping mirror insulation
	b. Fuel porosity, porosity inter-linkage, cracking/grain separation, liquefaction	b. Data on the deposit (fission product and control material) concentration map within the RCS	
	c. Fission product distribution in the microstructural features		
	d. Identification of processes that lead to release of fission products and control materials from severely damaged fuel		
2. Data needed to support model assumptions	a. Molecular forms of the individual fission products and other core component materials released from the core region	a. Data on deposit (fission product and control material concentration) variation with location in the RCS	a. None
		b. Data on variation with location of the chemical composition of deposits on RCS surfaces	
		c. Data on elemental composition variation with particle size for particles removed from various locations	
3. Data needed to support model development	a. Detailed examination of fission product and control material distributions in all core materials for the different core damage regimes	a. None	a. None
4. Data needed for code validation/benchmarking	a. Release data for indivudal fission product and control material elements from the various damage regimes in the core	a. Map of deposit concentrations on those surfaces that were uncovered during the core uncovery period	a. Data on the chemical composition and quantity of fission products and other core component materials in the drain tank debris and on the surfaces of the drain tank and associated piping
5. Data needed to support code input requirements	a. Temperature and temperature history data for the various damage regimes in the core	a. Temperature and temperature history data for materials and surfaces in the primary RCS	

Such data are particularly important to NRC because their decisions (e.g., severe accident rulemaking, siting criteria, emergency response procedures, engineered safety equipment requirements) are influenced very strongly by the facts and assumptions about fission product release and transport.

It can be argued that controlled experiments employing gas and liquid analysis, aerosol measurement equipment, on-line gamma spectrometers, time-sequenced sample collection, and other techniques allowing determination of a fission product mass balance are required. However, the greatest difficulty with such experiments is not one of collecting accurate data, but rather of extrapolating results of small-scale experiments to the much larger scale of a commercial reactor. Small-scale experiments probably do not adequately account for all significant phenomena occurring in a full-scale reactor accident such as TMI-2. This scale-up question leads to the most important role of the TMI-2 core examination plan: to help resolve the fission product release, transport, and deposition issues.

Chemical and physical transformations and leaching of soluble chemical forms will have altered the post-accident fission product deposition. Nevertheless, the fact that TMI-2 is a full-scale reactor means that it will become a data base for judging results of future experiments and computer calculations. Accordingly, the proper role of the TMI-2 core examination plan in resolving the issue of fission product release and transport is not to do exhaustive and detailed analyses of the exact distribution and chemical form of fission products; this is best done in well-controlled separate-effects tests. Rather, the objective is to selectively sample the TMI-2 primary system and its contents and characterize the amount, distribution, and current state of fission products present. Specifically, the TMI-2 core examination plan should address the following:

- The fission products retained in the UO_2 fuel (and thus, by calculation, the amount released). All forms of the fuel that are encountered (intact rods, broken rods, fragmented fuel, oxidized fuel, liquefied fuel, etc.) must be investigated.

- The apparent role of the reactor vessel internal structures and other primary system surfaces (particularly the high surface area in the above-core components such as the upper plenum and the fuel assembly upper end fittings) in fission product retention. This will require analysis of fission product compounds deposited on selected surfaces of these components. The goal here is to obtain some understanding of the relative role of various components in the fission product retention process.

The above scope of fission product behavior research at TMI-2 will complement experimental research on accident-caused fission product release, transport, and deposition, which is primarily in the form of out-of- pile separate-effects tests and small-scale, integral in-pile tests. The geometry employed in these tests is generally one- or, at most, two- dimensional. Information on three-dimensional fission product release, transport, and deposition from TMI-2 will provide a significant addition to ongoing source term research. The role of the upper plenum in the retention of fission products is expected to be especially critical and is a matter of considerable current uncertainty. Data from the TMI-2 core examination on the distribution of retained fission products will be an important benchmark for the source term experimental and modeling effort. Impact of the TMI-2 core examination on resolution of the fission product release, transport, and deposition issue will likely be very great. It will provide the basis for the industry to more convincingly apply its research conclusions to postulated accidents in commercial LWRs.

4.2 Core Coolability and Understanding Core Damage Processes

The issue of a loss of core coolability leading to core melt, vessel failure, and containment breach has been a safety concern for many years. The TMI-2 accident confirmed that even a severely disrupted reactor core could remain coolable. In the context of the general discussion of the categories of data needs discussed in Section 4.1.1, the current data needs for resolution of this issue are focused on clarification of the phenomenology involved (i.e., Categories 1 and 2). Specifically, unresolved questions remain as to exactly how the core reconfigured and whether that reconfigured core could have reached a noncoolable or difficult-to-cool geometry. The Rogovin report, in attempting to answer the question "How close to a meltdown?", concluded that massive UO_2 melting almost occurred on two occasions.[3] On the first occasion, about three hours into the accident, the core was probably only 30 minutes away from UO_2 melting. The report also concluded that fuel liquefaction (UO_2 dissolution by molten zircaloy) probably occurred, but the extent of liquefaction was difficult to determine analytically. Because the greatest core coolability concerns arise as a consequence of flow channel blockage by liquid material, the TMI-2 accident heightened the industry's sensitivity to the issue, while it demonstrated the ability to terminate a severe accident of a full-sacle LWR with no public health consequences from radiation.

The Rogovin report concluded that "present knowledge about the physical phenomena (associated with core disruption) are subject to considerable uncertainty." Thus, this nuclear safety issue extends beyond the question of coolability. Specifically the issue is this: For the range of core damage accidents, how does the core deteriorate, and can the core damage produce difficult-to-cool debris geometries? This issue is one for which a limited data base exists. Separate-effects and small-scale integral effects tests have provided researchers with an understanding of some core damage phenomena. The mechanisms of fuel rod failure (e.g., ballooning, oxidation, rod fragmentation, zircaloy melting, UO_2 melting, UO_2/cladding mechanical interaction, etc.) have been studied and modeled. The behavior and characterization of debris beds (heat transfer, fluid flow, dryout, compositions and configurations, etc.) also have been studied. One phenomenon that has not been extensively studied--fuel liquefaction--is now being addressed in both laboratory and reactor experiments. These individual aspects of core damage are, or shortly will be, better understood. Their integral behavior--how they interact to produce a predictable terminal damage state--is less well understood.

The type of data needed by the industry to resolve this core damage issue is one for which TMI-2 is particularly suited: large-scale data to confirm or modify the results of small-scale reactor tests. Small-scale reactor tests do an excellent job of defining the sequence of events by virtue of their instrumentation, and their postirradiation examination allows an in-depth analysis of individual damage structures. However, reactor tests do not very convincingly model the approximately 100 metric tons of material in a LWR core, nor do they necessarily reveal synergisms. Commercial reactor cores can be cooled in many ways (e.g., radiation, forced flow, natural convection, cross flow, top-down flow, and steam cooling) and a safe, stable state can be achieved even though a significant fraction of the core has been damaged. Thus the size and flow design of a power reactor provides accident termination options difficult to include in test reactor experiments. Propagation of damage, particularly molten material relocation, also is difficult to reproduce because the heat capacity of large masses and possibility of coolant bypass paths and local variations in heat transfer are not easily scaled down. Test reactor experiments are necessarily simplified and, therefore, only approximations of large cores. The role of Ag-In-Cd control rod alloy in the TMI-2 accident (some of which certainly melted and was released when the control rods failed) is unknown. The influence of this low-melting-point material early in the core damage sequence is unknown. Test reactor experiments have not investigated this phenomenon yet.

Many data requirements needed by industry from large-scale tests can be well met by the TMI-2 core examination plan. The subtleties of core damage phenomena can come from separate-effects and small-scale integral effects tests. The TMI-2 data are required to show how the core damage events developed on a large scale and whether or not there were unexpected phenomena. These data can be obtained best by a thorough sampling of the full range of core debris encountered during defueling. The first goal of this sampling will be to document the location and extent of damage features. Closed-circuit television camera inspections revealed a particle bed in the upper central region of the core. The character of the damage changes as one goes from the middle of the core (fine particles) to the outer radius (larger particles and recognizable fuel rods/assemblies). At some depth beneath the rubble bed, the character of the damage presumably changes. One is likely to encounter larger fuel rod pieces, remnants of damaged components such as spacer grids and control rods, zones of previously molten core materials and/or fuel liquefaction, and an underlying layer of fuel rod stubs.

Major differences can be postulated in the heat removal from such diverse structures during forced coolant flow. Peripheral regions should possess less tortuous flow paths than the extensively damaged, compacted central core. Core coolability may have been further complicated at TMI-2 by the formation of preferred coolant flow paths--leaving zones of unknown size more susceptible to UO_2 melting. Small-scale integral tests and out-of-pile experiments cannot approach a thorough description of such complex three-dimensional effects.

TMI-2 off-site examinations should concentrate on confirming the details of core damage (e.g., extent of oxidation, estimate of hydrogen generation, and fission product release because of fuel liquefaction). If such detailed examinations reveal unexpected phenomena, then these phenomena must be evaluated for their impact on safety, and, if necessary, experiments devised to understand them. Even though TMI-2 is only one in a broad spectrum of core damage events, it is likely to contain, on a large scale, many of the most potentially serious core damage features. Thus, TMI-2 will help provide the basis for judging the extent to which core coolability and severe accident phenomenology are understood and modeled.

4.3 Containment Integrity

Containment integrity is a major nuclear safety issue in the wake of TMI-2 for several reasons. First, some fission products bypassed the engineered containment systems and were released to the environment via the auxiliary building ventilation system of the reactor. Second, hydrogen gas released from the metal-water reaction in the core reached the reactor building, where it ignited. While the hydrogen burn did not threaten the integrity of the reactor building at TMI-2, it damaged some nonsafety-related equipment inside the building. Third, it appears some melting of non-fuel core materials occurred. If such melting occurred on a large scale, vessel breach concerns might arise. This has resulted in the need for data that will clarify the phenomena leading to vessel breach and also can be used to benchmark hydrogen generation estimates.

The nuclear industry has undertaken a sizeable effort to reevaluate the containment integrity issue. Specifically, the issue is this: For a broad range of accident types, what are the timings and modes of loss of containment integrity? The industry reevaluation covered a broad range of topics [e.g., new analyses of potential release pathways, review of hydrogen generation data and evaluation of improved systems for controlled hydrogen recombination, possible failure of safety-related equipment as a consequence of severe environmental conditions, and new analyses of the effect of "what if" scenarios (fuel melting, pressure vessel failure, core-concrete basemat interactions, etc.) on containment integrity].

Most of the containment integrity data needs of the industry are being satisfied by ongoing programs. These programs continue to generate basic data on potential containment integrity threats, such as hydrogen generation and effects of hydrogen burns on containment building equipment, accident-caused overpressurization of components, and consequent component failure. The potential contribution of the TMI-2 core examination plan in resolving the containment integrity issue is modest. The ongoing analyses of effects of the accident on the containment building environment (radiation levels, contamination locations, hydrogen burn damage, steam/water damage, etc.) and equipment and components will be the major TMI-2 contribution to the containment integrity issue. However, these activities are not within the scope of this plan. The role of the TMI-2 core examination plan is to analyze the fuel-related aspects of this issue. For example, identifying the extent and types of once-molten core materials (control, structural, and fuel materials) will help resolve vessel breach concerns. Thorough sampling of the TMI-2 core, followed by analyses of the extent of metal oxidation, will permit calculations of the amount of hydrogen generated to complement calculations based on the measured containment building pressure pulse during the hydrogen burn. The amount of hydrogen generated as a consequence of stainless steel oxidation also should be determined by sampling. Even though hydrogen generation from steel previously has been discounted, the large quantities of steel in and around the core and the stainless steel damage (indicative of high temperatures) revealed by the recent television camera inspections suggest that this phenomenon should be investigated.[4] Similarly, recent suggestions that the hydrogen gas release in TMI-2 may have been less than expected (as a consequence of hydrogen retention in the zircaloy cladding) can be studied by measuring residual hydrogen in core debris samples.[5]

4.4 Recriticality and Segregation of Fuel and Control Materials

Commercial reactors have extensive procedures to ensure criticality safety during operations such as fuel movement, fuel reloading, and spent fuel storage. The inherent design of the LWR core--low enrichment, fixed fuel geometry, presence of borated water--makes unexpected criticality very unlikely. The apparent extent of core damage in TMI-2, however, has caused reexamination and analysis of the possibility of recriticality during or following severe fuel damage accidents. Data on core material relocation phenomenology are needed to resolve this issue.

It must be emphasized that a recriticality did not occur in TMI-2, and the possibility of recriticality has been eliminated for all credible post-accident core geometries by the addition of extra boron to the primary system water. However, the possibility of recriticality was seriously examined when the magnitude of core damage became clear after the TMI-2 accident. Calculations indicating that fuel liquefaction and control rod failure probably occurred caused concern that fuel and control materials could separate, and moderated concentrations of fuel could accumulate (for example on the bottom of the reactor vessel). Thus, the recriticality issue is this: During severe LWR accidents, can core damage phenomena cause segregation of fuel and control materials, and can such segregation lead to recriticality?

Resolution of this issue will come from a thorough understanding of both fuel rod and control rod damage phenomena. Fragmentation, liquefaction, and melting of the UO_2 fuel could result in fuel relocation and accumulation away from control rod or poison materials. Similarly, control rod fragmentation or melting could cause separation of the control materials from the fuel assembly. The behavior of control materials should be carefully examined in TMI-2 because of the type of control rods used (Ag-In-Cd alloy clad in stainless steel). The Ag-In-Cd is the lowest melting component in the core (1060 K) by a substantial margin. Even though the stainless steel cladding melts at a substantially higher temperature, it is susceptible to oxidation and loss of ductility in the high-temperature steam environment of an LWR accident. Cladding failure could cause loss of the control rod alloy, either by molten material relocation or expulsion as a consequence of high vapor pressure of the alloy. The Ag-In-Cd control rod design is no longer the standard product of any U.S. reactor vendor. Even though this design has been supplanted by designs less susceptible to high-temperature deterioration, it will still be used in many LWRs for some time; therefore, it is important to determine its accident behavior.

Data needed to resolve the recriticality question will come from laboratory investigations, test-reactor experiments, and the TMI-2 core examination. Well-controlled experiments on Ag-In-Cd control rod behavior in an accident environment are needed to understand control rod failure mechanisms. Laboratory and test-reactor experiments, some of which are underway, will lead to an understanding of the tendency toward fuel segregation during an accident. Finally, the examination of TMI-2 will provide information on the integral and large-scale behavior of fuel and control material. During defueling, documentation of damage phenomena relevant to recriticality is important (e.g., molten material relocation, fragmentation, loss of geometry, and debris accumulation). Subsequent detailed examination of specific debris specimens will lead to an understanding of the possibility of recriticality during severe accidents.

4.5 10 CFR 50.46 Issues

The Code of Federal Regulations (CFR) describes the acceptance criteria for a LWR emergency core cooling system (ECCS).[6] CFR Title 10 Part 50.46 states a number of straightforward criteria (peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, and core coolability considerations) for the acceptable performance of an ECCS. An operating license applicant must evaluate performance of the proposed ECCS using analysis guidelines presented in Appendix K of Part 50 and demonstrate that it meets these acceptance criteria. Several technical issues underlie the ECCS acceptance criteria, but historically the two most significant have been zircaloy cladding oxidation and deformation.

The extent of oxidation during a design-basis LOCA has been the subject of considerable debate and analysis for several reasons. Zircaloy oxidation is an exothermic process. The heat generated heats the metal, which in turn increases the oxidation rate. Such a positive feedback phenomenon can make oxidation very rapid and, therefore, difficult to control or terminate. Oxidation of zircaloy by steam during an accident also releases hydrogen and embrittles the metal. Extensive embrittlement could lead to fracture of the cladding, and the consequent loss of rod-like geometry could produce a difficult to cool core.

The technical issue of zircaloy cladding deformation results from the phenomenon of cladding ballooning during a LOCA. Reactor depressurization can result in the primary system pressure being lower than the internal gas pressure inside the fuel rods. As the cladding heats up, it can swell (balloon) and thereby reduce the coolant flow path area between the fuel rods. Extensive ballooning could reduce this subchannel area enough to block parts of the core from receiving adequate primary system or ECCS waterflow, thereby contributing to increased local damage. Additionally, such cladding deformation could cause fuel rod rupture and release of fission gases, and alter the temperature and oxidation behavior of the core.

TMI-2 data could be useful in providing large-scale validation of current expectations in the areas of zircaloy cladding oxidation and deformation. Specifically, examination of TMI-2 could contribute to resolution of these issues because it could produce (a) the type of oxidation and deformation data that have never been generated (i.e., very large-scale data in which core-wide behavior can be studied) and (b) evidence of available three-dimensional flow paths for cooling in damaged portions of the core. This type of data is needed to help resolve remaining questions on these issues. The extensive laboratory and test reactor experiments on oxidation and ballooning have been limited by their size. As discussed previously, a large LWR can be cooled in so many ways that it is likely that cladding deformation will not prevent core cooling. Thus, the ballooning phenomenon may very well be localized and not lead to undercooling and oxidation of the core. In TMI-2, it should be possible to document both the extent and local variations of cladding deformation and oxidation, particularly in the outer regions of the core.

The five major safety issues discussed above and their underlying data needs are summarized in Table 2.

TABLE 2. MAJOR NUCLEAR SAFETY ISSUES AND THEIR UNDERLYING DATA NEEDS[a]

1. Fission Product Release, Transport, and Deposition

   a. Retention in fuel
   b. Chemical states (particularly I, Cs, Te, Ru, Sr, U, Pu)
   c. Aerosol generation
   d. Temperature distribution in the core and upper plenum
   e. Fuel relocation in the primary system
   f. Deposition on surfaces
   g. Deposition in balance of reactor coolant system and other parts of the plant

2. Core Coolability and Understanding Core Damage Processes

   a. Material relocation
   b. General debris characterization (permeability, porosity, packing density, stratification, etc.)
   c. Extent of oxidation
   d. Melting and liquefaction
   e. Fragmentation and embrittlement
   f. Deformation
   g. In-core instrument survivability

3. Containment Integrity

   a. Extent of oxidation
   b. Evidence of major accumulation of core materials in the lower plenum
   c. Integrity of lower reactor vessel head

4. Recriticality and Segregation of Fuel and Control Materials

   a. Location and configuration of fuel and control materials

5. 10 CFR 50.46 Issues

   a. Ballooning
   b. Oxidation

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a. The major nuclear safety issues listed are prioritized, based on their relative order of importance. Underlying data needs associated with each major safety issue are not prioritized.

5. SUITABLE APPROACHES FOR OBTAINING DATA FROM TMI-2

Having identified the types of data needed, the next step in developing technical data acquisition recommendations was to identify suitable approaches for obtaining data, using the criteria identified in Section 3. However, before identifying the specific _in situ_ measurements and/or sampling (for off-site analysis) approaches that are consistent with the many practical constraints on what can be done, it is useful to consider how the needed data could, in principle, be obtained. Table 3 summarizes the types of data needed for resolution of each of the major issues discussed in Section 4 and how each type of data could be obtained. This table shows that the needed data can be obtained only through a combination of _in situ_ measurements and sample acquisition for subsequent off-site analysis.

The remainder of this section discusses some of the information that is pertinent to developing specific approach recommendations using the criteria identified in Section 3. Specifically, the following subsections discuss (a) pertinent details of the TMI-2 accident sequence, (b) available information on the as-built and current damage state of the plant, (c) available and pertinent information on plant recovery plans, and (d) some techniques available for obtaining data from _in situ_ measurements and/or off-site analyses. Each area is discussed herein to provide some insight into the pertinent information environment in which the recommendations were developed.

5.1 TMI-2 Accident Sequence

Details of the TMI-2 accident sequence have been discussed in several reports,[3,7] as has interpretation of the sequence of events during the accident in terms of the core water level and associated core temperature transient.[8,9] Core damage and fission product release also have been discussed by several authors.[2,10] Herein, only a brief overview is

TABLE 3. SUMMARY OF HOW THE NEEDED DATA COULD BE OBTAINED

Anticipated Use	Type of Data Sought	How Data Could Be Obtained	Comments
Issue 1. Fission Product Release, Transport, and Deposition:			
Identify phenomena controlling release from fuel pellets	Fission gas bubble concentration, fuel porosity; porosity interlinkage, cracking; fission product distribution in fuel microstructural features	Sample fuel from core damage regimes and examine microstructure using metallographic techniques. Examine fuel samples using electron and/or ion microprobe	These data would lead to a better understanding of fission product release from the fuel UO_2 matrix during early phases of core degradation
	Identification of processes that lead to release of fission products or control materials from more severely damaged fuel	Sample visually significant debris material. Gamma scan and radiochemical analysis, SIMS, ESCA, Auger, etc., depending on inventories of materials involved	
Support core source term model assumptions	Molecular forms of fission product and core materials released from the core	1. Infer from molecular forms of residual materials in core samples (SIMS, ESCA, Auger, MOLE)	
		2. Infer from molecular forms of surface and debris material outside core region. SIMS, ESCA, Auger, MOLE analysis of the retrievable surface and debris samples	See Section 5.2 for types of samples available
Support core source term model development	Fission product and control material distributions and chemical forms in residual core materials	Gamma scan and radiochemical analysis of core material samples containing significant fission product inventories	Because TMI-2 was not a controlled experiment, it is not an ideal source of this information
Support code input requirements	Temperature and temperature history data for various core regions	Sample core materials, infer from material melting, microstructural changes, material oxidation, fission product distributions, using visual, metallographic, electron microprobe, and gamma scanning.	Much of the necessary data will have other uses
Identify primary system deposition and release phenomena	Fission product and control material elemental concentration map on the RCS surfaces	Gamma scan and/or sample RCS surfaces. Examine surface deposits for elemental concentrations using SIMS, SEM/WDX, and radiochemisty	Because of the extrapolation that would be necessary to infer deposit distributions from sample data alone, this should be supplemented with *in situ* gamma scanning where possible
	Vapor/surface and aerosol/surface interactions	Characterize deposit tenacity on surfaces. Optical and scanning electron microscopy. Obtain molecular forms and deposit depth profiles	

TABLE 3. (continued)

Anticipated Use	Type of Data Sought	How Data Could Be Obtained	Comments
Issue 1. Fission Product Release, Transport, and Deposition: (continued)			
Support assumptions in primary system transport and deposition models	Deposit variations on RCS surfaces	In situ gamma scans of surfaces. Sample surface elemental profiles from SIMS and SEM	
	Variations with position of the chemical composition of deposits	Molecular forms of surface deposits from SIMS and MOLE	
	Chemical composition of individual particles and variation with size	Ion or electron microprobe analysis of individual particles in surface deposits or debris	Should be done for deposits or debris removed from different locations
Validation or benchmarking primary system transport models	Deposit concentrations on those surfaces that were uncovered during the core uncovery period	Sample RCS surfaces. Analyze sample surfaces for elemental deposits using gamma spectroscopy, radiochemical analysis, or electron and ion microprobe. Conduct in situ gamma scans to supplement sample data	An extensive data base for RCS surface deposits is required to adequately map the deposit distribution. Also, thermal-hydraulic input data will be necessary to support code calculations
Support code input requirements	Temperature and temperature history data for RCS surfaces	Sample RCS materials. Infer temperature data from detailed examination of material properties (e.g., microstructure, melting, and oxidation)	
Identify natural deposition phenomena in the reactor	Vapor/surface and aerosol/surface interactions for the reactor building coolers and mirror insulation	Sample reactor building cooler coils and mirror insulation panels. Obtain deposit inventory and form (if possible) using gamma spectroscopy and SIMS	Current evidence would suggest that the deposit concentrations on reactor building surfaces are lower than the sensitivity of most nonnuclear surface analysis techniques
Benchmark pool scrubbing models	Deposit inventories on drain tank inlet and outlet surfaces. Elemental composition and quantity of drain tank debris	Sample surfaces and do surface analysis using gamma spectroscopy and SIMS, ESCA, Auger, SEM/WDX	
Check evidence that individual isotopes followed overall elemental behavior	Isotope ratios for the deposits and debris in RCS	Analyze surface samples for individual isotopes using gamma spectroscopy, surface elemental analysis, and SIMS	
Check fission product elemental groupings	Ratios of fission product elements in deposits at different locations	Analyze surface samples for individual elements	

TABLE 3. (continued)

Anticipated Use	Type of Data Sought	How Data Could Be Obtained	Comments
Issue 2: Core Coolability and Understanding Core Damage Processes:			
Identify phenomena that impact core coolability and damage progression	Material relocation	*In situ* macroscale photo/visual examination of the current core damage state. Metallography and composition of relocated material samples. Photo-visual examination of materials from lower reactor vessel plenum	Special attention should be paid to cladding, control material, and liquefied fuel relocation
	General debris characterization	Permeability of debris bed strata (stratifiction samples). Porosity of debris bed strata (stratification samples). Packing density (stratification samples). Photo/visual and metallography of stratification samples	
	Extent of oxidation	Sample materials from core damage regimes and measure oxide layer thickness	
	Melting and liquefaction	*In situ* photo/visual examination of core and lower reactor vessel plenum before and during cleanup. Photo/visual, metallography, and composition of removed samples	The *in situ* measurements should be done at selected stages in the cleanup (e.g., before and after core debris bed vacuuming)
	Fragmentation and embrittlement	Macrostructure, particle size, and composition of core samples. Oxidation, hydrogen content, and mechanical properties of core material samples	
	Deformation	*In situ* photo/visual examination. Macrostructure of samples from selected core damage regimes	
	In-core instrument survivability	Sample in-core instrument strings	Such samples also are needed for temperature information
Issue 3. Containment Integrity:			
Clarification of phenomena that could lead to vessel breach	Extent of oxidation	*In situ* photo/visual examination. Amount of oxide layers on samples of zircaloy and stainless steel removed from the various core damage regimes and selected plenum locations	Extensive sampling and analysis could be necessary to get a complete picture

TABLE 3. (continued)

Anticipated Use	Type of Data Sought	How Data Could Be Obtained	Comments
Issue 3. Containment Integrity: (continued)			
	Evidence of major accumulation of core material in lower plenum	*In situ* photo/visual examination of the lower plenum. Sample and obtain composition of the materials found	
	Lower reactor vessel head integrity	*In situ* photo/visual examination in lower plenum	
Issue 4. Recriticality and Segregation of Fuel and Control Materials:			
Clarification of phenomena that could result in fuel and control material segregation	Location and configuration of fuel and control materials	*In situ* photo/visual examination. Macrostructure and composition of large samples removed from various damage regions of the core. Composition, amount, and configuration of materials in selected primary loop locations (e.g., lower reactor vessel, lower OTSG plenum, upper OTSG tube sheet)	Requires both *in situ* photo-visual and sample acquisition
Issue 5. 10 CFR 50.46 Issues:			
Validation of current expectations	Clad ballooning	*In situ* photo/visual examination. Sample ballooned rods (preferably assembly portions of ballooned rods)	
	Oxidation	Sample fuel rod and other assembly components and measure oxide layer thickness	

presented of those accident sequence events that were particularly significant from the point of view of the five major safety issues discussed earlier.

The critical time period of the accident sequence contributing to core damage progression and fission product release and transport behavior generally is believed to be between 103 and 210 minutes after the reactor tripped on 28 March 1979 (see References 3 and 7). The 103-minute time corresponds to the beginning of core uncovery following phase separation of the primary coolant when the last of the reactor coolant pumps was turned off in Loop A at 101 minutes. The 210-minute time corresponds to the approximate time of core refill following the resumption of sustained high-pressure injection, which occurred at about 200 minutes and resulted for the most part in termination of the core heatup. This period, therefore, defines the period of interest from the point of view of obtaining data pertinent to the scope of the five issues presented in Section 4. During this period, several events occurred in the sequence that are pertinent to the scope of this document. At 131 minutes, the hot-leg temperatures went off-scale. At 135 minutes, the reactor building air sample particulate monitor went off-scale, indicating some core damage. At 142 minutes, the operators closed the pilot-operated relief valve (PORV) block valve. Following additional radiation detector responses which indicated significant core damage, Reactor Coolant Pump 2B was started and run for a short period, forcing water through the core. The PORV block valve was reopened for a period of approximately 5 minutes at 192 minutes. This sequence of events, which defines the accident time period of interest and identifies the escape pathways to the containment building, is summarized in Table 4.

Several estimates of the core time-temperature relationship have been developed (see References 3, 8, and 9). These will not be discussed here other than to point out that there are significant differences in the estimates. Since the time-temperature relationship in the core for the period of interest is very important in comparing core damage and fission product release data with calculations, it is important to obtain data on the time

TABLE 4. SUMMARY OF PERTINENT EVENTS IN THE TMI-2 ACCIDENT SEQUENCE

Time (min)	Event
101	Last reactor coolant pumps turned off in Loop A
131	Hot-leg temperatures went off-scale
135	Reactor building air sample particulate monitor response went off-scale
142	PORV block valve closed
145	The reactor building air sample (HP-P-227) iodine channel began to increase rapidly
174	Reactor Coolant Pump 2B was started and run until 3 hr, 13 min into the accident
192	The PORV block valve was opened for about 5 min, decreasing primary system pressure and pressurizer coolant level
200	Sustained high-pressure injection and core reflooded

and temperature history through the TMI-2 core examination. Temperature and temperature history data also are important for the reactor coolant system surfaces, because surface temperature is a key parameter in attempting to understand the fission product deposition and resuspension behavior for these surfaces.

5.2 As-Built and Current Damage State of TMI-2

Since it is clear from Table 3 that the needed data only can be obtained by *in situ* measurements, sampling for off-site analysis, or a combination of both, it is necessary to review the pertinent plant systems to determine which data acquisition activities are practical. Information concerning the current state of these systems can be inferred from the as-built state and available information on damage caused during the accident. The pertinent systems discussed below include the reactor vessel head and service structure, reactor vessel internals, core, hot- and cold-leg piping, once-through steam generators (OTSGs), surge line and pressurizer, drain tank and associated piping, and auxiliary systems. These are discussed to the extent necessary to identify retrievable samples and/or possible *in situ* measurements. This document assumes that samples will not be available that might compromise the integrity of the primary pressure boundary for possible future plant requalification. The following subsections provide a potential list of retrievable artifacts and *in situ* measurements. [Criteria applied in identifying retrievable artifacts were as follows: (a) the artifacts can be used to satisfy identified data needs, (b) integrity of the RCS not be compromised, and (c) acquisition and examination be practical.] Reference to these subsections can, therefore, provide a basis for identifying alternatives or additional useful artifacts should some of the recommendations presented in Section 6 not prove feasible or should additional resources become available.

5.2.1 Reactor Vessel Head and Service Structure

The parts of the reactor vessel (RV) head and service structure of interest to the TMI-2 core examination plan are the surfaces that were in contact with the primary coolant. Visual inspection has shown that the RV head is, at least on a macroscopic scale, essentially in the as-built condition. The RV head is constructed of carbon steel, the inside of which is covered with a weld deposit of austenitic stainless steel, which is the main material that constituted the bulk of the surface in contact with the primary coolant. The head contains 69 nozzles to which are welded the flanged adaptors constructed of austenitic stainless steel for attaching the control rod drive mechanisms (CRDMs). The head also contains eight thermocouple connections with the same type of construction and materials as the connections for the CRDMs. The major components of the CRDMs that were within the RV and exposed to fission product vapors or aerosols during the core uncovery period were the CRDM leadscrews and the leadscrew support tubes. The leadscrews, which couple to the control rod spiders, extended to the assembly upper end fittings during the critical core damage period of the accident. A schematic of a leadscrew is shown in Figure 2. Figure 3 depicts a typical CRDM and the associated leadscrew and support, or guide tube. The guide tubes are constructed of Type 304 stainless steel. Figures 4 and 5 show the general arrangement of the RV head and service structure components. The following retrievable artifacts that might be obtained for examination are:

- Thermocouples and flanges
- Leadscrew support tubes
- Leadscrews
- Main RV flange gasket
- Debris from the RV flange area.

In addition to samples or artifacts that might be available for off-site analysis, it appears that it is possible to obtain useful deposit-distribution data by gamma scanning the underside of the RV head after the CRDMs have been removed. Axial gamma scans of the leadscrews would help map deposit distributions in the upper plenum region.

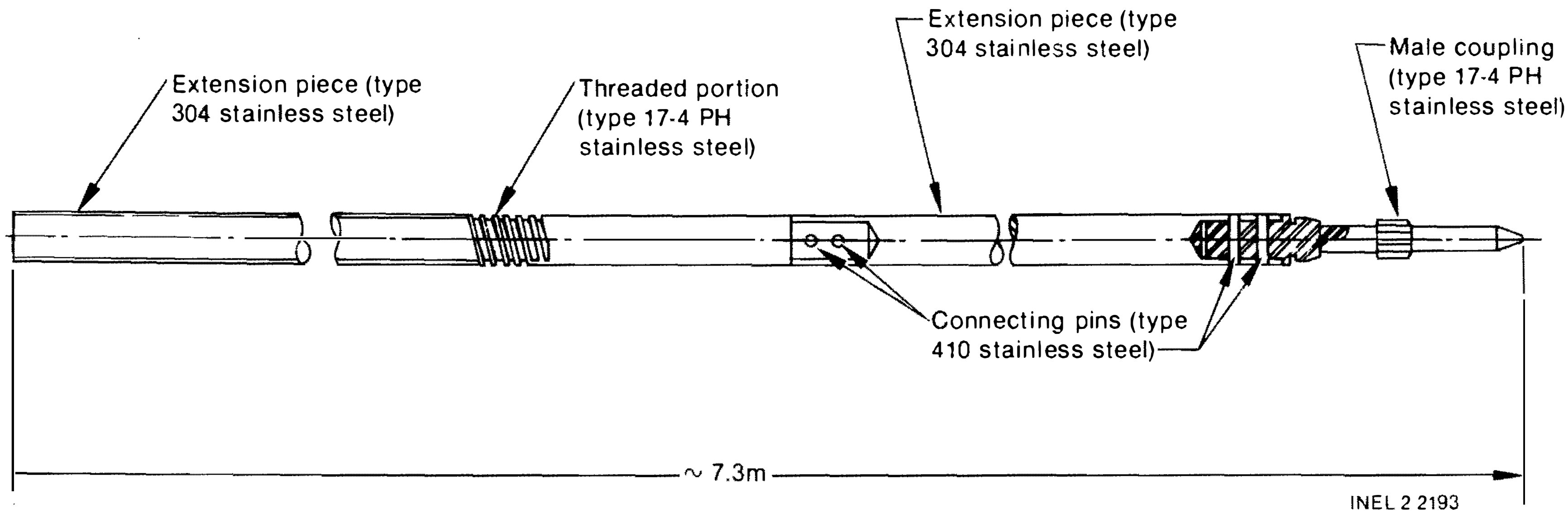


Figure 2. Schematic of a control rod drive leadscrew.

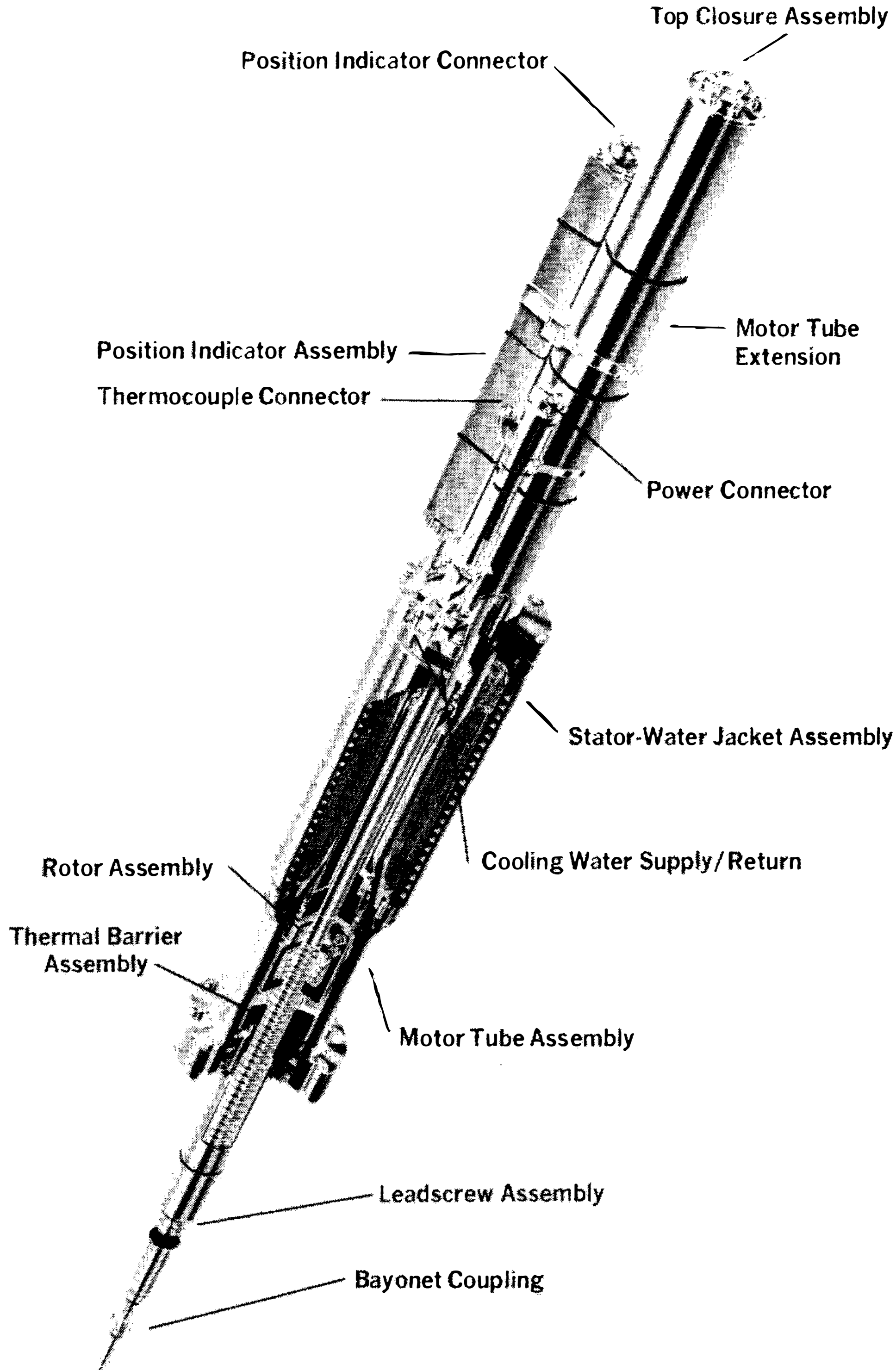


Figure 3. Typical TMI-2 control rod drive mechanism.

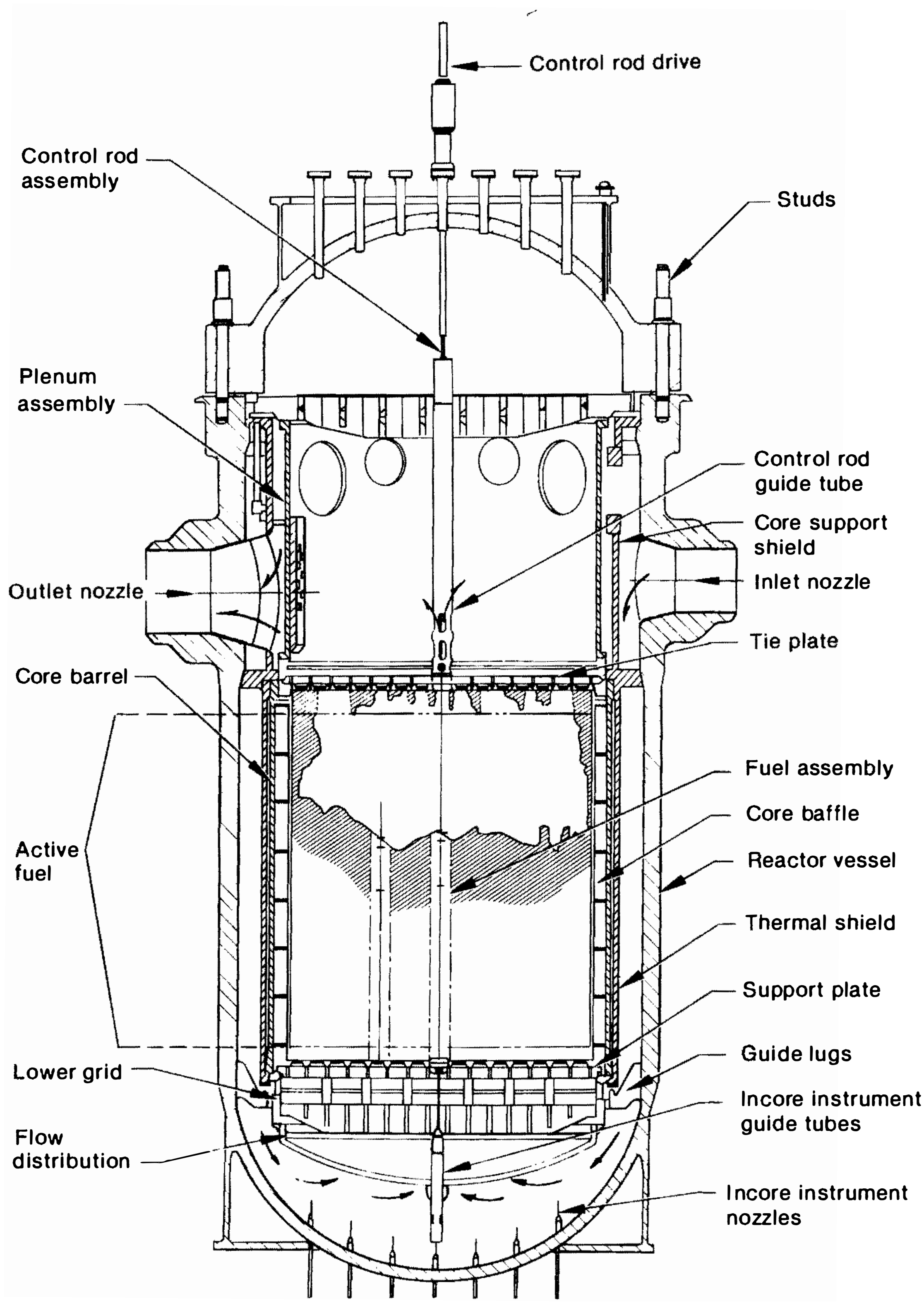


Figure 4. Longitudinal cross section of TMI-2 reactor vessel and internals.

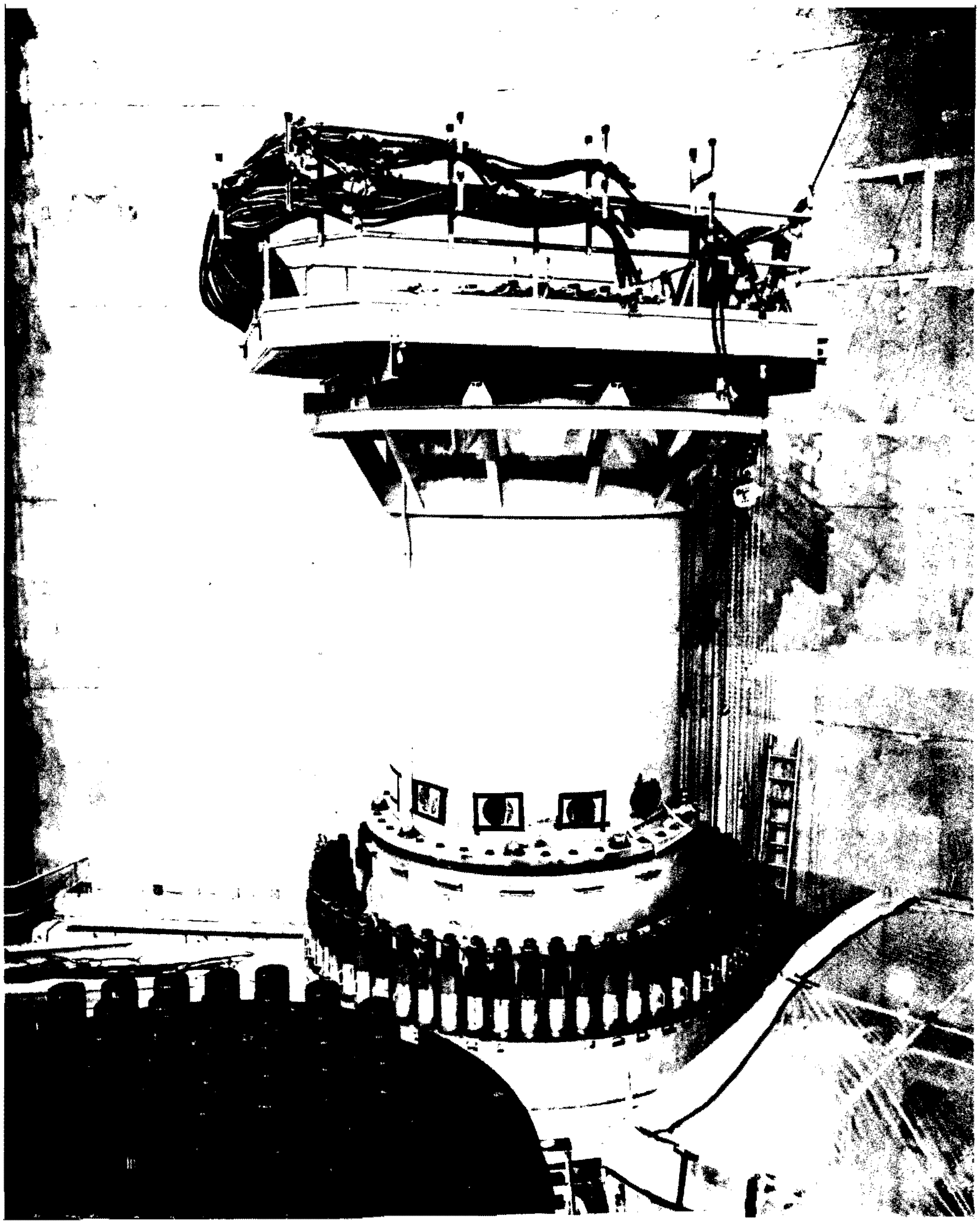


Figure 5. TMI-2 reactor head service structure.

5.2.2 Reactor Vessel Internals

The reactor vessel internal components include the plenum assembly and the core support assembly, which consists of the core support shield and the core barrel (including the lower grid, flow distributor, in-core instrument guide tubes, thermal shield, and surveillance capsule holder tubes). Overall arrangement of these components in the reactor vessel is illustrated in Figure 4. Visual inspections to date indicate that these components are in essentially as-built condition, at least on a macroscopic scale. Hence, the as-built condition is discussed below.

5.2.2.1 Plenum Assembly. The plenum assembly is located directly above the reactor core and is normally removed as a single component before refueling. Pictures of the plenum assembly from two different angles are shown in Figures 6 and 7. The plenum assembly, because of its large surface area, is a key component for understanding fission product behavior in the primary system. It consists of a plenum cover, upper grid, control rod guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is a series of parallel flat plates, intersecting to form square lattices, with a perforated top plate and flange; it is attached to the top flange of the plenum cylinder. The flat lattice plates are 2-inches thick and vary in depth and length from 5-1/2 and 87 inches to 18 and 160 inches, respectively. The perforated top plate is 1/2-inch thick and 124 inches in diameter and has 69 holes of 8.520-inch diameter. The plenum cover is attached to the top flange of the plenum cylinder by a 140-inch-OD flange, 5-inches wide and 1-1/4-inches thick. Three lifting lugs are provided for handling the plenum assembly. These lugs, 17-1/4-inches high, 3-inches thick, and 3-inches (minimum) wide, are welded to the lattice systems of the cover. The control rod assembly (CRA) guide tubes are welded to the plenum cover top plate and bolted to the upper grid. Control rod guide assemblies provide guidance and protect the CRA from the effects of coolant crossflow, as well as providing structural attachment of the grid assembly to the plenum cover.

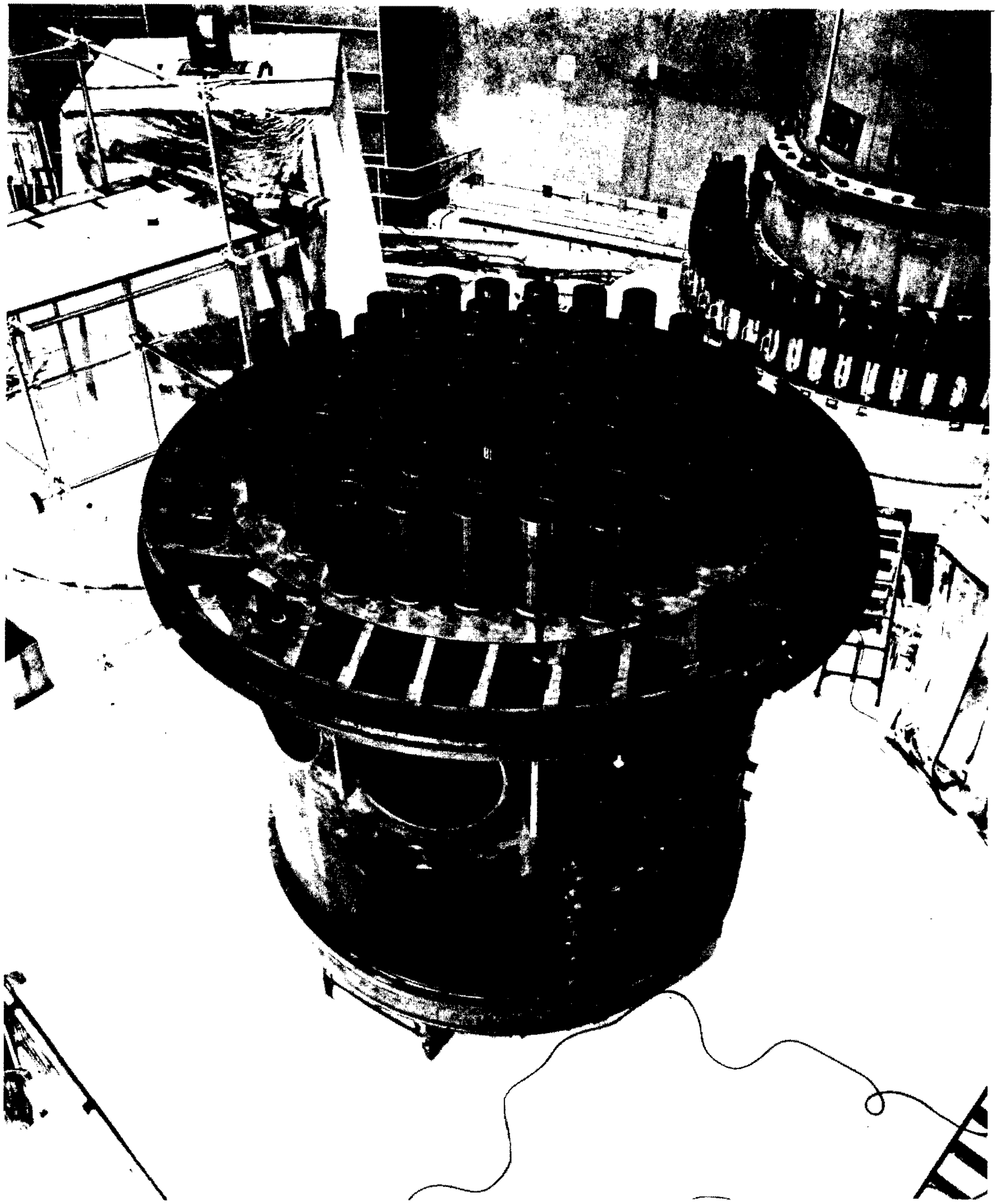


Figure 6. TMI-2 plenum assembly.

Figure 7. TMI-2 upper core tie plate and plenum assembly.

Each CRA guide assembly comprises an outer tube housing, mounting flange, 12 perforated slotted tubes (split tubes), and four sets of tube segments ("C" tubes), which are properly oriented and attached to a series of castings to provide continuous guidance and support to the CRA. The outer tube housing, 8-inch schedule 40 pipe, is welded to a 3/4-inch-thick mounting flange, which is bolted to the upper grid. Vertical and horizontal cross sections of the guide tube assembly are shown in Figure 8.

The plenum cylinder is a 130-inch-OD, 1-1/2-inch-thick cylinder with flanges on both ends to connect the cylinder to the plenum cover and upper grid. Six 34-inch-diameter holes and four 22-inch-diameter holes in the plenum cylinder provide a flow path for coolant water.

The upper grid is a rolled plate with machined holes that locate the lower end of the individual CRA guide tube assemblies relative to the upper end of the corresponding fuel assemblies. The grid is bolted to the plenum cylinder lower flange.

All materials used in construction of the plenum are either Type 304 or Type 304L stainless steel, with the exception of the spacers in the control rod guide assembly, which are stainless steel castings. The following retrievable samples that might be obtained for examination from the plenum assembly are:

- Debris from the plenum cover plate
- Debris from the spacer castings in the control rod guide tube assemblies
- Cuttings from the cover plate [Note: Some of these may be available if holes must be made in the cover plate to knock fuel assembly upper end fittings from the upper grid.]
- Guide tube protrusions above the cover plate

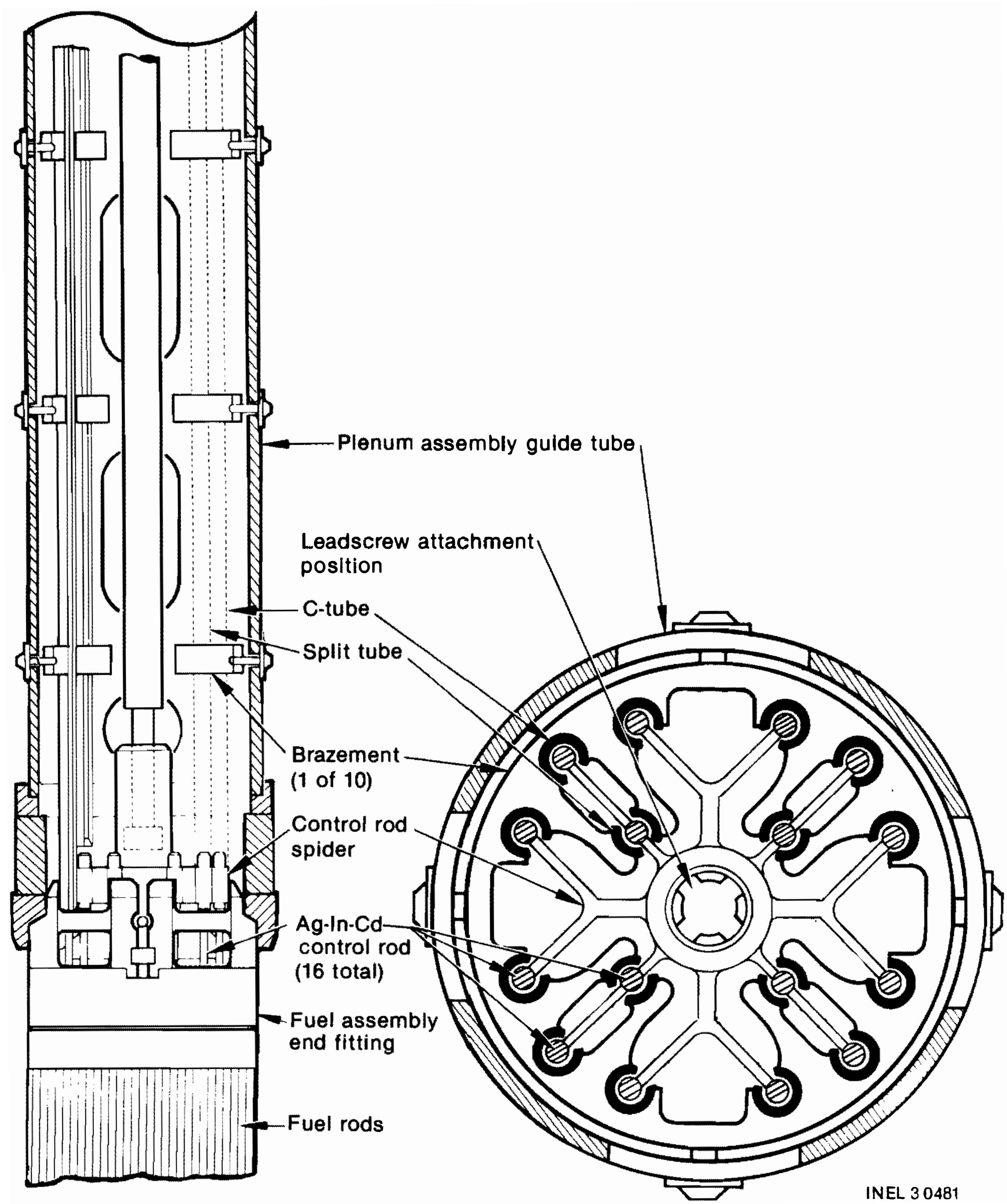


Figure 8. Vertical and horizontal cross section of plenum guide tube assembly.

- Complete guide tube assemblies

- Portions of the upper grid

- "C" or split tubes cut from the CRA guide tube assemblies.

5.2.2.2 Core Support Shield. The core support shield is a large, flanged cylinder that mates with the reactor vessel opening. The forged top flange, with a 146-inch ID and 10-7/16-inch thick, rests on a circumferential ledge in the reactor vessel top closure flange. The core support shield lower flange, with a 136-inch ID and a thickness of approximately 5 inches at the mating surface, is bolted to the core barrel.

The cylinder wall, 146-inch ID and 1-1/4-inches thick, has two nozzle openings for coolant flow. These openings are formed by two forged rings, 67-inch OD, approximately 47-inch ID, and 3-1/2-inches thick, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance during normal core support assembly installation and removal. At reactor operating temperature, the metal surfaces are in contact to form a seal without exceeding allowable stresses in either the reactor vessel or internals. The cylinder wall also has eight holes into which vent valve mounting rings are welded. These rings are approximately 7-1/2-inches thick and are 37-1/2 inches in OD.

The materials are Type 304 stainless steel, with the outlet nozzles being made of a stainless steel casting. The vent valves consist of cast stainless steel components with some 15-5 PH (precipitation-hardened) stainless steel and satellite materials.

The location of the core support shield is shown in Figure 4. Figure 9 shows the core support assembly of which the core support shield is the upper section. Retrievable samples that might be obtained from the core support

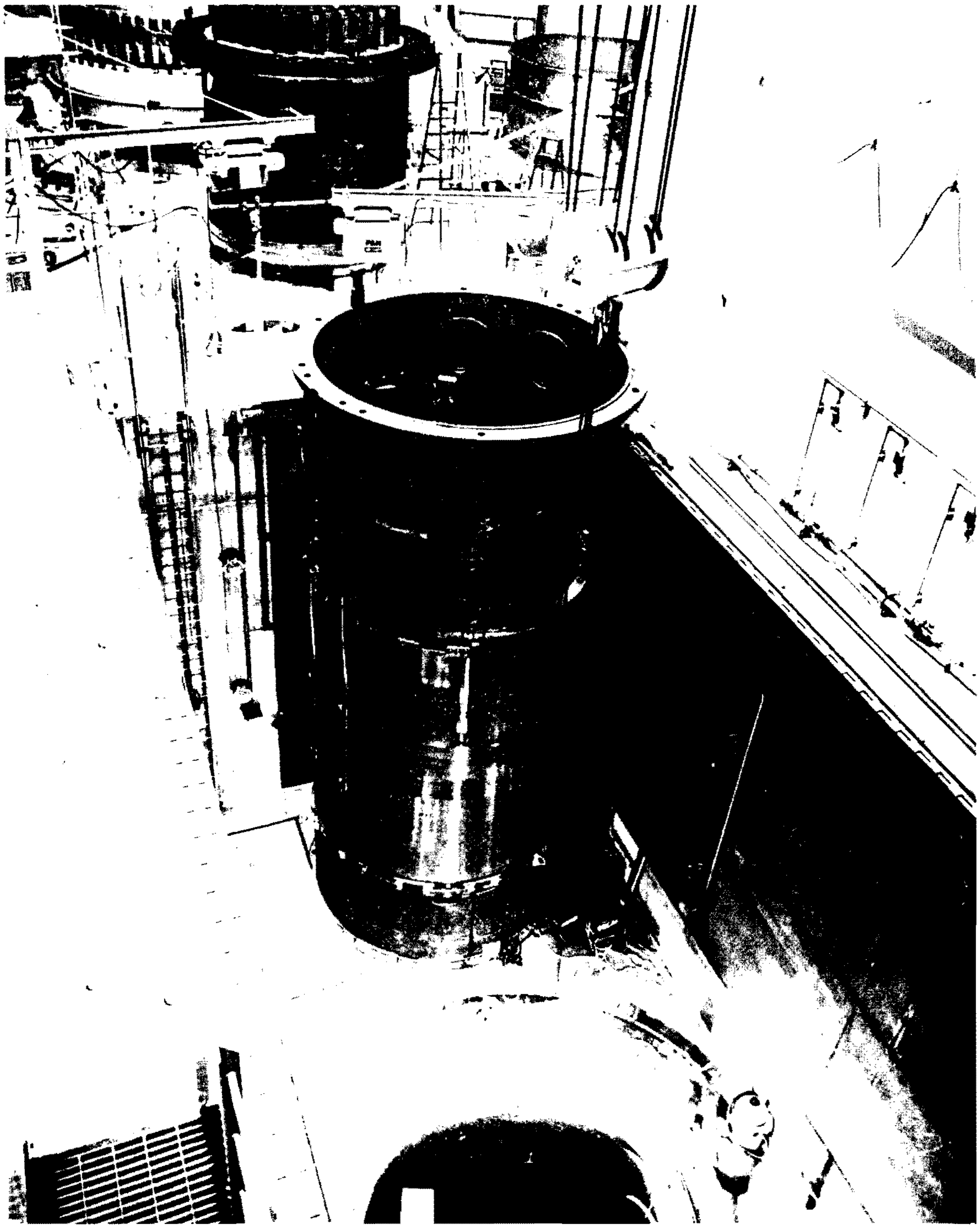


Figure 9. TMI-2 reactor internals core support assembly.

shield for examination are (a) vent valves and (b) outlet nozzle. In addition, in situ gamma scans of the core support shield cylinder after core removal may be feasible.

5.2.2.3 The Core Barrel. The core barrel consists of a flanged cylinder, a series of internal horizontal spacers bolted to the cylinder, and a series of vertical plates bolted to the inner surfaces of the horizontal spacers to form an inner wall enclosing the fuel assemblies. The core barrel cylinder has flanges on both ends and is 145 inches in OD and 2-inches thick. The upper flange of the cylinder is bolted to the mating lower flange of the core support shield assembly, and the lower flange is bolted to the mating flange of the lower grid assembly. All bolts are lock-welded after final assembly.

The horizontal spacers and vertical plates are 3/4-inch thick. Normally coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel. A small portion of the coolant flows upward through the spacer between the core barrel outer cylinder and the inner plate wall.

The lower grid assembly, flow distributor, and thermal shield are not discussed here but may be of interest to this plan if access to the components is possible. As shown in Figure 9, the surveillance capsule holder tubes are installed on the outer wall of the core support assembly to contain the surveillance specimen assemblies. The tubes extend from the top flange of the core support shield to the lower end of the thermal shield. The tubes are 3.5 inches in OD x 0.188-inch wall x 18.5-feet long with an ogee bend.

As is the case for the other reactor vessel internals, the core barrel components are fabricated from Type 304 stainless steel.

The following retrievable artifacts that might be obtained for examination are:

- o Core former plates

- o Reactor vessel surveillance capsule holder tube

- o Debris from the lower grid flow distribution plate.

In addition, it appears that in situ gamma scans of the core former plates may be possible after core removal and before or after the core barrel is removed from the reactor vessel.

5.2.3 Reactor Core

The TMI-2 core has 177 fuel assemblies (see Figure 10), which are arranged on a square lattice pitch to approximately the shape of a cylinder. Except for variation in fuel enrichment, all fuel assemblies are of identical construction and are designed to accept interchangeably any type of control assembly. The reactivity of the core under operating conditions is controlled by 61 CRAs and 8 axial power shaping rod assemblies (APSRAs). APSRAs are similar in physical configuration to the CRAs but have absorber material only in the lower portion of the rods. Except for the peripheral positions, fuel assemblies containing no CRAs or APSRAs have a burnable poison rod assembly installed into the fuel assembly guide tubes.

5.2.3.1 Fuel Assemblies. The fuel for the TMI-2 reactor is sintered pellets of low-enriched uranium dioxide inside Zircaloy-4 tubing (cladding). The cladding, fuel pellets, end caps, and fuel support components form a fuel rod. The basic fuel assembly is made up of 208 fuel rods, 16 control rod guide tubes, 1 instrumentation tube assembly, 7 spacer sleeves, 8 spacer grids, and 2 end fittings (see Figure 10). The guide tubes, spacer grids, and end fittings form a structural cage, which contains the rods and tubes in a 15 x 15 array. The center position in an assembly is reserved for in-core instrumentation. The fuel rods are supported at each spacer grid by contact points integral with the walls of the fuel cell boundary. The guide tubes are permanently attached to the upper and lower end fittings tying the assembly together.

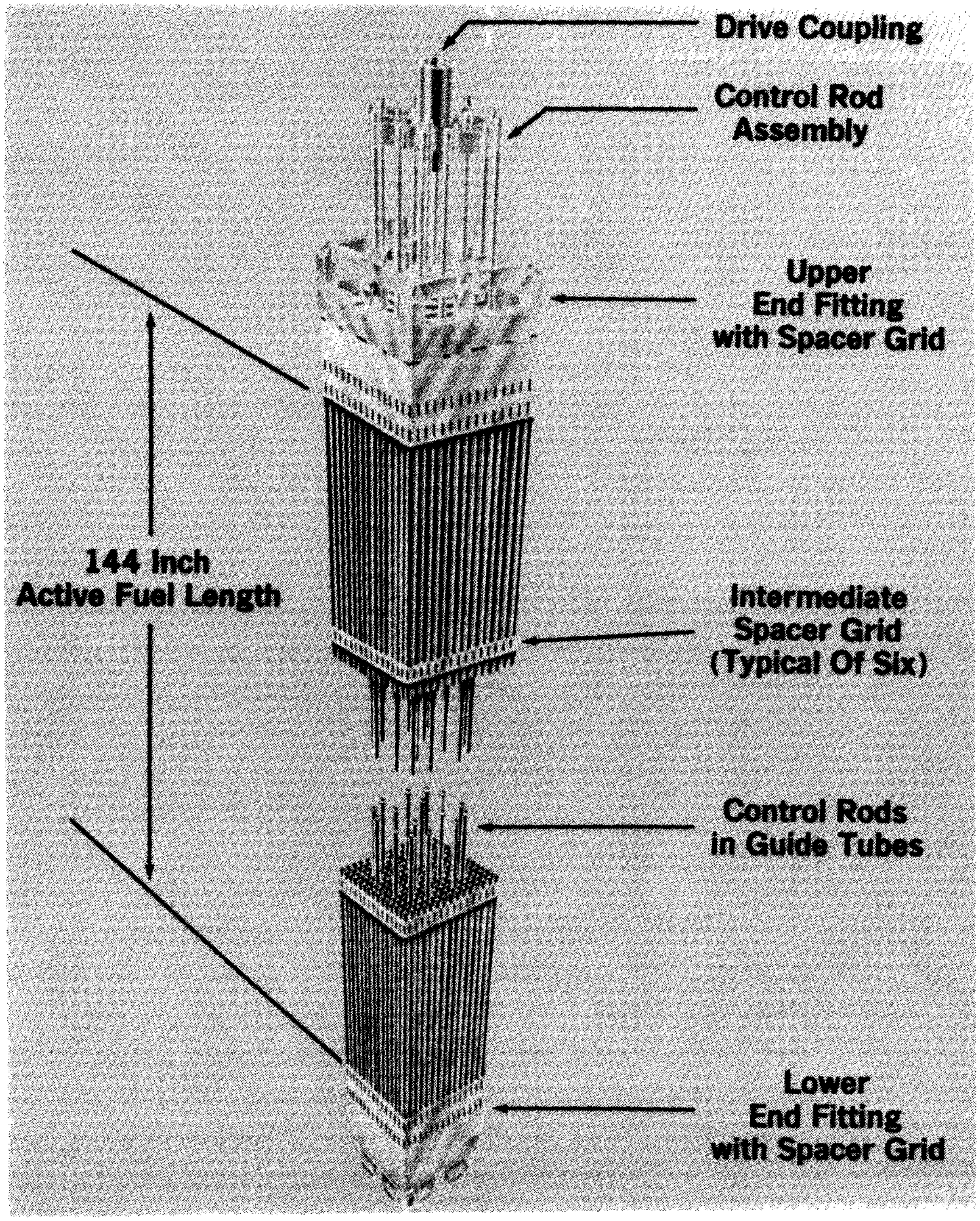


Figure 10. Typical TMI-2 fuel assembly.

The spacer grids are constructed from strips that are slotted and assembled into an eggcrate configuration. Each grid has 32 strips, 16 perpendicular to 16, which form the 15 x 15 lattice for the fuel rods. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the walls of each square opening are integrally punched dimples in the strips. The end spacer grids are longer axially than the intermediate spacer grids and are mechanically attached to the end fitting.

The lower end fitting positions the assembly when inserted into the lower core grid plate and supports the fuel assembly weight. The lower ends of the fuel rods rest on the grid of the lower end fitting. Penetrations in the lower end fitting are provided for attaching the control rod guide tubes and access to the instrumentation tube assembly.

The upper end fitting positions the upper end of the fuel assembly in the upper core grid plate structure and provides means for coupling the handling equipment. An identifying number on each upper end fitting provides positive identification during handling.

An internal hollow post in the center of the upper end fitting provides means for retention of the orifice rod assembly and burnable poison rod assembly. Attached to the upper end fittings is a fuel assembly holddown package, consisting of a spring and holddown spider, which provides a positive holddown margin to oppose hydraulic forces. Penetrations in the upper end fitting grid are provided for the guide tubes and instrumentation.

The guide tubes provide continuous guidance for the control assemblies when inserted into the fuel assembly during operation, and provide the structural continuity for the fuel assembly. A threaded sleeve and threaded plug are welded to the ends of the guide tube and are used to attach the guide tubes to the end fittings by lock-welded nuts.

The instrument tube assembly serves as a channel to guide, position, and contain the in-core instrumentation in the center of the fuel assembly. The instrumentation string is guided up through the lower end fitting to the desired core elevation.

Each control rod assembly has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to one side of the spider, and the female coupling is attached to the other side. The control rod drive connects to the CRA at the female coupling by a male-female connection. On the side of the spider opposite the control rod drive connection, the 16 control rods are tightly connected. When the CRA is inserted into the upper end fittings of the fuel assembles, the rods are guided by the guide tubes of the fuel assembly. Full-length guidance for the CRA also is provided in the upper plenum assembly by the control rod guide assembly.

Each control rod contains a section of neutron absorber material--an alloy of silver-indium-cadmium (Ag-In-Cd) clad in cold-worked, Type 304 stainless steel tubing. The Type 304 stainless steel end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The stainless steel tubing provides the structural strength of the control rods, prevents corrosion of the absorber material, and eliminates possible silver contamination of the reactor coolant. Above the absorber material is a spacer which keeps the absorber material in place during shipping and handling.

Each axial power shaping rod assembly is configured like a control rod assembly. The major difference is internal, with 3 feet of Ag-In-Cd poison material instead of 12 feet.

Each burnable poison rod assembly has 16 burnable poison rods, a stainless steel spider, and coupling mechanism. Each burnable poison rod has a section of sintered Al_2O_3-B_4C pellets whose ability to absorb neutrons decreases with exposure. The burnable poison is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material.

5.2.3.2 Television Camera Inspection of the Core. From the viewpoint of core data acquisition, the most important early core examination task has been the continued closed-circuit television (CCTV) camera inspections of the core condition. These inspections have been performed by lowering a small-diameter camera down through vacated control rod drive mechanisms into the core void region (Figure 11). The camera was then rotated from its straight-down position to nearly straight up at several radial orientations. CCTV inspections have been conducted at three locations (shown in Figure 12) and have yielded a wealth of visual information, both direct and inferred, on damage to the core and reactor internals. These data, together with the as-built core information, are essential to understanding the current core state and possible approaches to its examination.

5.2.3.3 Core Topography Measurements. In order to measure the core topography before alterations occur, an ultrasonic core topography system was built and operated in the TMI-2 reactor vessel. A transducer/detector range-finding system was inserted into the core cavity through the leadscrew opening in the central (H8) position. Using a scanning system, the range finder was moved axially within the core cavity (see schematic, Figure 13) and measured the height, depth, and location of topographic features with an accuracy of a few centimeters. The data was collected rapidly--taking a few hours for the entire TMI-2 core void.

The core topography data indicate that the void in the core region below the upper grid plate occupies 330 ft^3 (9.3 m^3) and extends radially into the peripheral row of fuel assemblies. Local variations in the nominal void radius range from exposed sections of core former wall to apparent standing fuel rods 12- to 14-inches inside the core former boundary. Significant quantities of core materials are suspended from the underside of the upper core support grid. The materials range from short stubs, interpreted as end fittings, to long, suspended lengths of fuel rods. The floor of the core void

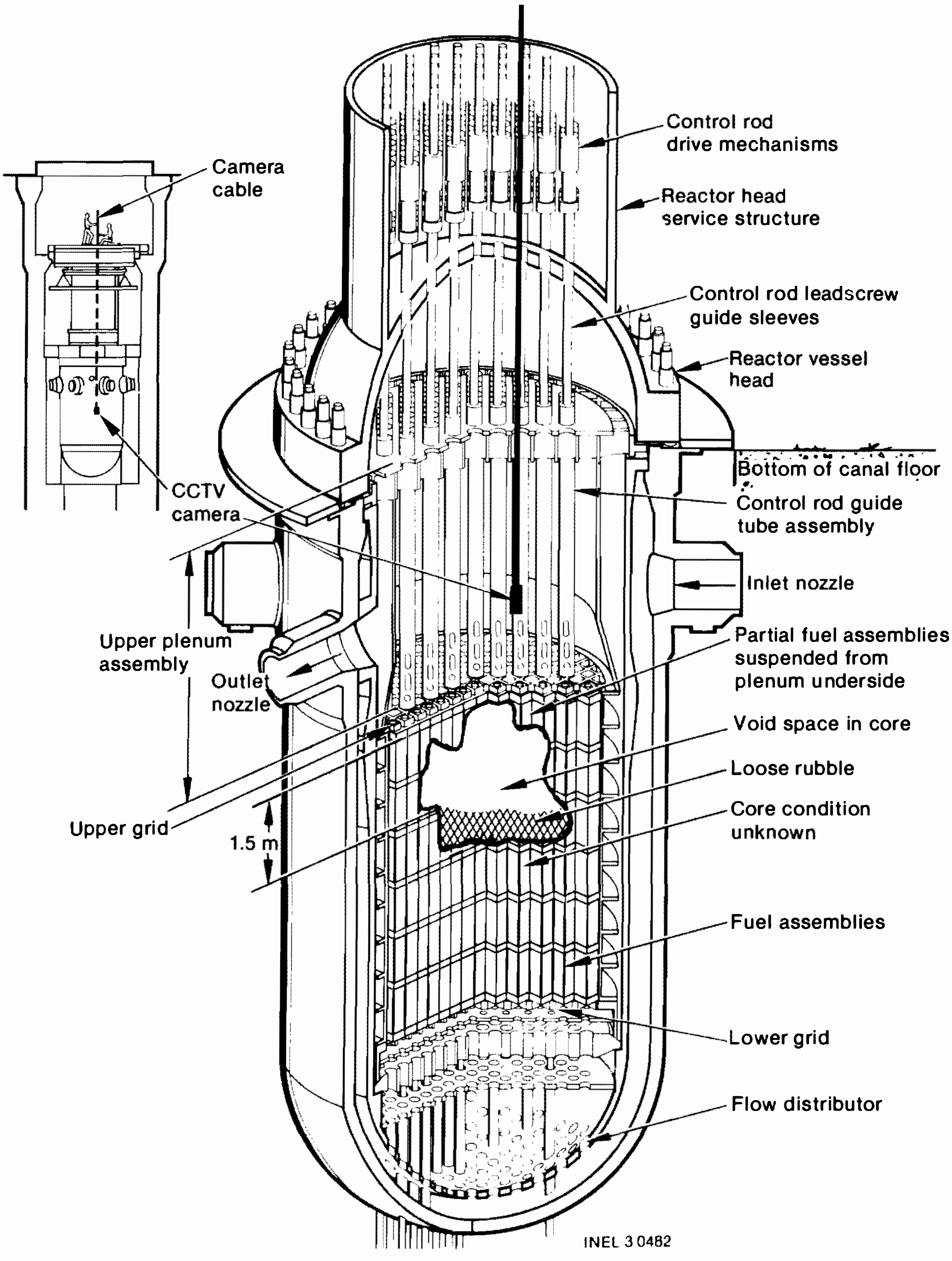


Figure 11. Closed-circuit television camera being lowered into the TMI-2 reactor.

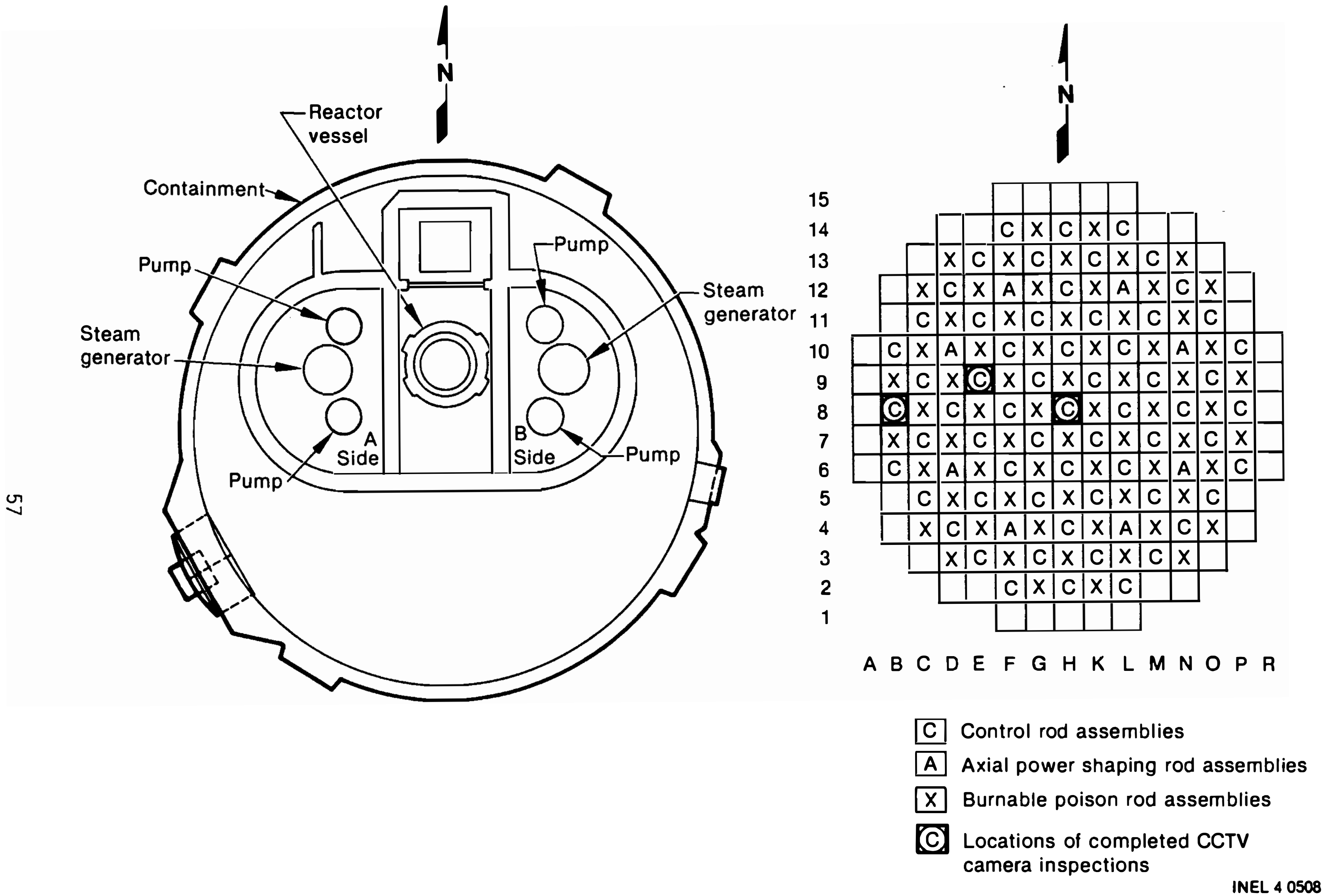


Figure 12. Locations of TMI-2 closed-circuit television camera inspection.

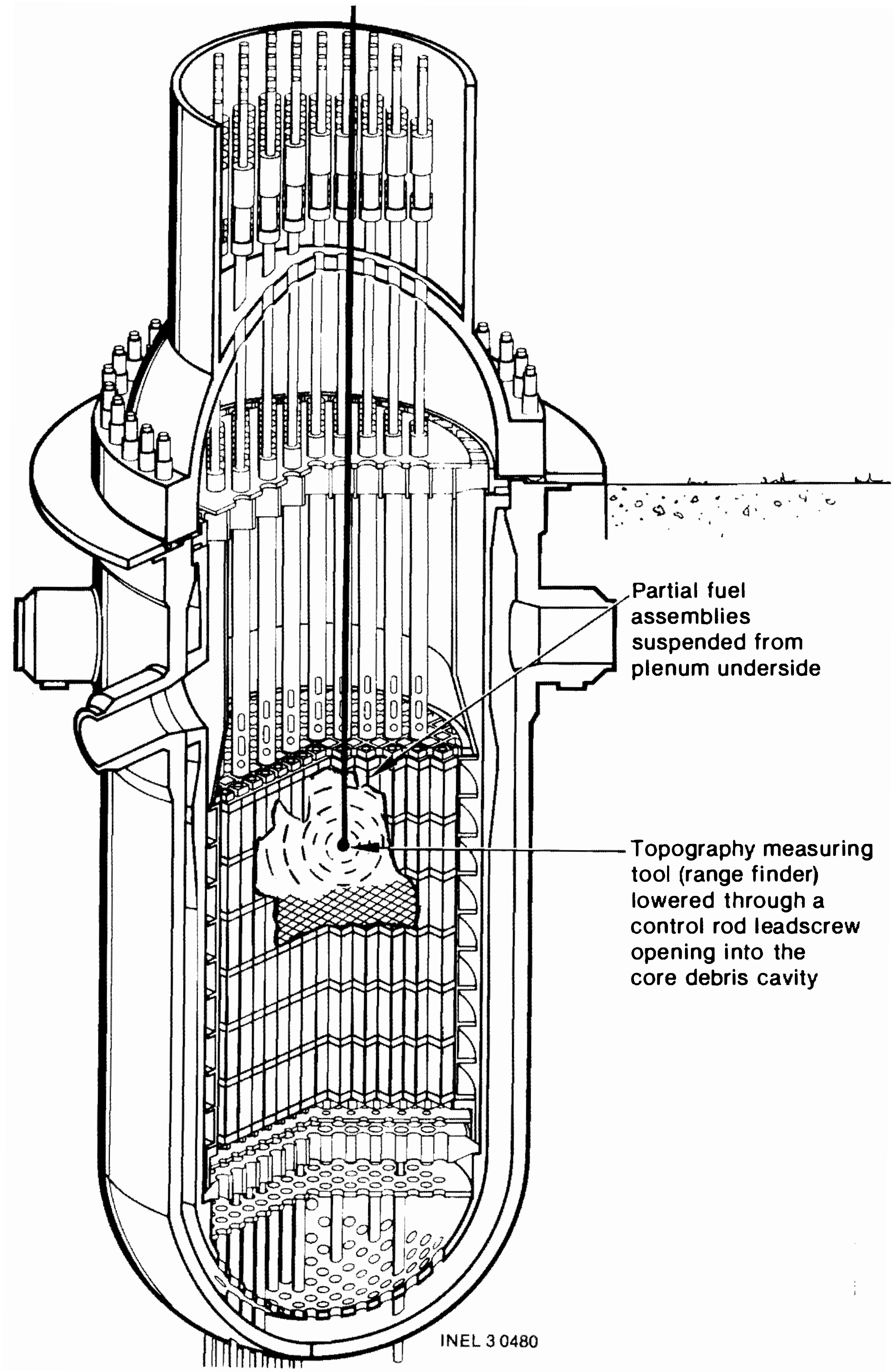


Figure 13. Schematic showing topographic measurement of core damage features.

shows a distinct depression on each side of the cavity, separated by a ridge of debris along the north-south axis. Interpretation of the topography data indicates that no more than two intact (i.e., full-height, full-width) fuel assemblies could exist in the upper portion of the core region.

5.2.3.4 In-Core Instruments. The in-core monitoring system measures core flux distribution and core exit coolant temperature profiles during reactor power operation. The measurements are provided by detector assemblies located at 52 preselected core radial positions. Each detector assembly, containing seven local flux detectors and a temperature detector, is installed in the central instrumentation tube of the fuel assembly. The local flux detectors are positioned to measure flux at seven core axial elevations. The temperature detector is located above the core to measure core exit coolant temperature.

The local flux detector consists of a rhodium emitter attached to a leadwire. The emitter and leadwire are surrounded by ceramic insulation and encased in a metallic sheath.

The temperature detector is a Chromel-Alumel, grounded-junction, metallic-sheathed thermocouple. The thermocouple is manufactured from special grade materials to comply with nuclear specifications. The thermocouples measure core outlet coolant temperature at 52 core radial locations.

Each detector assembly is inserted into the central instrumentation tube of the fuel assembly through guide tubing. As shown in Figure 14, the guide tubing extends from the bottom of the fuel assembly through the reactor vessel bottom head, completes two 90-degree turns, and terminates in the in-core instrument removal tank. The guide tubing termination and the detector assembly high-pressure closure assembly form a 2500 psi reactor coolant system seal just above the tank floor. The guide tubing is an extension of the reactor coolant system. When the reactor system is depressurized, the detector assemblies can be inserted and withdrawn through the guide tubing for installation or replacement.

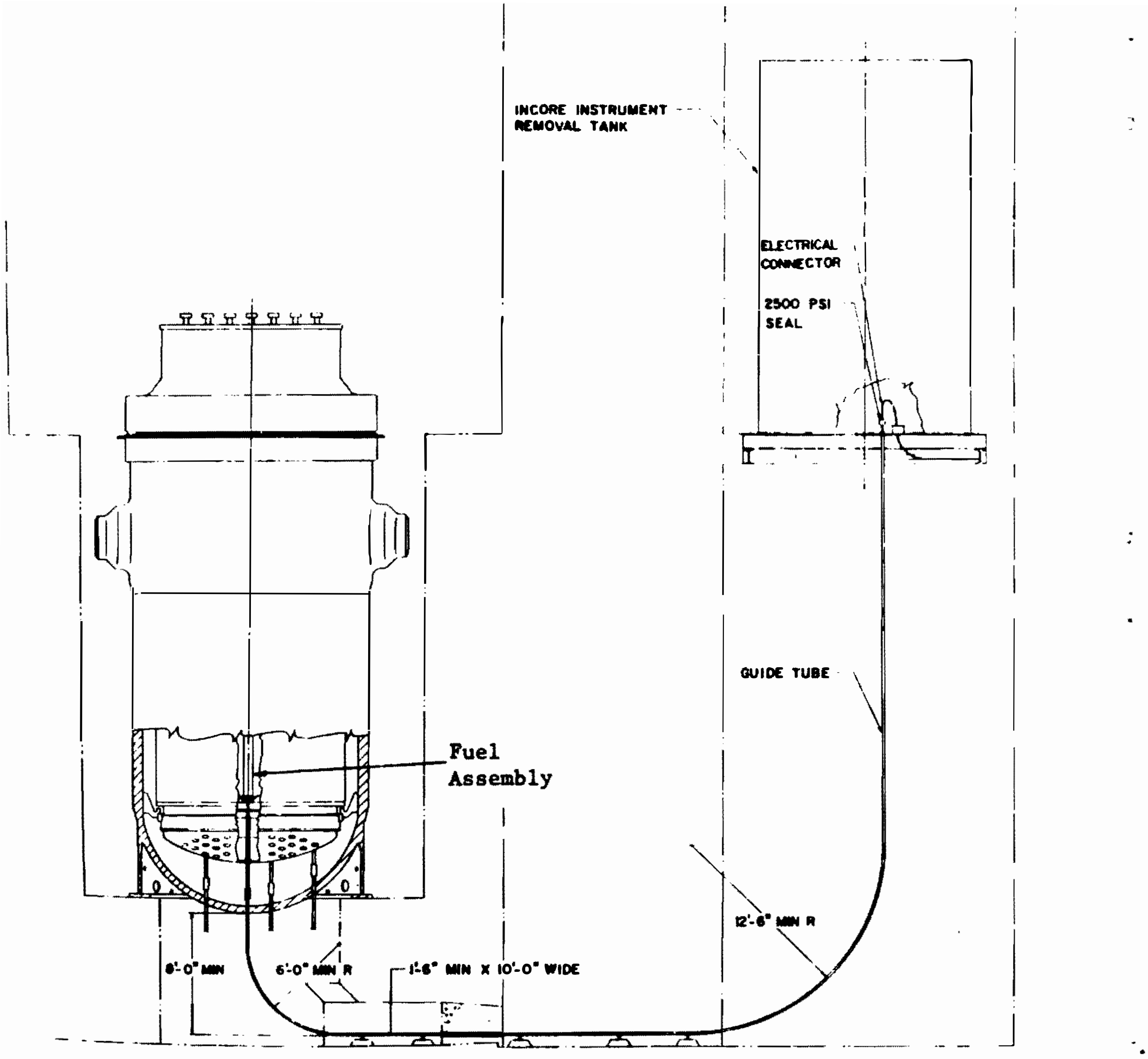


Figure 14. Schematic of a typical instrument detector assembly installation.

The current damage state of the TMI-2 in-core instrument assemblies is being assessed by EG&G Idaho (Reference 10). Because of the variety of well characterized materials in the in-core instrument assemblies and their documented responses during and after the accident, it appears that they offer the potential for obtaining the best core temperature and temperature history information should their recovery be possible.

5.2.3.5 Potential Retrievable Samples from the TMI-2 Core. A discussion of the as-built core and available information concerning its current damage state follows. Retrievable samples which might be obtained for examination are

- Rubble bed debris samples
- Hanging stub assemblies
- Portions of currently intact assemblies
- Debris bed stratification samples
- Miscellaneous core component pieces (fuel rod segments, spacer grids, end fittings, CRA spiders, springs, fuel pellets, etc.)
- In-core instrument assemblies.

5.2.4 Reactor Coolant System

Figure 15 shows a schematic of the reactor coolant system. It consists of four reactor coolant pumps, interconnected piping arranged in two heat transport loops, two OTSGs, and an electrically heated pressurizer. Pertinent features of these components are discussed below.

5.2.4.1 Reactor Coolant Pumps and Piping. The outlet piping leaving the reactor vessel is 36 inches in ID and runs in a horizontal plane for a few feet, undergoes a 90-degree bend, runs upward about 25 feet, enters a 180-degree bend, and enters the top plenum of the steam generator. At the

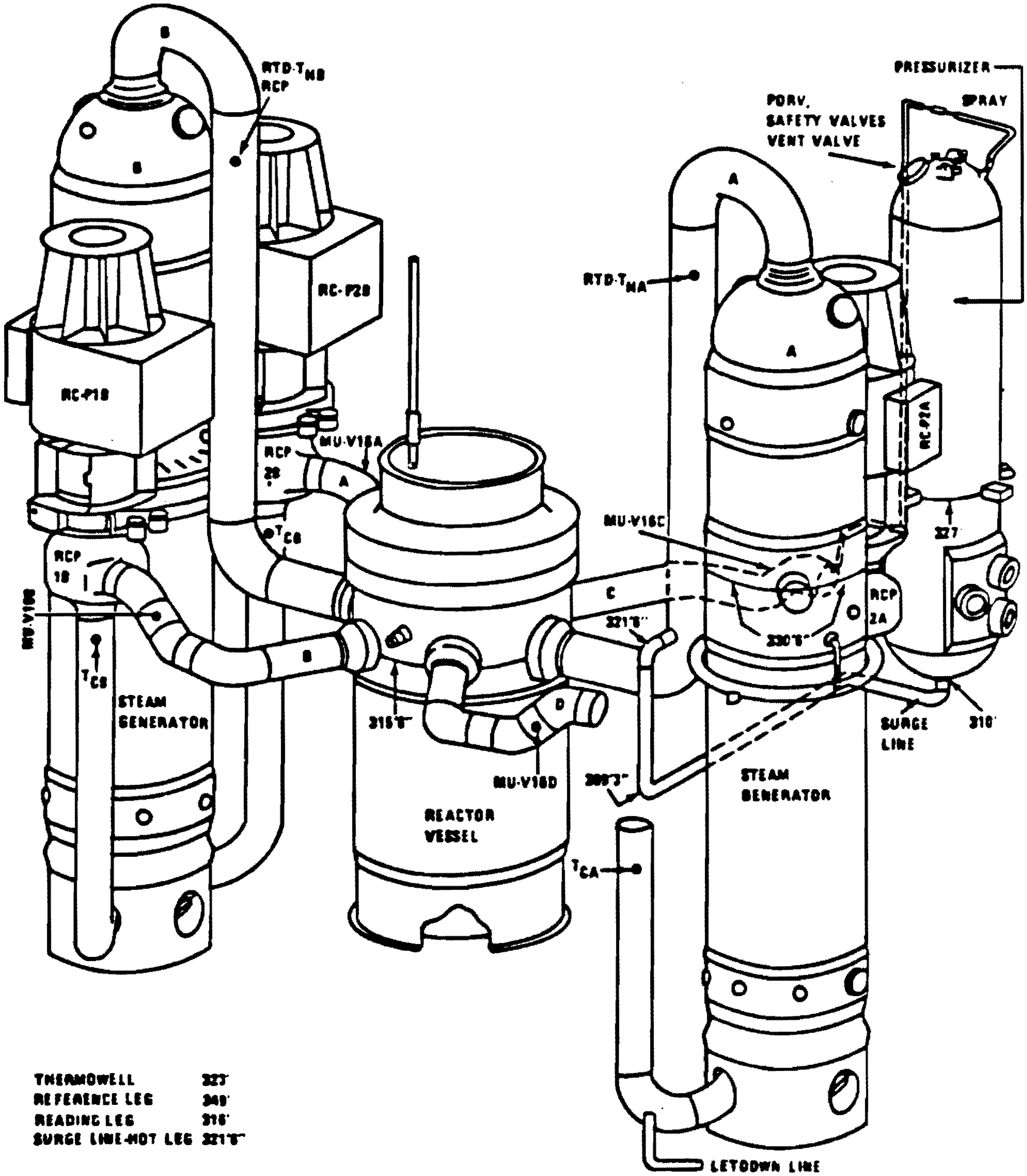


Figure 15. Schematic of the TMI-2 reactor coolant system.

steam generator outlet, each loop branches into two cold legs, each 28 inches in ID. Each cold leg leaves the steam generator through a bend, runs upward for about 10 feet, and enters the suction nozzle of one of the reactor coolant pumps. The reactor coolant pumps are located at an elevation of about 2-feet above the horizontal plane of the reactor coolant inlet nozzles. [This is important in assessing which portions of the cold-leg piping were uncovered during the accident sequence.] Upon leaving the pump discharge nozzle, the pipe drops down through two 45-degree bends into a horizontal run that enters the reactor vessel.

All surfaces in contact with the reactor coolant are constructed of austenitic stainless steel with some auxiliary system nozzle connections and instrumentation connections constructed of Inconel. The piping is carbon steel, with the austenitic steel (Type 304 or 316) cladding either weld deposited or explosively bonded. Retrievable samples that might be obtained for examination include the following:

- Resistance thermal detector (RTD) and thermowell connections on the hot legs (see Figure 16)
- Hot leg flowmeter instrumentation lines
- Portion of the vent line on top of each hot leg
- RTD and thermowell connections on the cold legs
- Reactor coolant pump impellers.

Also, it seems that with some development work, it would be possible to gamma scan the surfaces of the hot and cold leg piping after the fuel has been removed.

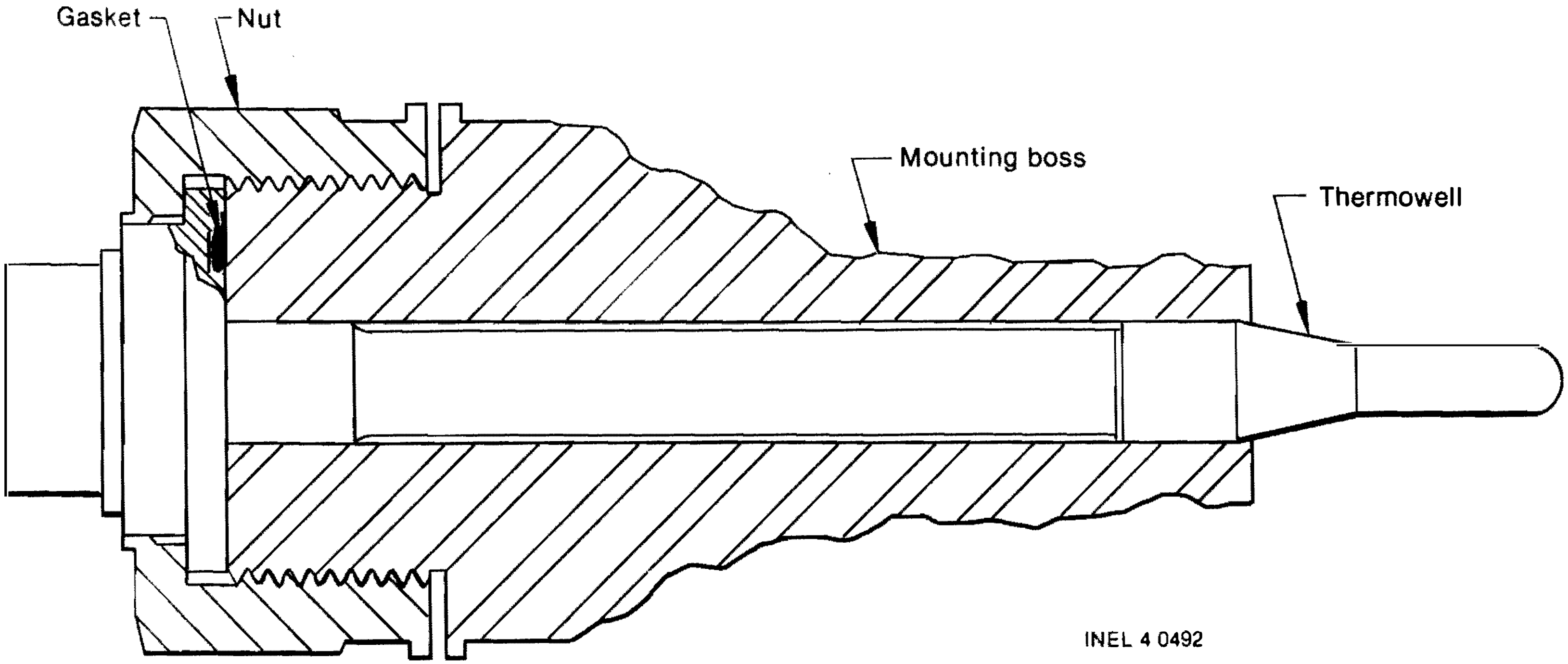


Figure 16. Cross section of an installed RTD/thermowell.

5.2.4.2 Once Through Steam Generators. The primary side surfaces of the OTSGs that were uncovered during the accident include the upper plenum surfaces, upper tubesheet surfaces, and upper portions of the tubes. The materials involved are stainless steel weld metal, Inconel cladding on the tube sheets, and Inconel-600 tubes. In addition, there are manway and inspection openings in the plenums which are flange sealed. Each of the manway and handhold covers has a Type 304 stainless steel backing plate which is in contact with the primary fluid.

Samples of interest that might be obtained for examination include the following:

- Backing plates for the primary manway and handhold openings
- OTSG tubes
- Debris from the tube sheets.

In addition, it appears that it should be possible to do *in situ* gamma scans of the plenum surfaces.

5.2.4.3 Pressurizer. The pressurizer vessel and its internals are shown in Figure 17. The vessel has penetrations for a manway, electric heaters, level and temperature measurements, spray line, vents, sample line, and relief valves. The main waterside surfaces, manway opening, and heater bundle openings are carbon steel clad with austenitic stainless steel. The spray line nozzles, pressure relief nozzles, and sampling and level instrumentation nozzles are also clad with stainless steel. The heaters are replaceable immersion heating elements. A Type 304 stainless steel ladder runs down the inside of the pressurizer vessel. The pressurizer is connected to one of the hot legs by a 10-inch surge line.

Retrievable artifacts that might be obtained for examination of the pressurizer and surge line include the following:

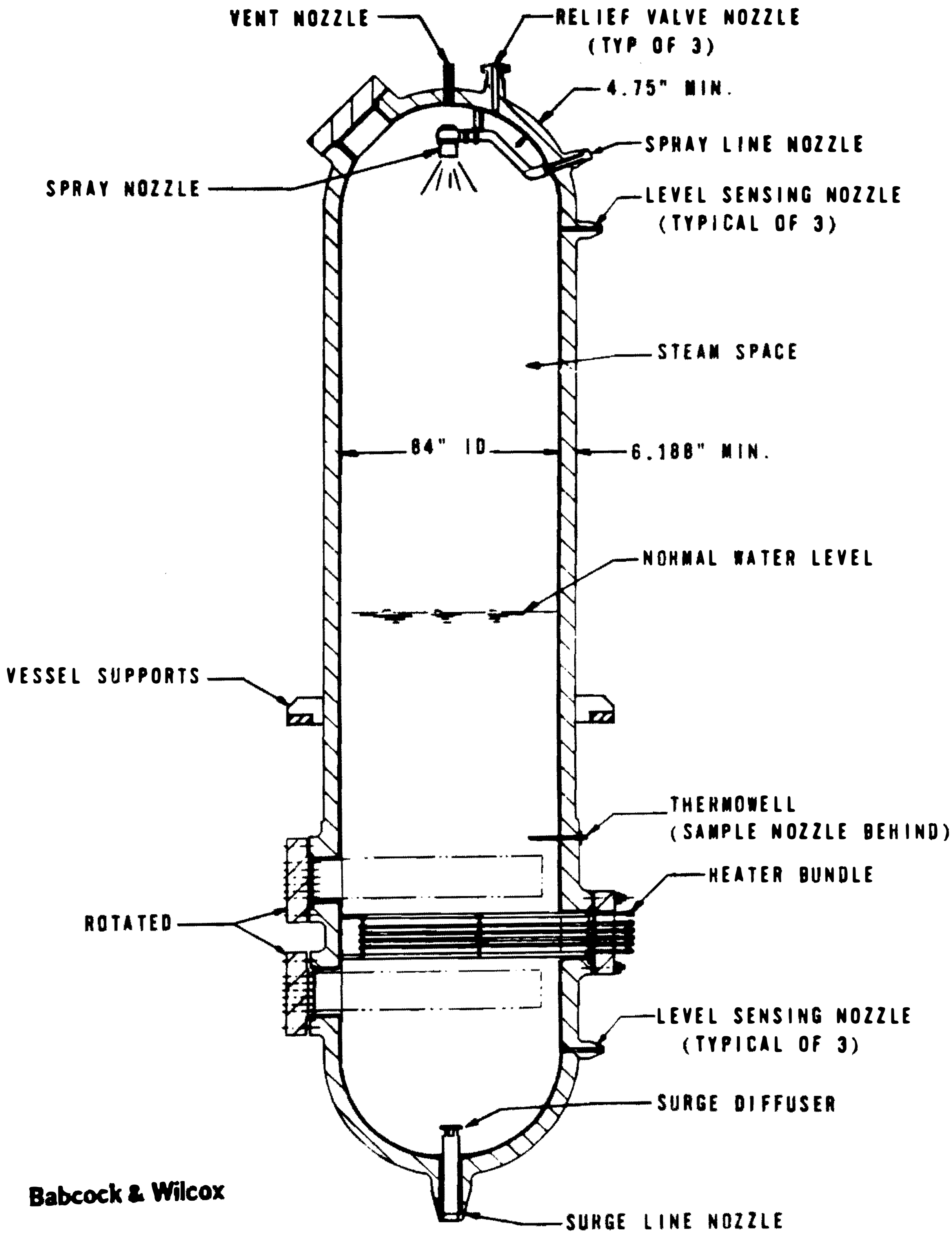


Figure 17. TMI-2 pressurizer vessel and internals.

- Pressurizer manway insert

- Heater bundles

- Selected ladder rungs

- RTD and thermowell penetrations.

In addition, because the surge line represented a segment of the major escape pathway to the containment, the surge line drain line may be examined for particulate debris.

5.2.4.4 The Reactor Coolant Drain Tank and Associated Valves and Piping. The reactor coolant drain tank received discharge from the pressurizer through the PORV and safety/relief valves. The discharge into the drain tank occurs through nozzles below the water level that are designed to quench flashing primary water and condense steam discharges.

Samples of interest that might be obtained for examination from the pressure relief valve discharge and drain tank include the following:

- PORV

- Drain tank rupture disk (if it can be located)

- Pieces from the rupture disk effluent piping

- Drain tank debris.

5.2.4.5 Miscellaneous Samples/Artifacts. Several miscellaneous samples/artifacts that might be examined include the following:

- Pieces from the reactor building coolers

- Panels of the primary piping mirror insulation

- Debris from the letdown coolers

- Debris from the in-core instrumentation tubes.

5.3 TMI-2 Recovery Plans and Schedule

The TMI-2 plant recovery plans and schedule have a major impact on the TMI-2 core examination. In light of this fluid situation, some major points of the defueling and recovery operations that have influenced the recommendations of the TMI-2 core examination plan are noted below:

- It has been assumed that the defueling operation will be aggressive (i.e., the defueling method and pace will be such that it is unlikely that *in situ* examinations and/or artifact recovery for the core region will be feasible during major defueling steps. Sample collection and *in situ* measurements should be done as early in the recovery operation as possible.

- It has been assumed that it will be possible to remove the plenum as an intact unit.

- It has been assumed that the hanging stub assemblies now attached to the plenum will be knocked off before plenum removal.

5.4 Analytical Techniques for TMI-2 Data Acquisition

In identifying the types of data that could be obtained from the TMI-2 core examination, it is necessary not only to identify the types of artifacts that could be recovered for examination, but also show that it is reasonable that the expected data could be obtained by examining these artifacts using current state-of-the-art techniques. Although a complete evaluation of techniques that could be used for obtaining TMI-2 data is beyond the scope of this report, such an evaluation should include consideration of expected TMI-2 conditions as well as the analytical sensitivity of available techniques. A preliminary scoping evaluation of selected analytical techniques and an evaluation of possible elemental concentrations in the core region and reactor coolant system deposits were used as a basis for recommending the types of measurements.

Table 5 summarizes material inventories and key properties for the as-built TMI-2 active core region.[11] In addition, Table 6 shows an ORIGEN-calculated radionuclide and elemental fission product inventory for the TMI-2 core after 4 years of decay (Reference 11). Data in these tables were used to estimate surface concentrations and activities for the TMI-2 primary coolant system (see Tables 7 and 8). Although admittedly crude, these surface concentration estimates, together with the available leadscrew examination data, provide a basis for the recommended surface analytical techniques. Likewise, the core inventories were used to recommend the analytical techniques for the core materials. These recommendations were made by comparing the estimated concentration of species of interest with the sensitivities of the analytical techniques under consideration.

Estimates of the approximate surface concentrations are based on the concentrations of selected elements in the core region, estimated core release fractions for those elements, and approximate surface area of the reactor coolant system (RCS). These data and the expected surface concentrations for selected elements are shown in Table 7. The total inventory of each element listed was obtained by summing the inventory of structural, cladding, and control materials and associated activation products with the inventory of fission products. The structural, cladding, and control inventory, as well as the fission product inventory and total core inventory for each of several selected elements, is shown in Table 7. The mass of each element released from the core was obtained from the total core inventory and the core release fraction.[12] The estimated surface concentration was obtained from the inventory of material released from the core and the surface area of the escape pathway in the primary system.[13] The surface concentration is expressed in g/cm^2. A number of confounding phenomena have not been

TABLE 5. SUMMARY OF MATERIAL INVENTORIES AND PROPERTIES FOR THE TMI-2 ACTIVE CORE REGION

Material	Inventory lb/(kg)	Composition[a] (wt%)	MP[b] (°F)
UO_2	205,140 (93,050)	--[c]	5,150
Zircaloy	50,770 (23,029)	Zr 98 Sn 2	3,350
Stainless Steel	3,550 (1,610)	Fe 65-75 Cr 18-20 Ni 8-12	2,600
Inconel	2,670 (1,211)	Ni 50 Cr 20 Fe 20	2,450
Ag-In-Cd	6,060 (2,749)	Ag 80 In 15 Cd 5	1,470
$B_4C-Al_2O_3$	1,380 (626)	--[c]	3,700
$Gd_2O_3-UO_2$	290 (131)	--[c]	4,980

a. Composition of the major elements only.

b. MP = Melting point (°F).

c. Quantities are either unknown or too small to list.

TABLE 6. ORIGEN-CALCULATED RADIONUCLIDE INVENTORY SUMMARY FOR THE TMI-2 CORE AFTER 4-YEAR DECAY

Nuclide	Half Life (yr)	Total Inventory (Ci)	Total Inventory (g)
		Fission Products	
$^{90}Sr/^{90}Y$	29	6.8E5	8.1E3
$^{106}Ru/^{106}Rh$	1.0	2.0E5	1.4E4
$^{125}Sb/^{125m}Te$	2.7	1.6E4	6.9E1
^{134}Cs	2.1	6.7E4	2.1E4
^{127m}Te	0.3	3.0E1	3.6E3
^{129}I	1.7E7	2.4E1	1.7E3
$^{137}Cs/^{137m}Ba$	30	7.4E5	--[a]
$^{144}Ce/^{144}Pr$	0.8	6.7E5	2.0E4
^{151}Sm	93	1.8E4	6.2E3
^{154}Eu	16	3.4E3	4.6E2
^{155}Eu	1.8	1.0E4	--[a]
		Transuranics	
U	--[a]	--[a]	8.3E7
^{239}Pu	2.4E4	9.4E3	1.6E5
^{240}Pu	6.6E3	2.0E3	--[a]
^{241}Pu	3	5.9E4	--[a]

a. Quantities are either unknown or too small to list.

TABLE 7. ESTIMATES OF SURFACE CONCENTRATIONS FOR THE TMI-2 PRIMARY COOLANT SYSTEM

Element	Total Weight of Structural, Cladding, and Control Materials (g)	Total Accumulated Fission Product (g)	Combined Weight (g)	Fission Product Fraction Released from Core Region	Total Fission Products Released from Core Region (g)	Surface[a] Concentration (g/cm^2)
Cs	--[b]	2.1E4	2.1E4	0.360	7.4E3	7.6E-4
I	--[b]	1.7E3	1.7E3	0.362	6.3E2	6.5E-5
Te	0.47	3.6E3	3.6E3	0.300	1.1E3	1.1E-4
Sb	5.9	6.9E1	8.5E1	0.204	1.7E1	1.8E-6
In	4.1E5	1.6E1	4.1E5	--[b]	--[b]	--[b]
Ag	2.2E6	1.9E2	2.2E6	0.300	6.6E5	6.8E-2
Cd	1.4E5	1.7E2	1.4E5	--[b]	--[b]	--[b]
Sn	4.2E5	2.3E2	4.2E5	--[b]	--[b]	--[b]
Mo	1.2E5	2.6E4	1.5E5	0.04	5.9E3	6.1E-4
Ce	--[b]	2.0E4	2.0E4	--[b]	--[b]	--[b]
Ru	--[b]	1.4E4	1.4E4	0.0030	4.1E1	4.2E-6

a. Based on a surface area of 967.4 m^2 (see Reference 8).

b. Quantities are either unknown or too small to list.

TABLE 8. ESTIMATES OF SURFACE ACTIVITIES FOR THE TMI-2 PRIMARY COOLANT SYSTEM

Isotope	Total Core Activity (Ci)	Fraction Released From Core	Surface Activity (Ci/cm^2)
^{120}I	2.54E1	0.362	9.4E-9
^{134}Cs	6.67E4	0.360	2.5E-3
^{135}Cs	2.25	0.360	8.7E-8
^{137}Cs	7.42E5	0.360	2.7E-2
^{127m}Te	3.01E1	0.300	9.4E-7
^{110m}Ag	3.15E1	0.300	1.0E-6
^{110}Ag	4.09	0.300	1.3E-7
^{113m}Cd	2.64	--[a]	--[a]
^{125}Sb	1.61E5	0.204	3.4E-4
^{89}Sr	2.12E1	0.033	7.2E-10
^{90}Sr	6.77E5	0.033	2.3E-3
^{93}Zr	1.62E1	7.0E-5	1.2E-10
^{95}Zr	1.29E1	7.0E-5	9.4E-11
^{103}Ru	4.25E4	0.0030	1.3E-13
^{106}Ru	1.95E5	0.0030	6.1E-5

a. Quantities are either unknown or too small to list.

considered in arriving at these estimates (e.g., dissolution of soluble deposits). Also, uniform deposition--regardless of material, temperature, or fluid flow--has been assumed.

Estimates of the surface activity of selected radioisotopes are shown in Table 8. The surface activities were calculated from the decay-corrected total curie content of the core, the elemental core release fractions, and an estimate of the surface area of the flow pathway in the primary system. The total activities shown were obtained from a simulation of the TMI-2 power history using ORIGEN-2. The simulation included a 1460-day decay period following shutdown. The fission product isotopes shown in Table 8 were selected on the basis of their presence at a minimum of one curie for the total core inventory. These activities provide a basis for assessing the applicability of radioassay techniques. However, these techniques do not consider the residual activated corrosion product isotopes (principally, ^{58}Co, ^{60}Co, and ^{54}Mn) or contamination from fuel debris.

Based on the above scoping analyses, the applicability of selected techniques that could be used for analysis of core materials and surface and debris samples are discussed below. Since the available analytical techniques have been discussed by other authors, the following discussion is brief and deals specifically with identifying the types of data in Table 3 that could be obtained using each technique.[14] In general, the analytical techniques used to acquire this information must have sufficient sensitivity to detect isotopes and chemical species of interest, be capable of yielding the desired information, and be cost-effective with respect to the type of information they provide. Application of these techniques must be commensurate with constraints on personnel radiation exposure.

5.4.1 SIMS, ESCA, and Auger

The SIMS, ESCA, and Auger techniques can be used collectively to obtain information on isotopic composition and chemical forms of surface species, as well as depth and area profiles. The sensitivity is on the order of 1 ppm and may involve 10^{-13} to 10^{-14} grams of material. SIMS (Secondary Ion Mass

Spectrometry) analyzes an area measuring 2 x 2 mm up to a depth of 60 to 80 μm. Set up as an ion microanalyzer, it can be used as an ion analog of a light microscope.[15,16] In this mode, it can provide both depth and area profiles of surface species. Examples of SIMS distribution maps are shown in References 15 and 16. ESCA (Electron Spectroscopy for Chemical Analysis) examines a surface measuring 1.0 x 1.25 cm, and Auger scans a surface with a 1 to 20 μm electron beam. The collective use of these techniques is attractive, partly because SIMS (especially when equipped with an inert gas ion beam) can effectively burn through oxide layers. After burning through a layer, the sample can be analyzed (using SIMS, ESCA, Auger, etc.) as required. This is an important feature, because TMI-2 samples may have corrosion film or other surface films which developed during the period between the accident and actual sample examination.

5.4.2 Electron Microprobe

The electron microprobe provides a useful method for ascertaining the elemental composition at a small area in a sample. It can be used for surface elemental analysis and for determining elemental distributions in surface deposits. For example, it is particularly useful for obtaining elemental distributions across grain boundaries. Sensitivities are generally on the order of 0.01 and 0.1 percent for wavelength-dispersive and energy-dispersive systems, respectively. Problems can be encountered with the analysis of highly radioactive samples; however, by using a shielded electron microprobe, quantitative and gradient analyses can be performed on materials up to 100 R/hr at 1 foot for elements with atomic numbers in the range 11 to 94.

5.4.3 MOLE

The MOLE (Molecular Optical Laser Examiner) can provide chemical specification information for Raman-active compounds on surfaces. This technique examines a 1-μm-diameter area and has a sensitivity of 1 to 10 $\mu g/cm^2$. To be useful, the surface examined should contain regions of pure materials. During surface analysis, each region of pure material would be examined individually. Many analyses would therefore be necessary to characterize a surface. It should be noted that while CsI is not Raman-active, a spectrophotoluminescent technique that would be sensitive to this compound is under development.

5.4.4 Gamma Spectroscopy

Gamma spectroscopy coupled with modern spectrum unfolding techniques is a very useful tool for the quantitative analysis of individual gamma-emitting isotopes in samples. With suitable collimation and shielding, it can have a wide range of applications, including both laboratory measurements and *in situ* measurements. This technique has, for example, been used extensively in such diverse applications as (a) precision gamma scanning of individual spent fuel rods in a hot cell environment to obtain quantitative maps of the radial and axial fission product distributions and (b) field scanning of deposit distributions in LWR piping. Because it can be applied (with some hardware development) to obtaining deposit distribution data for specific isotopes from *in situ* scanning measurements, it could prove very useful in the TMI-2 core examination for complementing the more detailed data that could be obtained from off-site analysis of retrievable samples. The specific isotopes for which gamma spectroscopy is expected to be most useful are ^{137}Cs, ^{134}Cs, ^{106}Ru, ^{125}Sb, and ^{144}Ce.

5.4.5 Wet Chemical Techniques

A variety of wet chemical techniques are available for surface leaching and species separation before analysis (e.g., for ^{90}Sr). It is beyond the scope of this document to discuss the number of techniques involved. In general, they can be used to improve the sensitivity of some of the techniques mentioned above. They also can be used in conjunction with more conventional isotopic and elemental analysis techniques such as neutron activation analysis (for ^{129}I) and atomic absorption.

5.4.6 Optical and Scanning Electron Microscopy

These techniques have been used extensively in metallographic analysis of such microstructural features as surface morphology and grain size and shape. In conjunction with image analysis techniques, they can give size and shape information for particles deposited on surfaces or in debris samples.

6. RECOMMENDED SAMPLE ACQUISITION AND EXAMINATION PROGRAM

After the needed data (see Section 4) and feasible approaches for obtaining it (see Section 5) were identified, specific recommendations were developed using the approach discussed in Section 3. Clearly, if the benefits and costs associated with each of the possible data acquisition activities could be reduced to a quantitative ratio, then the problem of prioritizing and selecting the TMI-2 core examination activities would be a trivial one of maximizing the benefit-to-cost ratio. However, one can only establish subjective measures of the value of data sought from each activity. Hence, the approach used to satisfy the objective of maximizing the useful data that could be obtained using the available resources was to rely on the collective judgment of the combined TEG membership. To support the TEG in making final recommendations, engineering feasibility and cost data for artifact acquisition, shipping, and examination were developed for the combined TEG recommendations. These consist of recommendations for *in situ* measurements and artifact (sample) recovery with subsequent off-site analysis. The *in situ* examination and artifact recovery recommendations that were developed are summarized in Table 9. The recommendations are discussed in Section 6.2.

6.1 Approaches for Acquisition and Shipping of Artifacts

Although most resources for the TMI core examination will be expended in acquiring and shipping suitable artifacts for subsequent off-site analysis, this document assumes that specification of artifacts to be recovered provides sufficient guidance to EG&G Idaho to execute this phase of the program. Detailed engineering, procedures, and coordination with other parties involved are program implementation activities and therefore are the responsibility of EG&G Idaho as the primary DOE contractor. Artifact acquisition and shipping is not discussed here other than to point out the importance of the following:

- o Properly identifying and documenting the location from which each artifact was removed

TABLE 9. RECOMMENDATIONS FOR TMI-2 SAMPLE ACQUISITION AND EXAMINATION

Reactor Sample or Component[a]	Sample Description	Proposed Examination Techniques	Expected Data	Principal Data Uses/Safety Issues	Comments
1. Subsurface debris bed samples	Eleven "grab" specimens from various depths from two radial positions, H8 and E9	Photo-visual, metallography, chemical analysis, SEM, radiochemistry, microprobe, gamma scan, surface analysis	Molten material relocation; fuel, control, and structural material reactions; extent of oxidation; peak core temperatures; control material relocation; retained fission products	Core relocation models, fission product transport codes, debris bed coolability models, H_2 generation estimates, source term determination, recriticality analysis	In progress. These samples should guide location and extent of core stratification samples
2. Reactor vessel internals	*In situ* documentation of core, upper plenum, and lower plenum condition. Inspections tied to removal of major core internals (2 topography and 4 CCTV examinations)	Photo-visual, closed circuit TV, acoustic topography	Core and internals damage symmetry, core void size, total volume and mass of debris, stub assembly elevations, extent of liquefaction, transition zone configurations, major coolant channels	Core heat up codes and relocation models, recrititicality analysis, core coolability models, mass balance determination, molten material relocation models, nature of debris stratification	Documents general condition of vessel and internals to aid data interpretation and update sampling and removal plans as necessary
3. RTDs and thermowells (surgeline, candy cane, and pressurizer)	Remove 4 accessible RTD thermowells	Surface analysis, gamma scan, radiochemistry	Fission product plateout, peak well temperature	Fission product transport codes, primary system temperature estimates	Some work in progress. Access to RTDs is RCS water level dependent. Four are accessible until approximately December 1984. The other 2 should be obtained following defueling.
4. Lower reactor vessel head debris profile and instrument location	Profile lower head through in-core instrument tubes	Piano wire and ion chamber	Character and extent of debris in lower head	Core debris relocation models, mass balance determination	Work is partially funded by EPRI. Study will provide early information on depth of core damage and may help determine sites for core stratification samples.
5. Fueled rod segments from known locations	Remove fueled rod segments from various locations (6 rod segments taken from 3 radial locations)	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, gamma scan, surface analysis	Cladding, fuel rod temperatures, extent of oxidation, extent of eutectic melting and fuel liquefaction, fuel rod fragmentation, UO_2 oxidation, fission product release from fuel	Core heatup codes, H_2 generation estimates, fission product transport codes, source term calculations, recriticality analysis	Rods should be taken from within the void region, at the periphery of the void. Detailed examination should be limited to 2 rods. Cursory examination only on the others.
6. Core stratification samples	Four core bore samples to the bottom of core region. The core bore should include several fuel rods and at least one control rod guide tube as part of the sample. Two samples will be taken at locations coincident with lower plenum inspection holes. One such position is at K9	Photo-visual, metallography chemical analysis, SEM, particle size, radiochemistry, microprobe, surface analysis, gamma scan, acid-base leaching, density	Fuel and structural material reactions, relocation of core materials, extent of fragmentation, extent of oxidation, retained fission products, nature of debris stratification, peak core temperatures, in-core instrument damage, control material relocation	Core heatup codes and debris bed coolability models, fission product transport codes, core relocation models, source term determination, H_2 generation estimates, in-core instrument survivability analysis, recriticality analysis	Sample acquisition should be coordinated with the lower vessel examination and sampling task. The samples should be continuous vertical bores through the entire core region. The outside diameter of the sample should be as large as can reasonably be obtained considering access, tooling, and subsequent handling after removal from the core.

a. The order of listing is organized according to estimated sample acquisition timing.

TABLE 9. (continued)

Reactor Sample or Component[a]	Sample Description	Proposed Examination Techniques	Expected Data	Principal Data Uses/Safety Issues	Comments
7. Lower vessel inspection and sampling	Samples from 2 accessible locations in the lower vessel head. To be performed in conjunction with Items 4 and 6 above	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, gamma scan surface analysis, acid-base leaching, density	Estimate of total quantity of debris, particle size distribution, extent of once-molten debris, fission product content, extent of damage	Core debris relocation models, fission product transport codes, vessel breach models, recriticality analysis, debris bed coolability	
8. Control rod leadscrew	Three leadscrews removed previously plus 7 additional leadscrews from various radial locations	Gamma scan, surface analysis, radiochemistry, metallography, SEM/STEM/microprobe	Fission product plateout, component temperatures, extent of oxidation	Fission product transport codes, plenum temperature estimates, core exit steam temperature calculations	Some work in progress
9. Leadscrew support tube	One support tube taken from core position H8	Gamma scan, radiochemistry, surface analysis, metallography, SEM/microprobe, acid-base leaching	Fission product plateout, metal temperatures, extent of oxidation	Fission product transport codes, source term determination, core internals temperature estimates	The tube has already been removed from the reactor vessel. Examination of the tube is being negotiated
10. Reactor coolant system	Scan of RCS hot legs, surge-line, SGs, and pressurizer	Gamma scan	Location of core material	Core debris relocation models, fission product transport codes, recriticality analysis, debris bed coolability	Samples of loose debris should be taken if large quantities of material are found
11. Steam generator handhole covers	Remove handhole covers from A and B steam generators. Might include samples from inside vessels depending on results of visual exam	Surface analysis and radiochemistry of handhole covers. Visual exam of vessel internals. Sample removal from vessel internals if visual exam deems it necessary	Fission product plateout, core relocation	Fission product transport codes. Core relocation, RCS thermohydraulic analysis	Must be accomplished while RCS water level is low enough to allow examination
12. In-core instrumentation	Recovery of 4 (minimum) in-core instrument assemblies at various radial locations	Photo-visual, chemical analysis, instrument reliabilities, metallography	Fission product plateout, peak temperature estimates, radial and axial temperature profile	Fission product transport codes, core relocation models, peak core temperatures, extent of core damage, core damage symmetry	Acquisition of the instruments is believed to be very difficult, however, temperature data that can be obtained through examination of the instruments emphasizes the critical need to obtain them

TABLE 9. (continued)

Reactor Sample or Component[a]	Sample Description	Proposed Examination Techniques	Expected Data	Principal Data Uses/Safety Issues	Comments
13. Distinct fuel assembly and control rod cluster components	Retrieve up to 40 specimens of cladding, CRs, spiders, spacer grids, end fitting hold down springs, in-core instruments, etc. from the debris bed. If possible hanging fuel assembly stubs should be obtained	Photo-visual, metallography, chemical analysis, SEM, radiochemistry, microprobe, gamma scan, surface analysis	Molten material relocation; control and structural material reactions; extent of oxidation; retained fission products; peak core temperatures; control material relocation	Core debris relocation models, fission product transport codes, debris bed coolability models, H_2 generation estimates, source term determination, recriticality analysis	Sample selection should be based on CCTV inspection. Spiders, spacer grids, and end fittings should be taken from the same (known) positions. Samples may be taken before and after removal of (vacuuming) the loose debris from the vessel. If the opportunity arises it is desirable to obtain damaged fuel assemblies from under the debris bed.
14. Reactor building basement solids	Determine the approximate volume of debris (m^3)	Measure depth of sludge at various basement locations. Relate fission product data to volume of material	Retained fission products, relocation of core materials	Fission product transport codes, core relocation models, source term determination, mass balance determination	Some work in progress
15. Filters	Sample debris from makeup and purification system filters	Photo-visual, metallography, chemical analysis, SEM, particle size, radiochemistry, microprobe, physical properties	Retained fission products, particle size distributions, fuel control material and structural material reactions, relocation of core materials	Fission product transport codes, fuel fragmentation models, core relocation models, source term determination, recriticality analysis	The work is completed. A final report is in progress.
16. Plenum cover debris	One sample from area of debris accumulation	Chemical analysis, SEM/microprobe, radiochemistry, particle size, surface analysis, gamma scan	Debris composition and particle size, fission product content of debris	Core debris relocation models, fission product transport codes, source term determination	Area selected was based on CCTV inspection. Samples are being analyzed
17. Reactor coolant drain tank	Sample tank sludge and vent pipe (also video exam)	Photo-visual, particle size, SEM, radiochemistry, surface analysis	Thermohydraulic transport, location of core material, fission product release	Fission product transport, core debris relocation, criticality analysis	The necessary work is in progress

- o Making provisions to ensure that sample acquisition and shipping procedures do not confound interpretation of data obtained in off-site analyses.

6.2 Recommended In Situ Measurements and Off-Site Artifact Examination

In general, the TMI-2 core examination plan recommends that relatively few detailed examinations or inspections be conducted at TMI-2 during reactor recovery. This is consistent with DOE policy of minimizing the impact of data gathering on reactor defueling.[17] *In situ* examinations generally are not specified unless failure to do so would result in irretrievable loss of vital information. This approach also is desirable because environmental conditions during defueling (e.g., water turbidity, background radiation levels, etc.) probably will not be conducive to conducting detailed examination work.

Much of the data regarding TMI-2 core damage will be obtained during off-site examinations. The TMI-2 core samples will be shipped to remote handling facilities at the Idaho National Engineering Laboratory (INEL), though smaller specimens may go directly to participating laboratories. Examination of the TMI-2 core samples will be conducted at INEL and other DOE and commercial hot cells. Because of the number of TMI-2 core samples, and the need to complete the examination in a timely manner, several available U.S. hot cell facilities will be involved in examining the TMI-2 samples.

The following sections present a general overview of the types of examinations and examination techniques that should be applied to the TMI-2 specimens. An effort has been made to include a brief discussion of each anticipated examination. However, as in many post-irradiation examinations, unexpected observations may change the course of some examinations. No attempt is made to specify details of the work breakdown or examination approach, since it is quite likely that those would quickly become obsolete as new data are obtained. This discussion is intended only to provide broad general guidance on the types of data that should be obtained and how off-site examinations should approach obtaining those data. The recommendations included in Table 9 fall into the following categories: (a) in situ measurements, (b) reactor coolant system surface samples, (c) reactor core samples, and (d) miscellaneous. Examinations recommended for each of these categories are discussed briefly below.

6.2.1 In-Situ Examinations at TMI-2

6.2.1.1 Closed-Circuit Television. From the viewpoint of core-data acquisition, the most important early core examination task has been the continued closed-circuit television (CCTV) camera inspections of the core condition. These inspections have been performed by lowering a small diameter camera down through vacated control rod drive mechanisms into the core void region (Figure 11). The camera then was rotated from its straight-down position to nearly straight up at several radial orientations. CCTV inspections have been conducted at three locations (shown in Figure 12) and yielded a wealth of visual information, both direct and inferred, on damage to the core and reactor internals.

An additional CCTV inspection of the core void area has been made from the center (H8) position before disrupting the present condition of the cavity. The additional inspection was made using a camera manipulator and controlled indexing system, which can maintain camera orientation during the inspection process. The entire core void, including the underside of the plenum, has been scanned.

After the reactor head is removed and direct visual access is obtained, some additional CCTV documentation should be conducted. CCTV inspection of the entire plenum cover, as well as other horizontal surfaces (e.g., plenum shield), should be conducted, and the types of debris documented.

Any significant damage to the plenum (e.g., distortion, slumping, melting, oxidation, etc.) will be documented to plan for plenum removal. This documentation is adequate for data acquisition purposes as well. Unless very unusual conditions are encountered, such as massive oxidation of the plenum, no metallurgical samples are warranted at this time. If such conditions are present, the concern over possible failure of the plenum during removal will dictate that metallurgical samples be taken. The samples then would be subjected to standard metallography and mechanical properties tests to determine their integrities.

With the head removed, the tops of all peripheral fuel assemblies can be examined readily with CCTV. Defueling personnel will probably require that peripheral assembly upper end fittings be thoroughly examined to confirm that proper tooling exists for their later extraction. During the course of examining these end fittings, it is important to document the range of debris deposition and any observable damage that is encountered and note any asymmetries. The location of such asymmetries, if present, should be correlated with the positions of the coolant inlet and exit nozzles, variations in the plenum damage, and variations in the plenum cover debris.

CCTV inspection of the lower reactor vessel is recommended to document the type and quantity of debris in this location. This inspection could be conducted in conjunction with core stratification sample acquisition.

The selection of TMI-2 core debris samples for off-site examination must be documented thoroughly so their initial appearances, locations, and orientations are known. Also, the general defueling must be documented so later estimates can be made of the relative amount and location of various damage features. The basic way to preserve this information will be the CCTV video tape with explanatory voice-over. There will probably be instances, however, when a better photographic record is needed. Accordingly, the TMI-2 core examination plan will require access to video image enhancement equipment to complement information acquired by the standard inspection equipment.

6.2.1.2 Core Topography Measurement. As stated in Section 2, providing data to help understand core damage processes during a severe accident is one of the objectives of the TMI-2 core examination. This refers to the entire spectrum of material behavior and movement during the accident. It includes (a) fuel rod fragmentation and settling of the consequent debris; (b) debris entrainment in the coolant, with subsequent deposition in the plenum or elsewhere in the coolant, and subsequent deposition in the plenum or elsewhere in the primary system; (c) slumping of portions of the core; and (d) axial and radial motion of liquefied materials. Detailed knowledge of core relocation must come from careful documentation of the post-accident core topography and measurements of the nature and extent of strata encountered during defueling.

In order to measure the core topography before major alterations occurred, an ultrasonic core topography system was built and operated in the TMI-2 reactor vessel before head removal. The information is summarized in Section 5.2.3.

The core topography measurement tool can be used in the defueling stage of TMI-2 recovery when the nature of the debris changes or a unique damage zone is encountered. Examples of additional uses of the core topography tool are as follows:

- A topography measurement made immediately after removing the plenum could be compared to the first topographic scan to see how much core debris relocation occurred as a consequence of plenum removal.

- When the layer of loose debris that forms the bottom surface of the core cavity is removed, the topography of the underlying structure should be documented. Therefore, topography measurements before and after removing the debris are recommended.

- During defueling, researchers will be looking for distinct changes in the nature of the debris, such as the presence of a once-molten zone beneath loose debris. As these distinct layers are encountered and exposed, their morphology can be readily mapped.

- After all loose and fused debris has been removed, a forest of standing, intact fuel rod stubs presumably will be encountered. The topography of this layer will tell exactly how deep the core damage penetrated, as a function of core location. Because of the speed of data collection, repeated use of the topography mapping tool is readily justified as a means of quantifying core material relocation.

6.2.2 Examinations of Reactor Coolant System Surface Deposit Samples

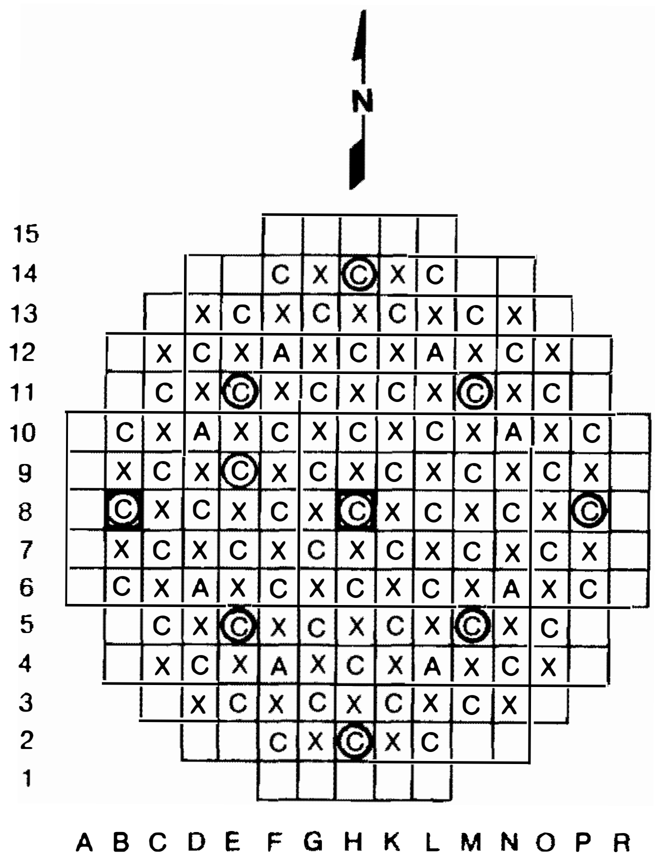
The data needs that will be satisfied through examination of reactor coolant system surface deposit samples (i.e., all reactor coolant system artifacts from locations other than the core region) include the inventory, distribution, and form of fission products and control materials on reactor coolant system surfaces, as well as the maximum temperature and temperature history that these surfaces experienced during the crucial 113- to 208-minute period of the accident.

6.2.2.1 Plenum Region. From the point of view of retention of fission products released from the core region, the retention on plenum surfaces is very important. In the recommended plan (see Table 9) the role of the plenum in capturing fission products and other volatile core component materials will have to be inferred from examination of leadscrews and a leadscrew support tube. [Note: It should be realized that major questions can be raised concerning the validity of extrapolation of the leadscrew deposit data to other plenum surfaces. This particularly is true for those plenum surfaces that are not within the control rod guide tube assemblies and have large horizontal surface areas that will collect debris.] The leadscrew and other surface deposit examinations should involve detailed chemical, radiochemical, and microchemical examinations of the surface deposits. These examinations should determine the surface elemental deposit concentrations (particularly for I, Cs, Te, Ag, In, Cd, Mn, and Sn), area and depth profiles for these elements, deposit concentrations of important radioisotopes (^{134}Cs, ^{137}Cs, ^{90}Sr, ^{125}Sb, and ^{144}Ce), and principal molecular species involved. In addition, the composition (chemical and radiochemical) and physical properties (particle size and morphology) of any loose deposit material should be characterized. The microstructure and microchemical composition of the base metal material, as well as its metallurgical properties, should be examined to extract information that would help define the maximum temperature to which it was exposed, and, if possible, the temperature history during the accident. Examination of the H8 and B8 leadscrews is discussed below to provide insight into the types of examinations that should be performed on all of the ex-core surface deposit samples.

The H8, B8, and E9 leadscrews--which drive the control rod assemblies--were removed from the reactor to provide access ports for the early CCTV camera inspections of the core. These components are now at INEL, and examinations of H8 and B8 are underway. The leadscrews are stainless steel components 7.3-m long (see Figure 2). The bottom section of the leadscrew is 17-4 PH (precipitation-hardened) stainless steel. This is an ideal material from which to reconstruct local temperatures because of characteristic changes in the intermetallic precipitates as a consequence of elevated temperature operation. The lower portion of the leadscrew also contains Types 304 and 410 stainless steel components, the microstructure of which also will be indicative of temperatures. Since the leadscrews were taken from different regions of the core, they will likely exhibit different microstructures. This will allow some very preliminary conclusions as to variations in the peak plenum temperature and, by implication, the peak core temperature. Preliminary metallographic results indicate strong temperature gradients along the axial length of the leadscrews.

Metallurgical examination of the leadscrews will be complemented by measuring fission product deposition on their surfaces. Preliminary analyses have revealed that two distinct layers, each 25- to 35-μm thick, formed on the leadscrews. The outer layer was easily removed by brushing; however, the inner layer containing significant quantities of cesium could only be removed by concentrated acid treatment.

In order to determine symmetry of fission product plateout and temperature profile in the upper plenum region of the reactor vessel, a cursory examination of 10 to 12 additional symmetrically located leadscrews is recommended. Figure 18 shows the recommended locations for leadscrew acquisition and examination. Data from the additional leadscrews will be compared to the H8 and B8 data to determine symmetry in the upper plenum.



C — Control rod assemblies

A — Axial power shaping rod assemblies

X — Burnable poison rod assemblies

Recommended locations for leadscrew acquisition and inspection

Locations of leadscrews presently being examined at INEL

Figure 18. TMI-2 core grid layout showing locations for recommended leadscrew acquisition and examination.

One leadscrew support tube has been removed from the H8 location for examination. Data from examination of this support tube and the leadscrews will provide information on how fission products were partitioned in the plenum region and what role various components played in retarding fission product release. The leadscrew support tube data will be compared to the leadscrew data to determine whether the leadscrews can be used to adequately represent all components in the upper plenum region regarding fission product deposition and plateout.

One sample of debris was taken from the plenum cover region and is being analyzed to determine the nature of the particulate debris on its horizontal surface.

Should examination of the leadscrews, support tubes, and plenum cover debris indicate significant differences in temperature or deposition of fission products, consideration should be given to obtaining guide tube(s) and/or "C" and split tubes. These items are not required as part of the recommended examination program, but are considered to be contingencies and should only be used if absolutely necessary or if convenient access to them is possible.

6.2.2.2 Ex-Vessel Reactor Coolant System. It is intended that similar examinations of other artifacts removed from various locations in the primary coolant system (ex-vessel) will provide information on how the fission products were partitioned in the primary system and what role they played in retarding fission product release. This will involve radiochemical analysis of the surface deposits, as well as detailed surface analysis for stable isotopes using the sophisticated techniques described in Section 6.2.2.1. Components recommended for examination are as follows:

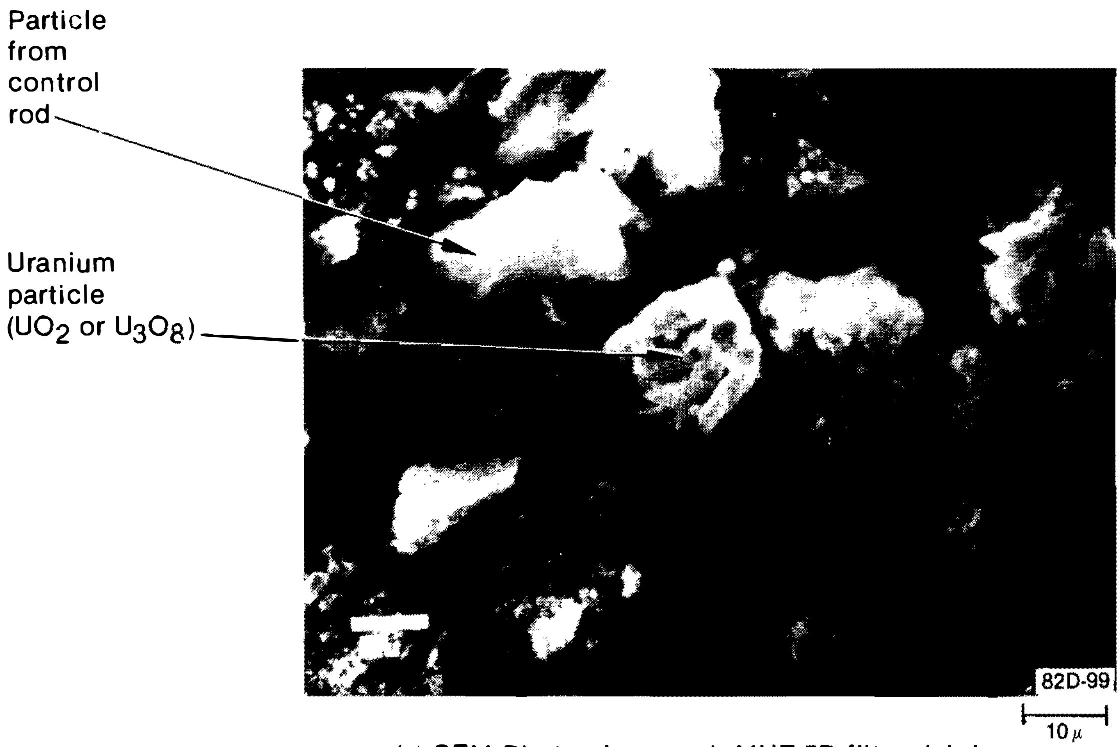
- RTDs and Thermowells--There are six RTDs and thermowells located in the primary system (surge lines, candy canes, and pressurizer), four of which are accessible before defueling. These components are prime candidates for determination of fission product plateout and peak well temperatures in the primary piping.

- Steam Generator Handhole Covers--These components from the A and B steam generators are easily obtainable and can provide valuable fission product plateout information regarding the steam generators. Following removal of the handhole cover, visual (CCTV) inspection of the inside of the steam generators should be conducted. If significant quantities of debris are observed in the bottom of the steam generator, samples should be obtained and analyzed.

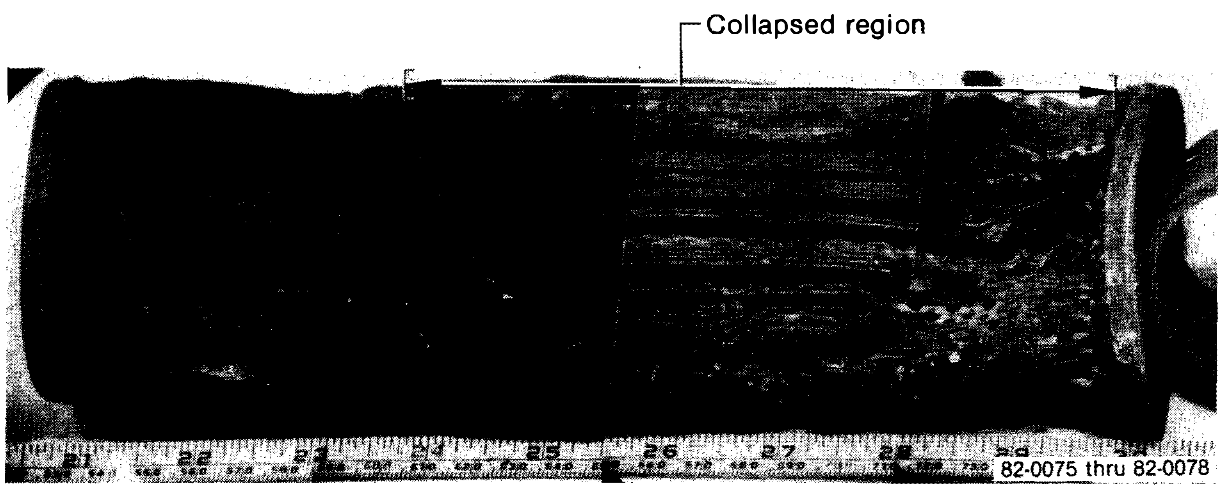
- TMI-2 Filter Debris--Ten makeup and letdown system[a] filters were received at INEL in April 1982. This system contains several pairs of filters, some of which were plugged by core debris flushed from the primary system. The filter debris is the first TMI-2 core material subjected to detailed analysis. The material is significant because of its core damage implications. The makeup and letdown system filters are located many meters from the core, and the debris had to pass through the steam generators to reach the filters. The presence of significant quantities of debris in the filters suggests the likelihood that large quantities of such debris remained in the core and also are dispersed within the primary coolant system.

  Analyses of filter debris indicate that the debris are only ∿6 wt% uranium, a relatively low fuel (UO_2) content.[18,19] The principal metal component of the debris is zirconium, presumably ZrO_2 from oxidized cladding. Significant concentrations (∿15 wt%) of silver, indium, and cadmium from the control rods are also present. The predominant radionuclides are ^{137}Cs, ^{125}Sb, ^{144}Ce, ^{106}Ru, ^{60}Co, ^{90}Sr, and ^{129}I. The particle sizes are less than 1 to 5 μm, with some larger (presumably) agglomerates of 2 to 50 μm. The mean particle size is ∿4 μm. One of the filters and a microstructural photograph of the filter debris are shown in Figure 19.

a. The makeup and letdown system recirculates and purifies reactor coolant and supplies water to the reactor coolant pump seals. The system also supplies high-pressure injection of borated water during an accident.



(a) SEM Photomicrograph MUF-5B filter debris



(b) Filter MUF-4A (collapsed as a result of differential pressure)

Figure 19. TMI-2 makeup and letdown system filter and debris.

6.2.3 Reactor Core Samples

This section presents recommendations for specific data-collection tasks, as well as guidelines that will be used to select core debris samples for subsequent off-site analysis. It is assumed that the actual core defueling will be primarily a remote handling operation, during which a number of CCTV cameras and underwater manipulating devices will be directed from a control station. Present defueling plans call for core debris to be retrieved and placed in canisters located underwater in the canal near the reactor vessel, or in the vessel itself. When full, the canisters will be drained, dried, and inspected. The canisters presumably will be stored at TMI-2 for an interim period, then shipped to INEL for storage--eventually to be disposed of.

Except for the core debris grab samples, selection of core debris samples for off-site examination and data acquisition should be done after the head and plenum have been removed. Most of the samples can be obtained before start of defueling. It is critical that all samples be identified properly and the location from which each artifact was removed documented. Based on previously established selection criteria (e.g., structure, location, appearance, etc.), samples retrieved would be diverted to off-site examination canisters. In practice, sample selection for data acquisition purposes may require somewhat different tools and techniques than defueling. For example, fragile oxidized specimens may require more careful manipulation if they are intended for subsequent off-site examination. In order to properly plan and coordinate both the defueling and sample selection for data acquisition purposes, it is essential that sample selection guidelines be prepared. It also is important that all requirements for sample preservation during handling and off-site shipment be established.

The core samples will consist of (a) small samples of loose debris from the rubble bed, (b) fueled rod segments, (c) core stratification samples, and (d) miscellaneous "pick and put" samples of core components (e.g., assembly end fittings, in-core instrumentation, holddown springs, control spiders, etc.). The types of examinations to be conducted on these artifacts are discussed briefly below.

6.2.3.1 Core Debris Grab Samples. Eleven samples of particulate debris from within the rubble bed have been obtained by lowering sampling devices through the H8 and E9 leadscrew openings. At position H8, samples were taken at the following depths into the debris bed: surface, 3, 11, 22, 27-1/2, and 30-1/2 inches. At position E9, the samples were taken at the surface, 3, 22, 29, and 37 inches into the debris bed. Figure 20 is a summary schematic showing the first group of six core debris grab samples taken from the core in late 1983. The samples are being examined to help define the following areas of interest:

- Physical and chemical forms of the particulate
- Identity and quantity of fission products released from the fuel and retained by the other core materials
- Release rates of radioisotopes from existing and freshly created surfaces (particularly cesium)
- Airborne potential for radioactive particles (fines)
- Unanticipated defueling equipment problems (i.e., filtration properties, settling rates, microhardness, crush strength, etc.)
- Hydrogen production as indicated by the extent of zircaloy and stainless steel oxidation
- Control material relocation
- Particle size distribution
- Peak temperatures experienced.

6.2.3.2 Fueled Rod Segments. The plan recommends six rod segments containing fuel, each 6- to 18-inches long, will be obtained from known locations at the periphery of the core void from either three radial and two

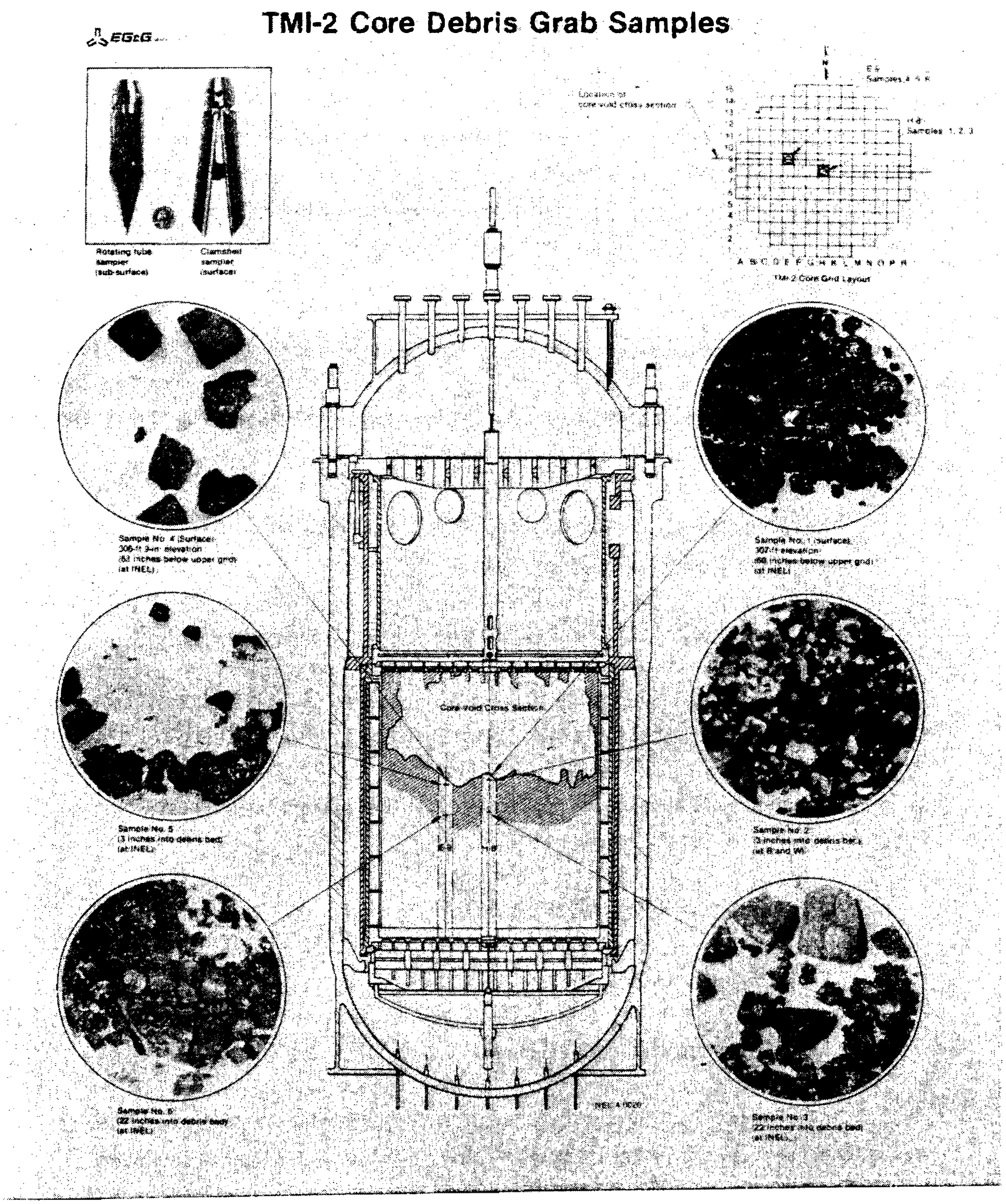


Figure 20. Summary schematic showing TMI-2 core debris grab sample acquisition.

axial locations or six radial and a common axial location. The main objective of the examination task is to characterize the fuel rods, which represent the boundary or transition zone between the melted/rubblized debris bed and the partially intact, standing fuel rod segments at the periphery of the core. It is desirable to obtain fueled rod segments that exhibit a gradation of damage. Such rods would present an ideal opportunity to investigate the sequence of damage events, as well as the type and extent of damage that precedes and leads to a loss of rod-like geometry. Such rods will require advanced preservation techniques to ensure that additional damage does not occur during shipping. Selected damage zones will be sectioned and encapsulated in epoxy for transverse metallography. The metallography will investigate the major damage features such as rod fragmentation, liquid phase formations, and the start of fuel liquefaction. The radial and axial migration of any liquid material and the nature of the UO_2-zircaloy interaction are other important research areas.

Radiochemical techniques will be used to measure fission products retained in various UO_2 structures (e.g., intact pellets, oxidized pellets, UO_2-zircaloy reaction products, zircaloy base metal, and liquefied fuel) as part of efforts to develop fission product partitioning data for the TMI-2 accident. Also, the residual structural integrity of the rods could be evaluated, where appropriate, by simple mechanical tests (e.g., ring compression). Peak fuel and cladding temperatures, and fuel burnup should be determined.

6.2.3.3 Core Stratification Samples. In order to properly characterize the extent of core damage, a study of the axial and radial condition of the core is necessary. The previous issuance of this plan recommended acquisition of fuel assemblies (e.g., intact, damaged, and stubs) from various positions within the vessel. These assemblies would have provided much of the stratification data needed to adequately define the damage conditions at the peripheral damage boundaries. It has since been determined that there can be at most only two intact standing fuel assemblies within the core, and they are positioned at the extreme outer periphery of the core region. Therefore, acquisition of fuel assemblies for off-site examination is no longer a major interest item.

In lieu of fuel assemblies, this revision of the plan recommends acquisition of four continuous, vertical core bore samples to the bottom of the core region. The core bores should be large enough in diameter and located so as to include several fuel rods and at least one control rod per core bore.

Two of the samples should be taken at locations coincident with access holes in the lower plenum, to allow inspection and sample acquisition of the lower vessel region. One such position is at K9. The samples must retain their undisturbed geometry as closely as possible to study the axial and radial stratification of the core. The samples are expected to exhibit every gradation of damage condition existing in the core (i.e., rubble, fragmented rods, eutectic formations, liquefied fuel, intact components, etc.).

A wealth of information is available from the core bore samples. The examination should be developed to assure maximum data is obtained. Virtually every major nuclear safety issue outlined in Section 4 can be addressed by examination of these samples.

If techniques to preserve debris bed stratification can be developed, the stratification specimens can be cross-sectioned at small axial intervals to reveal details of the unique strata. These strata can, in turn, be characterized for critical coolability factors such as (a) particle size, shape, and composition; (b) effective stratum porosity (packing density); (c) stratum thickness; (d) ratio of fuel-to-nonfuel debris; and (e) evidence of local dryout.

Metallographic examination also will be used to determine the behavior of the fuel assembly spacer grids. Eight Inconel spacer grids are located axially along each fuel assembly to provide lateral support and maintain rod-to-rod spacing. They may have helped hold the assembly together and delayed the onset of loss of rod-like geometry. Conversely, they may have acted as collection points for oxidized zircaloy debris and liquefied fuel. This would create localized regions of partial or total flow blockage with consequent high local temperatures. A zircaloy-Inconel eutectic reaction also may have occurred at the fuel-rod-to-spacer-grid contact points. The extent of this reaction, if present, should be documented, since it could represent a mechanism for rod failure very early in an accident.

The macrostructure and stratification of the core should be examined by metallography. The metallography will investigate major damage features such as rod fragmentation, liquid phase formations and their interactions with structural materials, and fuel liquefaction. The radial and axial migration of any liquid material and the nature of the UO_2-zircaloy interaction are other important research areas.

Radiochemical techniques will be used to measure the fission products retained in various UO_2-zircaloy reaction products and liquefied fuel, as part of the efforts to develop fission product partitioning data for the TMI-2 accident. Also, the residual structural integrity of the fuel assemblies and their components could be evaluated, where appropriate, by simple mechanical tests.

Individual specimens of fuel, cladding, or other assembly components should be removed for individual analysis after the integral examination is complete. The particular examinations to which these samples should be subjected depend on their appearance and the type of damage they exhibit. Such examinations are likely to include retained fission product analysis, metallography, SEM, burnup analysis, fuel oxidation state, UO_2 density, and mechanical properties measurements.

In examining fuel rod segments, gamma scans should be used to identify and quantify axial migration of volatile fission products (e.g., cesium) away from the hottest portion of the rod.

Ballooning of the cladding is best studied using transverse metallography. Sequential grinding through the encapsulated sections will reveal total circumferential elongation; cladding thinning; shape and axial extent of the balloons; cladding oxidation; cladding temperatures both axially and circumferentially; and the nature of fuel fragmentation, relocation, and washout.

Because Ag-In-Cd dispersal during core melt accidents is not well understood and is being studied in small-scale experiments, it is important to examine the behavior of the control materials. The extent of oxidation or damage of the control rods relative to the nearby fuel rods would be informative. The mechanism of the stainless steel cladding breach and failure, together with the axial and radial flow of the control materials and the associated propensity for blockage and breaching of the zircaloy control rod guide tubes, should be examined by investigating these components.

The preliminary examination of filter debris implies that the control alloy was widely dispersed. It is not known if this dispersal is a result of cladding failure due to oxidation and fragmentation, rod rupture from high alloy vapor pressure, or ductile failure of the stainless cladding at high temperature.

Determining the behavior of the nuclear control materials during the TMI-2 accident is an important data acquisition task because of the implications for flow blockage and reactor recriticality. Non-uniform dispersal of the control material in the core would imply that recriticality could be a concern in any future accident if control of coolant boron levels was lost. The examination should try to determine how the control rods degrade and how the Ag-In-Cd control material reacts and moves.

6.2.3.4 Distinct Fuel Assembly and Control Rod Cluster Components. CCTV inspection of the core void and the underside of the plenum revealed a myriad of fuel assembly and control rod cluster components (i.e., cladding, control rods, spiders, spacer grids, end fittings, hold down springs, etc.) either hanging from the plenum or lying loose in the core bed. The condition of the components varys from basically intact to severely damaged.

Approximately 40 such specimens should be retrieved from the core for subsequent examination. Sample selection should be based on CCTV inspection; spiders, spacer grids and end fittings should be taken from the same (known) positions. If possible, hanging fuel assembly stubs should be obtained.

The fuel assembly upper end fittings and their control rod spiders, as well as other fuel assembly components that exhibit a range of damage, should be obtained. The end fittings and spiders are stainless steel components originally located immediately above the fuel rods, although they may be found in a variety of locations in the core debris. As observed in the CCTV camera inspections, some of these components have remained attached to the underside of the plenum even though the oxidized fuel assembly beneath them fractured and fell away. Others have fallen into the core with the fuel assembly debris. The main damage to these components probably resulted from steam oxidation, although camera inspections indicated that temperatures may have been high enough to produce localized melting. Metallography will be the main examination technique to document the oxidation, melting, and other reactions of these components. Oxide-thickness measurements also will be used to estimate the hydrogen released during steam oxidation of these components, and thus their contribution to the total hydrogen generation and the containment integrity issue. In addition, fission product plateout on these surfaces will be measured to assess their role in radionuclide retention in the core. Peak temperatures should be determined for the components to help profile the maximum temperatures experienced in the core region.

6.2.3.5 In-Core Instrumentation. The 52 assemblies instrumented with in-core instrument strings [consisting of one coolant exit thermocouple and seven axially spaced self-powered neutron detectors (SPNDs)] have provided data to help understand the TMI-2 accident sequence. In an effort to improve this understanding, attempts will be made to obtain several debris specimens from within the core region, containing both reasonably intact and damaged sections of the instrument strings. Damaged instruments will be subjected to detailed metallography to determine the extent of oxidation, melting, eutectic formation, and chemical reaction.

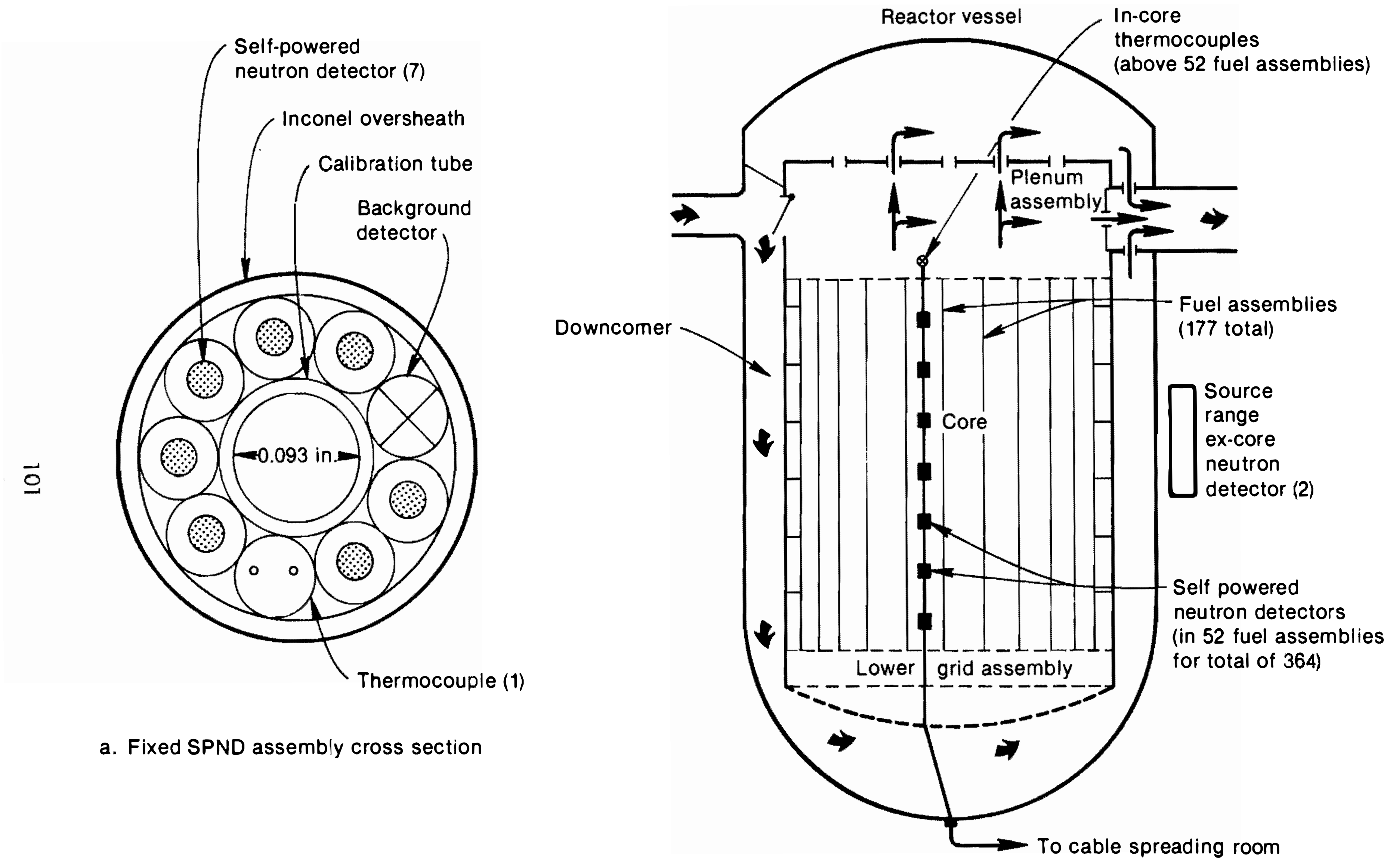
Each instrument string is comprised of one Chromel-Alumel thermocouple, seven SPNDs, and one gamma compensating background detector. Because of the variety of materials in the instruments and their individual reaction variances during the TMI-2 accident, the instruments can provide the best temperature profile data of any in-core component. Figure 21 is a schematic of an in-core detector assembly.

6.2.4 Miscellaneous Samples

This section presents recommendations for specific data collection tasks, as well as guidelines that will be used for samples taken from areas other than the reactor vessel and reactor coolant system. There are two tasks recommended to satisfy the needs of this plan: reactor building basement solids and reactor coolant drain tank. Both tasks are centered at determining the quantity and type of fission products that exited the primary system, as part of the overall source term calculations. A brief description of these tasks is outlined below.

6.2.4.1 Reactor Building Basement Solids. Four samples of debris have been collected from the reactor building basement. Examination included radiological and chemical elemental analyses of the solids and liquids, and particle sizing of solids in the slurry sample. This task is complete.

6.2.4.2 Reactor Coolant Drain Tank. CCTV examination of the reactor coolant drain tank and vent pipe has been completed. Samples of sludge taken from these areas will be examined using similar techniques used on the reactor building basement solids.



a. Fixed SPND assembly cross section

b. Instrument locations

Figure 21. Schematic of an in-core detector assembly.

7. SUMMARY AND CONCLUDING REMARKS

Specific recommendations have been made herein for obtaining data within the scope of the TMI-2 core examination plan that will support resolution of the major safety issues facing the nuclear power industry. However, the core examination activities should be reviewed periodically in the light of new information, as it becomes available. While the types of data identified herein will not change, recommendations on how to satisfy these data needs with data from this program possibly may change as new information is developed during recovery operations.

The following concluding points should be noted:

- Serious questions can be raised concerning the effects of processes that have occurred in the interim since the accident on the data of interest (i.e., the extent to which TMI-2 retains some "memory" of the phenomena of interest in the currently retrievable data is not clear). Although there is no shortage of expressed opinions on the extent to which interim effects have confounded the useful information that can be obtained, it should be remembered that much of this opinion is speculative since the interim phenomena are even less understood than the accident phenomena.

- Care should be taken in retrieving and shipping samples so as to avoid altering the surface deposits and other features of interest. Any carelessness could further compound the problems of interpreting the data to discover what happened during the crucial period of the accident. Also, the location from which each sample is removed should be clearly documented, as should the sample removal and handling techniques.

It is recognized that the TMI-2 core examination plan must develop strong interfaces with other organizations and other GEND-sponsored activities. These interfaces are essential because the core examination plan outlined herein clearly crosses currently established areas of responsibility. Examples of organizations and other GEND activities that must participate with DOE in the continued definition and performance of this plan are as follows:

- General Public Utilities Nuclear Corp. (GPU Nuclear)--Continued interface with GPU Nuclear and its contractors is essential to efficient scheduling of the core examination tasks and integration of those tasks into the reactor cleanup and recovery program. GPU Nuclear is the license holder and has ultimate responsibility for the TMI-2 plant.

- Electric Power Research Institute (EPRI)--EPRIs involvement in the cleanup and requalification of selected mechanical and primary system components means that some of its tasks will generate data relevant to the issues discussed herein.

- U.S. Nuclear Regulatory Commission (NRC)--NRC representation on the TEG and GEND advisory groups ensures their involvement in the program. NRC should act as a link between the core examination activities and the variety of severe fuel damage research programs that it sponsors. NRCs involvement in examination planning and research coordination will not compromise its regulatory function.

- Technical Evaluation Group (TEG)--This DOE-sponsored group should meet periodically to review the plan and its implementation.

- Technical Assessment and Advisory Group (TAAG)--This group, which advises GPU Nuclear on reactor recovery tasks, should be familiar with the plan.

- Fuel Examination Facilities--TMI-2 core examination management will continue to work with commercial and governmental fuel examination facilities to ensure cost-effective use of these resources.

- <u>Other TMI-2 Programs</u>--Several ongoing tasks have areas of technical overlap with this plan [e.g., Waste Immobilization Program (some data will be generated on the fission product content of selected types of reactor wastes)].

The TMI-2 core examination plan is extensive, yet when the examination is reduced to its constituent tasks, it will require few techniques and facilities beyond those now available for post-irradiation examinations or those being developed for reactor defueling. All on-site (<u>in situ</u>) examinations, which are intended to document the post-accident condition of the core, are designed to be compatible with the plant-recovery and reactor-defueling operations. The core examination emphasizes acquiring representative samples of core and reactor internals damage and ensuring that adequate precautions are taken to prevent damage or alteration of these samples during transit to remote handling laboratories. As the TMI-2 specimens are received and catalogued, detailed off-site examinations described herein can begin. While the specific TMI-2 core examination plan will likely change and evolve as knowledge of the core damage increases, the examinations always must be based on meeting specific examination objectives and having a cost-effective impact on the underlying technical issues. Only then will the program have served its purpose of having a permanent and positive impact on light water reactor safety and technology.

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