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Decontamination

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Final Report
September 1990

The Cleanup of Three Mile Island Unit 2

A Technical History: 1979 to 1990

Prepared by
Grove Engineering, Inc.
Rockville, Maryland

REPORT SUMMARY

SUBJECTS	Light water reactor safety / Risk analysis, management, and assessment / Radiation source term / Radioactive waste management
TOPICS	Reactor accidents Source term Degraded core accidents Radioactive waste management Decontamination
AUDIENCE	Nuclear utility managers and engineers

The Cleanup of Three Mile Island Unit 2

A Technical History: 1979 to 1990

The fuel damage and the release of fission products after the Three Mile Island unit 2 (TMI-2) accident required unprecedented decisions regarding the enormous cleanup operations. The rationale for those decisions will provide valuable information for other managers who may face similar situations. Planning and response procedures can benefit from the insights gained from the TMI-2 accident.

BACKGROUND Limited applicable experience hampers effective planning and response recovery operations in accidents involving fuel damage. TMI-2 demonstrated that effective cleanup responses remain unpredictable, determined largely by the specific situation. Managers must anticipate an extended period of plant recovery and cleanup following such an accident.

OBJECTIVE To describe the decision-making process during the TMI-2 cleanup, focusing on the project manager's area of responsibility and needs.

APPROACH Researchers gathered information by participating directly in the cleanup. Research by others and subsequent interviews with executives, managers, and workers supplemented their experience. The information collected was then analyzed, and substantive options, factors, and relevant decision-making data were identified and summarized.

RESULTS The report focuses on seven major aspects of the cleanup: cleanup management, stabilization, personnel protection, data acquisition and analysis, radioactive waste management, decontamination, and defueling. A chronological narrative identifies the major questions and challenges facing TMI-2 cleanup managers, describes the influencing factors and the options available, and presents the final decisions and their consequences. An extensive bibliography provides sources for more-technical details.

EPRI PERSPECTIVE EPRI sponsored this technical history project to preserve the logic and consequences of decisions made during the TMI-2 cleanup. The results may help streamline recovery operations from similar accidents involving damaged fuel and fission-product release. The history will assist others in anticipating the consequences of a fuel damage event and in training managers accordingly. The narrative also serves as a comprehensive source of information about the TMI-2 cleanup, providing a detailed chronology for those making a further study of the postaccident events.

PROJECT RP2558-8
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Nuclear Power Division
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The Cleanup of Three Mile Island Unit 2
A Technical History: 1979 to 1990

NP-6931
Research Project 2558-8

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Rockville, Maryland

ACKNOWLEDGMENTS

The TMI-2 history could not have been written without the strong cooperation of GPU Nuclear. Throughout the research and writing, corporate management provided insights and urged cooperation from all TMI-2 cleanup participants to ensure that the lessons of TMI-2 are available. Ed Kintner, executive vice president, supported the project from the beginning, reviewed each section, and gave the project important corporate and industry-wide visibility. Frank Standerfer, former director of the cleanup, was extremely interested in the work, both for the future and to provide insight into cleanup operations still in progress. His successor, Mike Roche, carefully reviewed the document and provided very useful comments. In addition, Jim Hildebrand, director of radiological controls for GPU Nuclear, and Greg Eidam of Bechtel National, Inc., contributed invaluable time and attention.

The authors would especially like to thank EPRI for its sustained support. Ray Lambert, our project manager, has patiently guided the work. John Taylor has continued to believe in the value of documenting the lessons of TMI-2. The EPRI TMI-2 site office under Ray Schwartz provided generous support in every aspect. Denise Gillin and Arlene Fischer made the research and review of this history as easy as possible.

The U.S. Department of Energy was also a strong supporter of the history project, making both office space and research information available. Willis Bixby and Dave McGoff backed the creation of the project, and Norm Klug continued an active role in its review. TMI-2 Program Manager Willis Young was an early and consistent supporter.

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ABSTRACT

The Electric Power Research Institute has sponsored a technical history project to ensure that the logic and consequences of decisions made during the Three Mile Island Unit 2 (TMI-2) cleanup are available for recovery from an accident involving damaged fuel and fission product release. The objectives of the history project are to identify the major questions and challenges facing management; describe the influencing factors and the options available; and present the final decisions and their consequences. This history of decision-making is intended to assist a project manager who must respond to a fuel damage accident, even if the scale is much smaller than TMI-2. The history has focused on decisions related to seven major aspects of the cleanup: cleanup management, postaccident stabilization, personnel protection, data acquisition, radioactive waste management, decontamination, and defueling. A detailed chronology and extensive bibliography accompany the text.

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INTRODUCTION

1.1 Objective

The cleanup of Three Mile Island Unit 2 (TMI-2) has been one of the most arduous and expensive programs conducted by the U.S. nuclear industry. Managers facing recovery from future fuel damage events, even if not as severe, will want answers to many questions regarding TMI-2: Why did it take so long to clean up after the accident? Why were certain technical strategies selected over others? What factors influenced these decisions? What were the consequences?

Recognizing this, EPRI has sponsored a technical history project to ensure that the logic and consequences of decisions made at TMI-2 are documented. This report is designed to be a primary, comprehensive source of information for future managers facing cleanup after an accident involving fuel damage and fission product release. It also provides a source of technical references for anyone who wishes to find out more details about what took place during the cleanup.

To anyone involved in the TMI-2 cleanup, the aftermath of the accident was a challenging time that presented a host of problems never before encountered in such magnitude. The mix of technical, economic, and institutional demands was fascinating, complex, and instructive.

Writing this history took on some of the aspects of the actual cleanup—unique R&D work of unexpected proportions. The accident and cleanup readily lend themselves to dramatic description and critique; for example, this emotional extract from the eminent *Three Mile Island: A Report to the Commissioners and to the Public:*

"The nine months since the accident have passed with surprisingly little change...Cleanup plans and efforts are mired in prolonged debate. The auxiliary building is slowly being cleaned up; the wastewater outside containment is finally being purified. A camera has been introduced to the inside of the reactor containment building. This long unseen space, where fiercely radioactive gas billowed and hydrogen burned during the course of the accident, appears on the video screen, still, shiny, dripping with humidity like a robot rain forest. TMI-2 seems suspended in time, still waiting to be opened, to be cleaned, or repaired, or torn down" (Rogovin 1981, p. 164).

From the perspective of the cleanup staff, the period described represents the culmination of nine months of hectic and successful stabilization work in an adverse technical and political environment.

There were also many differing views on why the cleanup took so long. The decision to depart from existing standards governing allowable releases and to use more stringent requirements and new regulatory processes not applied to other nuclear plants could be cited—as could the timeliness and quality of technical proposals, funding constraints, and the ability to meet construction schedules in performing work of unknown magnitude (Comptroller General 1981).

In addition, public mistrust and intervention—spawned in part by poor communication and a precautionary evacuation order following the accident—were factors. Finally, the initial management approach of planning the entire cleanup operation as if it were a design and construction project could be questioned. Instead, the cleanup could have been managed as a major unscheduled plant outage (Feinroth 1985).

The TMI-2 technical history does not resolve all of these perspectives nor second-guess the course of events. Its objectives are to identify the major questions and challenges facing management; describe the influencing factors and the options available; and present the final decisions with a summary of results. Paths not taken are also described—they are as instructive as those actually pursued. Every accident will be different, but the decision-making process that occurred at TMI-2 can be used to form the basis for a cohesive response.

This project is part of the role that the U.S. nuclear power industry has played in the TMI-2 cleanup, beginning immediately after the accident when senior technical staff was provided and continuing with the transfer of expertise and technology.

1.2 Approach

As the cleanup evolved, it developed four operational phases in which certain types of tasks were dominant:

Stabilizing the Plant	1979–1980
Waste Management	1980–1983
Decontamination	1981–1985
Defueling	1984–1990

There are no specific dates on which one phase ended and another began. Rather, these periods emphasized different activities; e.g., defueling planning actually began in 1979, while waste management will continue into the foreseeable future. Figure 1-1 depicts the major events of the TMI-2 cleanup in an overview timeline. Each major technical section of the history likewise has a timeline that expands on this summary-level one. Appendix A provides a detailed chronology of the entire cleanup.

Because most of the engineering, construction, and operational activities are related to the four phases introduced above, they have been chosen as the basic topics for the history. In addition, the topics of planning and management, personnel protection, and data acquisition and analysis have been included to provide a complete spectrum of subjects of interest to a future project manager.

The sections/topics of the TMI-2 technical history are:

- **Planning and Management**—A discussion of organizational structure, costs, funding sources, regulatory aspects, and outside resources.
- **Stabilization**—The first 15 months after the accident in which cold shutdown of the reactor was achieved, the containment was purged of radioactive gases, control was established over contamination and radioactive water, and access was gained to contaminated areas essential for future recovery operations.
- **Personnel Protection**—The programs and techniques required to protect workers under extraordinary conditions of radiation, heat, and humidity.
- **Data Acquisition and Analysis**—The acquisition and vital importance of data about plant and reactor vessel conditions to ensure safety and to support planning, operations, and the transfer of information to the outside world.
- **Waste Management**—The extended efforts to process water contaminated by the accident and during defueling, and the immobilization, packaging, storage, and shipment of radioactive waste.
- **Decontamination**—The decontamination strategies and methods to reduce worker exposure in support of defueling and general access.
- **Defueling**—The evolving development of strategies and techniques for removing and shipping highly damaged reactor fuel, with its unknown conditions and technical challenges.

One other subject is interwoven with these seven—the post-defueling status of the plant, including cleanup end points. This topic is introduced in Section 1.4 and discussed in Section 2.

1.3 Background

Three Mile Island is located in south central Pennsylvania on the Susquehanna River, approximately 16 kilometers south of Harrisburg, the state capital. Unit 1, an 800-MW, pressurized water reactor (PWR), was constructed first and went on line in September 1974. At the time of the accident, it was shut down for a planned outage; it was not allowed to restart until 1985. Unit 2, a 900-MW PWR, was added to the rate base in December 1978.

Both units are owned jointly by Metropolitan Edison (50%), Pennsylvania Electric Company (25%), and Jersey Central Power & Light (25%); they are subsidiaries of the General Public Utilities (GPU) system. GPU serves 1.7 million customers in a land area covering one half of Pennsylvania and New Jersey, making it the 16th largest investor-owned utility in the United States. GPU Nuclear Corporation was incorporated in 1980. In 1982, it succeeded Metropolitan Edison as the licensed operator of the nuclear plants in the GPU system (TMI-1 and -2, and Oyster Creek in northern New Jersey).

The TMI-2 accident began at 4:01 a.m. on Wednesday, March 28, 1979. It was terminated approximately four

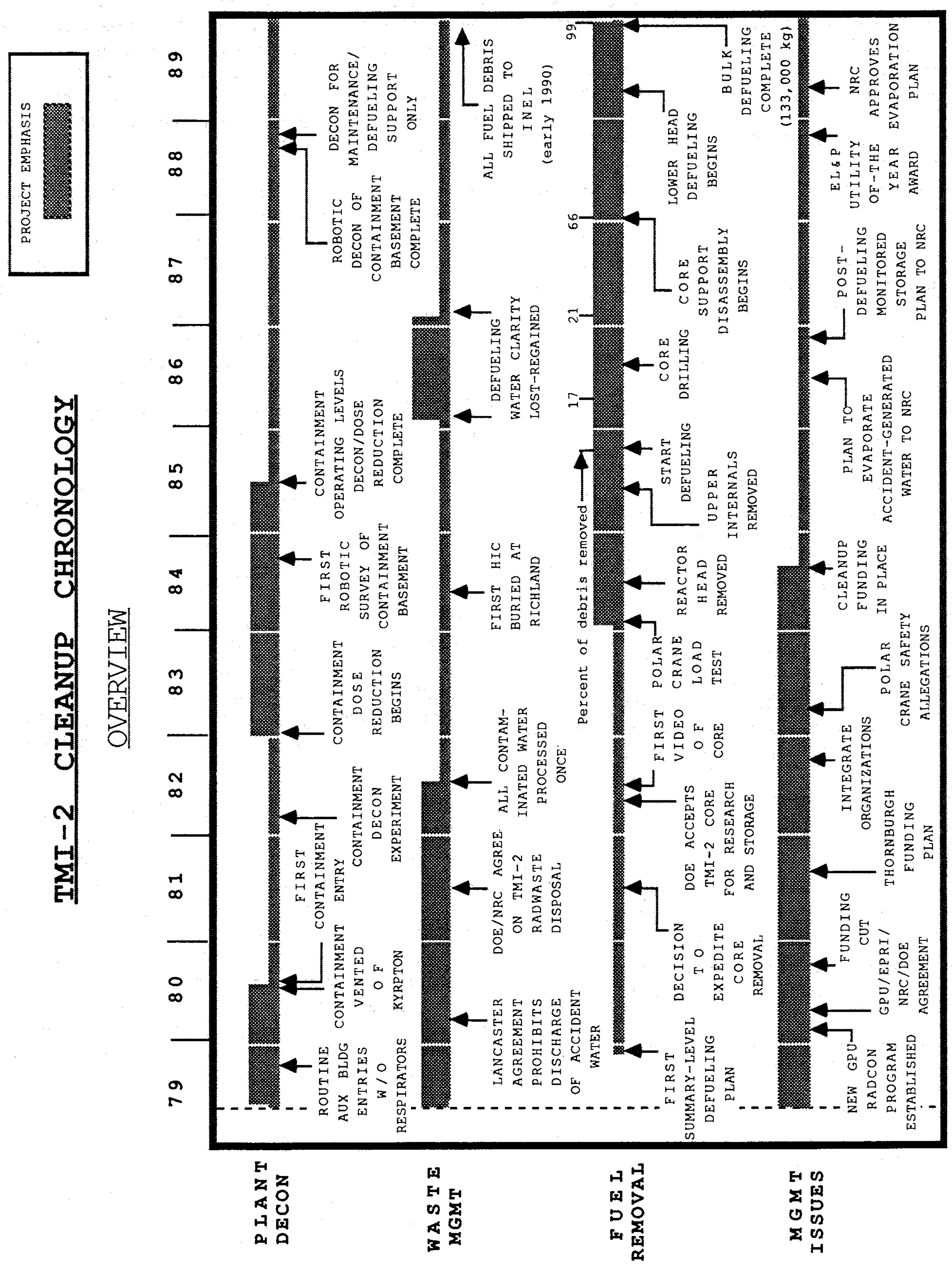

Figure 1-1. TMI-2 Timeline: Overview

hours later, although full control of the reactor was not established for some time afterward. The accident itself is not discussed in this history because it is so thoroughly documented elsewhere (e.g., NSAC 1979; Kemeny, et al. 1979; Rogovin, et al. 1981; Tolman, et al. 1986; ANS 1989).

The narrative begins with the efforts to establish and ensure long-term control of the reactor, and to deal with the enormous consequences of the accident. The damage conditions inside the reactor vessel are described in Appendix B so that the reader can understand both the difficulty of the cleanup and the contrast between the anticipated and actual conditions in the vessel. Photo 1-1 shows Three Mile Island in the early 1980s; Figure 1-2 depicts the layout of the site before the accident. Figures 1-3 through 1-5 show different computer-generated views of the plant.

1.4 Post-Defueling Plant Status

Beyond the strategies and techniques for cleaning up TMI-2 lay the question of the plant's final disposition. For some time after the accident, before the actual damage conditions were well understood, GPU envisioned returning the unit to service. Extensive planning efforts were directed toward this end.

By 1981, a shift in thinking had occurred, and over the next three years reached the status of a formal program strategy in which cleanup work would be conducted without regard to the final disposition of the plant (DeVine and Negin 1984). The determining factor was the extensive damage that had been revealed. In turn, this led to a growing understanding of the resources, equipment, and plant modifications required to remove the core debris and eliminate any threat to public health and safety.

In 1986, GPU proposed a TMI-2 post-defueling monitored storage (PDMS) condition to the NRC. PDMS would place the plant in a condition of safe, secure monitored storage that would pose no hazard to the public and would permit TMI-2 to be decommissioned at the same time as TMI-1 (GPUN 1986). GPU's primary logic for PDMS was that, with the plant safe in terms of the public, additional exposure to workers conducting a total cleanup was not justified. Deferring dose-intensive decontamination work until the future will lower exposure rates because the radioactivity will decay and more efficient decontamination techniques will be available (e.g., robotics).

The development of the PDMS concept followed an in-depth survey of existing regulations and documentation, which uncovered no specific regulations for a facility in a postaccident condition and very little guidance for a nonoperating and defueled facility. Several major licensing issues were identified as pertinent: safety-related equipment, quality assurance, security, fire protection, emergency planning, effluent limits, and natural phenomena (Smith and Byrne 1989).

The resulting review of these issues vis-a-vis 10 CFR Part 50—which was an awkward fit—was formally docketed in a proposed license amendment and PDMS safety analysis report submitted in 1988 (GPUN 1988). The NRC performed an environmental impact review of PDMS and found it acceptable (US NRC Supplement 3, 1989). In addition, GPU obtained NRC approval for a change in technical specifications that permitted the plant to progress through three facility modes to a fourth—the PDMS status. Modes 1 to 3 included defueling, onsite storage, and shipment off site of greater than 99% of the original core inventory.

NRC approval of the license amendment, safety analysis report, and Facility Mode 4 (PDMS) technical specifications was not received at the time of this writing. Considering the potential for extended regulatory and public review, PDMS approval and implementation is in the future.

Although formal approval has not been received, developing the concept of PDMS brought cleanup project goals and emphases into focus—it gave direction and an achievable end point for the cleanup. Within the framework of working toward PDMS, the majority of resources were expended on defueling operations, which eliminated the possibility of a recriticality.

With defueling completed in early 1990, resources could then be focused on: 1) completing decontamination to end point levels; 2) quantifying residual fuel in the plant; and 3) completing the draindown, modification, or layup of systems and equipment for long-term storage (Fonner 1989). Decontamination efforts will remove and/or stabilize residual contamination to prevent release to the environment and to minimize occupational exposure to workers conducting necessary plant monitoring, maintenance, and inspections. Section 7 presents the decontamination end point objectives. The potential for offsite exposure from TMI-2 will be well below the normal NRC guidelines for operating plants (10 CFR Part 50, Appendix I).

A more detailed description of PDMS is beyond the scope and timing of this history; however, the references cited above provide ample guidance to the development and implementation of the monitored storage condition.

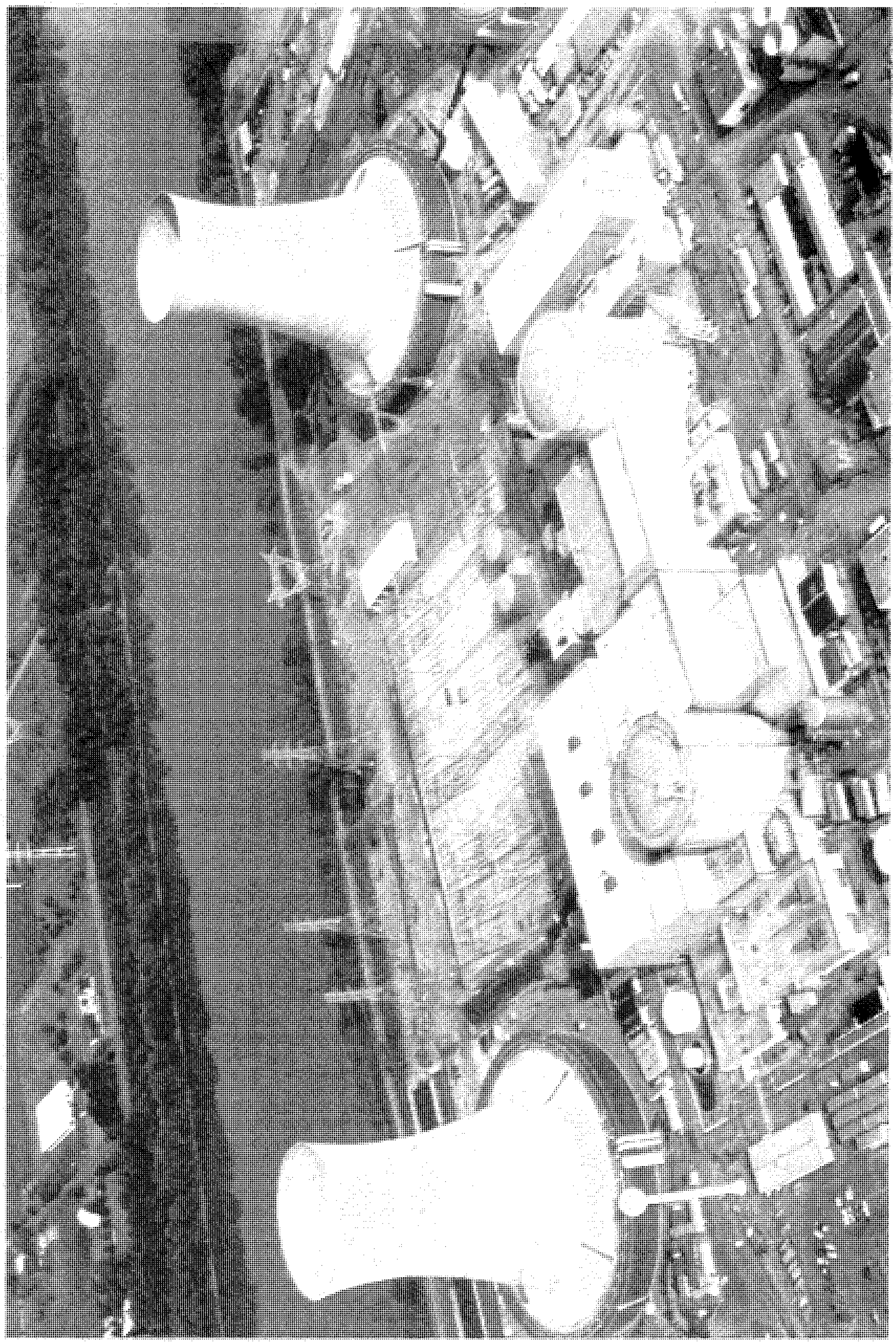

Photo 1-1. Three Mile Island (Looking East — Unit 2 on the Right)

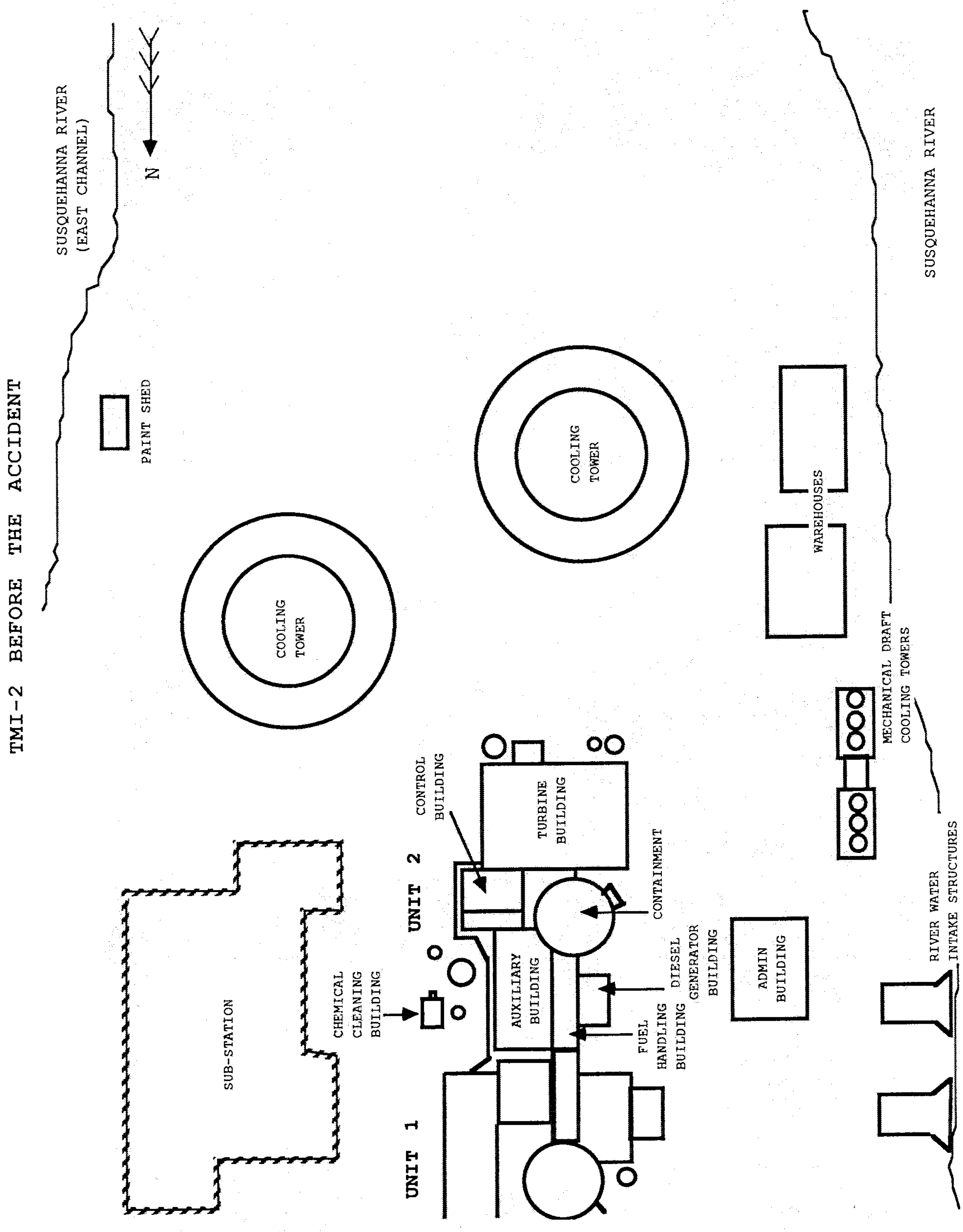

Figure 1-2. TMI-2 Before the Accident

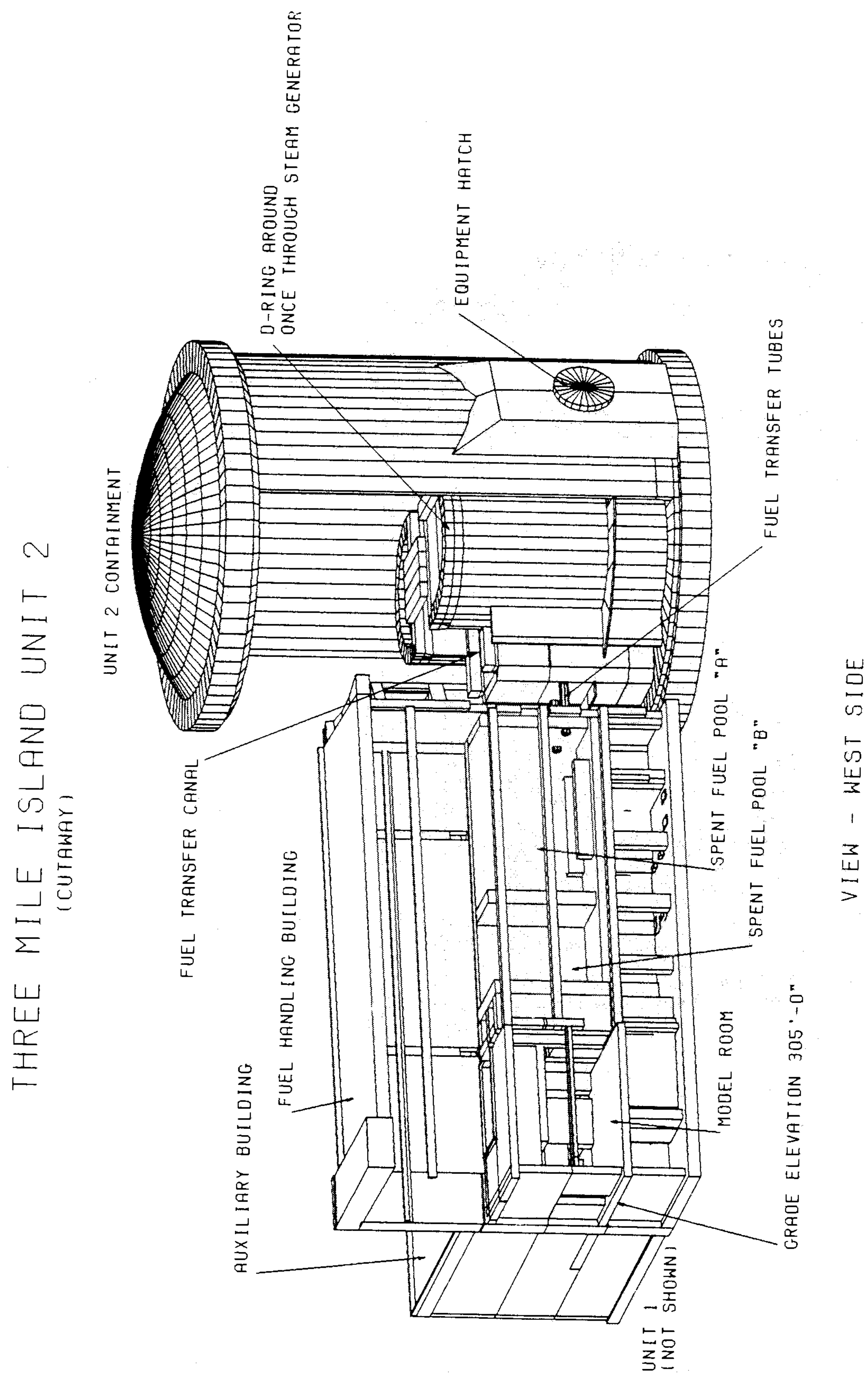

Figure 1-3. TMI-2 (View — West Side)

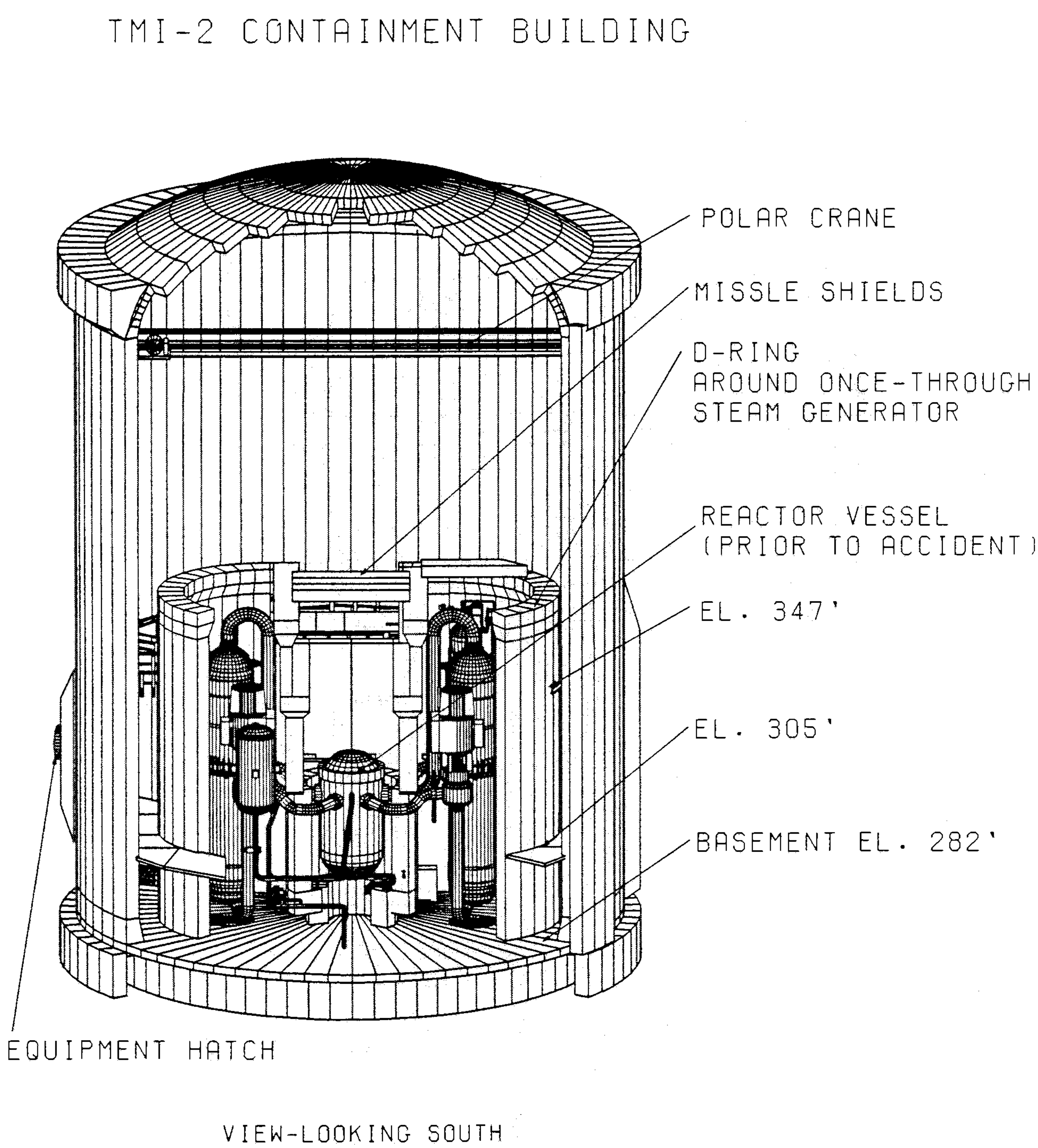

Figure 1-4. TMI-2 Containment Building (View — Looking South)

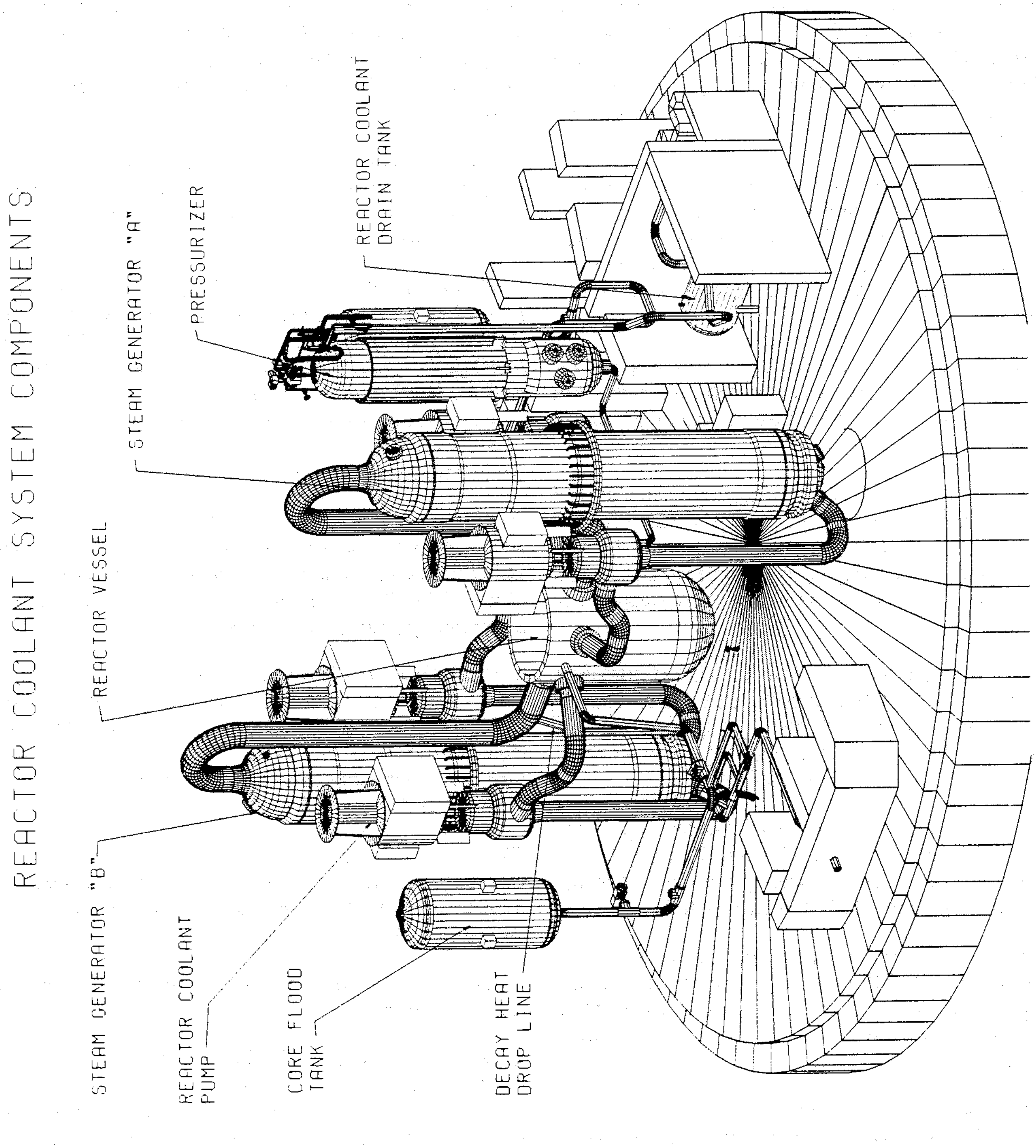

Figure 1-5. TMI-2 Reactor Coolant System

1.5 References

Many reports and papers document the 11-year cleanup. In most cases, however, the reason for choosing a particular course of action was not clearly recorded because: 1) the choice was clear at the time and needed no further justification; 2) the decision resulted from an evolving process rather than being a single step; or 3) the staff was just too busy implementing the decision. Consequently, many of the insights in the history are based on interviews and first-hand knowledge.

The bibliographies accompanying each section provide references to both the technical details and the documents that served as the primary planning tools at various stages of the cleanup. The following list identifies several of the most important references for any extended study of the cleanup:

Three Mile Island: A Report to the Commissioners and to the Public (Rogovin, et al. 1980)—An excellent research effort and well written account of people and technology during and shortly after the TMI-2 accident.

Nuclear Technology, Vol. 87 (ANS 1989)—Contains the full text of 138 papers presented at the 1988 ANS/ENS Topical Meeting on the TMI-2 Accident: Materials Behavior and Plant Recovery Technology. A comprehensive source on both the accident scenario and many of the cleanup operations.

TMI-2: A Learning Experience—ANS Executive Conference (ANS 1985)—A collection of papers presented to industry executives midway through the cleanup summarizing strategy, operations, and management, and the effects of the accident/cleanup on the industry.

Defueling the Three Mile Island Unit 2 Reactor: Approaches, Techniques, and Lessons Learned (Owen and Bentley 1990)—An EPRI-sponsored report cataloging the tools and equipment used to defuel the TMI-2 reactor vessel.

The TMI-2 Data Acquisition and Analysis Experience (Urland and Babel 1990)—An EPRI-sponsored report on techniques and equipment used to characterize conditions at TMI-2, emphasizing new methods or the novel use of proven techniques.

The TMI-2 Waste Management Experience (Deltete and Hahn 1990)—An EPRI-sponsored report providing detailed descriptions of the water processing systems at TMI-2 and their operating experience. It also details radioactive waste storage and shipment, and the shipment of the reactor core debris to Idaho.

The TMI-2 Decontamination Experience (Urland, Pearlstein, and Schwartz 1990)—An EPRI-sponsored report of decontamination techniques and equipment used during the cleanup.

Historical Summary of the Fuel and Waste Handling and Disposition Activities of the TMI-2 Information and Examination Program (1980–1988) (Reno and Schmitt)—An overview of the DOE role in the cleanup and a bibliography for further reading.

TMI-2: Lessons Learned by the U.S. Department of Energy—A Programmatic Perspective (Reno, Owen, and Bentley 1990)—A listing of TMI-2 cleanup lessons learned from the perspective of the DOE.

Final Programmatic Environmental Impact Statement Related to the Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979, Accident at Three Mile Island Nuclear Station, Unit 2 (USNRC 1981)—A broad study of Three Mile Island and environs plus evaluation of many potential cleanup technologies. Three supplements evaluate worker exposure, disposal of mildly radioactive processed water, and post-defueling options for the plant.

Post-Defueling Monitored Storage Proposed License Amendment and Safety Analysis Report (GPU Nuclear 1988)—The utility licensing document describing the proposed condition of the TMI-2 plant for the period after the cleanup and pending final decommissioning.

Defueling Completion Report (GPU Nuclear 1990)—Contains a summary of the accident and cleanup, but is mainly focused on the post-defueling status of the plant and the characterization of remaining fuel debris. Its primary purpose is to demonstrate that a recriticality cannot occur at TMI-2 during Mode 4.

TMI-2 Annual Safety Advisory Board Reports—Issued to the public by a group of independent advisors specializing in the fields of health physics, nuclear engineering, risk analysis, and community relations.

In addition, several other types of documents serve as basic reference sources:

GPU Nuclear documents—Technical plans, planning studies, data reports, and technical bulletins were issued; safety analysis reports, technical evaluation reports, system descriptions, and procedures were the licensing/work documents.

GEND and national laboratory reports—The GEND reports are a series of detailed reports issued as a result of the GPU, EPRI, NRC, DOE agreement. GEND and national laboratory reports are available through the National Technical Information Service (NTIS).

EPRI—Numerous full reports, technical briefs, and nuclear notes were issued throughout the cleanup; videotapes summarize the accident and cleanup. Available from EPRI or NTIS.

Data bases—GPU Nuclear maintains an extensive microfilm record of all important plant documents in its CARIRS computerized data base. A comprehensive computerized record of cleanup-related documents and reports from the U.S. Department of Energy's Technical Information Office is available at the INEL technical library in Idaho Falls, ID.

1.6 Corporate Affiliations

The TMI-2 cleanup was completed through the cooperation of many distinct organizations and individuals. Developing a nomenclature that would fairly represent the role of each party proved an impossible task. To illustrate: over 150 organizations/companies responded within two weeks to the accident. The integration of GPU Nuclear staff with its many contractors created one cleanup organization in 1982, further blurring distinctions.

Occasionally, some of the major companies or organizations must be singled out in order to provide context or to make a point. In general, however, the responsible organizations will be designated as follows: "GPU" will be used when the reference is specific to the utility (e.g., Radiological Controls, licensing issues, or funding). "The cleanup project," "the project team," or "management," will be used in reference to operations/decisions that embraced the entire cleanup effort (e.g., defueling, decontamination).

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2 MANAGEMENT

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2

MANAGEMENT

2.1 Overview

The management of the TMI-2 cleanup was uniquely demanding. It not only comprised a complex technological mixture of necessary innovation and unfamiliar safety issues, but saw many of the technical decisions influenced by nontechnical factors.

The technical decisions and course of the cleanup were inextricably bound with issues of management organization, planning, funding, a skeptical public, media spotlighting, and regulatory investigations and restraints. Most technical decisions involved debate reflecting these issues. The challenge of this section is to sort through to the real management and planning decisions related to the technical progress of the cleanup.

2.1.1 Basic Management Decisions

Questions about how to do almost everything had to be evaluated in light of the unprecedented postaccident situation. How to organize? How to fund? What were the overall objectives? The answers resulted in a new company, a unique cleanup organization, shared funding of the work, and novel forms of federal and industry involvement.

In programmatic terms, the following major management decisions were made:

- **Survive**—In a fundamental decision, GPU elected to fight the threat of bankruptcy and potential federal takeover of the cleanup. The utility created a subsidiary devoted strictly to nuclear matters (GPU Nuclear) and a division within it devoted solely to the cleanup of TMI-2. It also created a support division chartered to concentrate on radiological controls. By physically and operationally separating Units 1 and 2, GPU removed one potential argument against the restart of Unit 1, which was essential for corporate health. The credibility and progress of the Unit 2 cleanup minimized the potential of it being used as an issue in the Unit 1 restart proceedings.
- **Ensure Utmost Safety**—The decision to perform the work with safety paramount guided the cleanup. Although at times this was carried to conservative extremes—adding technical difficulties, expense, and time—no alternative was acceptable. In fact, the cleanup was carried out at a personnel radiation exposure level within the NRC's estimate (less than 6500 person-rem) and with an OSHA lost-time accident rate better than at many operating plants. The overall goal of the cleanup was to establish a condition of stability and safety such that there was no risk to public health or safety.
- **Use Experts**—GPU immediately realized that many aspects of the cleanup were beyond its expertise. The use of resources from the government, other utilities, national laboratories, and universities brought a sophisticated technical presence to the cleanup. In particular, the U.S. Department of Energy laboratories had skills and special facilities that did not exist elsewhere. Combining these outside resources with the onsite workforce was difficult, but the combination brought much needed technical and financial support, new ideas, and a channel to the worldwide technical community.
- **Hire a Contractor**—In making this decision, GPU hired Bechtel, the largest nuclear power A/E-constructor in the world; Bechtel had resources and expertise, or access to them. An alternative would have been for GPU to have increased its staff and hired subcontractors—a drain on resources that GPU was not in a position to undertake. By hiring a contractor, GPU could get back on its corporate feet while performing normal plant operations. In 1982, GPU decided to integrate with Bechtel staff and other subcontractors to form one cleanup organization. In itself, this was a major and essential step that caused some painful adjustments and delays.

- **No Restart**—For some time after the accident, GPU envisioned returning Unit 2 to service or at least did not preclude a restart. Public opposition to restart would have been intense. As the extent of damage to the reactor core and expense of plant refurbishment became evident, the decision was made to focus on the defueling effort and work without regard to the final disposition of the plant. At first, this was difficult to accept for engineers and operators trained in maintaining or improving an operating power plant. The overwhelming advantage was that the decision focused available resources on near-term issues. Eventually, GPU decided to place the plant in a long-term monitored storage condition after fuel removal.

- **Pursue Flexible/Parallel Approaches**—No one knew how hard the cleanup would be or how long it would take. No one knew what the conditions were inside the reactor vessel or what defueling tools would work. In this situation, project management found, time and again, that schedules and plans were quickly outdated. The only practical approach became to establish an overall strategy and then take steps one at a time. Financial restraints played a role; but more importantly, the unique nature of the damage and the need to evaluate conditions before expending resources too quickly dictated that the strategy would be to "eat the elephant one bite at a time" (Dieckamp 1983). Flexibility required parallel and sometimes redundant approaches until an effective method was found. (Since the failure of one plan or piece of equipment could stall the work for months while another was developed, the policy made sense. It also meant that if one action was stymied by public or regulatory debate, progress could still be made.)

- **Challenge the System**—The project team struggled in a difficult regulatory and public environment. Since NRC rules had not been written for postaccident conditions, attempting to fit existing rules was often burdensome. By continually showing that plant conditions were safe, management slowly reduced the burden of specific operating plant requirements to reflect the stability of TMI-2 and the progress of the cleanup.

2.1.2 Management Insights

In terms of managing the cleanup, several general insights stand out:

- The details of any postaccident scenario are unpredictable and specific to the situation. Responding to them requires elements that are alien to conventional utility and service management organizations. Managing the TMI-2 cleanup required an entirely different philosophy and approach from that used to design, construct, or operate a plant.

- Before beginning much of the cleanup work, the theoretical parameters of a postaccident situation should be understood. At TMI-2, this was necessary before developmental work could be performed to prepare the way for production defueling work.

- A heavily project-focused approach is more effective than a large functional organization of engineers and designers responsible for small bits of several projects. The personal involvement and the direct knowledge that this approach created were invaluable assets at TMI-2.

- While redundancy in organizational functions is expensive and difficult to manage, some degree of redundancy is prudent to ensure that all options and potential problems are considered.

- In the early phases of cleanup, a centralized, high-priority effort is needed to provide data on actual physical conditions. Visual observations are essential to understanding and efficiency. Visual observation was often necessary at TMI-2 before unexpected or hypothesized conditions were accepted as real.

- Proceeding on arbitrarily conservative or optimistic assumptions may be counter-productive because the real situation will likely be different. Emphasis must be placed on having hard characterization information before building systems and facilities.

- Insights from experienced senior technical advisors are invaluable. Although difficult to integrate with the workforce, their third-party review is essential in fields where new ground is broken.

- The onsite location of production staff and experts leads to increased efficiency and a pragmatic understanding of conditions.

2.1.3 Beneficial Circumstances

Just as there were impacts from sources beyond the influence of the project team, there were several aspects of the TMI-2 plant that made decisions easier. The implicit warning to other nuclear plants is that these circumstances may not exist in a future accident:

- **Limited Quantities of Fission and Activation Products**—Had the plant been operated for over a year, the presence of cobalt-60 would have made access to many plant areas much more difficult (e.g., areas containing surface films and crud). At the time of the accident, the plant had been operated continuously at power for only a few months, resulting in little accumulation of cobalt-60. At TMI-2, cesium-137—a significantly lesser external radiation hazard than cobalt-60—was the primary radionuclide of concern. Furthermore, the quantity of cesium-137 was considerably less than at a nuclear power plant that had been operating for several years.

- **Empty Spent Fuel Pools**—Because the plant was new, the two spent fuel pools had yet to be filled with water. Both large pools were in a seismically-qualified, shielded, protected building contiguous to the auxiliary building from where connections could be made to waste processing systems. They provided excellent locations for installing equipment (initially water storage tanks and a high-activity-level water processing system). Ultimately, one pool served as a staging area for core debris canisters awaiting shipment.

- **Chemical Cleaning Building**—The plant was built with a steam generator chemical cleaning building containing two large tanks adjacent to the auxiliary building and thus provided an excellent location for a water processing system. The building was also close to a site road, which was convenient for moving processing vessels.

- **Resources from Two Units**—Even though the adjacent Unit 1 was eventually separated operationally and administratively, it was a source of plant expertise and labor for Unit 2 operations. Its containment arrangements were a direct analog of TMI-2 and thus could be used for orientation and job planning before the TMI-2 containment was accessible.

2.1.4 Impacts Of External Factors

There were many institutional aspects of the postaccident period that are important because they were significant diverting influences. Two such significant issues were evolving radioactive waste regulations and public concern.

2.1.4.1 Evolving Radioactive Waste Regulations

The TMI-2 accident occurred at a time in the history of nuclear power coinciding with a transition in the rules for disposal of low-level radioactive waste. This transition, which was heightened in importance by TMI-2, was marked nationally by adverse occurrences such as burial vessels containing liquid, the failure of urea formaldehyde as a solidification media, ground water contamination at the Maxy Flats site, resistance by the leaders of the three states with burial sites to taking the entire Nation's waste, crises at nonnuclear waste disposal sites, and the separation of civilian and defense waste programs.

The transition culminated in the Low-Level Waste Policy Act of 1980, and NRC's subsequent efforts leading to the adoption of 10 CFR Part 61 in 1982. These regulations prescribed the allowable nuclide concentrations and packaged forms of waste that could be buried as low-level radioactive waste.

A few examples of the impacts of this situation on the TMI-2 cleanup were:

- The NRC linked permission to start up the first water processing system to the development of a resin solidification method. However, project management would not commit to meeting the requirements for a "homogeneous monolith" for which no standard existed and in the absence of solidification data for the ion exchange medium being used.

- The project had to construct staging (i.e., temporary storage) facilities in order to decouple the need to process water and the inability to ship the resulting concentrates because of restrictions imposed by commercial burial grounds.

- A memorandum of understanding was required, in 1981, between the DOE and NRC to provide for disposing of highly loaded ion exchange wastes (and core debris) that did not meet commercial burial requirements.

The consequences of this coincidence between a changing national picture and the TMI-2 cleanup were exacerbated by the public perception that TMI-2 wastes were somehow different than those produced at operating facilities (see Section 6).

2.1.4.2 Public Concern And Media Attention

A difficult-to-quantify yet significant influence on the conduct of the cleanup started with the breakdown in communications that occurred during and shortly after the accident (Rogovin 1980, Friedman 1985). As a result, TMI-2 became a focal point for many anti-nuclear organizations and politicians. Especially in the early years, management attention was frequently diverted to address public issues or respond to concepts proposed by people not close enough to the project to have considered all factors. For a certainty, every anniversary of the accident would be marked by demonstrations outside of the plant gates. During the prior week and on March 28, some degree of management attention was inevitably diverted from the work in progress.

Such things as the shipment of radioactive waste and fuel debris across country also had to be considered in the context of public concern. In addition, much time and attention were spent on developing testimony that related to the accident, defended proposed actions, or responded to accusations—issues that contributed nothing to the progress of the cleanup.

2.2 Organization

The ideal management organization for the cleanup was often debated. The challenges were extremely difficult. First, most of the constraints of an operating plant remained, which required an organizational structure for plant operations; i.e., a departmental-style, horizontal organization with each member attuned to a specific well-defined functional discipline. In contrast, the cleanup work itself required a task-focused, project management, development-oriented organization. Mix these contradictions with a matrix-managed organization commonly used by A/E companies such as Bechtel to design and build a new plant, and the complexity is evident. A major theme of this organization section is the tension inherent in the competing demands and how the cleanup managers chose between or blended them at different times.

The organizational structure was frequently modified as the cleanup progressed. This caused inefficiency as personnel adjusted to the new structure but was thought necessary to meet new circumstances. The organizational changes are presented in terms of four major (though simplified) organizational structures. Table 2-1 compares the type of organizational structure employed with the major tasks of the cleanup; i.e., stabilization; waste management; decontamination; core removal and shipping; and preparation for storage.

2.2.1 Initial Organization

Immediately after the accident, management at TMI-2 was faced with the double operational emergency of gaining control of the reactor and preventing or minimizing the uncontrolled release of contaminated water and gas. This had to be accomplished with a damaged core containing unknown and potentially worsening conditions, an inaccessible containment, and a highly contaminated auxiliary building (see Section 3).

Table 2-1 TMI-2 Project Organization Versus Task

Dates	Primary Type of Management Organization	Primary Focus of Technical Operations
1979	Project	Stabilization
1980–1985	Departmental	Waste Management, Decontamination
1982–1985	Departmental	Waste Management Decontamination, Defueling Preparations
1986–1989	Departmental/Project	Core Removal/Shipment, Preparation for Storage

The initial threat demanded an all out effort without any regard to cost. To handle the immediate effects of the accident, an onsite staff approaching 2,000 was created within three weeks. It provided round-the-clock, seven-day-week coverage. In the following years, many who worked at TMI-2 would regard this period as one of the most effective and rewarding of the cleanup. There was an urgency not present during later phases and the organization was markedly different in form and character. It was primarily project-oriented, with top-to-bottom direct lines of responsibility. This was extremely effective in Herculean efforts to stabilize the reactor and to put several new systems in place within a few months. Figure 2-1 represents the organization shortly after the accident.

The organization was formed on April 4, 1979, by Mr. H. Dieckamp, President of GPU Service Corporation (GPUSC), who had arrived on site to take personal command of the situation. He had established telephone contact with senior managers and specialists throughout the nuclear industry. The resulting mobilization involved representation by nearly the entire U.S. nuclear industry. The development of the initial organization, the individuals and companies involved, and the effectiveness of the organization are described in excellent detail in *Three Mile Island: A Report to the Commissioners and to the Public* (Rogovin, et al. 1980).

Industry leaders arriving on site immediately began to form an ad hoc think tank of engineers, scientists, and experts, which later became known as the Industry Advisory Group (IAG). It had a distinctly R&D flavor. By April 1, 30 people from 10 organizations had arrived at the site to form the nucleus of the IAG. This group met at the Air National Guard Headquarters near the Harrisburg Airport, where they could be one stage removed from the hectic and confusing pace of site work. The group was headed by M. Levenson of EPRI. Eventually over 100 persons participated in the IAG, although never more than 40 at one time. Although members were from competing companies, all were focused on a common purpose.

As the group first assembled, experienced utility personnel were dispatched directly to the plant site to assist in plant functions. Initially, group assignments were divided into four general categories:

- Damaged core status
- Cooling mode problems
- Options to take the plant to cold shutdown
- Waste management problems.

It was recognized almost immediately that the waste management team needed to be an implementing organization and thus it moved on site. The other three teams continued as the IAG.

As a think tank, the IAG functioned in parallel with all ongoing activities. It was not part of the implementing organization, although people went back and forth between the two. On its own initiative or by request, the IAG assessed potential tasks based on experience and judgment as opposed to detailed engineering review or to new calculations. All the while, it kept aware of the conditions of the plant and the core, as best as could be determined. The achievement of core cooling by natural circulation on April 27, removed much of the urgency for the IAG and it was disbanded on May 6, after five weeks of operation. Many of its members returned to their home offices to continue working on projects related to TMI-2.

The Technical Working Group (TWG) was established to provide an upper-level decision-making staff. The TWG met on a scheduled basis first thing in the morning and last thing in the evening, and on an unscheduled basis as necessary. Its membership included highly experienced people from Metropolitan Edison, GPU Service Corp., Babcock & Wilcox (the reactor system supplier), Burns & Roe (the plant architect/engineer), EPRI, and others.

The TWG received information (e.g., plant status, data, recommended courses of action) from the Technical Support Department, Waste Management Project Group, Plant Operations Department, and the Plant Modifications Project Group. The IAG coordinator, as a member of the Technical Working Group, could provide information and assistance from the nuclear industry. The TWG then set the technical goals for the recovery project. Depending on the scope of the task, the TWG direction was translated into specific design criteria by either the Technical Support Department or Waste Management Project Group. These criteria were then designed and erected by the Plant Modifications Group. The Plant Operations Department remained essentially unchanged from before the accident and continued with its responsibility for operation of all plant equipment.

This first organization was structured for crisis management and served very well in that role. It was organized

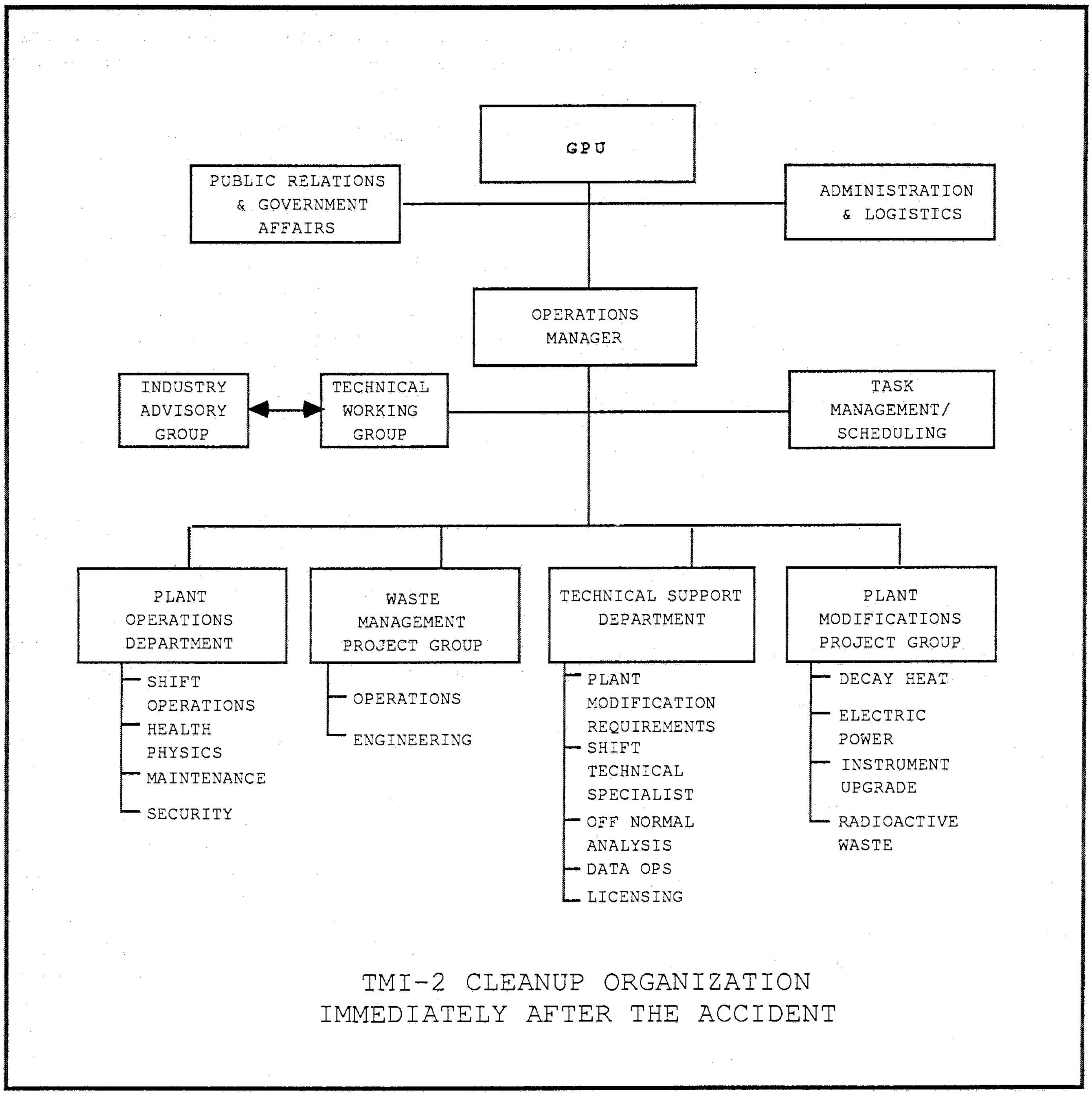

Figure 2-1. TMI-2 Cleanup Organization Immediately after the Accident

to have clear division of responsibility and to operate around-the-clock for an indefinite period, which meant that one or more deputies with full authority were required for each group.

2.2.2 Second Organization

By 1980, the recovery organization had become departmental in structure. Figure 2-2 shows the organization as it basically existed in 1980–1981. During this time, Mr. R.C. Arnold, President of GPU Nuclear, assumed general control of the project with Mr. G. Hovey of GPU Nuclear as Director, TMI-2. Bechtel became the primary contractor for cleanup work. Two Bechtel organizations were brought on site: Bechtel Power Corporation for engineering, construction, and system startup management; and Bechtel National, Inc. for technical and administrative support and cleanup planning.

Of significant note, the radiological controls function was elevated to report at the director's level instead of at the operational level; in terms of the GPU corporate structure, radiological controls now existed as a separate division. This was primarily because the personnel protection issue required much more attention than at a normal plant (see Section 4). Reporting at the director's level ensured operational independence for the radiological protection objective of minimizing personnel exposure.

The Waste Management Project Group in the initial organization was distributed to other parts of the organization. Water processing and waste disposal were placed in Operations. A special process support project group was retained to provide support for these operations because methods were still evolving. The engineering portion of the initial Waste Management Project Group became a Recovery Engineering Project Group with broader responsibilities, including those new systems put in place to control the reactor. This group was eventually incorporated into the Recovery Programs Department.

A Recovery Programs Department was established to encompass the Bechtel scope of work. The objective of the separate Recovery Programs Department was to focus on decontamination, reactor disassembly, and defueling. This organization operated basically in a traditional utility-A/E construction project relationship, with GPU performing oversight of contractors. Many other projects were being worked off site by GPU, Bechtel, Burns & Roe, Babcock & Wilcox, Allied-General Nuclear Services (AGNS), Gilbert Associates, several national laboratories, and others.

2.2.3 Third Organization

The third organization reflected the growing sophistication of project management in terms of understanding the requirements for recovery; the overwhelming organizational need to make the project work efficiently; and the fact that, with the plant in effective cold shutdown, the need for redundant organizations was eliminated. By this time, the postaccident crisis was an event of the relatively distant past and the organization was adopting more formal procedures.

In this context, the existing organizational elements had become too complex and GPU saw the need to develop a stronger team identification. In particular, having the utility and its principal contractor operating as independent units was inefficient. The procedural difficulties alone were stressful (see Section 2.5.3) and GPU needed to concentrate its resources on completing the cleanup and supporting the restart of Unit 1. In September 1982, GPU officially integrated its TMI-2 personnel with those of all its onsite contractors into a unified departmental structure. Mr. B. Kanga, a Bechtel Power vice president, became Director, TMI-2. The resulting organization is shown in Figure 2-3.

In addition to integrating staff, a coordinated planning organization was set up: the Technical Planning Department. This department's charter was to provide planning for waste management, decontamination, and defueling. In addition and most importantly, it contained a data management and analysis group, providing a centralized location for information. In this way, the bases for proceeding on specific projects would be systematically gathered, analyzed, and trended data. The Technical Planning Department's output included data management, engineering concept development, and evaluations of alternatives for a variety of projects.

A separate Licensing and Nuclear Safety Department was also set up to provide licensing support and safety review. As with Radiological Controls, it was a member of the onsite organization. A separate safety review group within the department was also established. (Before this, safety review had been conducted on site by the Site Operations' Programmatic Operations Review Committee, and off site by Technical Functions' General Review Committee).

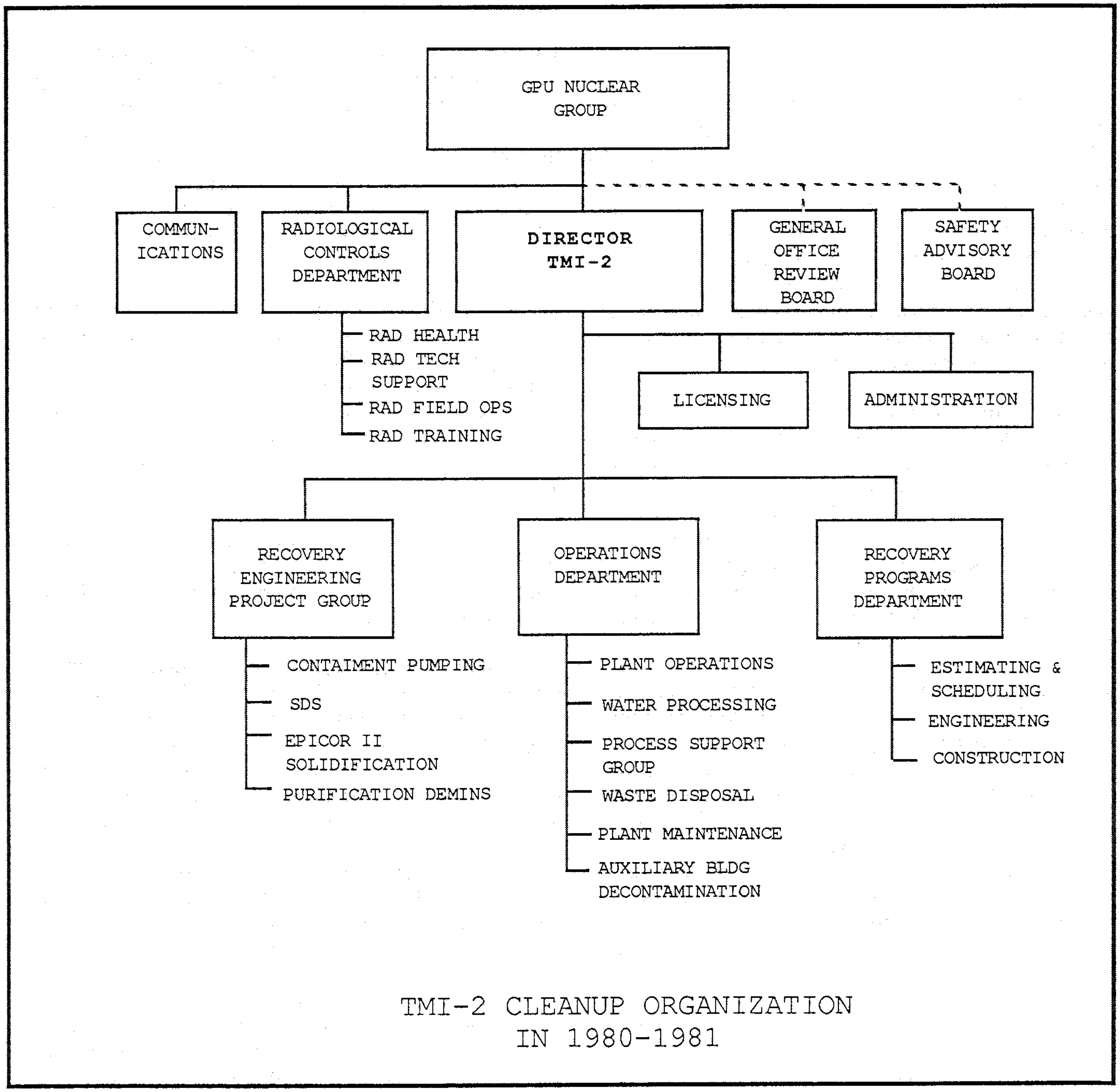

Figure 2-2. TMI-2 Cleanup Organization in 1980–1981

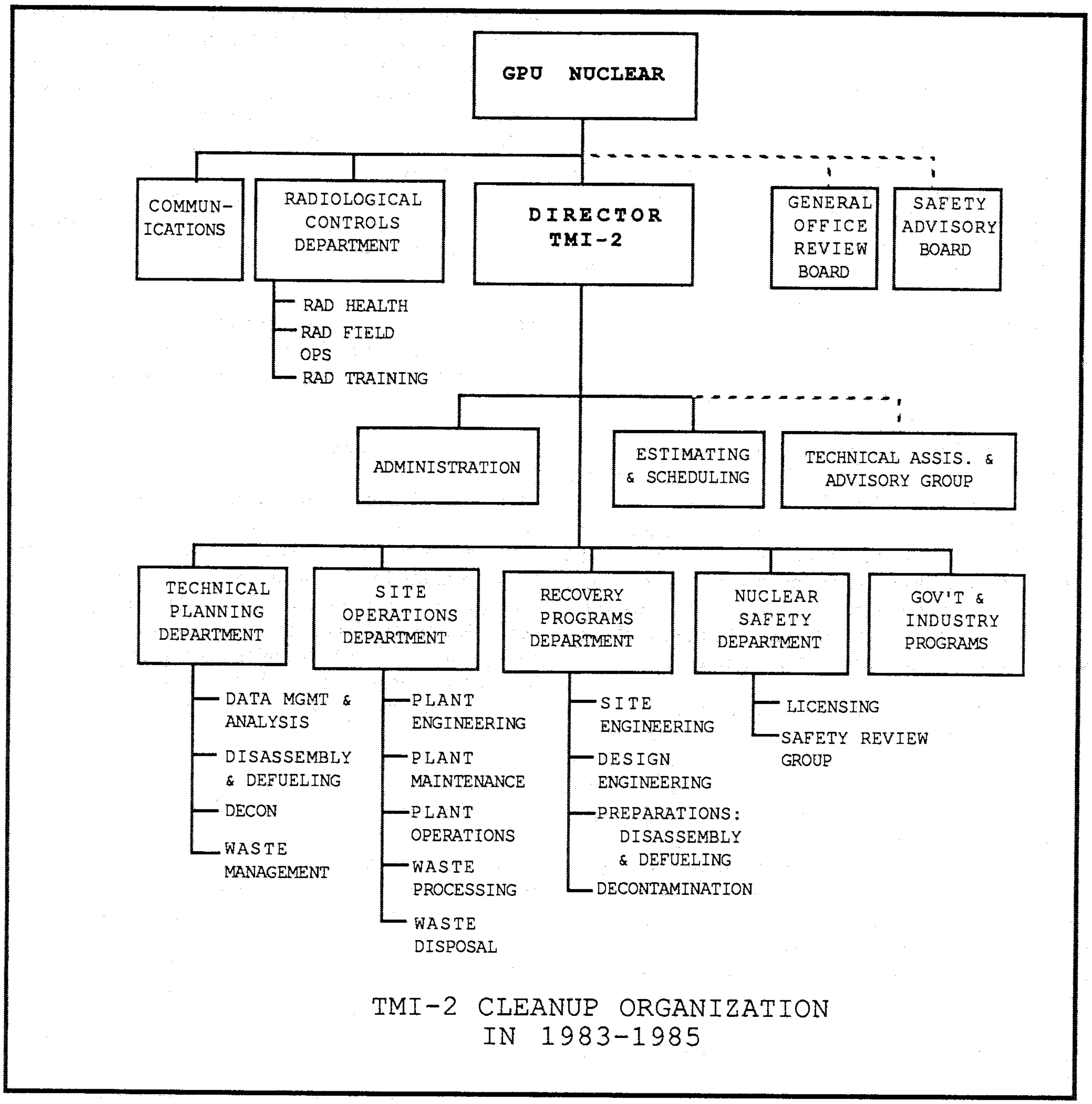

Figure 2-3. TMI-2 Organization in 1982–1985

The last addition to the new organization was the Government and Industry Programs Department. This small group was needed to coordinate outside sources that were supporting and funding the cleanup and to support the independent advisory groups discussed in Section 2.2.5.

2.2.4 Fourth Organization

The previous organization operated essentially unchanged through 1985—the start of defueling. During 1984, GPU had begun to re-assume a more dominant role in the technical direction of the cleanup. At the corporate level, Mr. E.E. Kintner, GPU Nuclear Executive Vice President, increased interactions with DOE and the nuclear power industry regarding TMI-2. On site, GPU Nuclear vice presidents F.R. Standerfer (1984–1988) and M.B. Roche (1988–1990) directed the project.

When defueling was about to begin, the project management believed that much of the R&D nature of the program had been completed. The main focus now would be on production-type defueling of the reactor vessel. Essentially all facility and major equipment engineering, construction, and fabrication appeared to be complete and much of the plant was characterized (or, at least, most of the uncertainties were understood). However, as defueling began, a multitude of new operational challenges appeared and dictated a change back to a more project-structured organization.

This organizational structure was not created immediately nor was it necessarily ideal. The history of the previous six years, with its integration of companies, in-place procedure system, and limited resources dictated that no radical change to the organization could be made. Yet, shortly after the start of defueling in late 1985, project management could see that the task was far more formidable and would demand a greater concentration of resources than anticipated. Actually, it would require many of the same capabilities that the organization had displayed immediately after the accident. The intervening years had been consumed with stabilizing the plant, gaining access to the reactor, and planning and preparing for fuel removal. Now, as if a curtain were lifted after months of rehearsal, the real work began with a complex life of its own.

Directing the work required more than one senior manager, a handful of specially trained senior reactor operators, and auxiliary operators and engineers borrowed from other departments. This was the structure that had first been set up in 1985 to make use of the existing organization; however, it had to be changed quickly. Figure 2-4 shows the new form it took, with the heavy focus of resources in a Defueling Department.

Because fuel removal was the critical path to the end of the program, the department contained task groups that concentrated resources across all departmental lines. The task groups developed the overall sequence for defueling operations, evaluated alternatives, and controlled the scope of work. The task groups worked with other sections in the same department to coordinate engineering; training; tool design, fabrication, and repair; and control of daily operations (which were performed by auxiliary operators from the Site Operations Department).

This organization was essentially the one to complete defueling, with the major modification being the concentration of all engineering functions into one department.

2.2.5 Advisory Groups

The initial advisory groups were the Industry Advisory Group and the Technical Working Group, which are described in Section 2.2.1. With stabilization, these groups were disbanded and replaced over time with others. The new groups were less involved in the day-to-day running of the cleanup and focused more on overall technical and safety issues. The two major such groups were the Technical Assistance and Advisory Group (TAAG) and the Safety Advisory Board (SAB); they are discussed below.

In addition, many ad hoc task force efforts were conducted as the need arose; e.g., the Defueling Tooling Advisory Group and the Water Clarity Task Force. GPU also used its own General Office Review Board, which contained senior members of the corporation and outside experts and was charged with corporate overview of all safety-related issues.

2.2.5.1 Technical Assistance and Advisory Group

The DOE had supported the creation of a Technical Advisory Group (TAG) early in the project to assist in choosing sound technical approaches, particularly in developing methods of processing highly contaminated water. The DOE had provided a similar group to assist in waste management in the summer of 1979. The TAAG was an expanded, more permanent version of this group resulting from the 1980 GEND coordination agreement (see Section 2.7.1) between GPU, EPRI, NRC, and DOE.

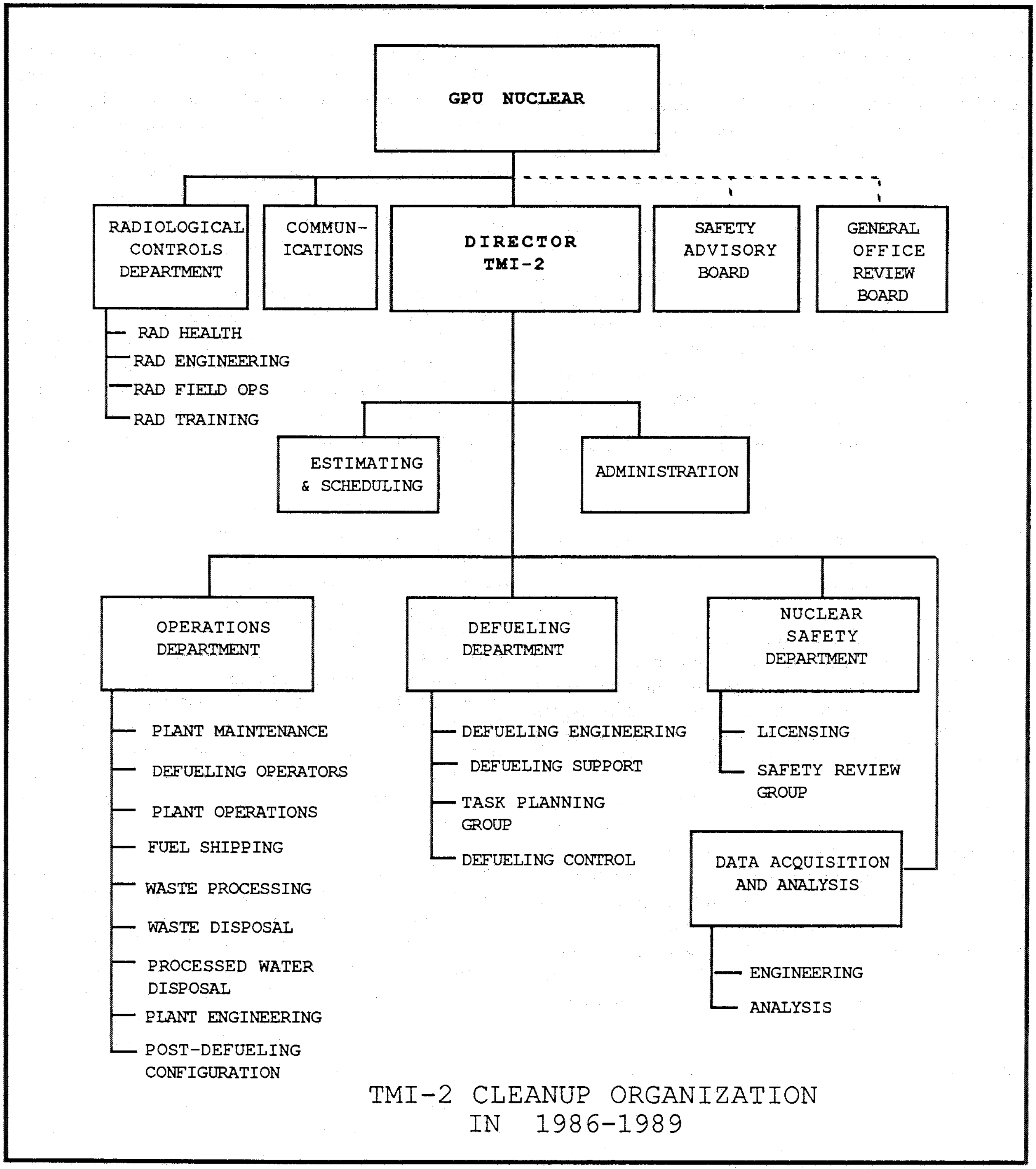

Figure 2-4. TMI-2 Cleanup Organization in 1986–1989

In the summer of 1981, the DOE began discussions with GPU and the NRC regarding how best to spend the funding that had been authorized for core damage assessment and investigation. In November 1981, the Technical Assistance and Advisory Group was formed (it was initially called the TMI-2 Technology Assessment and Advisory Group). The membership was agreed to by GPU, NRC, and DOE and funded through EG&G Idaho, Inc. by the DOE. Later, EPRI sponsored those members of TAAG that were not part of the DOE infrastructure. The chairman of TAAG was Mr. W.H. Hamilton, a former head of Bettis Laboratories. The members of TAAG were well experienced persons from the utility service industry, nuclear submarine shipyards, and government laboratories, and they were able to assist in providing access to many of national facilities and sites. Their experience covered refueling, radioactive maintenance and repair, water processing, analytical investigations, analysis techniques, personnel protection, and, in general, the practical side of resolving difficult technical problems.

In the next five years, TAAG met over 50 times and discussed all technical aspects of the recovery. TAAG provided technical reviews, technical reports, draft procedures, and draft safety evaluations, and loaned personnel to the project to participate in specific task forces. TAAG meetings also served the unchartered but useful purpose of providing a forum for an integrated technical look at the whole program. The presentation portion of these meetings was often attended by TMI-2 managers and staff to gain a broader perspective on the cleanup status than existed during day-to-day operations.

TAAG meetings were held on a monthly basis throughout much of its existence. These meetings consisted of presentations by site personnel involved with an aspect of the cleanup under review by TAAG, presentations by TAAG members describing the results of independent tasks or reviews, and presentations by independent experts invited to speak to TAAG. These presentations were often punctuated with lively debate. TAAG would then meet in executive session to develop recommendations and to write reports.

Although not as deeply integrated into the project as previous groups, TAAG's contributions to the work often helped provide a technical basis for management directives; e.g., contributing to a change in the approach to defueling (see Section 8). The group stressed the need to gather information rather than proceed from conservative and somewhat arbitrary assumptions and the importance of proceeding stepwise as new information was gained rather than trying to pre-plan the whole of defueling from the start.

At the beginning of 1987, with the major planning efforts in place and DOE and EPRI priorities shifting away from direct involvement in TMI-2, TAAG was disbanded. Afterward, TAAG was replaced by a smaller advisory group—the Defueling Review Group—that more narrowly focused on defueling alone. This smaller group was funded entirely by GPU. Its membership included some of the TAAG members from the private sector and provided advice until defueling was almost complete.

2.2.5.2 Safety Advisory Board

Beyond specific technical review and assistance from external sources, GPU sought a programmatic review that focused on issues related to health and safety. Given the uniquely hazardous conditions in the plant and public trepidation, GPU funded, in March 1981, an independent review group comprising members of national repute.

The resulting Safety Advisory Board appraised the cleanup in terms of how the work related to public and worker health and safety. In general, the SAB focused on regulations, risk assessment, project organization and financing, procedures, planning, and public communication and conflict resolution. The first chairman of the SAB was Dr. J.C. Fletcher, a former Administrator of the National Aeronautics and Space Administration. He returned to head NASA in 1986, and was succeeded by Dr. R.Q. Marston, who had directed the National Institutes of Health. Board members were drawn primarily from universities and were specialists in nuclear sciences, engineering, risk analysis, government, and medicine.

Four times each year the SAB met in full sessions that reviewed the work of panels assigned to monitor the areas of fuel removal, community relations, radioactive waste control, and radiation hazards. Closed sessions followed in which members debated issues. Unanimous recommendations were then made directly to the executive management of GPU Nuclear and published annually in public reports. The Board also met annually with the GPU Board of Directors.

Recommendations during the first several years stressed the need for secure funding to support an expeditious cleanup and the creation of an effective management team. The SAB was concerned that both the NRC and GPU were overly conservative in interpreting regulations and the TMI-2 Programmatic Environmental Impact Statement (PEIS) (see Section 2.6.1), thus potentially slowing down the cleanup and adding other, nonradiological hazards to the work. When funding was secure and major decontamination had been completed, the SAB continued to look closely at defueling and post-cleanup planning. The SAB remained in existence until December 1989.

2.3 Program Planning and Policy

Cleaning TMI-2 was not a logical sequence of steps known from the beginning or a methodology applied for reaching a predetermined goal. Plans evolved in view of new data, available technology, regulatory guidelines, and financial restraints. The strategic direction of the project evolved over time.

During the course of the cleanup, many specific task force efforts and planning studies evaluated options and provided recommendations. However, until conditions were generally understood, the specific steps and technical objectives of the cleanup were often wishful thinking. This is illustrated by a 1979 estimate that called for restarting the TMI-2 reactor in 1985.

During an intense, initial planning effort from April 25 to July 1, 1979, the project prepared a plan for containment entry and decontamination in preparation for reactor head removal (Bechtel Power Corp. 1979). The plan was heavily qualified by emphasizing the unknown conditions in the containment. That study was followed by similar engineering studies for future steps (Bechtel Power Corp. 1980; Bechtel Northern Corp. 1982). These plans illustrated the initial, step-by-step approach of:

1. Decontaminate the plant to near-normal levels
2. Disassemble the reactor and remove the fuel
3. Requalify the plant for commercial operation.

As, over time, the extent of damage was discovered and the effects of limited resources grew, the approach changed to:

1. Stabilize conditions and gain access to characterize the containment and reactor vessel
2. Disassemble and defuel the reactor, with supporting dose reduction and decontamination
3. Place the plant in a safe, secure monitored storage condition.

The change began in 1981 with a management decision to seek the early removal of the reactor core and culminated in first the program strategy document in 1984 and then the plans for post-defueling monitored storage, which were submitted to the NRC in 1986. The major strategies in this approach are discussed below.

2.3.1 *Summary Technical Plan*

In the fall of 1979, the project team began to develop broad "top down" strategies and policies to identify:

- Performance objectives
- A logical, operationally focused sequence
- Completion criteria (quantitative, where possible)
- Priorities among and between waste management, decontamination, and defueling
- Tasks that could be deferred.

The need to document an overall plan was catalyzed by a request to present such a plan to Senator Hart's congressional subcommittee on TMI-2 in December 1979. This resulted in the *Summary Technical Plan for TMI-2 Decontamination and Defueling* (Met Ed 1979). The plan developed an overview of key activities, including:

- A basis for ensuring the reactor would remain under control and monitored
- A statement that the technology was available for decontaminating the auxiliary, fuel handling, and containment buildings
- A general approach to defueling—not detailed because there was very little information
- Plans for processing liquid and solid radioactive waste
- A listing and brief description of facilities needed to support the cleanup
- A discussion of the importance of and methods for personnel protection
- Appendixes that addressed assumptions, key decisions, required NRC approvals, and R&D work.

As the cleanup evolved, the actual conduct of the project differed from this initial strategy; however, much of the general philosophy survived. This plan provided a succinct platform from which to proceed and satisfied the political community that there was an overall plan. As it turned out, the document was invaluable for communicating information about the program both internally and externally—over 500 copies were printed.

2.3.2 TMI-2 Program Strategy

By late 1983, with the project stretching out, funding concerns, and better information about conditions, a more detailed overall strategy was needed. The resulting *TMI-2 Program Strategy* (DeVine and Negin 1984) was published in June 1984. It put forth a logic depicted in Figure 2-5 and defined the recovery program in three phases: Stabilization, Fuel Removal, and Cleanup. The Program Strategy stated that the stabilization phase was complete, and that the fuel removal phase must be conducted without regard to the ultimate disposition of the plant.

The specifically stated purposes of the Program Strategy were to: 1) provide a concise overview for corporate and program management; 2) establish fundamental program priorities; 3) establish a logical hierarchy of policy and technical guidance; and 4) provide a vehicle for communicating the program to individuals and organizations not directly involved in the project.

The document provided the approach for more detailed planning and, as with the 1979 strategy document, proved invaluable as a means of describing the overall program. Many of the directions were used as bases for program cost estimates and yearly budgets.

The program strategy provided guidance for resolving technical and managerial issues by establishing policies addressing: 1) generic issues, 2) characterization, 3) fuel control, 4) methods and end points for defueling, 5) disassembly, 6) dose reduction, and 7) waste management. Some were obvious or required by law, others eliminated alternatives. The policies themselves had often evolved over the course of the cleanup, were implicit in past actions, or were decided during drafting of the strategy.

Generic policies were:

- **TMI-2 Program Objectives**—The overall objective of the cleanup was defined as "reducing to an acceptably low level the radiation hazard resulting from the accident." This objective would be achieved when fission products were immobile, fuel was shipped off site, other radioactive waste was packaged and shipped or stored, and radiation levels were low enough to permit examinations to support decisions on the plant's eventual disposition. The plant would not be returned to pre-accident conditions and all waste would not necessarily be shipped. The objective was stated very carefully to avoid establishing impractical objectives whose accomplishment could be subjected to future debate. (This policy was later modified to include plans for long-term monitored storage.)
- **Definition of Program Phases, Priorities, and End Points**—Resources would be expended on current priorities; any decision related to restart versus decommissioning was deferred and decoupled from current activities. For example, chemical decontamination of the primary system was delayed and then abandoned since bulk chemical decontamination would only be needed to prepare for ultimate disposition of the plant.
- **ALARA**—All managers would make the as-low-as-reasonably-achievable (ALARA) concept a major factor in decisions; i.e., it was not sufficient to merely satisfy the text of the regulations. The point was to reemphasize existing policies. Further, the NRC's PEIS was not to constitute control of project tasks nor was it to be construed as a dose budget. It was to be viewed as an estimate based on how the cleanup might be conducted.
- **Use of Remote Technology**—Remote technology would be assessed on a task-specific basis before using this rapidly growing technology. This policy was needed because of the potential that esoteric projects might consume substantial resources without specific objectives.
- **Application of Regulations and Regulatory Guidance**—Regulations, rules, and guidelines that did not fit the TMI-2 situation would be assessed and, where specific compliance would not be logical, relief would be sought. The rules were not written for the cleanup situation and the project team would not hesitate to question their applicability.
- **Preservation of Plant Equipment and Structures**—Planning and resources would only be spent on preservation for safety or achieving project objectives, although actions precluding refurbishment would be avoided if possible. In hindsight, this should have been stated in more detail because the reluctance of engineers and workers to act against their training

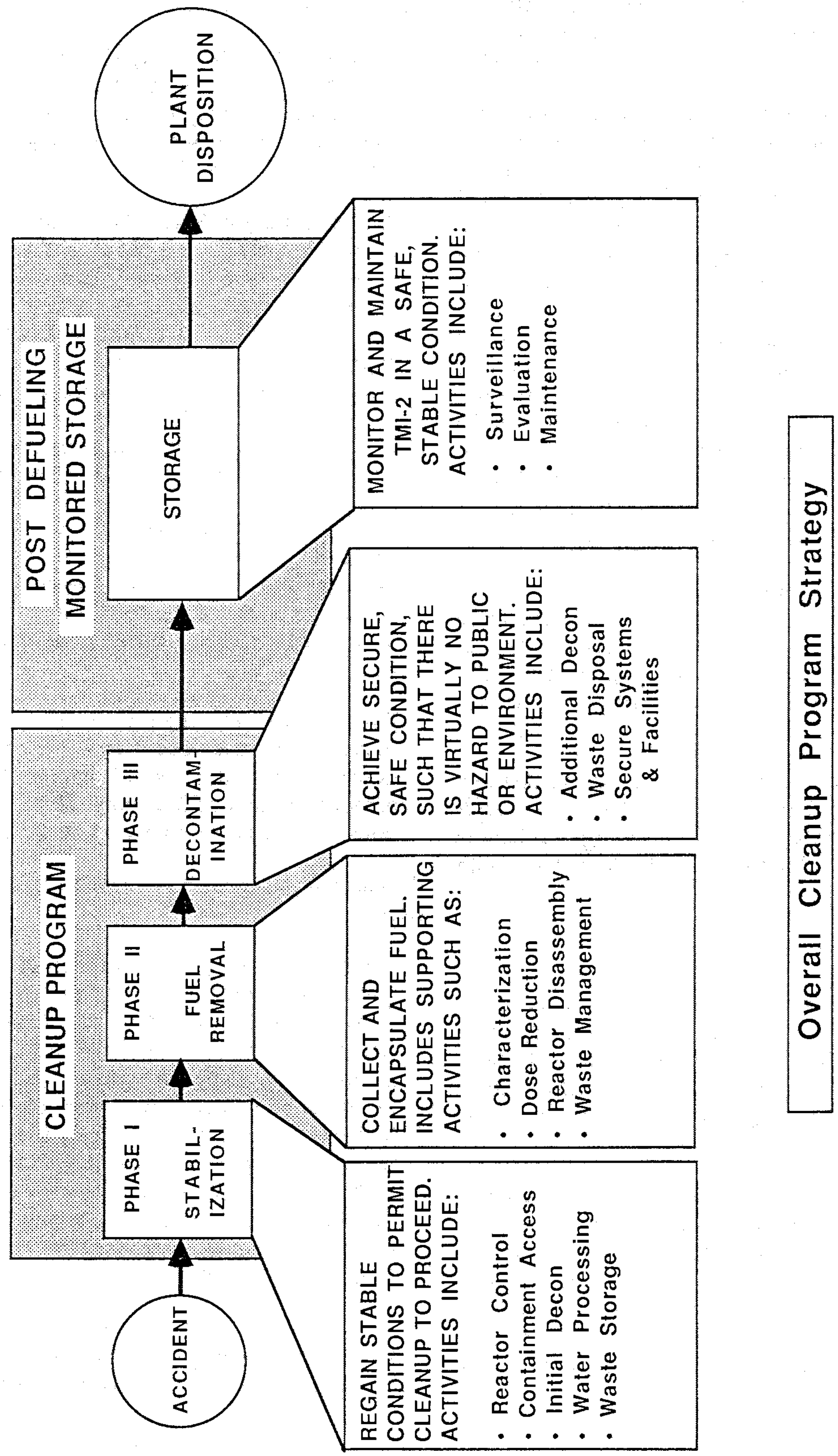

Figure 2-5. Overall Cleanup Project Strategy

continued for several years regarding preservation of the plant's original configuration and maintenance of the QA chain for non-safety related work (Long 1987).

- **Permanence of Recovery Facilities**—Facilities and systems for cleanup work would not be designed for a 40-year plant life. Without this policy, well-meaning engineers might naturally overdesign with eventual restart in mind. Early in the recovery, there was a difference in the design requirements between temporary systems (recovery systems) and modifications to existing operational plant systems, most notably for seismic design requirements. The logic for not using seismic design criteria on recovery systems was that they would be in place for only a short time and the likelihood of a seismic event was small. As the cleanup progressed and the return to normal operation without a massive rehabilitation decreased, most new systems and modifications to original plant systems were classified as "for recovery purposes only."

- **Facilities and Systems Shared by TMI-1 and TMI-2**—The facilities for the two units would not be shared except for a few limited cases where separation was illogical and sharing would not affect operation of Unit 1 (e.g., the truck bay and certain radwaste storage facilities).

- **Opening Containment**—If needed for the cleanup, opening of the containment to the outside was acceptable. This position was aimed at avoiding establishing any agreement with political institutions that would preclude such action.

Characterization policies were:

- **Importance of Characterization**—Characterization was vital to avoid the pitfalls of proceeding from assumptions and models, which were comfortable techniques for conducting accident analyses and safety reviews. Data gathering efforts, however, were costly and would be carefully considered.

- **Objectives of Characterization**—Data gathering efforts were to support the TMI-2 cleanup project; data for scientific purposes that did not also provide such support was considered on a case-by-case basis. This statement did not affect the agreement with the DOE for work to support accident analysis information, most of which was conducted at national laboratories.

Fuel control policies were:

- **Criticality and Reactivity Control**—Boron was one, but not the only, method for preventing a recriticality. Separation, quantity control, and geometry control could be used as well, depending on the circumstances (see Section 5).

- **Accountability of Fuel**—Although accountability for the regulatory purpose of preventing material diversion was really not an issue, compliance with the law was required. Conventional means of accountability could not be used; therefore, weighing and/or item counting after removal were to be the primary means of measurement, with approximate results ("one core, more or less"). This policy was later modified to measuring what was left after the fuel was removed as a means of final accounting (see Section 5).

- **Definition of Core Waste**—Core wastes did not just comprise fuel located within the original core envelope, but included structural elements and other materials mixed in with the fuel. This definition was used in the fuel shipping and disposal contract between DOE and GPU.

- **Fuel Storage and Disposal**—Core wastes were not to be stored on site for more than an interim period. This policy reinforced a commitment to the public and complied with the PEIS, which stated that TMI-2 was not suitable as a long-term waste repository.

Policies regarding methods and end points for defueling were:

- **Defueling Strategy**—The actual method of defueling was not fully determined at the time the Strategy Plan was published in 1984. As later defined, the defueling strategy was to proceed from those areas with the greatest amount of fuel to those with lesser quantities, recognizing the need for operational flexibility. The highest priority was removing fuel from the core region, then proceeding directly to the lower portions of the core support assembly and reactor vessel. Fuel that had escaped the vessel was to be removed later unless (as occurred) parallel efforts were more efficient (see Section 8).

- **Completion of Core Region, Vessel, and Ex-vessel Defueling**—Initially, the project team wanted to avoid elaborate requirements and equipment to measure minute quantities of residual fuel; consequently, visual inspection was proposed as the primary method of ensuring fuel removal. Some fuel contamination would not be practical to remove. Eventually, residual fuel was measured by a combination of radiation measurements and visual examination (see Section 5).

Disassembly policies were:

- **Storage of Large Components**—Efforts to remove and ship components such as the reactor vessel head and upper internals could consume a substantial amount of resources. Instead, local storage with some combination of shielding, isolation, or decontamination would be sufficient during the fuel removal phase and while decisions for future disposition were being considered.

- **Primary System Integrity**—The ability to re-close the primary system and maintain its structural integrity were to be maintained. This was needed in case resources were curtailed and the system had to be sealed to maintain isolation. Thus, two of the three barriers (fuel cladding, primary system, containment) between fission products and the environment would be maintained.

Decontamination/dose reduction policies were:

- **Phase II Decontamination**—Decontamination during the Fuel Removal Phase was performed only to support defueling and safety/end point objectives, and not to clean areas or systems for the sole purpose of making them clean. Decontamination work to achieve project end points was conducted to the extent that resources permitted. In late 1988, project management postponed most of the remaining plant decontamination until the completion of defueling. (Defueling was found to be recontaminating some previously cleaned areas and so only decontamination that directly supported defueling operations was performed.)

- **Criticality Prevention During Decontamination**—During decontamination operations such as sluicing or flushing, criticality should be prevented by a combination of analyses and procedural or physical restrictions. The policy was intended to caution about the possible relocation of fuel during such operations; it also allowed the theoretical use of unborated water in the right circumstances.

- **Worker Efficiency**—Worker efficiency should be improved by taking steps that would reduce the amount of protective equipment that had to be worn, consistent with good personnel protection practices. Striking the balance between industrial safety and radiological protection was the objective.

- **Reflooding the Containment Basement**—Some advisors were promoting the idea of reflooding the containment basement for shielding and as a potential decontamination-through-leaching method. Project evaluations showed that the impacts would be substantial relative to the benefits, and so specific direction was given that the containment would not be reflooded without a program-level management decision. This put the issue to rest in 1984, although it resurfaced at later times (see Section 7).

Waste management policies were:

- **Storage of Waste and Closure of Disposal Sites**—The project recognized the potential for a disposal site moratorium as a result of the Low-Level Waste Policy Act. Interim storage could be planned, but it had to be modular to prevent over-building.

- **Noncommercial Waste (Abnormal Waste)**—Efficient waste processing was stressed to achieve products as radioactively concentrated as possible, consistent with handling limits. Some non-fuel wastes at TMI-2 did not meet waste form or content requirements for standard burial. Processing these to meet burial requirements could produce excessively large amounts of waste and, in the case of having to solidify wastes already generated, could result in personnel exposure that contradicted the ALARA principle. The project management reached an agreement with DOE in which they would take custody of such wastes where it was not prudent to make them conform to commercial disposal requirements.

- **Re-use of Processed Water**—Water should be recycled. The discharge of processed accident-generated water was prohibited for most of the cleanup; therefore it was important to minimize the accumulation of any new water requiring disposal. Considering the final total of approximately 7.6 million liters, this was an important policy.

- **Segregation of Water at TMI-2**—Water should be segregated for only two reasons: to maintain boron control and to minimize mixing of new, nonaccident-generated water with water that could not be released. Water should not be segregated by tritium concentration because this would take resources and require hardware for no value.

2.3.3 *Post-Defueling Planning*

In the fall of 1985, defueling plans and preparations were essentially in place and GPU began to consider seriously what to do when defueling was completed. Thus began an intensive planning stage that continued throughout 1986 and culminated in a plan for monitored storage. A task force provided the technical bases and initial recommendations from which to proceed (TMI-2 SPTF 1986). In December 1986, its three-volume study was compressed into a one-volume document that was submitted to the NRC (and the public) to outline the company's plans for the future (GPUN 1986). Eventually, the position became the basis for a formal safety analysis report, including new technical specifications; emergency, security, and QA plans. This new condition—called Facility Mode 4 or Post-Defueling Monitored Storage—was the fourth mode in a progression of technical specification modes designed to place the plant in a long-term monitored storage condition (see also Section 2.6.2).

The three dominant characteristics of the plant condition were:

- The reactor vessel and the RCS would be defueled and the core material shipped off site (less than 1% would remain).
- Decontamination would be complete to the extent that further major decontamination programs were not justified on the basis of worker dose.
- A condition of stability and safety was established such that there was no risk to public health and safety (GPUN 1988).

2.4 Staffing and Financing the Cleanup

The resources required to conduct the cleanup reflected the large scale of operations and the first-of-a-kind work involved.

2.4.1 *Personnel*

As has been well documented (Kemeny, et al. 1979, Rogovin, et al. 1980), the accident brought a flood of engineers, scientists, workers, and regulators. Three weeks after the accident, 1,964 people had checked in at the TMI Observation Center near the plant. Some only stayed days; a few stayed the full course of the cleanup. The initial arrivals represented approximately 150 companies/organizations (McIntire 1979).

The initial mobilization by GPU and the industry provided most of the personnel for controlling the plant, eventually bringing it to a cold shutdown condition within the first few months. Decontamination was performed in large part by trained volunteers from within the GPU system. Hundreds of these volunteers (clerks, lineman, janitors, etc) worked in shifts for approximately one year performing a vital function. No special incentives were provided. The decontamination labor pool provided a broad range of capabilities that was extremely valuable in addressing the many skills necessary for various decontamination tasks; i.e., specialists were available when needed.

The GPU voluntary pool started with 20 persons and eventually used over 400. As volunteers were recruited, they were immediately given hands-on training and then placed in a two-week rotation. Seven days a week, 50 people worked in three 10-hour shifts. The use of people from outside the nuclear-related departments allowed plant auxiliary operators and engineers to concentrate on stabilizing plant systems rather than reaching their exposure limits performing decontamination tasks.

The approximate number of full-time personnel associated with the TMI-2 project at the end of each year is shown in Figure 2-6. (The numbers for 1979 and 1980 are estimates.) In 1981, the uncertainties of funding resulted in a substantial reduction in the contractor force, primarily from Bechtel and Catalytic Construction, Inc. The subsequent resolution of the cleanup budget was followed by a buildup through 1985 to support the design, construction, and installation of defueling equipment. Thereafter, most efforts were concentrated on fuel removal and preparation for long-term storage, gradually decreasing the need for much of the support that had existed.

After the initial volunteer effort by the unions and considering the working conditions and publicity surrounding TMI-2, GPU was very concerned that it would be unable to attract the craftworkers and laborers needed for the cleanup. A special approach was needed. In March 1980, the affected unions and GPU (with Metropolitan Edison as the interface) reached an unprecedented agreement that recognized the importance of recovery from the accident to the entire nuclear power industry and the Nation. The agreement recognized that the work was specialized and highly demanding, requiring large-scale capital outlays, and exacting measures to protect public and worker health and safety. The unions also recognized that the company's existence was at risk because of its fragile financial health.

Consequently, the unions and management agreed to a cooperative relationship. This included a mutual ban on any form of work stoppage or lockout, and a high standard of radiological safety practices (Arnold and Georgine 1980). As a result of this agreement, labor problems were resolved by arbitration—technical progress was not affected by either labor grievances or the necessity of planning for them.

2.4.2 *Cost and Schedule*

Estimating the cost of the cleanup while it was in progress was as elusive as pinning down its technical scope. The cost estimates varied in direct relationship with the uncertainty and novelty surrounding the work. They were plagued by a lack of data and uncertainty in the regulatory environment within which the cleanup activities were to take place (Comptroller General 1981). The scope of work for each estimate varied much more than the dollar figures as project management learned the condition of the plant, conducted work, and defined a reasonable stopping point. Many estimates were made, and the following discussion shows only the general course of that process.

The American Nuclear Insurers had estimated $140 million of damaged insured property in April 1979. Another preliminary estimate in July 1979 reckoned that the plant could be returned to power in four years for approximately $430 million. As the real scale of the undertaking was revealed, a series of formal program estimates was made (GPUN 1985). In August 1980, after one containment entry, $855 million was the estimated total cost. The projected work scope envisioned defueling complete in March 1983, and then cleanup through reconstruction to pre-accident conditions, refueling, testing, and commercial operation, which was to have begun in late 1985.

An estimate in August 1981—for $1034 million—stressed the need for significantly more data and did not include the costs of restart. The cleanup experience of the past year was reflected in the increased estimate, as was the temporarily reduced level of effort resulting from the

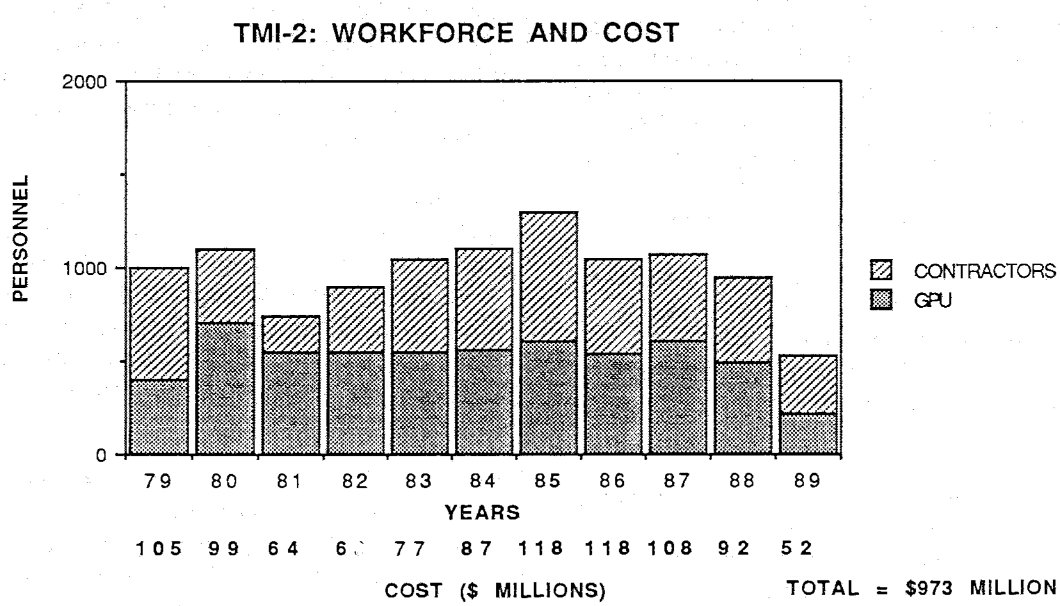

Figure 2-6. TMI-2: Workforce and Costs

uncertainty of funding sources. This estimate projected January 1985 to complete defueling and December 1986 for completion of the cleanup project, assuming that secure funding could be obtained.

Not until December 1982, with considerably more data about reactor vessel conditions available, was a cost estimate made that approximated the ensuing reality. Then, based on 130 containment entries and limited remote camera inspections of the reactor core, the estimate was $975 million; defueling complete in July 1986; and the cleanup project complete in June 1988. This cost estimate (although not the scope) held fairly steady, only changing to a late 1988 completion date. In October 1988, with the technical difficulties of disassembling the reactor internals slowing progress, the schedule was revised to a 1989 date for completing defueling. Bulk defueling of the reactor was actually completed in December 1989, with final cleanup and core debris shipment in the first part of 1990. The final direct cost of the project was $973 million.

2.4.3 *Financing*

Within weeks of the accident, the character of difficulties faced by GPU expanded from a strictly physical crisis to a fiscal crisis. The company faced bankruptcy (Kuhns 1985). The immediate cash costs of the accident were substantial, and the cost of replacement power was $25 million per month, which GPU (Metropolitan Edison) did not have. (Regulatory authorities ultimately recognized the cost of replacement power, but only after approximately $150 million of these costs had been deferred for future recovery.) Establishing a line of credit and arranging long-term financing and reasonable purchase of replacement power became emergency issues.

GPU stock dropped from approximately 17 in March 1979 to 3 3/8 in March 1980, its all-time-low following the accident. The company did not pay dividends from February 1980 to April 1987. (By 1988, financial recovery was established, leading to the award of "Utility of the Year" by *Electric Power & Light.*)

In mid-1979, it appeared that the cost of the cleanup, while in excess of insurance, represented a difficult, but manageable problem. The onsite property was insured for the maximum available of $300 million. This represented less than one-third of the eventual cost. Expenditures during 1979 and 1980 of $200 million were largely covered by this insurance.

The rationale for making up the $700 million shortfall was that the burden should be shared with: 1) those who were collectively insured, since the insurance was inadequate; 2) those who had an interest in ensuring an expeditious cleanup; and 3) those who would benefit from the lessons learned during the cleanup. With this approach, GPU began an effort to achieve consensus among the various parties that their participation was appropriate and to develop a cost-sharing formula that was equitable.

Adding to the urgency was the action by the Pennsylvania PUC in September 1980 to deny Metropolitan Edison's request for emergency rate relief. The effect on the cleanup schedule was immediate, as most Bechtel and Catalytic workers were demobilized and only $66 million of a desired $175 million were available in 1981. (In any event, the full $175 million would have been difficult to use because of a lack of sufficient plant knowledge to support cleanup plans.)

The financial crisis continued until July 1981, when Pennsylvania Governor Thornburgh proposed a specific formula allocating costs among ratepayers, stockholders, the utility industry, and the federal and state governments. Three years of negotiations, discussions, and hearings were required to put the plan fully in place.

The Edison Electric Institute established a utility voluntary program to contribute $150 million, which became available to the project at the rate of $25 million per year beginning in 1985. The DOE contributed a $105-million multi-year authorization for R&D related to the accident. (Of this, approximately $76 million was for direct support and $29 million indirect.)

With the U.S. government and the utility industry participating, the state regulatory agencies were willing to consider customer funding. Both the Pennsylvania PUC and the New Jersey BPU allowed cleanup funding to be included in customer rates at the level recommended by the Thornburgh plan. In 1982, the governments of Pennsylvania and New Jersey authorized a total of $42 million from the state treasuries.

After several fruitless years of trying to obtain a significant international involvement, a consortium composed of Japanese utilities, engineering companies, vendors, and the government agreed to provide direct funding of $18 million in return for information and the opportunity for more than 40 engineers to gain first-hand experience by participating in the cleanup.

Figure 2-7 shows the funding sources for the cleanup.

In addition to the funding shown in the figure, considerable research funding was committed that could not be considered as being of direct benefit to the cleanup (e.g., laboratory work). EPRI provided approximately $11 million in funds to the cleanup and to technology transfer activities. (This does not include the R&D funds spent by the Source Term Program or in-house costs such as NSAC work.)

2.5 Other Administrative Issues

Several other issues are worthy of discussion because of their unique roles in the postaccident situation.

2.5.1 Emergency Materials Management

Ensuring a steady flow of equipment and supplies was, of course, an intrinsic part of the cleanup. The initial effort at mobilizing procurement to support the cleanup was outstanding and contributed greatly to the early successes. The day of the accident, the plant purchasing staff reverted to an "Emergency Procurement Mode" that had previously been used during outages and weather-related emergencies. A temporary warehouse was set up at the Crawford Station plant, about five miles from TMI. This operation was staffed 24 hours a day to handle requests for supplies, equipment, and services, and to act as a marshalling area to receive material and equipment.

The day after the accident a Materials Management Task Force was also established. The task force combined the existing personnel at the Crawford Warehouse with TMI contract administration and added a home office team consisting of buyers, contract personnel, and transportation personnel. An interface with corporate procurement was also established. The task force pursued water storage and processing equipment and services, boration and decontamination equipment, and facilities for support personnel; e.g., setting up "Trailer City" at the observation center, about one-half mile from the plant. Several days later, Burns & Roe purchasing personnel were sent to Crawford to coordinate receipt of material specified by Burns & Roe and to otherwise assist as needed.

Within a week, project management realized that this arrangement was insufficient for the situation and the role was essentially turned over to GPU Service Corp., which had been responsible for construction of the plant and had until recently been responsible for purchases. The procurement organization processed over 1,000 purchase orders within approximately 40 days.

A few highlights of this initial effort:

- A diesel generator was delivered to the site in two days, with the cooperation of four railroad companies who flagged the shipment as an emergency.
- A large-system HEPA and charcoal filters were provided for use on the roof of the auxiliary building. These were transported from a western utility by the U.S. Air Force in six C141s and one C5A.

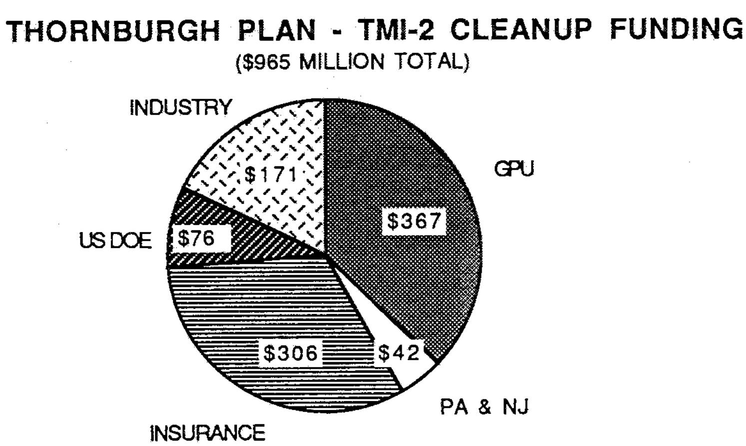

Figure 2-7. Thornburgh Plan — TMI-2 Cleanup Funding

- With the cooperation of the Pennsylvania State Police, overwide and overweight tanks were shipped without difficulty.
- When solicited for special pressure gauges for which the delivery time was normally 16 weeks, the vendor committed within four hours to fabricate and deliver the gauges within two days.
- A supplier of piping materials and fittings sent an engineer who sat with plant staff while they designed piping and spool pieces and arranged for air shipment from Louisville and Houston. The supplier also located and arranged for immediate, straight-through shipment of a 0.2-m3/s air compressor and a large heat exchanger, which weighed over 60,000 kg and was grossly overweight for truck shipment. With help from the NRC, who worked with the Interstate Commerce Commission, permits were granted for its transport. With two drivers, it was delivered over a weekend.
- A vendor modified and shipped boron injection tanks within three days of receiving a purchase order.

These are only a few of the numerous cases of extraordinary work and cooperation to address the initial recovery efforts (McIntire 1979).

As the situation stabilized, a more normal mode of purchasing began. Much of the recovery and defueling-related equipment was procured through Bechtel, which had the contract for the equipment.

2.5.2 *Separation of Units 1 and 2*

Because the Unit 1 license was suspended by the NRC in the summer of 1979, much of GPU's attention and resources were spent over the next six years in satisfying new requirements, improving operating conditions and training, and addressing the hearings for Unit 1 restart. In the course of this, GPU separated the two units to the extent practicable. The separation extended to facilities, plant staff, and procedures.

For hardware, this primarily affected liquid waste systems. The method of separation was usually by double-valve isolation, removal of spool pieces, and, in some cases, severing of pipe. In a few limited cases, shared use of facilities could not be avoided or clearly would not affect operations; e.g., interim storage of low-level radioactive waste and staging facilities for waste shipment. The fuel handling building shared a truck bay, receiving area, and bridge crane. This meant that considerable coordination was necessary between the two units to make sure that varying technical specification and operating procedures did not conflict. After TMI-1 began operating in 1985, an outage would constrain the use of these facilities by Unit 2 (and limit the number of technicians and auxiliary operators available to support the cleanup).

2.5.3 *Procedures*

The paperwork associated with the cleanup posed an additional dilemma for project management and caused as much delay and confusion as many technical issues. The dilemma was finding creative engineering solutions within the strict procedural structure of a licensed operating plant. How should GPU mesh the methods of an A/E-engineering company (i.e., Bechtel)—which historically built plants within the constraints of a construction permit—with those of a utility that must operate within the constraints of an operating license? The resulting paperwork mismatch was to prove a major source of confusion and resulted in a substantial impact on the schedule. The two big issues that combined to delay the project were: 1) allegations regarding procedure violations, and 2) the need to create a new procedure system.

Procedural control immediately after the accident was sometimes ad hoc and based primarily upon standard operating procedures of GPU and, initially, Burns and Roe (the original A/E of the plant). Until Bechtel assumed the contractual responsibility for the cleanup in 1980, it and the other contractors had little difficulty performing according to GPU direction in their limited roles. As Bechtel assumed greater responsibility, the difficulties increased. The GPU administrative procedures in particular were a source of confusion because they were alien to a construction project and time consuming. GPU management recognized the problem, which became one of the primary impetuses for formally integrating the two companies in September 1982 (see Section 2.2).

The period of integration—summer 1982 to spring 1983—was a period of uncertainty about organizational responsibility and procedural adherence requirements. During this period, the containment polar crane was being refurbished and plans were underway for a load test (see Section 8). In March and April 1983, the inherent organizational conflict came to a head when allegations sponsibility and procedural adherence requirements. During this period, the containment polar crane was being refurbished and plans were underway for a load test (see Section 8). In March and April 1983, the inherent organizational conflict came to a head when allegations were made of procedural violations related to the polar crane refurbishment. GPU vigorously contested the allegations and thus consumed an enormous amount of management attention (Steir 1983).

Until the issue was resolved and the safety of refurbishment established, the NRC refused to approve the use of the polar crane, which was essential for proceeding with the project. The period of NRC investigation was one of intense scrutiny by both the public and regulators. Load testing of the polar crane was suspended for many months while the allegations were investigated; however, other work did continue and the hiatus provided the opportunity to complete planning and preparations for other defueling-related work.

The second procedure-related issue delaying the cleanup was brought into focus by an inadvertent release of radioactive material during maintenance work on a HEPA filter. At the subsequent Region I Enforcement Conference in July 1983, the president of GPU Nuclear committed the company to a complete overhaul of cleanup administrative procedures by January 1, 1984. This commitment resulted in a new policy and procedure system that was adopted at the plant and corporate levels. Cleanup work was greatly affected while the energy of managers and engineers was devoted to writing, reviewing, and approving the new procedure system.

The resulting system combined elements of existing administrative and operating procedures with a unit work instruction (UWI) system, used at other reactor facilities where many tasks were one-of-a-kind and required versatility. An engineering change authorization (ECA) was adapted as the basic design document; in it were assembled the baseline documents for modifying a structure, system, or component.

The consequences of implementing this new procedure system, aside from the schedule delay, were a consistent approach to procedures by all departments (hence a true integration), the systematic identification of all TMI-2 regulatory commitments, and an increased professionalism (Kelly 1987). A complete system of paperwork for cleanup operations emerged and carried the project through the cleanup. Conceptually, it resembled Figure 2-8.

2.6 Licensing & Safety—NRC Involvement

The accident brought enormous regulatory changes that affected every nuclear power plant. The effects were felt differently at TMI-2 during the cleanup because, though it had spawned the changes, it was no longer an operating plant.

2.6.1 NRC Organization and Interaction

There were three branches of the NRC involved in the cleanup: Nuclear Reactor Regulation (NRR), Inspection and Enforcement (I&E), and Waste and Transportation. Immediately following the accident, both NRR and I&E were involved, with the latter being detailed from the NRC Region I office in King of Prussia, PA. The Chief of NRR, Mr. H. Denton, arrived at the site within days of the accident. During the first few months, the NRC staff at the site, composed of both NRR and I&E staff, was an important part of the crisis teams putting emergency systems in place.

The normal chain of communications for NRC review and approval proved inappropriate. Since significant license changes and safety reviews would be required throughout the cleanup, a special field office reporting to NRR was set up at the site: the TMI-2 Program Office (later Cleanup Project Directorate). By 1984, this office had 14 full-time staff on site and 7 at NRC headquarters (NRC TMI Program Office 1984). In early 1988, with major decisions behind the project and plans in progress for the post-defueling phase, the NRC TMI-2 staff was down to two full-time staff members on site.

A major accomplishment of the project office team was to oversee preparation of the TMI-2 Programmatic Environmental Impact Statement (PEIS 1981), which became the basis for satisfying NRC's responsibility under the National Environmental Policy Act (NEPA) for the cleanup program. The PEIS was an overall study of the activities necessary to decontaminate, defuel, and dispose of wastes. The available alternatives considered ranged from implementing a full cleanup to no action other than continuing to maintain the reactor in a safe shutdown condition. The PEIS was considered necessary because of the ever-growing needs of the cleanup to perform actions with potential effect on the environment.

Issued in March 1981, the PEIS had taken almost a year and a half to prepare. Yet when finished, because of its wide enveloping bounds, it provided a single reference

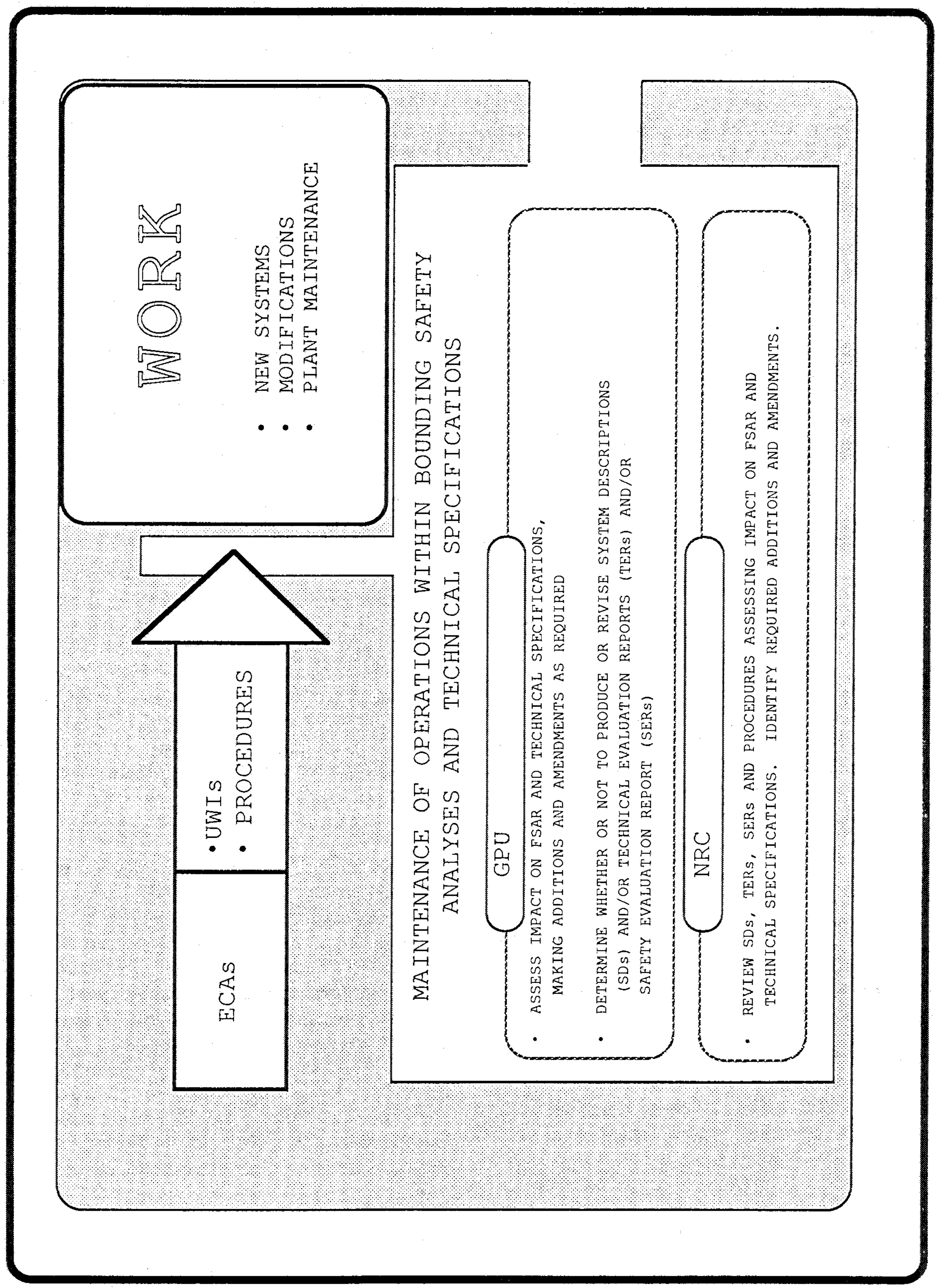

Figure 2-8. Interrelationship of TMI-2 Documents/Work

by which NRC could approve specific cleanup project technical and safety evaluations and other proposed actions. Supplements to the PEIS were later issued on the subjects of personnel exposure, accident-generated water disposal, and post-defueling monitor storage. In a policy statement accompanying the PEIS, the NRC staff was given authority to approve cleanup activities that fell within the PEIS scope. The significant exception was disposal of accident-generated water—a decision that the NRC Commissioners reserved to themselves.

The existence of this onsite office proved beneficial. First, it was an NRR responsibility and so removed the inspection and enforcement group (I&E) from continuous interactions with the cleanup project. This was appropriate because the uniqueness of the TMI-2 situation bore no resemblance to a normal power plant situation, which was the primary scope of I&E. Second, it eliminated the need for the utility to interact with Washington headquarters. The local NRC staff could represent NRC's responsibilities in an expeditious manner.

2.6.2 FSAR and Technical Specifications

The TMI-2 final safety analysis report (FSAR) was not revised to accommodate changes to cope with the accident. Instead, a system was created in which technical evaluation reports (TERs) and safety evaluation reports (SERs) were written by the project for a system or operation. A TER had many of the elements of a mini-FSAR; however, it was narrowly focussed on an issue. A TER could contain, for example, a system description and a safety review.

At the end of the project, the FSAR was not updated, but it was applied in a limited way to TMI-2. In particular, the bounding conditions in the TMI-2 FSAR were used to judge the acceptability of changes, tests, and experiments, and the attendant determination of unreviewed safety questions. It was also applied to those areas not addressed by the post-defueling monitored storage safety analysis report (GPUN 1988)

The technical specifications were gradually modified to accommodate the radically different plant situation. In general, the changes in technical specifications can be seen as a response to the decreasing hazard posed by the plant and the desire by GPU to minimize the resources used to maintain the plant for no real benefit. Since the original technical specifications were of limited value to a damaged reactor, a new set of 30 technical specifications were proposed and then issued via an NRC Order in February 1980.

The Order mandated that the plant be maintained in accordance with the proposed specifications until the operating license was formally amended to include them. The Order also permitted interested parties to intervene—three did. Several years passed while the contentions raised were resolved in a process of meetings and exchanges of information. Meanwhile, the proposed technical specifications were modified by more NRC orders to reflect changing cleanup conditions. In late 1985, the last contention was resolved and in January 1986, the license was finally changed to incorporate the proposed specifications (Byrne and Rogan 1989).

The early modifications for the most part addressed effluent releases and the new systems for cooling the reactor. As the cleanup program progressed, technical specifications were eliminated in several succinct steps because they no longer applied to the plant conditions. For example, after the reactor head was removed, it was no longer necessary to maintain systems for pressurization. Similarly, when the decay heat level was sufficiently low to remove all heat through the coolant system and by convection, there was no technical or safety need to maintain decay heat removal systems.

A long series of change requests were applied to the technical specifications. One of the most important of these was Technical Specification Change Request 53, approved in 1988, which identified three "facility modes". It initiated the concurrent reduction and deletion of technical specifications pertaining to systems and equipment that were not required for a subsequent mode:

- Mode 1—The period during which reactor vessel defueling and other major decontamination and waste shipping tasks were in progress.
- Mode 2—The period after completion of defueling and before completion of the core debris shipping program.
- Mode 3—The period after the last shipment of core material off site.

In August 1988, GPU submitted a proposed license change to "Possession Only" and technical specifications for a Mode 4 (post-defueling monitored storage) condition, in which the following activities would be conducted:

- Monitoring and surveillance

- Decontamination
- Radioactive waste processing
- Special nuclear material accountability
- Water processing.

2.6.3 NRC Advisory Panel

In response to concerns of local residents and politicians, the NRC established and funded an independent review group in November 1980. The NRC Advisory Panel for the Decontamination of TMI, Unit 2 was chartered to consult with and provide advice to the NRC on major activities required to complete the cleanup. It acted as a conduit to convey local public concerns and opinions, and as a forum for intervener groups in the area; e.g., the Susquehanna Valley Alliance (SVA) and TMI Alert (TMIA).

Membership on the Panel varied but usually consisted of 10–12 scientists, local citizens, and/or local politicians who all served without pay. Mr. J. Minnich, Chairman of the Dauphin County Commissioners, served as Panel Chairman until December 1983, when he was succeeded by Lancaster Mayor A. Morris. The Panel met approximately seven times per year and also held meetings with the NRC Commissioners in Washington, D.C. At each local meeting, GPU, NRC, and other involved federal and state agencies would make presentations on the current status of the cleanup and other pertinent issues.

The utility and government agencies often found themselves on the defensive in this public forum, but the meetings provided an outlet for the fears and concerns of the public. The issues that commanded public attention were funding, schedule, reactor vessel head removal, defueling plans, NRC enforcement actions and investigations, fuel shipping, post-cleanup conditions in the plant, and the disposal of accident-generated water.

2.7 Involvement of Others

From the outset, recovery from the accident required skills, specialized equipment, and institutional connections beyond those possessed by an individual utility. Both the utility industry and the federal government understood that if industry and national resources were not used to resolve the situation at TMI-2, then the rest of the industry would be severely impacted (even more than it was). From a management perspective, this involvement represented an extension of the resources available to GPU, as well as the challenge of effectively using these resources.

The major external contributors to onsite activities were the DOE and EPRI. The U.S. Environmental Protection Agency (EPA) also played a role designated by the Executive Office of the President as the lead federal agency for conducting a comprehensive long-term environmental radiation program. Its role was limited, consisting of environmental monitoring expertise, particularly in the stabilization phase when there was extreme public apprehension regarding releases. EPA's role effectively augmented the NRC's responsibility (US EPA 1980).

Internationally, the Organization for Economic Cooperation and Development (OECD) contributed through subcommittees on models and material behavior. Models of the accident were developed, compared to computer codes, and adjustments made. Information was gathered from around the world on material behavior/interaction to help predict postaccident conditions at TMI-2 and support defueling tool design; e.g., eutectic reaction behavior. When defueling was completed, the OECD, NRC, and EPRI co-sponsored a sampling program in the reactor vessel lower head (see Section 5.5.3.2).

2.7.1 U.S. Department of Energy

The most significant extraordinary participant in the cleanup was the U.S. Department of Energy. There were several reasons why the federal government's role was essential. First, the accident generated much political activity that was beyond the jurisdiction of any individual locality or state, thus federal involvement was needed in such matters as effluent release and transportation and disposal of waste. Second, from a technical perspective, expertise and resources existed in the national laboratory system that did not exist elsewhere or were extremely limited, particularly in terms of facilities to handle and examine radioactive material.

DOE's participation during 1979 and early 1980 included an emergency response team, technology transfer (often via advisory groups), technical support for a wide variety of data acquisition tasks, and a local citizen's program for monitoring the release of airborne activity. Late 1979 saw the beginning of what became a major and essential involvement. In January 1980, DOE formally initiated the TMI Information and Examination Program to secure important R&D data that might be of value to the industry and the NRC.

This was furthered in March 1980 when the four-party GEND coordination agreement was signed by GPU, EPRI, NRC, and DOE. The agreement set up policy and technical planning mechanisms and defined objectives and areas of common interest among the parties. The Joint Coordinating Group and the Technical Working Group were established and a Technical Integration Office was set up at the TMI-2 site[1] (and INEL) and operated by EG&G Idaho, Inc (AGNS 1980, Comptroller General 1982). The site office, in its heyday, comprised 48 engineers and support staff. At INEL, up to 25 people worked on the cleanup at any one time.

In early 1981, the new administration in Washington committed the substantial funding necessary to implement large-scale work over the course of the cleanup. A fundamental point of this commitment was that it was not to bail out the utility but to benefit the Nation. The program was predicated on the need to resolve the impasse that existed over how the cleanup would be carried out and funded, and to determine the accident scenario and progression so as to ensure the safety of nuclear power. DOE officials believed that their offer to assist in the cleanup through an R&D program would have several advantages:

- Encourage other parties to modify their positions sufficiently to resolve the impasse
- Expedite the cleanup, thereby reducing total costs and minimizing further deterioration of equipment with possible public and occupational health repercussions
- Limit the possibility that the federal government would eventually be required to assume total responsibility for the cleanup
- Enhance the regulatory agencies' and industry's knowledge of the causes, effects, and prevention of a similar accident and thereby improve safety at other nuclear plants
- Enhance DOE's knowledge of high-level radioactive waste disposal (Comptroller General 1982).

[1] The Technical Integration Office was actually established on site in October 1979, and fully operational by January 1980.

The DOE focussed on two areas: 1) data acquisition and analysis, which was a generic DOE responsibility authorized by Congress and strongly recommended at TMI-2 by the Kemeny Commission (Kemeny, et al. 1979); and 2) research and development, which centered on high-level radioactive waste immobilization and shipment; reactor core access, removal, and shipment; characterization of the molten core, damaged core structures, and reactor vessel; and analysis and studies to understand the accident scenario. By accepting the TMI-2 fuel core for research and temporary storage, DOE made a vital contribution to the cleanup and resolved the dilemma of what to do with the debris (see sections 6 and 8).

2.7.2 *Electric Power Research Institute*

The nuclear power industry, primarily through EPRI but also through direct contributions of managers and engineers, was technically involved in the cleanup in three phases. The first phase started the day following the accident and consisted of personnel and equipment donated or loaned to address the immediate crisis. The direct contribution of personnel from utilities in the initial postaccident period was one of the most valuable and substantial forms of participation.

The second phase began with the establishment of a permanent EPRI site office in November 1981. This resulted from the GEND agreement and eventually contributed to research projects on decontamination and dose reduction, mechanical component survival, primary system pressure boundary characterization, and robotics. Most of these were tasks identified as being of potential value by the GEND agreement (AGNS 1980, IEAL 1982).

The site office consisted of only a few engineers but was able to contribute by: 1) subcontracting research off site before onsite demonstration; and 2) the innovative use of specialists who were sponsored by EPRI but worked within the cleanup project organization. Using these specialists to both support the cleanup and transfer the technology to the utility industry proved mutually beneficial. In addition, EPRI also supplied its own in-house expertise or sponsored one-time work by experts to address such things as robotics and water filtration problems. With other organizations, it assisted in areas that might not otherwise have been funded; e.g., R&D related to robotics. The commercialization of technology developed for the cleanup was supported and promoted by EPRI as a practical means of transfer to the industry.

The degree of EPRI involvement in the cleanup is reflected in contributions made to several projects:

- **Mechanical Components Examination**—What started as a broad scope program ended up focussing on the polar crane refurbishment. This was a very important early activity because the polar crane was key to many activities necessary to proceed with disassembling and defueling the reactor vessel. In cooperation with others, the crane was evaluated, refurbished as necessary, and requalified for use. Because of a lack of apparent damage, most other mechanical components were believed to be relatively too expensive to retrieve from within containment.

- **Portable Spectroscopic Detectors**—A compact (35-kg) scintillation spectrometer was developed by New York University. The value of this project was to provide a unit that was much easier to use than larger units that were state-of-the-art at the time.

- **Chemical Decontamination Studies**—There was early interest in the possibility that chemical decontamination would eventually be needed to clean the primary system. EPRI first sponsored a review of candidate reagents and combinations for a chemical decontamination process. A followup project bench tested the prime candidates and estimated the magnitude of the reagents and waste processing that would result.

- **Nonchemical Decontamination Techniques**—After studies showed that chemical decontamination would mean significant waste problems, EPRI helped sponsor the demonstration of several devices such as flexhones, ultra-high pressure water jets, and others devices. One task developed an articulated arm for decontaminating the underside of the reactor vessel head with water spray should it be needed during head removal. This was unnecessary; however, the device was later used to determine the mobility of the particles in the bottom of the reactor vessel displacing them.

- **Remote Technology**—EPRI participated in the development and testing of several remotely controlled vehicles and devices. These were primarily used for assessment and decontamination work in the highly contaminated containment basement.

- **Cool Suit**—EPRI sponsored development of an ice vest that was used extensively during containment entries to reduce heat stress to the workers.

- **Analyses and Evaluation of Accident Scenario**—EPRI participated in studies that used data from the degraded core to validate analysis methods to predict severe accident characteristics and consequences (the Modular Accident Analysis Program).

There were other specific technology development projects sponsored or partially supported by EPRI. They included software, hardware, and expert consulting. Most of these were related to waste management and decontamination challenges per the original GEND agreement.

In mid-1984, with the announcement of direct contributions to the cleanup by the Edison Electric Institute, EPRI's role changed. The focus then became specifically to evaluate and transfer cleanup technology and severe accident R&D results to the U.S. utilities via reports, demonstrations, presentations, and workshops. This change was formalized by a memorandum of understanding signed by EEI, EPRI, and GPU in February 1985 (MOU 1985).

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3 STABILIZATION

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STABILIZATION

3.1 Overview

As the accident ended, the pressing need was to stabilize conditions throughout the plant. How to keep control of the reactor? How to regain access to vital but highly contaminated areas of the plant? How to contain and manage large quantities of radioactive water and gas?

The months following the accident—when these questions were answered—were later defined as the "Stabilization Phase." It was:

"...directed toward achieving a reasonable degree of control for reactivity, water inventory, reactor coolant system temperature and pressure, mobile radioactive fission products, and radioactive wastes, as necessary, to permit longer term tasks to proceed safely" (DeVine and Negin 1984).

These goals were basically clear from the beginning, but the terminology and duration were not. Many thought they were in a "recovery phase" that would put the plant back on line in a few weeks or months. Others felt it was a time of hectic yet successful efforts to plug the dike.

When the initial crisis period ended on April 4, 1979, the engineering and operations challenges appeared staggering. While individual tasks had past analogs, the magnitude was far greater. The situation facing the staff was:

- The extent of damage to the reactor was unknown.
- Reactor coolant system pressure and volume control were being maintained by non-safety-related equipment, leading to concern about long-term reliability.
- Decay heat was being removed by non-safety-related equipment because using the installed safety-related system would have resulted in much higher radiation levels in parts of the auxiliary building.
- The auxiliary building had recurrent air contamination problems and the containment held approximately 64,000 curies of krypton-85.
- High dose rates inside the plant limited access and threatened to damage important equipment (dose rates ranged up to several thousand R/h).
- Liquid radioactive waste was being generated faster than it could be processed and threatened to exceed available tank capacity (approximately 3.8 million liters of contaminated water existed and the volume was steadily on the rise, especially in the containment basement).

The first one to two months were an emergency period of gaining control; the following months saw that control consolidated. By the end of 16 months, the project had addressed the overriding questions of safety and control posed by the postaccident conditions:

- The threat to the workers, public, and environment was vastly reduced.
- Removal of decay heat from the reactor was passive.
- Radioactive gas was no longer a significant concern.
- The influx of water was minimal and water processing systems were either in operation or under construction.
- Waste storage was adequate.
- Access to most plant areas, including the containment, had been gained and the work of detailed characterization, planning, engineering, and cleanup could begin.

Figure 3-1 shows the major milestones of this phase.

Because of the emergency nature of the stabilization tasks, parallel paths were usually taken to the same end.

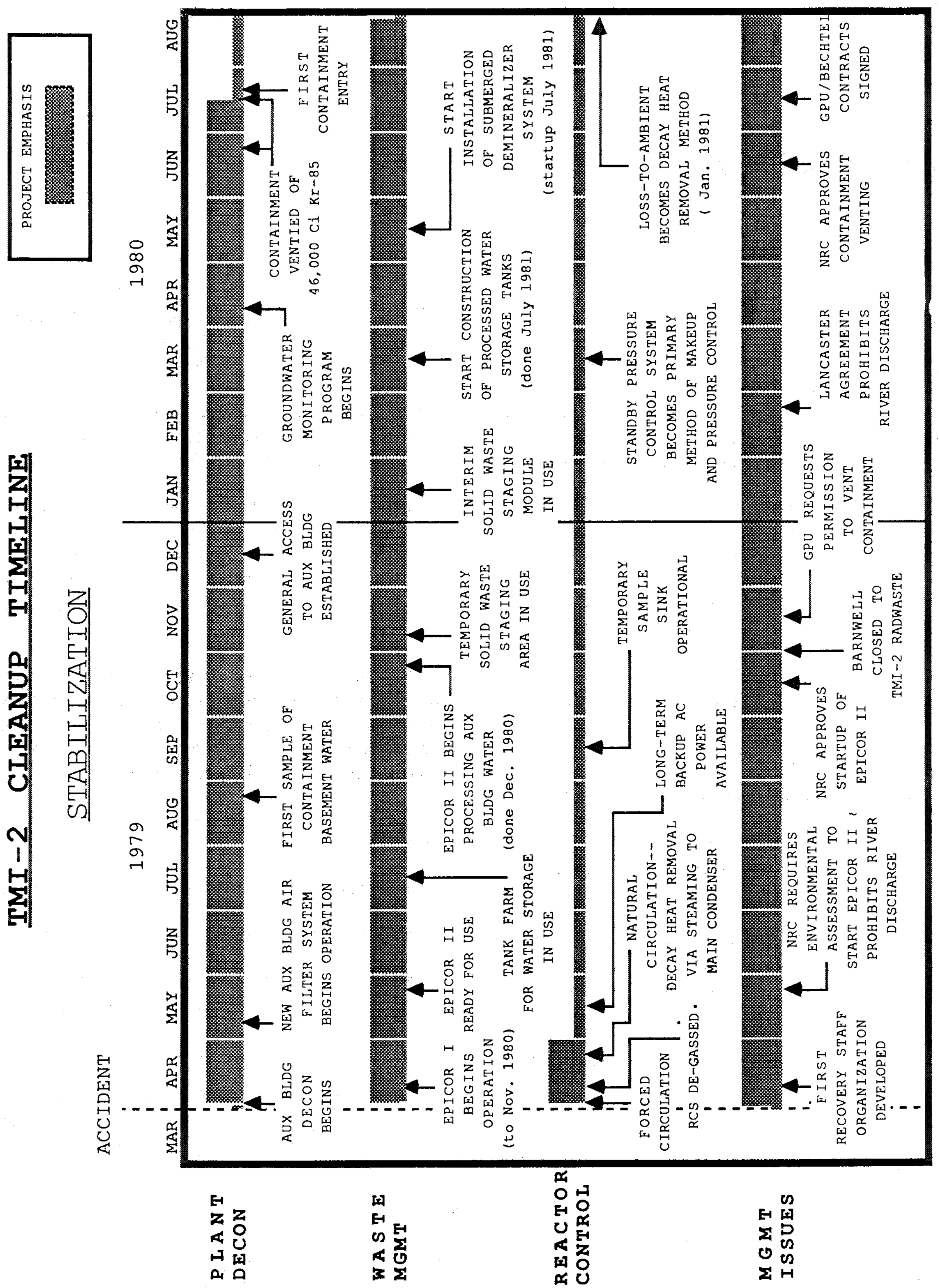

Figure 3-1. TMI-2 Cleanup Timeline — Stabilization

Whichever path first led to the desired result was used. The others were either abandoned or finished as a backup. Because of time and procurement constraints, designs often had to be fashioned from the available material and equipment rather than from the optimum choice.

The organizational structure itself was a major contributor to meeting the challenges successfully (see Section 2). It was a project organization with a direct line of responsibility from the top to the workers. A small group of engineers and designers was assigned to each task; they had the responsibility and authority to get the work done. Close cooperation among operators, engineers, craft labor, and suppliers permitted problems to be addressed as they occurred. Within ten months of the accident, this organization had installed and made operational:

- A complete nuclear-qualified ventilation and filtration system
- Two diesel generators and an emergency power system
- Over 380,000 liters of new storage tank capacity
- Two ion exchange-based radioactive water processing systems
- A secondary closed cooling water decay heat removal system
- A passive reactor pressure control system and many lesser systems.

Much of this was built in less than two months. Some of these systems, such as the diesel generators and the secondary side cooling system, were never needed. However, others were major systems used for years after the accident.

This section describes the stabilization phase first in functional terms of how the reactor was controlled, radioactive gas was contained, and electrical and other support systems were built. Decontamination and waste management efforts are discussed, but only for completeness and to note those systems or facilities unique to the first months (sections 6 and 7 provide more complete descriptions). The purging of krypton-85 from the containment in the summer of 1980 concludes the section as it concluded the first phase of the cleanup (Section 4 discusses the effort to re-enter the containment).

3.2 Reactor Control

Even at a plant as damaged as TMI-2, certain essential operational and control functions were considered necessary to keep the reactor cool. These were:

- Maintaining reactor coolant flow
- Maintaining reactor coolant system (RCS) heat removal
- Maintaining RCS water inventory
- Controlling RCS pressure
- Ensuring that the core remained subcritical.

These functions will be used to address reactor control operations during the stabilization phase.

For perspective, the conditions shortly after the accident and at the beginning of the stabilization period were:

- Reactor coolant system pressure—~1165 psig
- Reactor coolant temperatures—~389 to 394 K (240–250°F)
- Incore thermocouple temperatures—~589 to 644 K (600–700°F)
- Reactor coolant flow—forced circulation using the "A" pump in the "A" once-through steam generator (OTSG) loop
- Heat removal method—steaming to the main condenser from steam generator "A"
- Containment temperature—~320 K (117°F)
- Containment pressure—negative 0.2 psig
- Dissolved gases existed throughout the RCS
- Unknown core cooling conditions
- No reactivity control.

3.2.1 Reactor Coolant Flow

The severely damage core and resulting large quantity of noncondensible gas threatened to block coolant flow. The project team had to act quickly to gain control, and then, over the course of several months, establish the long-term reliability of coolant flow.

3.2.1.1 Noncondensible Gases

The most immediate challenge to stabilizing the reactor was the large quantity of noncondensible gases, primarily hydrogen, that had collected in the reactor coolant system high points. The so-called "hydrogen bubble" was never in danger of exploding because it had been formed by the zirconium-water reaction and thus no oxygen was present in the gas.[1] The primary threat was that gas would create disturbed flow conditions (i.e., accumulate and block the hot leg "candy cane" pipe to the reactor vessel) and perhaps lead to pump damage. These pockets of gas had to be removed in order to maintain sufficient coolant flow to ensure decay heat removal.

The operations staff quickly developed procedures to control and remove the noncondensible gases from the primary system. This degassing operation was conducted by: 1) maintaining the system pressure high (900–1100 psig) to force the gas in the loops and reactor vessel into solution; and 2) by continually adding reactor coolant makeup water. The solution was then sprayed into the pressurizer. By maintaining the pressurizer temperature considerably higher than the coolant system, the gas effervesced and was then partially removed by venting the pressurizer into the containment. It was also removed by depressurizing the letdown water to the purification system. Six days were required for the hydrogen to be effectively removed—it was one of the first major recovery tasks successfully completed.

3.2.1.2 Forced Circulation

Eight days after the accident, the only then-operating reactor coolant pump in loop "A" stopped because of vibrations; the other loop "A" pump was started two minutes later. Because of concerns with potential leakage in the "B" once-through steam generator, that generator was not used for heat removal and, consequently, the loop "B" reactor coolant pumps were not used. They also had indications of high vibration.

To operate the reactor coolant pumps required a system pressure of approximately 175 psig to prevent cavitation. Without reliable instrumentation, it was possible to inadvertently reduce the pressure below that required for pump operation. Therefore, even if a pump were operating satisfactorily, loss of the instrument functions could effectively reduce the operators' confidence to the point of stopping the pumps, thus causing potential loss of forced circulation. To avoid using the "B" loop in the event that the remaining "A" loop pump was stopped, plans were developed for natural circulation heat removal without reactor coolant pump operation.

3.2.1.3 Natural Circulation

Natural circulation is defined as occurring when the flow through the reactor coolant system is driven solely by the thermal buoyancy (density difference) of water heated in the reactor and cooled in the steam generators. Natural circulation was expected to be reliable (and not dependent upon mechanical pumps), but there were questions about whether the transition to it would be successful. The available alternatives to natural circulation cooling, including high- and low-pressure injection or recirculation, did not offer the same long-term assurance of reliability and fission product containment.

On April 27, 1979, the sole operating reactor coolant pump ("A") was stopped because of the failure of the third and last pressurizer level transmitter. With this event, operators made the planned switch to natural circulation. Two immediate advantages were gained:

- The system was more reliable because fewer mechanical and electrical components were operating
- A large source of reactor coolant heat was removed when the reactor coolant pump was shut down.

At that time, the energy input from the reactor coolant pump was approximately two-thirds of the total reactor coolant heat load, with reactor decay heat providing the other third (the heat was then being removed by the steam generator). Thus, when the pump was stopped, the reactor coolant system began to cool.

Cold shutdown conditions were established on the evening of April 27, 1979.[2] At first, both once-through steam generators were used with steam directed to the main condenser and the turbine turning gear running. Within a day, the "B" steam generator was isolated when there was a radiation level increase detected in the station vent. This was later determined to be the result of changing the station vent filters; however, the "B" steam generator remained isolated throughout the following years.

The decay heat load continued to decrease, and so did the natural circulation flow rate. The operators were concerned that flow would be lost and temperatures rise as smooth natural flow diminished. However, time and observation of natural circulation behavior answered this question satisfactorily.

[1]This should not be confused with the hydrogen burn (a pressure spike of at least 28 psig), which occured in the containment atmosphere on March 28, 1979.

[2]Cold shutdown, the most passive state for a nuclear plant, is a condition defined as negative reactivity, zero power, and a reactor coolant temperature low enough to avoid boiling, usually specified as below 366 K (200°F).

One unanticipated occurrence was a flow transient in the reactor coolant system—commonly known as the "B" loop "burp". It is illustrated in Figure 3-2. (This phenomenon occurred in both steam generators, but "B" is used here as an example.) The fluid in the "B" steam generator and the "B" loop isolated cold legs gradually cooled until the density was high enough to initiate natural circulation flow in the "B" loop. The flow was only sustained until the warmer fluid from the reactor vessel displaced the cold fluid in the "B" steam generator and cold leg. Hence, the phenomenon was more a repositioning of fluid of different densities to achieve hydraulic balance. The fluid was then stationary for several days until another "burp" occurred.

The observed behavior of the isolated "B" steam generator provided insights on what to expect when natural circulation ceased altogether and it removed fears that the core might overheat once circulation stopped for the first time. When the temperature difference between the reactor vessel and the steam generators was sufficient, natural circulation would restart. Continuous natural circulation stopped in October 1979. As expected, circulation periodically re-established itself without the need for active measures by the plant staff. The periodicity of the surges was initially at about four hours per burp of hot water into the cool steam generators. The period gradually increased to over 200 hours. In January 1981, the loss-to-ambient mode of decay heat removal (convection and conduction to the containment environment—see Section 3.2.2) superseded natural circulation.

3.2.2 *Heat Removal and Achieving Cold Shutdown*

Getting to and ensuring cold shutdown did not follow the normal course. Many alternative decay heat removal systems were investigated, several were completely designed, and one was completely installed. All of them were conceived and developed with an eye on the then-current emergency conditions; i.e., decay heat in the thousands or several hundreds of kilowatts. But that production level continued to diminish over the summer and autumn of 1979. In the end, the reactor cooled via natural circulation and, eventually, loss-to-ambient. The over-design of some of the alternative decay heat removal systems consumed resources but seemed prudent at the time.

The normal scheme of decay heat removal for the TMI-2 pressurized water reactor (PWR) used the once-through steam generators to cool the reactor (initially to approximately 395 K or 250°F) and then used a medium-pressure decay heat removal system to achieve cold shutdown.

The TMI-2 decay heat removal system's design pressure was 350 psig. Because of the need to maintain RCS pressure between 900 and 1100 psig for degassing operations, reactor system pressure was necessarily greater than that allowed by the design of the decay heat removal system. Consequently, the decay heat removal system was not used during the first week after the accident. That turned out to be extremely fortunate because the very high specific activity (approximately 10,000 μCi/ml) of the coolant water made using the decay heat removal system something to avoid at almost any cost. If the decay heat removal system had been used, the dose rates near the system would have been prohibitively high—approximately 11,000 R/h adjacent to where the pumps were located. In addition, leakage from the existing pump seals would probably have caused severe contamination problems.

The only viable installed alternative to the decay heat removal system was to use the steam generators to achieve and maintain cold shutdown. This meant operating the secondary side equipment indefinitely, and relying on non-safety-related components to cool the reactor. At the beginning of the recovery, no one could be certain how long that method of decay heat removal would work. If any of the major components failed, there would be no choice but to use the installed decay heat removal system (see Appendix C). Figure 3-3 illustrates the method used to remove decay heat (and to force circulation) shortly after the accident.

By April 4, 1979,with decay heat production down from 160,000 kW to 5400 kW, the crisis phase of the accident was over. The noncondensible gas pockets had been removed from the reactor coolant system and the general situation had stabilized. A concerted effort then began to cool the reactor enough to achieve cold shutdown. To achieve cold shutdown while steaming with the steam generators, operating temperatures had to be below 373 K (212°F) in order to remove stored energy in case of a leak. This required operation of the steam generator shell side and the main steam piping at vacuum conditions. The main condenser vacuum pumps were used to reduce the pressure in the steam system. The plant was cooled from 413 to 386 K (283 to 235°F) between April 13 and April 19, 1979. On April 19, the main turbine started turning as a result of opening a bypass valve to increase steam flow to the condenser. The reactor coolant temperature decreased to 360 K (188°F) and cold shutdown was initially achieved on April 27, 1979 (the temperature later briefly rose above cold shutdown conditions).

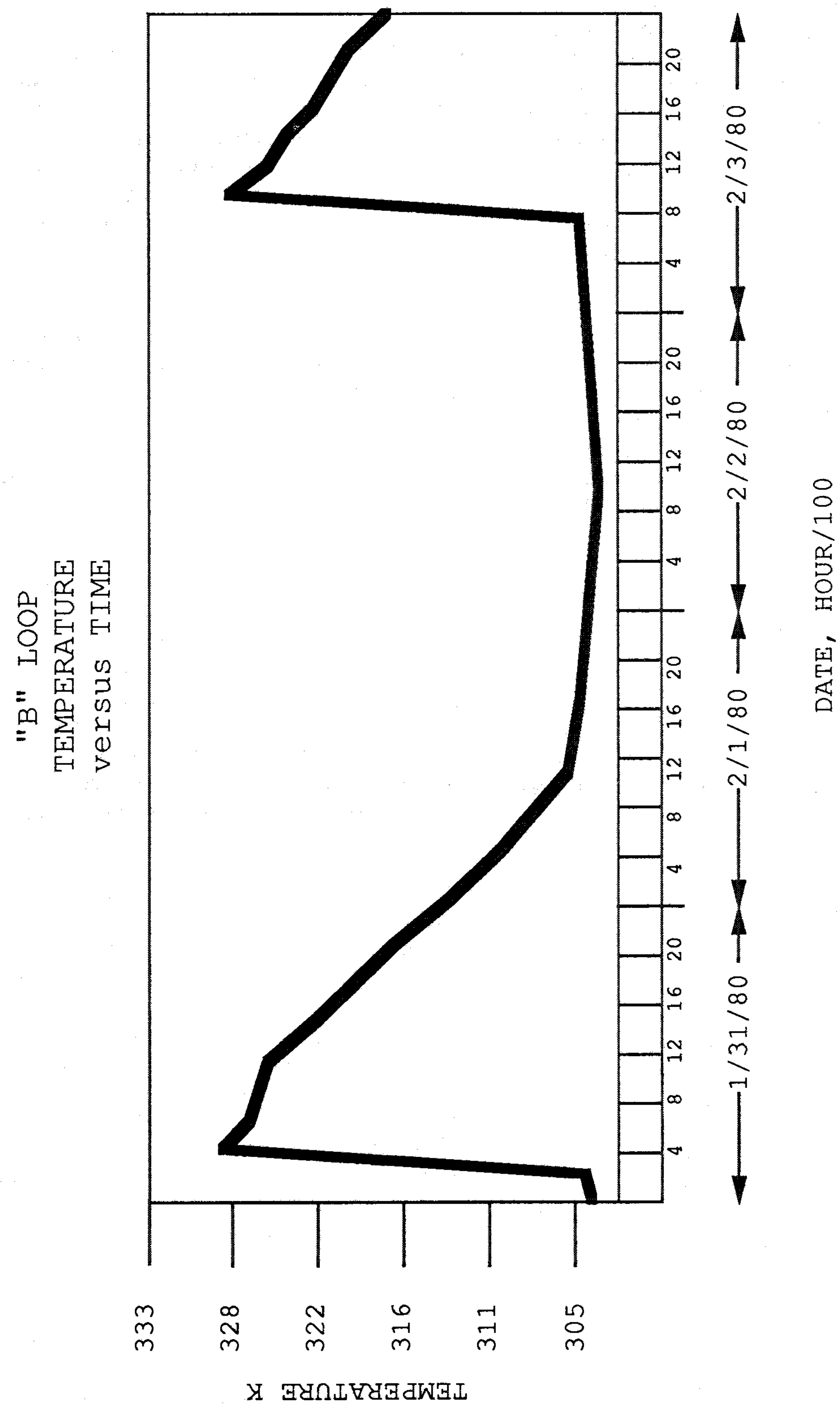

Figure 3-2. "B" Loop Temperature versus Time

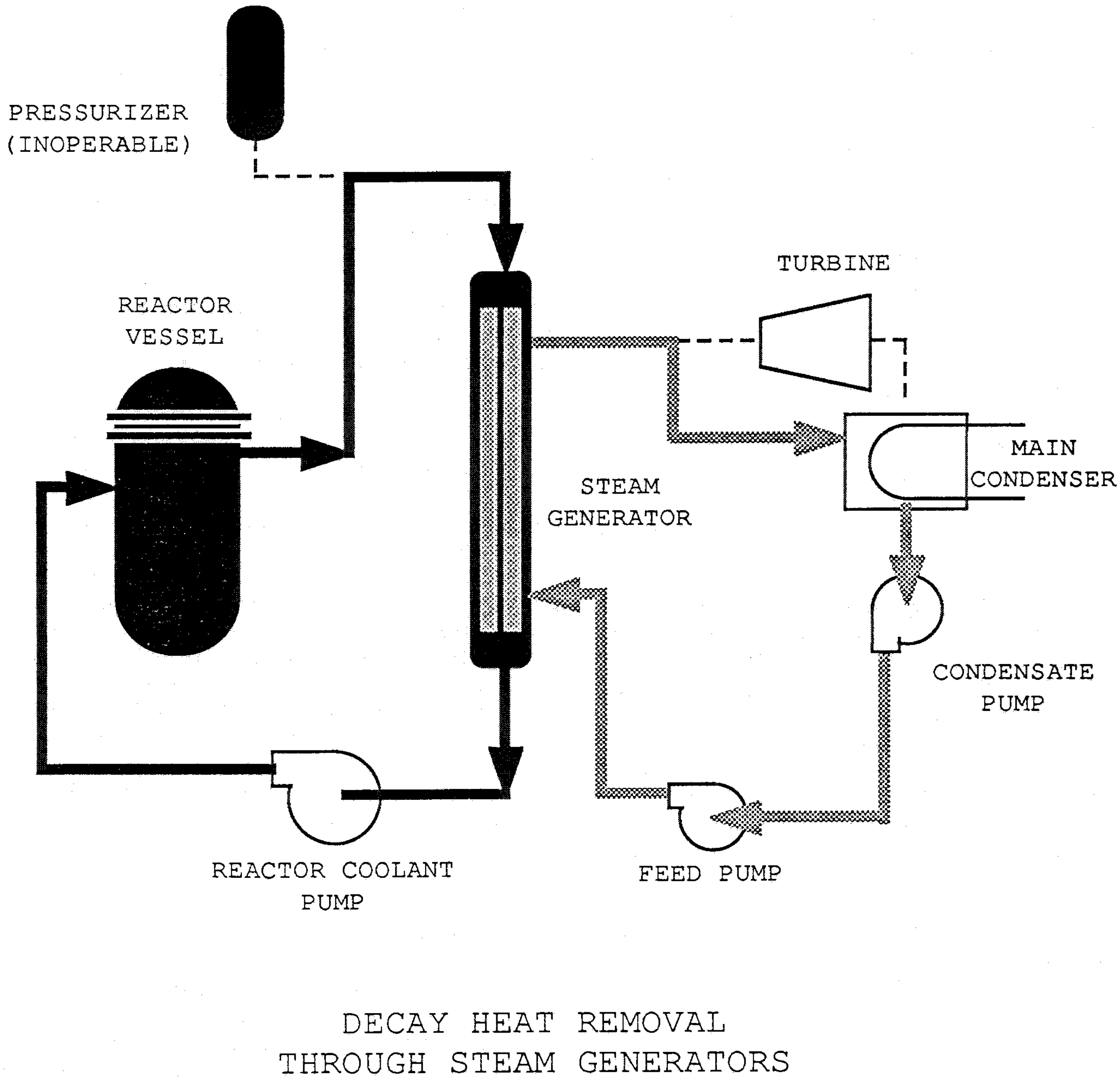

Figure 3-3. Decay Heat Removal through Steam Generators

Because of the potential vulnerability of secondary side heat removal and the undesirability of using the installed decay heat removal system, the project team decided to develop backup methods to remove the decay heat from the core. Several concepts were defined that included short- and long-term approaches. Ultimately, these concepts evolved into three general methods of implementation, each with its advantages:

- **Install new primary side systems**—The main advantage of installing a new primary side decay heat removal system was that it could be more operationally versatile; i.e., could also be used for chemistry, inventory, and pressure control.

- **Install new secondary side systems**—Secondary side approaches were more expedient because the bulk of the heat removal equipment was already installed (i.e., steam generators, main steam piping, condensate piping, the main condenser, vacuum pumps, etc.) and most of the work could be done in the turbine building, which was barely contaminated by the accident. Furthermore, the steam generators isolated the reactor coolant to within the containment—this permitted the use of conventional, nonnuclear components, which simplified the procurement, design, and construction efforts.

- **Improve reliability of existing secondary side systems**—This approach was the immediate response and, because the systems were not classified as safety-related, was open to questions of long-term reliability. No commercial PWR had ever operated its secondary side systems long enough to achieve cold shutdown. The systems were not designed for the low flow rates and low heat loads associated with achieving cold shutdown. No one could be certain how long, chronologically or in terms of performance, secondary systems would continue to work.

The project team decided to continue using the steam generators for heat removal while proceeding in parallel with the following specific alternatives:

- On the primary side, an alternate decay heat removal (ADHR) system was begun outside the fuel handling building—it was never completed because it was too ambitious, posed serious environmental risks, and was oversized (see Appendix D).

- On the secondary side, a long-term "B" cooling system was partially installed—it remained unused initially because of some doubt about its reliability and eventually because loss-to-ambient proved to be a satisfactory method of heat removal (see Appendix E).

- When the project team realized how long it would take to build the ADHR, a smaller, safety-grade seismic system was built, tested, and turned over to the plant operations staff. Although never used, this system —called the mini-decay heat removal (MDHR) system—was intended for the long haul at low decay heat generation rates. It remained in standby after installation (see Appendix F).

- Because of the potential failure of the existing condensate pumps, a backup mini-condensate system was installed (see Appendix G).

As it turned out, until decay heat became sufficiently low to rely on direct heat transfer to ambient, the only decay heat removal path used after the accident was via steam generator "A" and the main condenser.

By 1980, the reactor appeared able to cool itself via heat loss to the atmosphere and via portions of the primary system submerged in the flooded containment basement. In November 1980, a test of the loss-to-ambient mode of cooling was completed. No significant increase in reactor coolant system or containment temperature was noted. In January 1981, the turbine bypass valve from steam generator "A" to the condenser was closed, isolating the primary system from all active cooling modes. Loss-to-ambient became the sole mode of decay heat removal. (From January to December 1981, decay heat generated by the core decreased from 95 kW to 50 kW; by 1983, it was less than 30 kW.) The ability to use the loss-to-ambient method for heat removal was a significant plus for the project because it simplified a task that could have been far more complex.

3.2.3 Primary Coolant Volume and Pressure Control

The systems for reactor coolant system volume and pressure control were lost as a result of the accident. Although operators were able to work around the problems temporarily, the construction of a new pressure control system was necessary.

When the accident fragmented the core, the letdown to the makeup system became loaded with solids. On the day of the accident, the letdown flow was lost because of blockages in the system. Letdown flow was reestablished the next day, at a reduced flow rate of approximately

9.5E-04 m^3/s compared with the normal 2.8E-03 m^3/s. This flow was achieved by bypassing the letdown filters and the purification demineralizers, leaving the reactor without a cleanup system.

The normal method of controlling the reactor coolant pressure was to increase pressure with the pressurizer heaters and to reduce pressure with water sprayed into the pressurizer steam space. Water for volume control was supplied by the makeup pumps, which were also the high-pressure safety injection pumps. The letdown system removed water from the system via filters and demineralizers, which purified the water before it was used again for makeup.

The makeup system was in operation during the accident. However, blocked letdown flow mandated throttling both makeup flow and seal injection flow. The makeup pumps provide cooling and seal water to the reactor pump seals in addition to providing makeup water to account for the leakage from the system. However, the chemistry and purity of this water was suspect. The dose rates near the makeup system were extremely high—in excess of 1000 R/h in places—which made normal surveillance and maintenance impossible.

As described in Section 3.1.1, the reactor coolant system had to be maintained at a high pressure because of the presence of noncondensible gases in the system. The pressurizer heaters and their electricity supply had to be relied upon under adverse environmental conditions in the containment, through which the heater electric cables are routed. The insulation on the power cables feeding the pressurizer heaters was not intended to operate in the high-radiation environment inside the containment. In addition, most of the transmitters for the pressurizer instruments were located in the basement and were threatened by the amount of water being added and a rising basement water level.

Because of the makeup system's inaccessibility and the threat to pressurizer control reliability, the project team decided that a new method was needed to control the inventory and pressure of the reactor coolant system and to provide a means of water chemistry control. An engineering project was begun immediately to design a new pressure/inventory control system.

As long as the pressurizer heaters and instruments survived, the pressurizer could be used. The pressurizer pressure and level instruments ultimately proved to be the weak link. Readings were suspect almost from the beginning. The operations staff tried to develop a backup level measuring system based on a highly precise pressure gauge on the letdown line outside of containment. This did not prove to be sufficiently accurate because the level changes were on the order of less than a meter of water (i.e., differential pressures of 5 to 10 psi), while the pressure was several hundred psig. Consequently, any changes were very difficult to measure.

While work proceeded on the new system, plant operators had to maintain pressure with the available systems. After the last of the pressurizer level instruments failed on April 27, 1979, the pressurizer could no longer reliably be used to maintain pressure. The pressurizer was then filled with water, and the reactor pressure and inventory were controlled by balancing the makeup and letdown flows. This was predominantly done by balancing the choked letdown by manually throttling the seal injection manual (leaking) valves and completely closing the makeup valve. This method of pressure control was very sensitive because it relied on the compressibility of water and elasticity of the system boundaries for a cushion. Nevertheless, because the heat input to the system was not changing very fast, controlling the pressure control in this manner generally worked well.

The main disadvantage of operating the reactor coolant system solid was that it relied on active components over a long period of time. The makeup pumps had to be kept on-line and the valve used to throttle the letdown flow had to be regulated. Small leaks in the makeup system piping added to the contamination problems in the auxiliary building and were a primary pathway for the release of radionuclides. Eventually, a different way of operating would be necessary.

Two concepts for inventory, pressure, and chemistry control were pursued:

- An active pressure/volume control system similar to the existing makeup system
- A passive accumulator system similar to a pressurizer.

The two concepts were subsequently combined into one system called the standby pressure control system (SPCS), which consisted of pumps and accumulator tanks using nitrogen gas for surge suppression (see Appendix H). In April 1979, three 3400-L surge tanks (originally for a fast boron injection system), two large and one small reciprocating charging pumps, and 12 large nitrogen bottles were procured. These were connected to create a standby pressure control system. The SPCS, although not complete, was available to go on line by June 1979, if a need developed. The system was essentially completed by the end of 1979. It was connected to the reactor coolant system in early 1980.

Figure 3-4 shows a schematic of the SPCS, which is also shown installed in Photo 3-1.

The operators preferred using the existing makeup pumps because the pumps were familiar to the operators and could be operated from the main control room. This could not last, however. In February 1980, a fitting broke on makeup pump "1B" and 3200 liters of reactor coolant spilled into the auxiliary building. This resulted from starting the "1A" pump while the "1B" pump was operating. The resulting water hammer ruptured a small compression fitting on one of the pumps. The consequent krypton-85 release forced the evacuation of the auxiliary building and interrupted decontamination activities. The broken fitting was isolated and the system was restarted to maintain reactor coolant pressure.

This event reinforced the need to change the method of pressure control. The makeup system had operated too long without routine maintenance and so the makeup pumps were stopped in March 1980, and never restarted. At the time of the 3200-L leak from the makeup pumps, the SPCS system had been in standby and floating on line "in test" for several months. After this incident, the SPCS was formally placed in operation and became the primary method of controlling the pressure and inventory of the reactor coolant system. It remained in operation from March 1980 through the summer of 1984, when the reactor was depressurized before vessel head removal.

3.2.4 *Reactivity and Boron Control*

Given the unknown severity of core damage, the control rods were not relied on in any way for reactivity control and assurance of shutdown. The operators had no way of measuring the criticality margin so, to ensure that the core would not become critical again, a high boron concentration was maintained in the coolant.

Later knowledge showed that the control material was essentially gone from the core region as a result of melting. Still, recriticality was virtually impossible because of the non-optimum configuration of the agglomerated fuel in relation to the surrounding water. This was unknown at the time and the project staff then judged it imprudent to rely on analyses or models when a method assuring shutdown with boron was available.

The normal boron concentration in the TMI-2 reactor coolant system was 1000–1500 ppm. Just before the accident, a routine sample of the reactor coolant contained 1026 ppm boron. A sample taken shortly after the reactor trip indicated a boron concentration of only 700 ppm. This caused concern, especially when, two hours later, another sample showed a boron concentration of approximately 400 ppm. At the time, the operators believed that this was evidence of a boron dilution accident. In fact, it was due to reflux boiling in the core caused by low pressure and high temperatures. Much of the water in the coolant sample, which was taken from the loops, was condensate that contained no boron; whereas because of boiling, there was a higher than normal concentration in the core.

Immediate steps were taken to raise the boron concentration in the RCS because of the low concentration samples and the higher than normal neutron flux readings from the source range monitors. As a result of these efforts, boron concentration increased to 1750 ppm. Because of the uncertainty regarding the extent of damage to the core, the boron concentration was then raised to over 3000 ppm and this limit incorporated into the technical specifications. The limit was eventually raised to 4,350 ppm to support defueling (see Section 5.5.1 for more discussion of recriticality issues).

Boron concentration was controlled primarily by limiting the sources of makeup water containing the requisite concentrations. Boron dilution events due to possible operational error were avoided by ensuring that only approved sources of makeup water were available for injection. Reactor coolant samples were analyzed weekly to confirm this method of control. The routine was that at 4 a.m. on Monday, 500 cc of water (at 8 R/h on contact) were withdrawn, sealed, staged, packed, doubly packed, taken to the airport, flown by chartered flight to the B&W laboratories in Lynchburg, VA, analyzed, and results reported by noon.

This approach had the disadvantage of not being able to detect rapid changes in boron concentration resulting from possible equipment failures. For this reason, a boronometer was ultimately added to the temporary sample sink (see Section 3.5.2). The boronometer provided on-line boron concentration readings that would indicate any boron dilution events. After the boronometer was put into operation in 1980, this method of reactivity control by boron concentration was not changed.

3.3 Controlling Radioactive Gas

Because of the need to work more efficiently and the seriousness of offsite releases, controlling radioactive

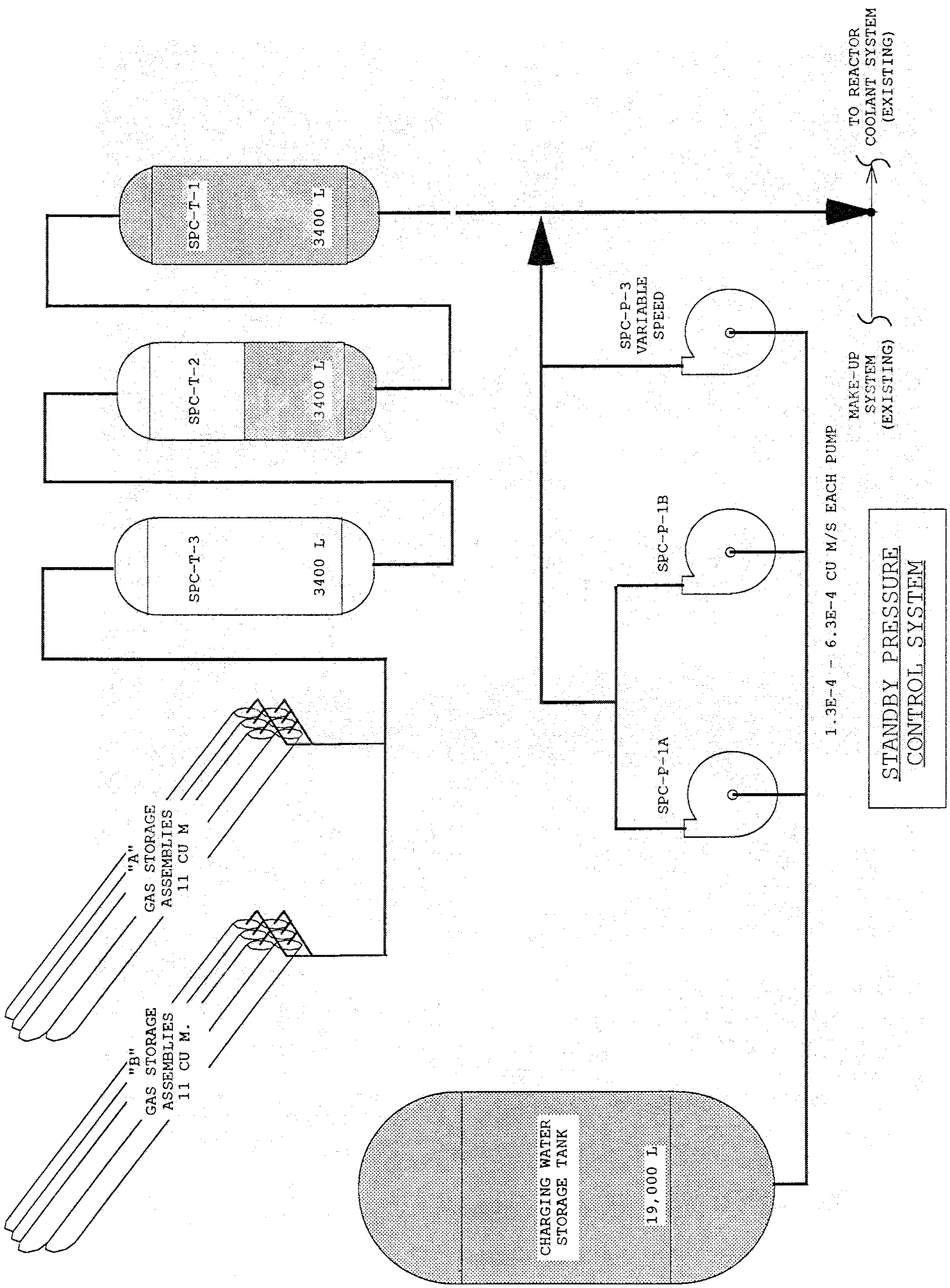

Figure 3-4. Standby Pressure Control System

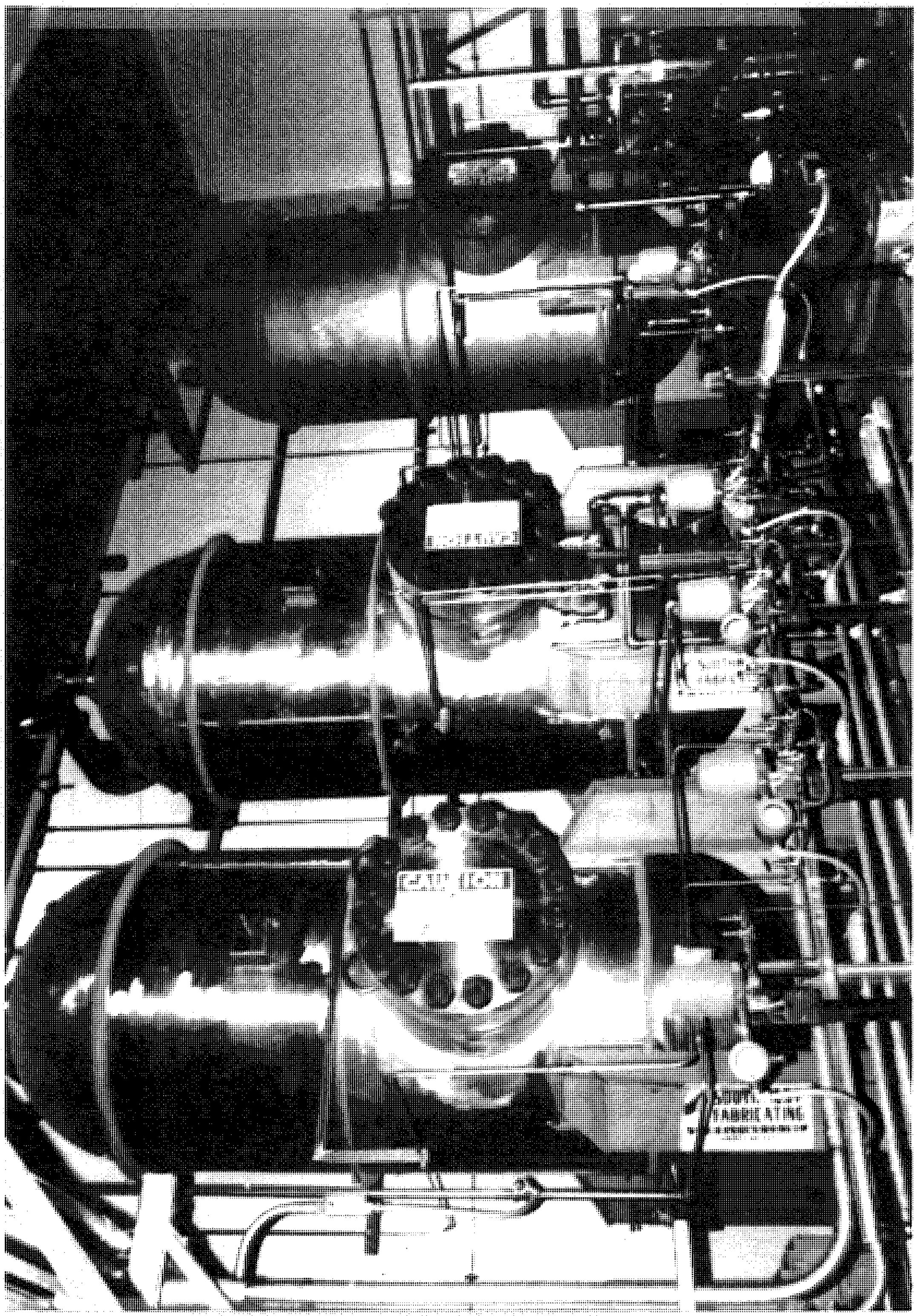

Photo 3-1. SPCS Pressure Control Vessels

gases was a high priority task of the early recovery. Several courses of action were initiated to address the problem. These included:

- Releasing the contents of the waste gas decay tanks into the containment
- Finding the source of the leakage into the auxiliary building
- Replacing the charcoal filters when the dose rates subsided
- Designing a charcoal bed adsorption system
- Designing and constructing a new AFHB ventilation system
- Purging the containment of krypton-85 (see Section 3.7).

To understand why these actions were necessary, it is first necessary to understand how radioactive gas was handled during the accident. Normally, waste gas was transported via a vent header, compressed into storage tanks, held up for decay, and filtered before being released to the environment.

During the accident, ruptured fuel released substantial quantities of gaseous fission products and the rapid oxidation of zirconium cladding released large quantities of hydrogen, all into the reactor coolant. As a result of RCS letdown to auxiliary building systems, these noncondensible gases were carried to various low pressure tanks in the liquid cleanup system, where they came out of solution. (These gases were the primary source of problems.)

On the day following the accident, plant operators began periodic manual venting of the makeup tank to the waste gas collection manifold in order to eliminate the tank back-pressure. However, because of the efforts to reduce the gas bubble in the reactor coolant system, the pressure buildup in the makeup tank could not be controlled by periodic purging. On the second day following the accident, the pressure in the makeup tank opened the relief valve and all of the water in the makeup tank was discharged to the reactor coolant bleed holdup tanks (three 300,000-L tanks in the auxiliary building used for primary system inventory control). Because the makeup tank was the direct source of water for primary system makeup, the water level had to be restored and the operators had to manually open the makeup tank vent in order to restore it.[3]

These gaseous releases were causing serious airborne contamination problems inside the auxiliary building. Increased airborne contamination also resulted from releases from the waste gas compressors located across the hall from the detector. (Direct exposure from the makeup tank behind a 1-m-thick reinforced concrete wall also contributed to the reading.)

There was increased pressure in the makeup tank as the result of offgassing the letdown water it received; this in turn interfered with reactor coolant system letdown operations. In addition, small leaks in the vent header that were of only minor concern before the accident became a direct pathway for release of fission product gases to the auxiliary building. Surveys on March 28, 1979, showed that the dose rates in the auxiliary building basement (El. 282') were 35–40 mR/h, while the dose rates on the grade floor (El. 305') were 300–400 mR/h. This indicates that the major contributor to the area dose rates was airborne contamination. The gaseous releases forced all personnel activities in the auxiliary building to be performed wearing self-contained air packs, which greatly reduced efficiency and increased fatigue.

The radioactive noble gases were unaffected by the plant ventilation system filters and were released to the environment. Based on data then available, approximately 2.5 million curies of noble gases and 7.5 curies of iodine-131 were released during the first week of the accident (Rogovin 1980).

3.3.1 *Releasing Waste Gas Decay Tanks*

The effects of the leaks into the auxiliary building were so pervasive that the sources were not easy to identify. Every time the waste gas compressors started, the airborne contamination increased. The most likely source appeared to be the waste gas decay tanks—the explanation was believed to be that operation of the compressors was opening the tank relief valves.

To reduce tank pressure, the gas was released to the containment. This proved to be acceptable after analysis

[3]The makeup vent valve was opened at approximately 07:10 on the morning of March 30, 1979, and left open for two hours. A 1200 mR/h beta-gamma exposure rate was measured from a helicopter 40 meters above the Unit 2 containment approximately 50 minutes after opening the makeup tank vent valve. This measurement and the ensuing confusion about its meaning were major factors in convincing Governor Thornburgh to issue his limited evacuation advisory for pregnant women and children.

showed that the amount of hydrogen in the waste gas decay tank, which was approximately 50% hydrogen, when mixed with the 56,000 m^3 of air in the containment, would remain well below the 4% operating limit for flammability. To conduct this operation, a temporary hose was rigged to a containment penetration instrument tubing to provide a path for venting the tanks. For five days, the contents of the waste gas decay tanks were transferred to the containment. The pressure in the tanks was reduced considerably, but the effort did not reduce the dose rates in the auxiliary building.

3.3.2 *Waste Gas Leakage*

After the contents of the waste gas decay tanks were vented to the containment, the dose rates and airborne contamination were expected to decrease to near normal. In fact, neither condition improved noticeably after the tanks were emptied. The gas releases continued at irregular intervals. The releases normally occurred in the form of a sudden increase (spike) in the gas activity measured by the auxiliary building monitors. The peak activity typically occurred about 30 minutes after the onset and then decayed back to equilibrium in one to six hours.

In late April 1979, a "Find-the-Leak" task force was formed. Its leak detection method was to isolate sections of the waste gas decay system. Radiation measurements were then taken to determine if there was any measurable effect on airborne contamination. When the leaking portion of the system was identified, more precise detection techniques were used to locate the exact leak locations.

Spikes in the airborne activity that apparently correlated with large changes in the water level in the makeup tank caused suspicion to center on either a leaky vent valve or relief valve on the makeup tank. A controlled test involving changing the water level in the tank showed poor correlation between the level changes and radioactive gas releases.

The task force then focused on the compressors. Finally, the leaks were traced to the "A" compressor, which had a hole in its casing—a result of operating without cooling water flow. The compressor was removed, repaired, and reinstalled, and the major gas leak was eliminated.

There were other gas leaks in the system. Several of the diaphragm valves in the waste gas system had ruptured or torn diaphragms. Radioactive gas leaked through the valve stems of these valves directly to the building. The reactor bleed holdup tank valves and/or fittings also leaked.

In addition to the direct gas leaks, radioactive iodine contamination was occurring. Two months after the accident, a test was performed to determine the direction of air flow in the building. This test showed that the air was migrating upward from El. 305' of the fuel handling building and then into the auxiliary building through the doorways and wall penetrations. This meant that the source of the iodine contamination was on the ground level of the fuel handling building (Graber 1979).

The largest source of iodine was identified in the makeup valve gallery in the fuel handling building, where some of the valves had leaked. The floor was covered with deep deposits of boron crystals that resulted from the evaporation of reactor coolant. These crystals were contaminated with iodine and other fission products. This area was decontaminated to reduce the iodine source term, but by that time there was already enough airborne contamination in the AFHB to maintain the equilibrium concentration far too high for access without respirators. Not until a general decontamination effort was completed was the requirement for respirators in the auxiliary building removed.

3.3.3 *Condenser Air Extraction Filtration System*

TMI-2 used three mechanical vacuum pumps to maintain the vacuum in the main condenser. Two of these pumps were normally required for routine operation and the third was maintained in standby with automatic starting capability. The discharge of the vacuum pumps was routed to the auxiliary building stack downstream of the filtration system because the air removed from the PWR condenser was normally nonradioactive. The fact that TMI-2 used mechanical vacuum pumps rather than steam jet air ejectors made it relatively easy to maintain vacuum in the condenser and, later, in the entire secondary side of the plant because to do so did not require steam to drive air ejectors. As discussed in Section 3.2 on decay heat removal, the secondary plant, and, thus, the vacuum pumps, were operated until the plant went to loss-to-ambient cooling in January 1981.

Early in the accident, when releases from the AFHB were in excess of technical specifications, much attention was directed toward the effectiveness of the building air filtration system. However, there was concern that the releases might be coming from the main condenser because the condensate was known to be contaminated and the condenser air extraction system discharged downstream of the auxiliary building filtration system.

This concern led to a decision to install a system to filter the vacuum pump discharge. The system was designed, procured, installed, and tested within two weeks of the accident. The filtration unit was located in the turbine building basement just to the west of the "C" condensate booster pump. It was a 0.9-m^3/s packaged unit with a prefilter, a HEPA filter, a charcoal filter, and a HEPA filter. It was operated effectively until the condenser air extraction system was secured.

3.3.4 Charcoal Bed Filtration Systems

Work was conducted on a contingent method of dealing with the gas in the waste gas decay tanks. Rather than vent the contents into the containment, a design was initiated for a radioactive gas treatment system based on the use of large activated charcoal beds, similar to those used in boiling water reactors (BWRs) for offgas holdup. This task became moot when the contents of the waste gas decay tanks were successfully vented into the containment.

Engineers also worked on a variation of this concept that was a recycle system intended to purify the containment atmosphere by controlled purging of the system through a charcoal bed cleanup system, and then returning the processed gas to the containment. Compressors would pump gas through charcoal filters, which would absorb radioactive iodine, hold up noble gases to allow their decay, and trap their solid radioactive daughters. The gases would then be passed through particulate filters to remove charcoal dust before the gas was routed to the stack or back into the containment.

After the waste gas decay tanks were vented to containment, the urgency for this system diminished. Administratively, the NRC had decided that an environmental impact statement was required before any approved releases would be allowed. More importantly, the recovery staff had to contend with the need to process and store liquid wastes. These influences resulted in termination of the design effort for the charcoal beds. The concept was later used in an evaluation of the alternatives available to purge the containment.

3.3.5 AFHB Filtration System

Enhancing the existing AFHB filtration system became a priority in order to support decontamination and to control releases to the environment. The installed air handling system for TMI-2 consisted of separate push-pull heating and ventilating systems for each major building. Each system consisted of a supply air system and an exhaust air system, which provided once-through ventilation with no recirculation.

When the radiation detector upstream of the filtration units detected excessive radiation, the bypass damper was closed and the filtration unit damper was opened. If the radiation detector downstream of the filter detected excessive amounts of radiation, the supply system was isolated, which reduced the flow rate through the filtration system and caused the building to operate at subatmospheric pressure, ensuring all leaks were inward.

Because of a temporary alignment that had been in place for several months while pre-accident ventilation startup deficiencies were being resolved, all of the welding smoke, dust, paint fumes, and other construction-related pollutants had passed through the charcoal bed adsorbents and the HEPA filters. The result was that the charcoal adsorbents had been exposed to far more contaminants in one year than they normally would have been for many years of operation.

On April 4, 1979, the dose rates near the filter units in the auxiliary building were approximately 300 mR/h, dominated by the noble gases flowing through the units. When the releases in the auxiliary building decreased and natural decay had occurred, the dose rates near the units were reduced to 50–120 mR/h. This permitted access for replacing the charcoal cells in the filtration units—an essential step in reestablishing control of offsite releases.

Only 80 spare charcoal filter trays were available at the site. Within three weeks of the accident, 80 more charcoal trays had been purchased and 200 additional trays had been located in vendor warehouses. Actual work replacing the charcoal trays began two weeks after the accident. Twenty of the filter cells were replaced. Each cell had to be removed individually, bagged to prevent the spread of contamination, and placed in a waste disposal container. All of the work had to be performed in anti-contamination clothing and respirators in general area dose rates of 50–120 mR/h. Over the course of a week, the filters were replaced in the ventilation systems.

Many of the charcoal trays installed in April had been impregnated with potassium iodide and triethylenediamine to enhance iodine absorption. The performance of these charcoal filters was monitored closely and found to be decreasing in effectiveness. The charcoal filter trays were again changed in the autumn 1979 and lasted through the stabilization phase of the project.

3.3.6 *Auxiliary AFHB Air Filtration System*

To ensure that the releases of iodine were controlled, the project team decided to install a temporary backup system in parallel with the effort to replace the charcoal trays in the existing system.

Within a week of the accident, the criteria for the backup air filtration system were under development. Figure 3-5 shows the temporary AFHB filtration system as it was eventually tied in with the existing air filtration system. Functionally, the system duplicated the installed filter system with an air capacity of 57 m3/s, redundant fans, and a fire protection system. In addition, a heater/dryer, airborne radioactivity detectors, and radiation dose rate detectors not included in the installed plant filter system were needed. Installation was required within days.

The first challenge was to locate the equipment. Units of the size and capacity required were not available from suppliers on short notice and the only chance to meet the schedule was to find suitable units that had already been constructed for another nuclear power plant. An intensive search was begun of manufacturers and owners of plants under construction—eventually the units were located at the Washington Public Power Supply System site.

Alternative locations and tie-in points for the filtration system were studied. Installation inside the AFHB was impractical because of the size of the equipment and because the radiological conditions were unfavorable. The best location was the roof of the building, which was designed to withstand the impact from an airplane crash and had 1.2-m-thick reinforced concrete walls and roof (see Photo 3-2). The existing ventilation system exhausted through the auxiliary building roof. The tie-in point was above the roof where the combined ventilation exhaust ducts entered the station vent stack. The system was installed on the roof adjacent to the stack. All of the exhaust flow was forced through the new system by capping the station vent stack.

The system was used until after the containment was vented in the summer of 1980. After the plant vent was uncapped following the venting, this temporary system was no longer needed.

3.4 Electrical System Improvements

As described in Section 3.2, balance-of-plant (BOP) systems were first used to cool the reactor—these systems were powered from BOP electrical power sources. The lack of redundancy in the BOP power system meant that any interruption in electrical power would stop the systems being used to cool the plant and potentially force the use of the decay heat removal system, which was undesirable.

In addition to needing a reliable BOP power system, power was needed by the many new recovery systems. These new systems needed to be powered by a BOP electrical system because safety-related electrical systems were still required by the plant's license. The new electrical loads had to be added while the existing systems were being made more reliable.

An overview of the TMI-2 electrical distribution system is shown in Figure 3-6. Auxiliary electrical power was brought to the site by two 230-kV circuits from the Middletown Junction switchyard of the electrical grid. The Middletown Junction was located approximately 2.5 km from the plant and was a major substation in the Pennsylvania-New Jersey-Maryland (PJM) Interconnection. An autotransformer tie interconnects the 230-kV substation with the 500-kV substation from TMI-2's generator. An intertie to the TMI-1 generator was also provided.

On site, electrical power was stepped down in voltage to 6.9 kV for the reactor coolant pump motors and to 4.16 kV for all of the major power distribution buses in the plant. Four of these buses were safety-related and backed up by two redundant 3-MWe diesel generators. The rest of the 4.16-kV buses formed the BOP electrical system and were not backed up in case the electrical grid failed.

3.4.1 *Reliability of the External Grid*

The existing electrical system supplying TMI-2 was very reliable. It was interconnected with the PJM Interconnection, which was interconnected with other power pools.

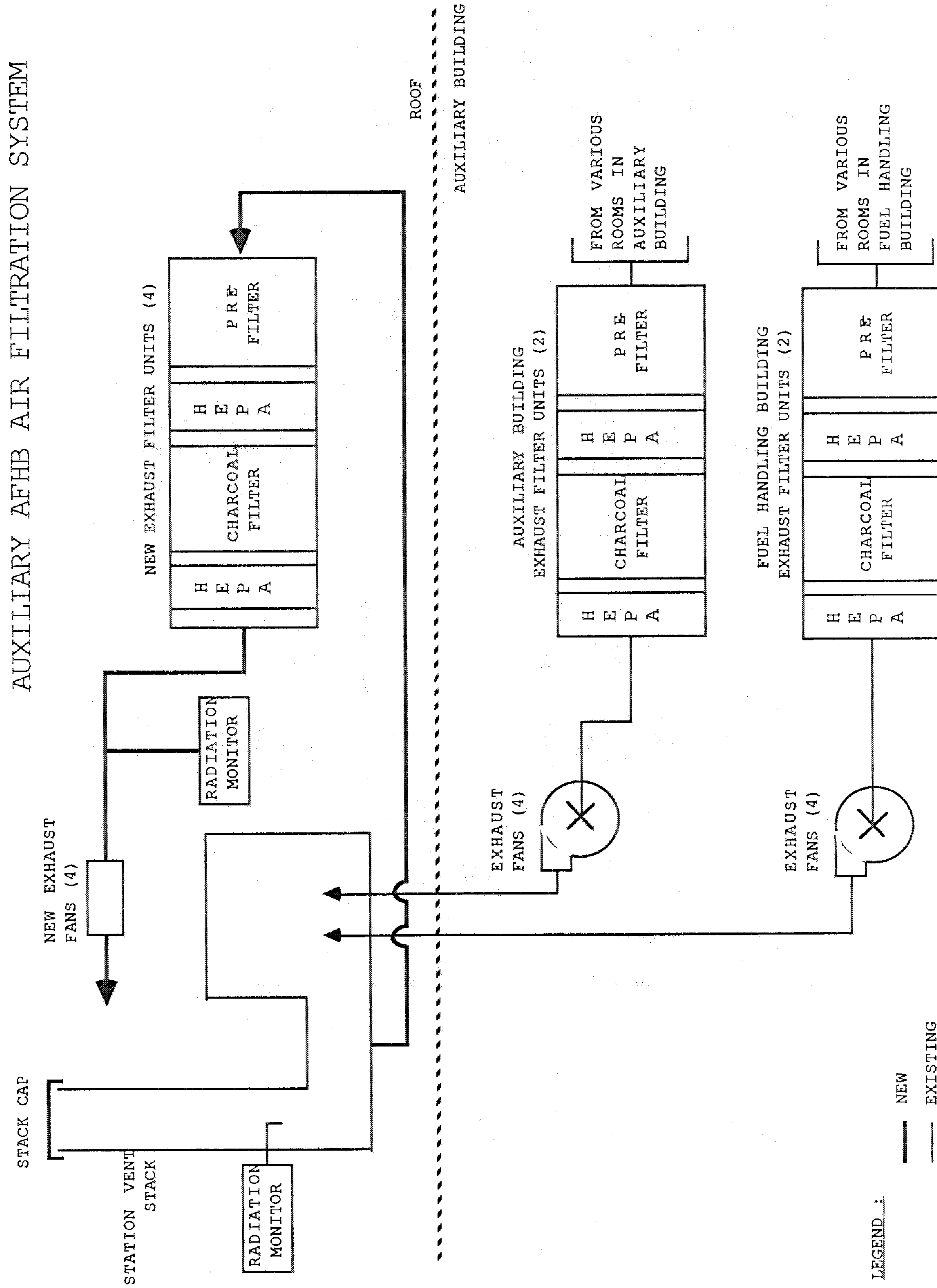

Figure 3-5. Auxiliary AFHB Air Filtration System

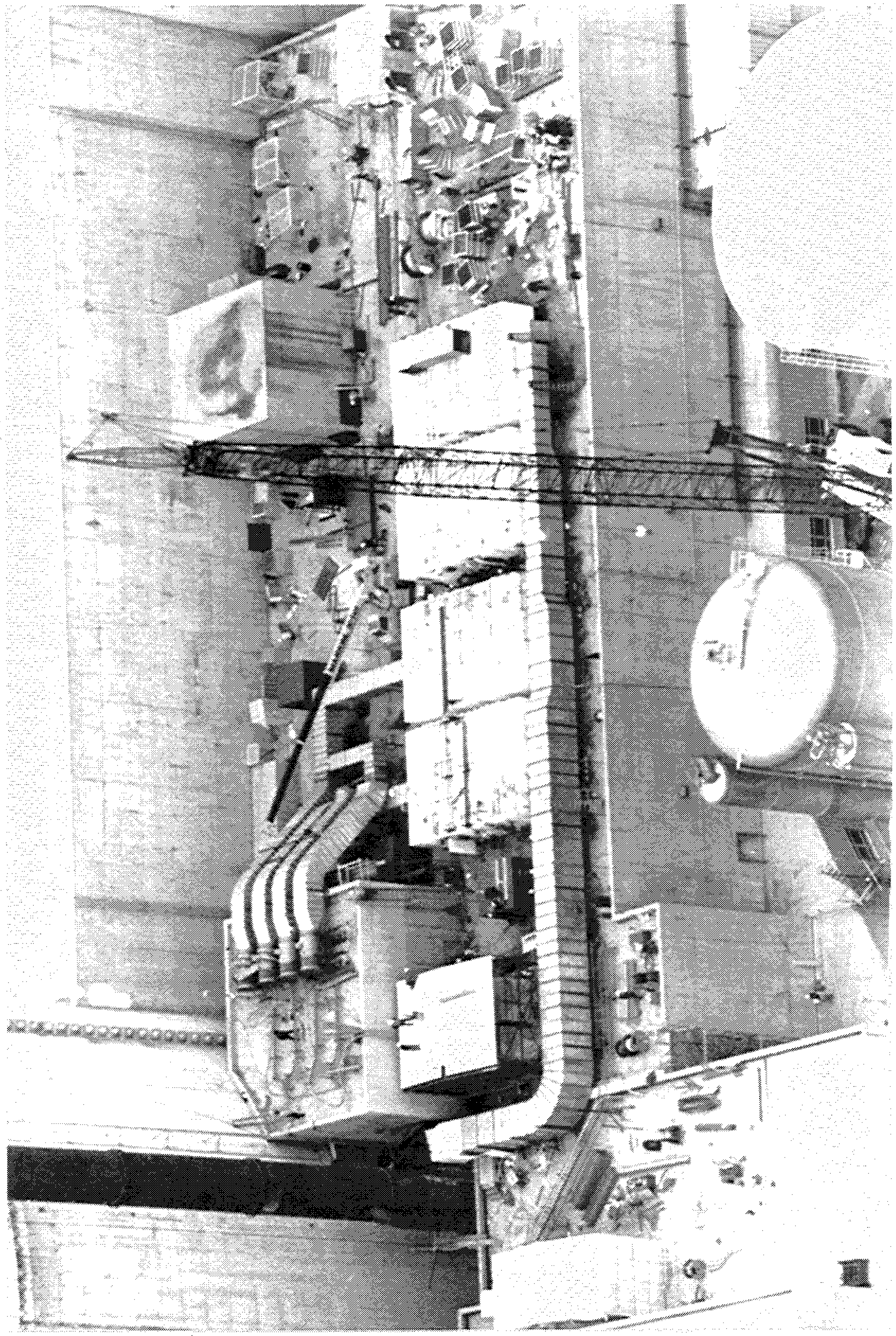

Photo 3-2. Installation of the Auxiliary AFHB Air Filtration System

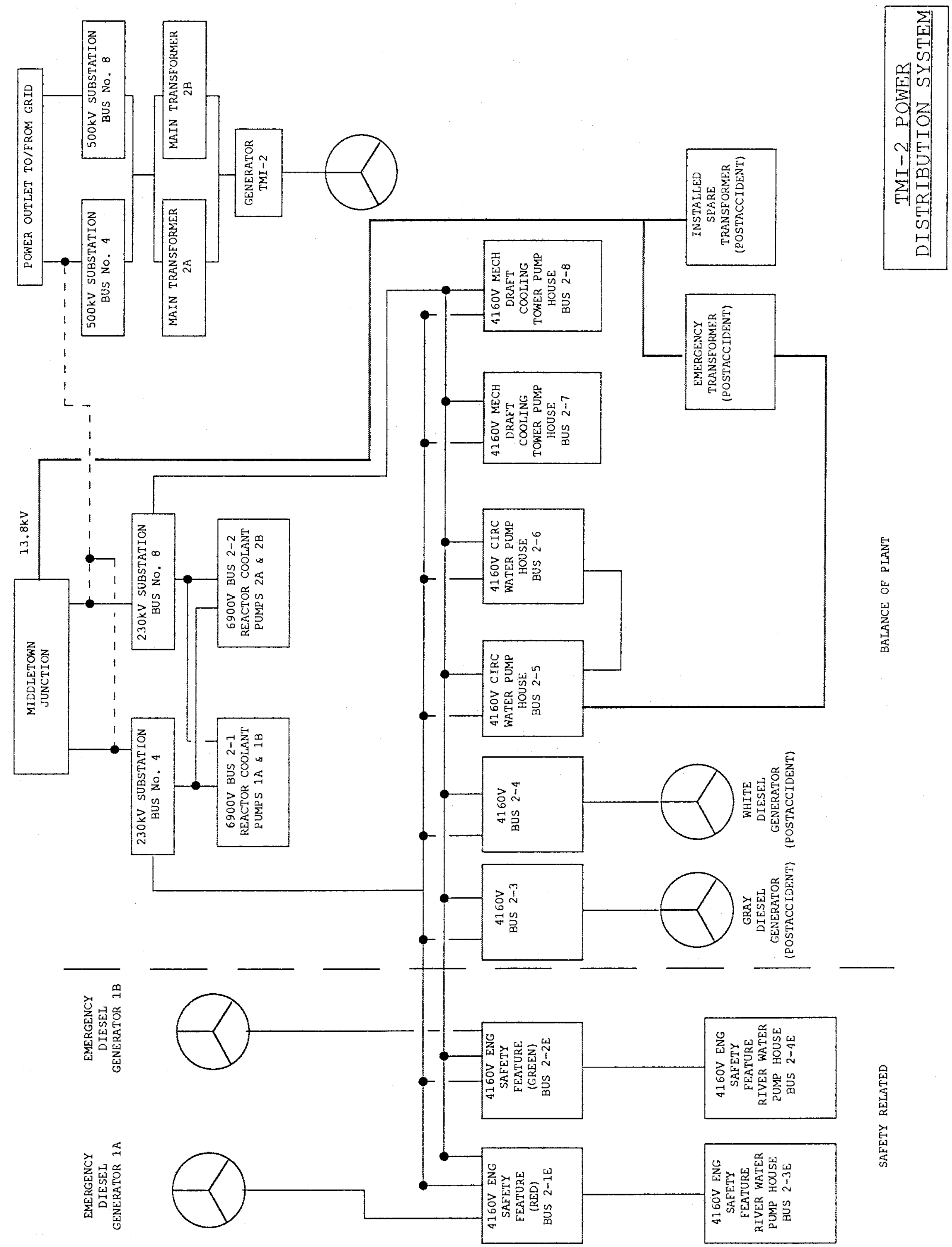

Figure 3-6. TMI-2 Power Distribution System

In order to improve the reliability of the electrical power during the cooldown period, Metropolitan Edison and the other power pool utilities undertook the following precautions:

- All routine maintenance work was curtailed on substations or lines in the area. This avoided the possibility of an inadvertent loss of electrical power to the plant.
- Operators were assigned to the Middletown Junction and at the TMI-2 substations to back up automated systems. The Middletown Junction and the TMI substations were normally controlled remotely from the Lebanon Dispatch Center. Soon after the accident, these substations were manned around the clock to correct any malfunctions and to reset any breakers that might trip.
- The TMI-2 substation and auxiliary transformers were barricaded to eliminate any vehicular threat.
- Daily random foot patrols were established to monitor the 230-kV circuits in the area around TMI to ensure that the proper clearances existed and to prevent/ detect sabotage.
- Combustion turbines in Metropolitan Edison's Western Division that were connected to the 115-kV network would be dedicated to the recovery and would be run if any problems occurred at the Brunner Island Plant (a 1,500-mW coal-fired plant owned by Pennsylvania Power & Light Co. located 5 km to the south of TMI-2). If it failed, the voltage drop in the area might cause TMI-2 to lose power. The Western Division combustion turbines would be run in the event of thunderstorms in the area.
- All Metropolitan Edison maintenance supervisors in the area were instructed to provide rapid maintenance in case of any equipment malfunction.

These precautions remained in effect until the crisis passed. In spite of the initial concerns, the electrical power to the site was not interrupted during the recovery. When the plant achieved cold shutdown, normal BOP electrical system and PJM power pool operations were reestablished.

3.4.2 *Balance-of-Plant Diesel Generators*

A study of methods to improve the reliability of the TMI-2 electrical supply systems determined that the source reliability of four 4.16-kV BOP buses needed to be increased. Two of the buses required 2,500 kW. An urgent search located two 2,500-kW diesel generators that could be delivered to the site in a few days; they arrived on April 9, 1979. Both generators were skid-mounted and equipped for outdoor service (see Photo 3-3).

One ("gray") diesel was originally intended for nuclear application at a BWR. It was equipped for remote, automatic fast starts and had a solid-state fast-response voltage regulator and other auxiliaries that made it ideal for the application.

The second ("white") diesel was designed for residential and light commercial applications and had exclusively manual controls. Extensive modifications were made to the starting system to allow remote starting and to enable it to automatically start and pick up load within two minutes. Because each diesel was required to be able to start the 520-kW condensate pump, a test was performed to demonstrate that the "white" diesel generator was adequate. This test was conducted because analyses were inconclusive.

The units were set up to start automatically upon loss of voltage and to close their own breakers after reaching speed and voltage. They would re-energize their respective buses that would be stripped of load upon loss of voltage. The diesels were to be manually loaded. This combination of automatic start and manual loading was chosen because it allowed the operators to quickly power whatever system they needed, but only when needed. Devising an automatic loading sequence would have been difficult because of the continuously changing conditions, addition and modification of systems, and the inability to predict a precise course of events should power be lost. Furthermore, because the core conditions were not changing rapidly, there was no need for automatic, instantaneous load sequencing.

Both diesels were installed and successfully test loaded within six weeks of the accident. The diesels remained on standby for approximately one year. There were no power interruptions at TMI-2 after the accident so they were never called into service. Eventually, they were sold and, in December 1987, shipped to a Brazilian power authority.

3.4.3 *Increasing Reliability of Offsite Power*

Concurrent with the backup diesel generators project, a search began for a way to improve the reliability of the other two busses, which powered the circulating water pumps and required 4,000 kW each. It was impractical to add two more diesels for several reasons:

- Diesel generator sets rated for 4,000 kW were usually special order items that required long lead times for delivery.
- They were also quite expensive and, if one could be found, extraordinary premiums would be needed to buy it.
- It would be difficult to find the space and the time to install two more diesel generator sets.

The only remaining alternative was to install another independent circuit to the site to power these buses. A 115/13.8-kV transformer at the Middletown Junction Switchyard was selected and isolated from all other connections except the 115-kV bus. The 115-kV bus was backed up by the Western Division combustion turbines and could supply TMI-2 with electrical power approximately 20 minutes after a complete station blackout.

Two 13.8/4.16-kV, 10-megaVolt-ampere-transformers were brought to the site and installed adjacent to the circulating water pump house, which was a short distance from the circulating water pump buses. The transformers were energized and connected to the bus on May 7, 1979. The electrical transformers remained on standby for approximately one year. Since no power interruptions occurred after the accident, they were never called into service.

3.5 Miscellaneous Support Systems

In addition to the operations discussed above, there were several others that were of lesser importance, but nevertheless needed.

3.5.1 Sample Sink

The nuclear sampling system at TMI was a shared sample sink located in TMI-1. Reactor coolant samples after the accident measured over 10,000 µCi/ml, and as a result the dose rates near the 1.3-cm sample lines were quite high (1 to 5 R/h). These sample lines were routed in open areas and were causing elevated area dose rates in the TMI-2 AFHB and in the TMI-1 auxiliary building, and raising concerns about the spread of contamination. To reduce these dose rates and to prevent the shared sample sink from affecting the operation of TMI-1, a temporary nuclear sample sink was installed for the TMI-2 postaccident samples. The new sink provided for recirculating and sampling water from points that included the reactor coolant system and tanks that received water from the system and from the containment basement. The sink was also the means for monitoring the reactor coolant boron concentration. It was located in the fuel handling building at grade level because all of the sample lines from TMI-2 passed through a room adjacent to that location. This permitted all of the new sample lines to be short and direct. All tie-ins were made in the same general area.

Installation of the system began in May 1979; it was operational by September 1979. In 1984, all of the sample lines from TMI-2 were routed to this temporary sample sink, converting it to the permanent sample sink.

3.5.2 Hot Chemistry Lab/Sample Data Management

The amount of radioactivity in samples of the reactor coolant and the water that was accumulating in the containment basement was so much higher than usual that normal analytical methods, equipment, and procedures were inadequate. The hot chemistry laboratory for the two TMI plants was located in TMI-1 and samples taken from TMI-2 made the laboratory effectively inoperable and, sometimes, uninhabitable.

In May 1979, a laboratory design was proposed that was very comprehensive in terms of analyses that could be conducted. However, the proposed system was much too complex and expensive to be built in the required time. Therefore, a more rudimentary chemical laboratory, built inside a truck trailer, was modified for nuclear applications. It was placed adjacent to the turbine building and was initially used to package samples for offsite analysis.

This trailer was sufficient for the urgent need for a nuclear chemistry laboratory but was inadequate for detailed analyses. In October 1979, a gamma spectrometer trailer was procured. Both trailers were tied together with the air filtration system and with the necessary electrical and utility connections. This laboratory began operation in December 1979, and operated throughout the stabilization period.

Photo 3-3. Gray and White Backup Diesels

These two trailers were subsequently replaced by a more sophisticated trailer-mounted system provided through the DOE and placed inside the fuel handling building. This facility, known as MERL (mobile emergency radiochemical laboratory) or MRL (mobile radiochemistry laboratory), provided an onsite analytical capability to support the recovery effort. It was used to conduct much of the initial research on many highly radioactive samples, including makeup demineralizer resins, control rod drive leadscrew surface scrapings, and containment sump water samples. Its arrival enabled the other two trailers to be used for other analyses. The faster turnaround with MRL was very important for minimizing delays in cleanup operations that required radiochemistry analyses for planning or execution. A more complete description of sample gathering and analysis following the accident is contained in *The TMI-2 Data Acquisition and Analysis Experience* (Urland and Babel 1990).

3.5.3 Temporary Auxiliary Boiler System

The only method of cooling the reactor relied on the main condenser (see Section 3.2.2). As the decay heat load diminished with time, the pressure in the entire secondary system was lowered below atmospheric in order to continue steaming. Operating at this pressure required gland sealing steam to be maintained for the turbine shaft seals to avoid air leaking in. This was accomplished by using the auxiliary boiler located to the east of the TMI-1 auxiliary building.

Using this boiler eventually became a concern because of water usage and the policy of separating the two units. Therefore, a packaged, oil-fired boiler was purchased and installed in the yard near the containment. It was available for operation in October 1979, and operated through the winter of 1979–80. The auxiliary boiler became unnecessary when loss-to-ambient cooling showed that gland sealing steam was no longer required.

3.5.4 Winterization

In the summer of 1979, the project team realized that the supplemental air filtration system built on the roof of the auxiliary building would be required to operate well past its original end-of-life date of September 1979. As the system was not designed for outdoor service during the winter months, it had to be "winterized" to prevent condensation and possible freezing.

A heated sheet metal building was erected on top of the auxiliary building roof to enclose the fans and filter units. Heaters were provided to keep the structure warm. Electric heat tracing was added to exposed lines that were not routed inside the new structure.

Several other newly installed facilities were recognized as inadequate for winter operations. EPICOR II (Section 6.2), the temporary radwaste staging area (Section 3.6.2.3), and the valve pit for the ADHR system (Section 3.2.2), among others, were modified to include insulation and either heat tracing or space heaters in order to continue operating throughout the winter.

A 1979 analysis of the heat loads in the containment indicated that under certain winter conditions, freezing temperatures could be reached inside the building. This was a concern because power had not yet been restored to the containment. The closed cooling water system used for the containment coolers had a small outside evaporative cooling tower that happened to have electrical heaters in the basin to prevent basin freezing. To heat the containment, this tower was enclosed in polyethylene and the water circulated in the basin with the heaters turned on. This proved to be adequate for keeping the containment from freezing during the winter.

3.5.5 Groundwater Monitoring Program

In July 1979, when it became obvious that the water in the containment basement would be present for a long time, the NRC requested that a groundwater monitoring program be considered to provide assurance that any leaks would be detected. The project team was skeptical of the need because of the design and construction of the containment (with its high quality liner), although the auxiliary building was a potential source.

To ensure detection, the Groundwater Monitoring Program was implemented in the spring of 1980. Eight monitoring wells were dug in April—seven around the periphery of TMI-2 and one in the north parking lot, several hundred feet away. Each well was approximately 9 m deep and contained a submerged electric well pump to draw samples. Weekly samples were drawn from each well and analyzed.

The samples from these wells had perplexing anomalies in tritium concentrations. The concentrations varied widely with no apparent correlation to the accident or to water in the containment. Seven additional observation wells were sunk in May 1980—six near TMI-2 and one in the south parking lot. Beginning in May 1980, samples from the 15 wells and a test pond were taken weekly.

Evaluation of the samples identified a source of radioactivity coming from the borated water storage tank to the east of the auxiliary building. The tank had slightly contaminated water in it because it had been filled from the TMI-1 fuel pool after the accident as an emergency procedure. Since 1979, water leaking from this tank and associated equipment had been periodically identified and the sources eliminated. Soil samples near the tank confirmed the presence of small amounts of cesium-137 and cobalt-60.

Such findings reconfirmed to the project team the need to remove the water from the containment basement so as to eliminate the potential that the 2.3 million liters of contaminated water would be viewed as a reservoir for contaminating the soil. Although many minor sources of radioactivity were identified by the Groundwater Monitoring Program, no leakage was ever detected from the containment. The wells were used throughout the cleanup to monitor site conditions.

3.6 Decontamination and Waste Management

In parallel with controlling the reactor and the release of radioactive gases, the cleanup team had to pursue a vigorous campaign to mitigate the effects of widespread surface contamination and to regain control of vital plant areas. The overall campaigns of radioactive waste management and plant decontamination are described in sections 6 and 7, respectively. The following subsections provide a sketch of the actions taken immediately after the accident.

In terms of decision-making, the decontamination actions were driven by very specific and short-term goals: support the construction of new systems, maintain access to important equipment, and establish and expand areas where protective gear was not required. The resulting campaigns cleaned the auxiliary building to the point where it could serve as the base for entering the containment.

3.6.1 Auxiliary Building Decontamination

A combination of events extensively contaminated surfaces in the auxiliary building—the primary cause was the overflowing of the auxiliary building sump (see Section 3.6.2). The contamination was exacerbated by the ventilation system flowpaths. In addition, tanks and sumps were full and floor drains backed up. Many valves had small leaks that became major sources of contamination.

Within two weeks of the accident, limited decontamination efforts had begun, primarily to support construction of backup decay heat removal systems and to recover general use of the auxiliary building. By the end of April 1979, large quantities of construction materials had been brought into the building for all the temporary systems and modifications. This material included lead bricks, concrete blocks, scaffolding, waste drums and boxes, power and hand tools, compressors, welding machines, hoses, and other miscellaneous items. (These, too, became contaminated.).

All entries to the building required air breathing packs and double or triple anti-contamination clothing. All normal change areas within the building were contaminated and much of the contamination was beta-emitting fission products (see Section 4 on personnel protection practices).

The planning horizon for decontamination tasks was one to two weeks, with daily meetings to assess progress. The first objective was to gain easier access to permit plant operations and then to proceed to clean up. To accomplish this, three beachheads were established in the building:

- The basement of the diesel generator building, where access was required in case offsite power was lost
- A rollup door from grade-level in the auxiliary building, which provided access to the radwaste system control panel that was essential to operations
- The normal access point from the service building.

At the start of the effort, in order to make even the briefest of entries to the radwaste control panel, a time-consuming process of dressing in protective clothing was required. To eliminate this need, a ventilation-tight tunnel of herculite and scaffolding was constructed between the rollup door and the panel. This provided a major morale boost for the plant operators by eliminating the requirement to spend 30 minutes to an hour accomplishing what normally would be a five-minute task.

The overall building decontamination work began from the service building access at grade elevation. First, the top floor, which was the least contaminated, was decontaminated; then the middle or grade elevation floor; and finally the basement level, which was heavily contaminated. Open areas were decontaminated first, followed by cubicles, the most contaminated of which were bypassed. The general approach was to decontaminate from ceiling to floor, working on all flat and exposed surfaces including piping, lighting fixtures, cable trays, valves, ceilings, walls, and floors.

Most of the effort was by manual wiping. A variety of other methods and tools were eventually used, including wet vacuums, water lances, high-pressure water spray, strippable coating, floor scabbling, and steam/vacuum surface cleaners. Because the water storage capacity was limited, much of the initial decontamination was performed with minimal water—a handicap to which the work crews adjusted. Eventually, flushing was conducted in many areas, particularly where boric acid had crystallized around minor leaks.

By November 1979, after six months of decontamination, the cleanup had progressed to the point that respirators could generally be used instead of self-contained breathing apparatus. Approximately one year passed before general area entries could be made into the upper elevations of the auxiliary building without respirators.

The principal contaminating radionuclides in April 1979 were iodine-131 and cesium-137, and the general radiation level was 1 R/h, with contamination levels of over 1E+07 dpm/100 cm². Within two months, the general radiation level was between 2 and 12 mR/h and contamination levels were on the order of several hundred thousand disintegrations per minute. By September 1979, general radiation levels ranged from <1 to 3 mR/h and contamination levels were 2E+03 dpm/100 cm² or less. Much of this decline was due to the decay of the short-lived isotopes.

Progress can be characterized as two steps back for every three forward. There were approximately 100 recontaminations from a variety of sources such as spills, decontamination activities, and unknown causes. On one occasion, bleeding moisture from the air system into floor drains caused water to spurt out of floor drains in other cubicles. On another, the building sump backed up, recontaminating cubicles via the floor drains.

One of the most significant problems was the vagaries of the ventilation system. For example, when various combinations of doors were opened or closed, local flows would change and puffs of contamination would occur. This led to the institution of door controls, in which signs would be posted warning against opening certain doors while specific decontamination operations were being conducted. Within the first several months, such airborne recontaminations occurred five to ten times, leading to substantial recontamination. Consequently, much of the time, the ventilation system was operated with only the exhaust fans running.

3.6.2 Water Processing and Interim Waste Storage

The hundreds of thousands of liters of contaminated water in the auxiliary building were a major obstacle to cleanup. Getting it processed, discharged, and/or stored were major accomplishments and gave the project confidence that it could later deal with the far more highly contaminated water in the containment basement and reactor coolant system.

The existing liquid radioactive waste processing system at TMI was a system shared between the two units. Most of the processing equipment was located in TMI-1 and was used for both units. TMI-2's liquid radioactive waste system consisted primarily of collection, sampling, and chemical neutralization capabilities. The system's components were located in the auxiliary building

After the fuel cladding failure, the reactor coolant became extremely contaminated. Reactor coolant samples taken the day of the accident had an iodine-131 concentration of 13,000 μCi/ml and a gross cesium radionuclide concentration of 500 μCi/ml. During the accident, reactor coolant was discharged from the pressurizer relief valve to the reactor coolant drain tank (RCDT), which was located in the basement of the containment. The RCDT relief valve and later the RCDT rupture disk permitted this water to overflow into the containment sump.

Seven and a half minutes after the accident began, the first containment sump pump turned on because of the rising water level in the sump. Three minutes later, the second sump pump turned on. The containment sump pumps were aligned to pump water to the auxiliary building sump tank. This tank had a blown rupture disk that was awaiting repair. It also contained about 9,000 liters of pre-accident water, leaving only 3,000 liters of capacity available.

The containment sump pumps sent approximately 30,000 liters of water to the auxiliary building sump tank. The auxiliary building sump overfilled and the water backed up through the floor drains into the building basement. Both containment sump pumps were in operation for approximately 30 minutes before operators realized what was happening and turned them off, although not before some highly contaminated reactor coolant (i.e., containing recently released fission products) had also been transferred to the auxiliary building.

Consequently, most of the water in the auxiliary building soon became contaminated by the reactor coolant. The activity in the water varied from 1 to 100 μCi/ml, which aggravated the existing auxiliary building problems of airborne activity and surface contamination.

The day after the accident, the TMI-2 operators transferred pre-accident water (i.e., not contaminated with fission products) to TMI-1 to provide additional capacity. The installed auxiliary building tankage could only accommodate approximately 190,000 liters of excess water. In-leakage to the auxiliary building was obviously going to exceed this capacity.

Not only was the water storage capacity inadequate for the new demands, the existing radwaste water treatment system was inadequate for processing the highly contaminated water. Consequently, over the next few months, new processing systems were needed:

- A portable ion-exchange system (EPICOR I) was brought in to process less contaminated water (<1 μCi/ml)—commencing operation in April 1979.
- A second ion-exchange system (EPICOR II) was designed, installed, tested, and placed in operation to process auxiliary building water of intermediate contamination (1–100 μCi/ml)—commencing operation in October 1979.
- Intensive design work began on a third system (submerged demineralizer system) to process highly contaminated water (>100 μCi/ml) in the containment basement and reactor coolant system—commencing operation in September 1981.
- Intensive design work began on an evaporation/solidification facility to process highly contaminated water—project cancelled.

Projects were also begun to store and dispose of the by-products resulting from operation of these water processing systems and other decontamination activities:

- New water storage tanks were installed.
- Temporary storage modules for expended water processing vessels were built.
- Temporary storage space for contaminated clothing and tools was created.

The immediate and temporary solutions found by the project team in the months after the accident are described below. The systems, equipment, and buildings that became an intrinsic part of the 11-year cleanup are covered in Section 6.

3.6.2.1 EPICOR I

EPICOR I carried the plant through the initial response stage until the more powerful and flexible EPICOR II went into operation. EPICOR I was a portable, temporary, lower activity level radioactive water processing system that had been operated at the TMI site before the accident. (The system was also referred to as Cap-Gun.) Earlier in 1979, during the TMI-1 outage, EPICOR I had been used to process water in the TMI-1 auxiliary building sump in support of sediment removal activities. By March, it had been removed from the site. The day after the accident, it was returned.

Although its water processing abilities were limited to water containing less than 1 μCi/ml, the importance of EPICOR I stemmed from several other aspects:

- It was portable, so it could be quickly and easily transported to the site.
- The vendor operating the system was familiar with the site and so could quickly integrate into the recovery work.
- Its operation (including taking samples) by subcontractors freed cleanup workers to support other TMI-2 emergency response activities.
- It processed and released pre-accident water, and so freed up storage space.
- It supplanted the existing TMI-2 evaporator, which was unusable because of high radiation levels.

The system was installed outdoors at grade level on the reinforced concrete roof of the Unit 1 decay heat vault. It began processing water immediately. The basic system is depicted in Figure 3-7, which shows the flow paths of water from a source, through a 1.2- by 1.2-m prefilter and 3.6-m^3 mixed bed demineralizer, and then through a postfilter into one of two 75,000-L Haliburton monitor tanks. The disposable ion exchange vessels were modified radwaste transportation vessels filled with custom-blended ion exchange resins.

The pumps were pneumatic-diaphragm, which became standard at TMI-2 because of their ability to readily pump slurries and water-bearing solids. The water could be either recirculated if it did not meet specifications

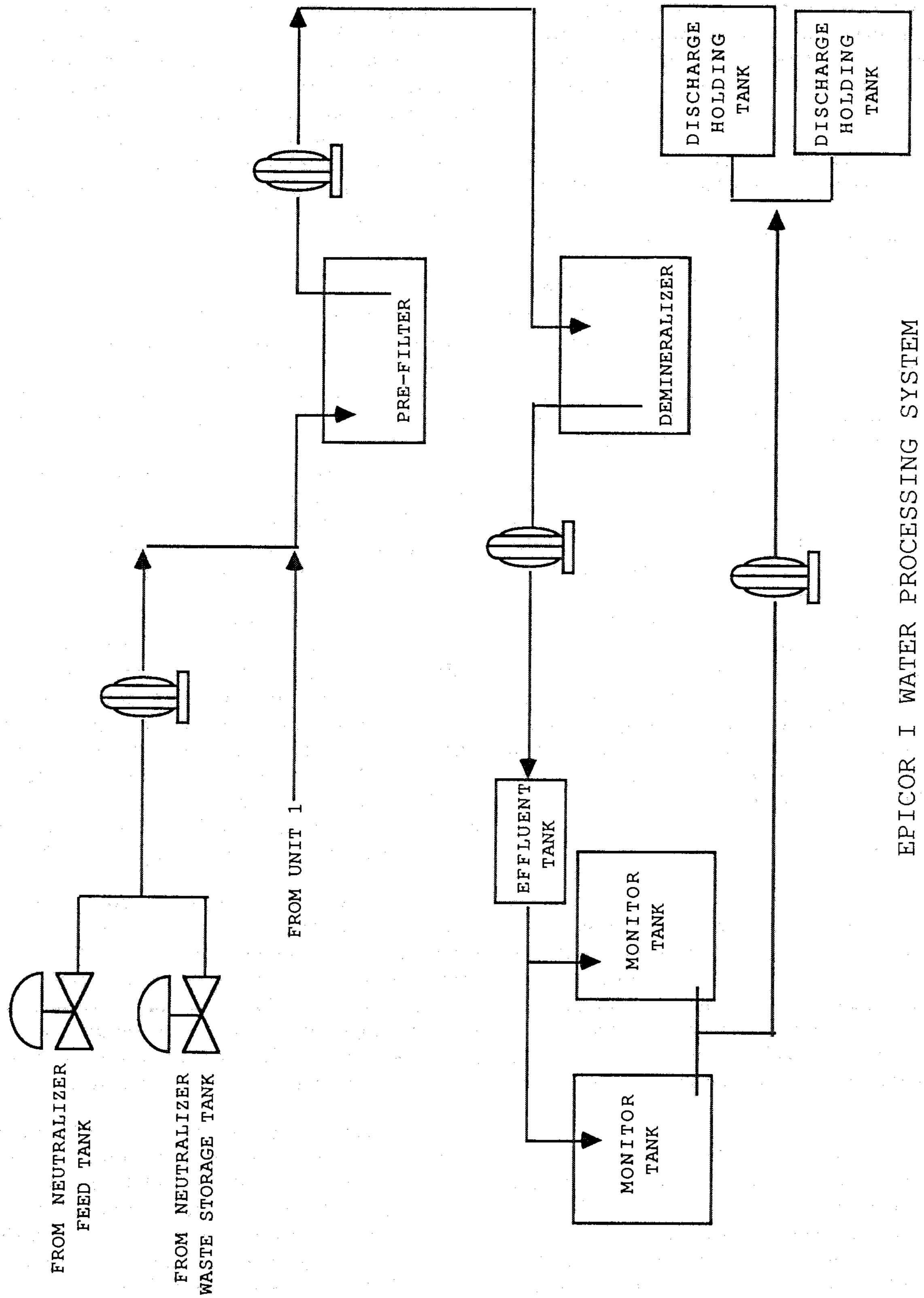

Figure 3-7. EPICOR I Water Processing System

for purification, or sent to discharge holding tanks for reuse, retention, or discharge to the river through Unit 1. The system was eventually encased in a temporary cover and, although consideration was given to moving it, remained atop the decay heat vault until its removal from the site.

After first passing through the system, the water went into one of two receiving tanks and was sampled. If the water had not been decontaminated sufficiently to permit release in one treatment cycle, then the filter and mixed bed demineralizers were changed and a second decontamination run was made. After the second pass, the water was routed to the second receiving tank. All batches of water treated by this system required two passes to meet river disposal specifications in the initial days after the accident. The first successfully processed batch was released to the river on April 11, 1979. By June 6, a total of 344,000 liters had been treated and released.

EPICOR I performed well as an immediate and temporary solution to accident conditions. The system was used to process over 5 million liters of lower activity level water from both Units 1 and 2. Thirty-eight processing vessels were generated for burial as low-level radioactive waste. EPICOR I was disconnected from Unit 2 in January 1981, when the pre-accident water had been processed and the cleanup organization had stabilized; i.e., TMI-1 personnel were no longer required to support TMI-2 work. The TMI-2 staff was then able to store or discharge whatever nonaccident-related water existed while processing all accident-related water through either EPICOR II or, later, the submerged demineralizer system.

3.6.2.2 Water Storage

With the rapid accumulation of accident-generated water, an immediate need developed to add storage tank capacity. The day after the accident, a search was started throughout the industry for available prefabricated tanks. The perceived need for temporary water storage was so great that for a short while, the project team actively pursued a plan to locate and obtain railroad tank cars to provide emergency capacity. Two tank cars were purchased, although they were never actually used for TMI-2 water.

The first location considered for placing temporary tanks was on top of the auxiliary building roof. It had the advantage of being out of the way and easy to shield. It also provided a direct route for tapping into the existing radwaste system. This approach was abandoned when the temporary AFHB air filtration system took precedence for the use of the roof.

The next location considered for the new tanks was the "A" fuel pool. Since TMI-2 had only been in operation for three months, the spent fuel pools contained no fuel and no water at the time of the accident. The "A" fuel pool was the larger of two fuel pools, was closest to the containment, and contained all of the fuel transfer mechanisms as well as empty fuel racks. It was also a large space in a safety-related building that had surfaces amenable to decontamination and was out of the way of ongoing recovery tasks. Lastly, it had a large-capacity (100-mt) overhead crane that could be used for construction.

Work on the fuel pool liquid waste storage facility—known as the "tank farm"—started in early April 1979. The initial urgent schedule targeted the tanks to be installed and operational within one week. A survey of the multitude of tanks that had been sent to the site identified four 57,000-L tanks and two 95,000-L tanks that were suitable for storing radioactive wastes. These tanks represented the maximum amount that could be physically placed in the fuel pool. The tanks were sent back to the manufacturer for necessary hydrostatic testing; they were returned in less than 24 hours.

The one-week design and construction schedule quickly proved to be unachievable. In addition to design changes and procurement problems, worker efficiency in the fuel handling building was severely affected by radiological protection procedures, resulting in only a few welds each day. Fortunately, the urgency for the tank farm was diminished by the general reduction in leakage rates. The tank farm began accepting radioactive waste water in July 1979, and eventually held 350,000 liters of auxiliary building water pending processing by EPICOR II.

Figure 3-8 and Photo 3-4 show the arrangement of the tank farm. The tanks were connected by manifolds to form two separate storage volumes: a 190,000-L volume consisting of two 95,000-L tanks and a 227,000-L volume consisting of four 57,000-L tanks. Each level of tanks had separate supply and discharge headers and standpipes. Steam eductors provided the mechanism by which water was emptied from the tanks or recirculated.

Care was taken to spread out the load to the entire structure. The steel supporting the tanks could neither be welded to nor penetrate through the fuel pool liner. Also, supporting steel could not be rigidly attached to the existing building structures—this restriction eliminated the original plan of filling the pool with water for shielding, as the tanks would float.

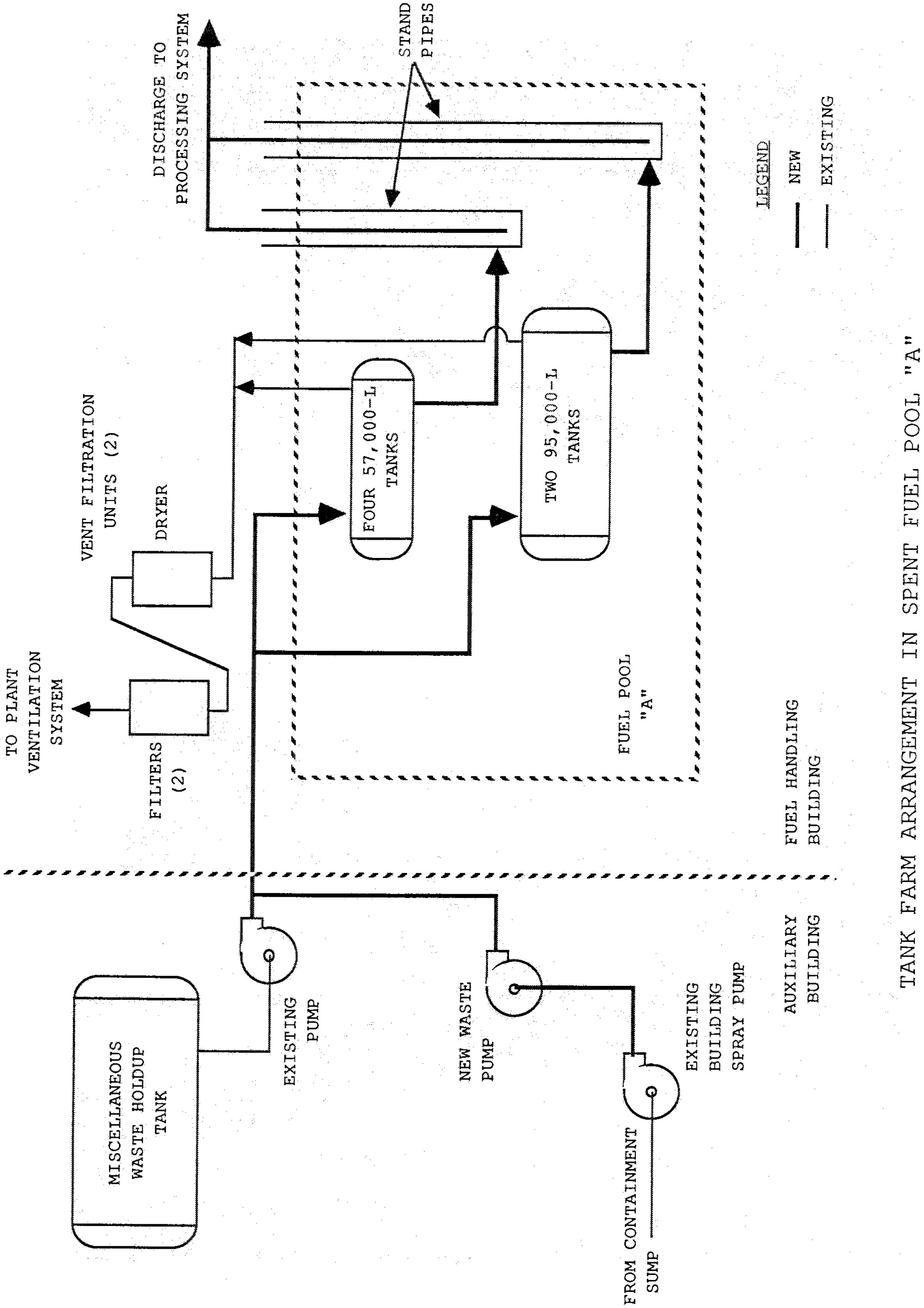

Figure 3-8. Tank Farm Arrangement in Spent Fuel Pool "A"

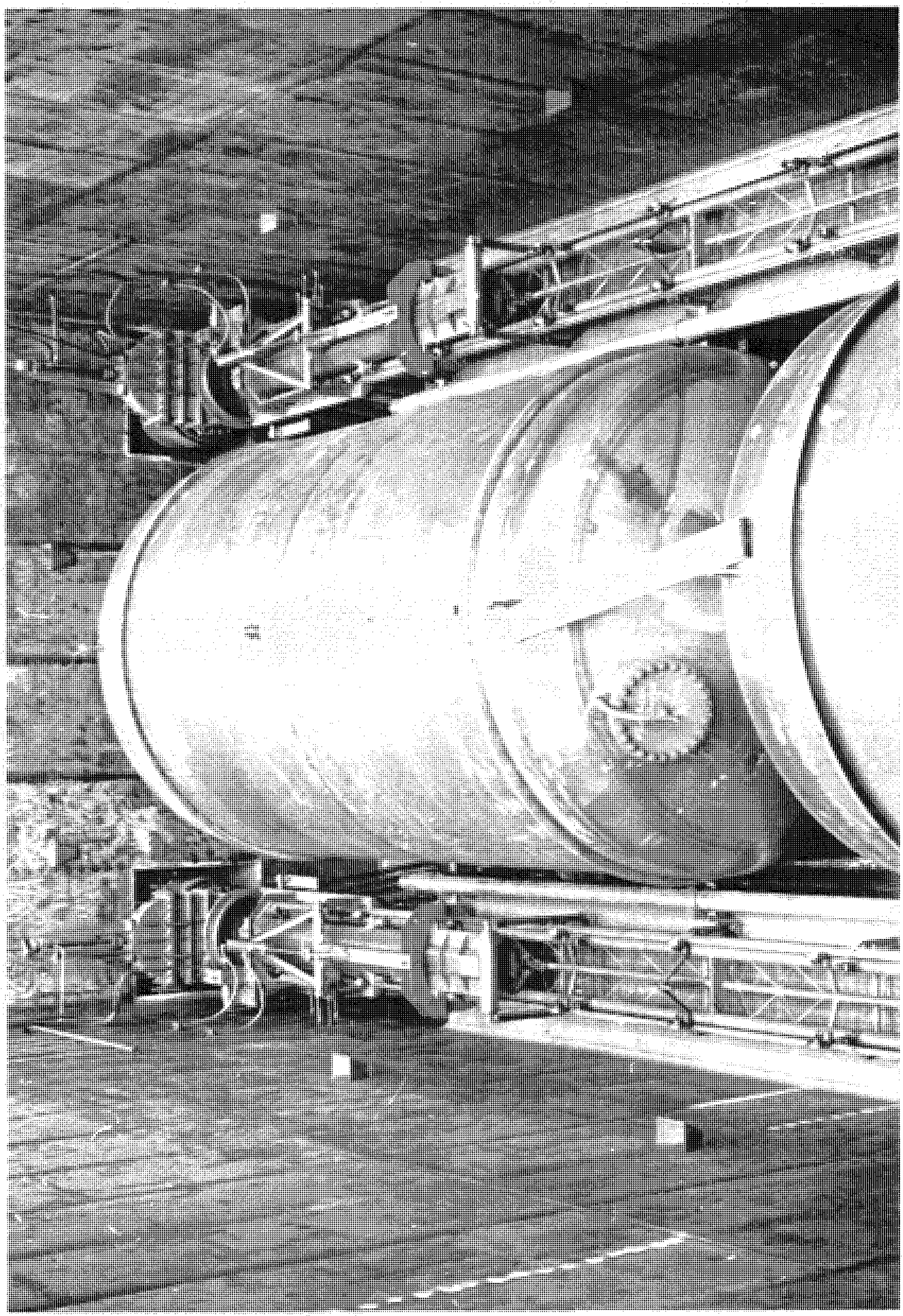

Photo 3-4. Tank Farm in Spent Fuel Pool "A"

An additional concern with filling the pool with water was that a leak into the shield water might render the fuel pool area uninhabitable. Therefore, the tanks were shielded with large, interlocked, 0.6-m-thick concrete slabs that spanned the pool width. These were supported above the pool on the same steel structure that supported the upper four tanks. The steel was capable of supporting a second layer of 0.6-m-thick slabs, but this proved unnecessary. The gap between these slabs and the top of the pool wall was shielded with 0.6-m-thick solid concrete block walls placed around the sides.

After EPICOR II processed the initial contaminated auxiliary building water, the tank farm was used to store miscellaneous liquid waste that was eventually processed through the submerged demineralizer system (SDS). The SDS started processing water from the containment in 1981, using the tank farm for feed staging. By that time, the water had been significantly reduced in radionuclide concentration by decay and the diluting effects of decontamination activities inside containment. The tank farm performed flawlessly as a part of the SDS.

The tank farm was removed in 1983 to make the "A" fuel pool available for staging of the core debris canisters. This removal required a substantial decontamination effort.

3.6.2.3 Temporary Radioactive Waste Storage

Two types of solid radioactive waste needed immediate attention: mildly contaminated trash and water processing concentrates in the form of spent resins, and several cartridge filters that were removed from water systems.

Much of this waste could not be shipped for disposal because of the adverse public and political pressure. Concern with the potential quantity and type of waste prompted the governors of South Carolina and Washington to ban TMI-2 waste at Barnwell and Richland. These states contained the two primary locations available for radioactive waste disposal, and so created the need for temporary onsite storage or staging. (See Section 6 for a more complete background on the national waste management scene and its effects on TMI-2). Consequently, several temporary storage areas were created.

The first project was conversion of a paint shed into an interim storage facility for drums and boxes containing low-specific-activity (LSA) wastes. This building was actually not a shed, having been used during construction for the safe storage of paints. All of the anti-contamination clothing and decontamination and contamination control material were stored in this interim facility until the wastes could be shipped. The building had a usable floor space of approximately 140 m2 . The yard around it was also used to stage contaminated equipment, which was otherwise covered. Its use was limited after completion of other staging areas.

Interim storage of vessels containing demineralizer resins used in water processing required a more substantial effort. While the EPICOR I vessels were not particularly radioactive and could be stored outside in open concrete culverts, the EPICOR II vessels were too radioactive. And the operation of EPICOR II was, in part, contingent on a rapid solution to the ability to store radioactive concentrates.

Two projects in 1979 addressed this type of waste.

- **A quick one**—A temporary radwaste staging area was located adjacent to the TMI-2 cooling towers in a relatively isolated area within the flood protection dike. The facility consisted of 14 large-diameter drainage culverts welded to a steel endplate. These culverts were placed vertically in the ground and the area was backfilled. A 1-m-thick concrete plug covered the top of each storage cell. This staging facility was taken out of service in 1980.
- **An engineered storage facility (solid waste staging facility or "waste acres")**—This facility consisted of two identical storage modules with a shared drainage sump. It went into service in January 1980 (see Section 6.4 for details).

3.7 Containment Venting and Initial Entry

Regaining worker access to the containment was the culmination of the stabilization phase. With access, the project team was at last in a position to evaluate fully the damage to the plant and to work directly on the systems and equipment that had been most affected. Access to the containment was limited by an atmosphere with levels of krypton-85 that were unacceptably high for any sustained occupation (1.04 μCi/cc shortly before venting). To proceed with the cleanup safely and quickly and to reduce the potential for unpredictable and uncontrollable leaks to the environment, the gas had to be removed.

Project management was prepared to enter the containment with or without krypton-85 venting, but preferred to reduce the hazard. (A attempted pre-venting entry was unsuccessful because of an equipment problem.). After a long process of preparation and review, the atmosphere was vented of approximately 46,000 curies of krypton-85 and the first entry was made in July 1980—almost 16 months after the accident.

In May 1979, the Containment Assessment Task Force was established to determine, from outside containment, its inside environment. With that information, the task force planned the initial entry into the building. The resulting efforts included air and water samples, radiation readings, gamma scans, video recording, determination of the availability of power and lighting, and then venting. Appendix I contains a more detailed description of the venting itself. Descriptions of the personnel preparations, monitors, training for entry, and the first entry are to be found in Section 4.2.2.

From the first request for permission to purge in November 1979, over seven months of intensive licensing and legal effort were required to obtain the NRC's approval to vent (approval to enter the containment was granted earlier). During this period, the NRC prepared an environmental assessment, the DOE established a citizens' monitoring program that recruited area residents for monitoring activities during the purge, and the public was involved in a number of ways.

The public process resulted in at least three alternative schemes proposed by a U.S. congressman and other organizations. One was selective absorption with charcoal, another was a scheme for a balloon-supported sleeve, and a third was for jet-assisted boosting of the vent effluent to higher elevations.

Also in the public arena was the attempt by S.C. Sholly and People Against Nuclear Energy (PANE) to block the purge using various legal challenges. Following a Court of Appeals ruling requiring the NRC to hold a hearing before issuing a license amendment, an appeal was filed with the U.S. Supreme Court—an action which stayed the so-called "Sholly Decision" of the Court of Appeals. At the same time, the NRC submitted proposed legislation to Congress clarifying the Atomic Energy Act so that the NRC could issue a license amendment without prior hearing if the issue constituted a "no significant hazards" consideration. After the legislation was enacted in January 1983 (Public Law 97-415), the Supreme Court vacated the Appeals Court ruling.

While the NRC staff was reviewing the purge request, another option was presented for ridding the containment of krypton-85. Lauded by the Commissioners, the Selective Absorption Process, under development at Oak Ridge National Laboratory (ORNL), was the only alternative seriously considered by the regulators. An independent technical evaluation of the process was conducted and stated that purging was preferred in all respects, including feasibility, effectiveness, practicality, health and safety, psychological stress on nearby population, schedule, and cost (SAI 1980).

TMI-2 project management noted the risks of an uncontrolled release of krypton-85 to the environment and an unacceptable increase in personnel exposures as their justification for the purge request. The NRC staff concurred.

After the purge was completed, the "Heidleberg Report" was issued, arguing that particulates released during the purge could lead to radiation exposures much higher than those possible from the krypton-85 if purging were not done. Much of the data supplied by the licensee in its original request were used by the NRC staff to refute this document (US NRC 1980).

The NRC Commissioners requested that the staff consider the use of a more rapid venting schedule to minimize the stress and other psychological impacts on the surrounding public. The project team had not requested an expedited schedule because this would have required a temporary change to the technical specifications on radioactive material releases. However, the NRC staff determined that this was in order, and issued a temporary technical specification change to allow venting to be performed as quickly as possible.

The order for Temporary Modification of License required that none of the following limits be exceeded for any of the 16 (22-1/2°) sectors centered on the TMI-2 containment: a) 15 mrem skin dose; b) 5 mrem total body dose; and c) 20% of the limits in (a) and (b) were not to be exceeded over a one-hour period. The revised technical specification allowed the use of real-time meteorological data to compute offsite doses. This permitted the project team to take advantage of optimum dispersal conditions by increasing the release rate when meteorological conditions allowed, and thus complete the venting more rapidly while still meeting the requirements used to compute estimated release limits in compliance with 10 CFR Parts 20 and 50, Appendix I.

Two existing systems were used for the purge.; they are depicted in Figure 3-9. The hydrogen control system, which was modified with a higher capacity fan, new controls, and interlocks, was used to vent at a rate up to

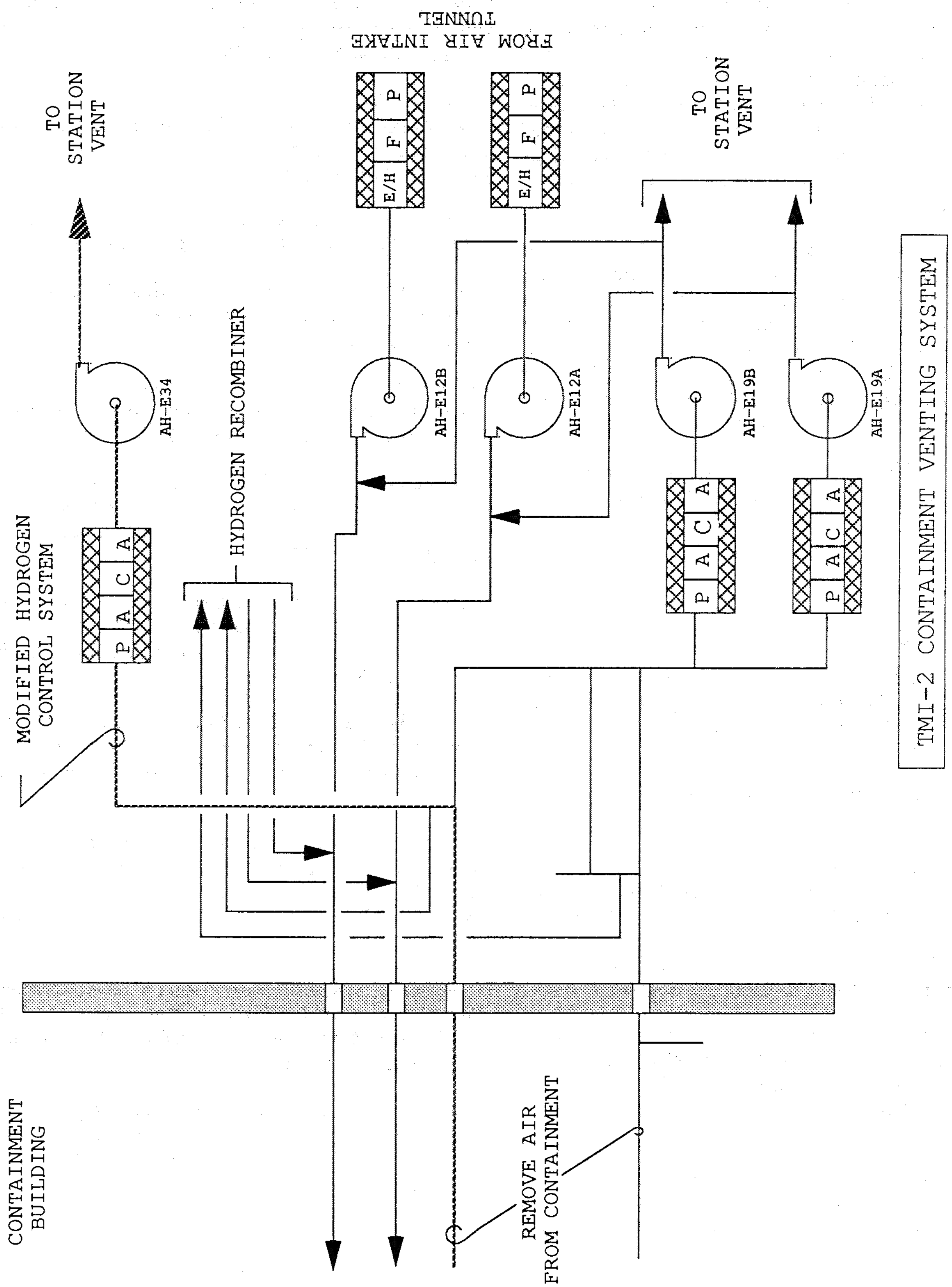

Figure 3-9. TMI-2 Containment Venting System

0.28 m^3/s, while the containment atmosphere was rich in krypton-85. The "B" train of the containment air purge and purification system was used for rates up to 8.73 m^3/s during later stages. The flow rate was controlled based on the offsite integrated dose criteria. All releases were through the station vent, which contained monitoring instrumentation. An extensive offsite network of monitors and samplers was established for the purge.

Slow rate purges were conducted over an 11-day period followed by four days of fast purging. The operation was accomplished without incident. During this period, the krypton-85 concentration within containment dropped from approximately 1 to approximately 6E-05 μCi/cc. (There were also a number of later purges of smaller magnitude to vent the krypton-85 subsequently released from the water in the basement.)

On July 23, 1980, two technicians, heavily laden with protective gear and instruments, made the first postaccident entry into the dark and dripping wet containment. More than 2000 entry days were to follow.

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4 PERSONNEL PROTECTION

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PERSONNEL PROTECTION

4.1 Overview

Now that a decade of intense, industry-wide efforts to improve worker protection practices has passed, it may well be difficult to picture the scene facing project managers immediately following the TMI-2 accident. The accident had released highly concentrated, mixed-fission products from the reactor core throughout the containment and into the auxiliary and fuel handling building (AFHB). The high-energy beta component was up to 100 times the gamma component (Rich, Alvarez, and Adams 1981), producing conditions in which beta-particle (skin) dose rather than gamma (whole-body) dose was often the limiting parameter in many areas of the plant.

On the one hand, it was necessary to regain control of the AFHB and to enter the containment and begin collecting data for recovery planning. On the other hand, it was crucial that no one receive unnecessary radiation exposure in the process. There were no precedents as the project team embarked on a course that would help redefine radiation protection planning and program management practices.

Seven major hurdles related to personnel protection faced the recovery team early on:

1. Reassessing the GPU radiation protection organization
2. Controlling the dose from high-energy beta radiation
3. Providing for ALARA/dose reduction in hostile radiological environments
4. Responding to unique respiratory protection needs
5. Controlling heat stress
6. Applying robotics to achieve ALARA/dose reduction objectives
7. Overcoming the negative impacts of the media on worker attitudes (Hildebrand 1985c).

Of these, the technical hurdles were significantly easier to surmount than those associated with workers' attitudes. This was particularly true for workers' complacency over time regarding very high beta exposures and for workers' fears that some management decisions threatened worker safety and, to a greater degree, threatened the safety of their families; e.g., when management decided that respirators were not always required in containment (Hildebrand 1989a).

As this section will show, the TMI-2 cleanup program was very successful in ensuring worker safety. The cumulative occupational dose associated with the cleanup was less than 6500 person-rem. This falls comfortably within the original (1981) NRC Programmatic Environmental Impact Statement (PEIS) estimates of 2000–8000 person-rem, and well below the 1984 revised PEIS estimates of 13,000–46,000 person-rem (US NRC 1981). The annual collective doses at TMI-2 were comparable to those at most operating nuclear power plants during the same 10-year period (see Figure 4-1).

4.1.1 Background

The Health Physics (HP) profession knew, more than a decade before the TMI-2 accident, that state-of-the-art beta dosimetry likely would not be adequate for plant survey or personnel monitoring requirements associated with a loss-of-coolant accident (LOCA) recovery (TMI-2 TI&EP 1981). The state-of-the-art in personnel monitoring amply addressed the needs of operating plants at the time, but no provisions for a recovery/cleanup program of the magnitude of TMI-2 had been made. The regulators and the nuclear power plant industry had focused on actual rather than postulated needs, and health physics practice was considered adequate for safe nuclear power plant operations.

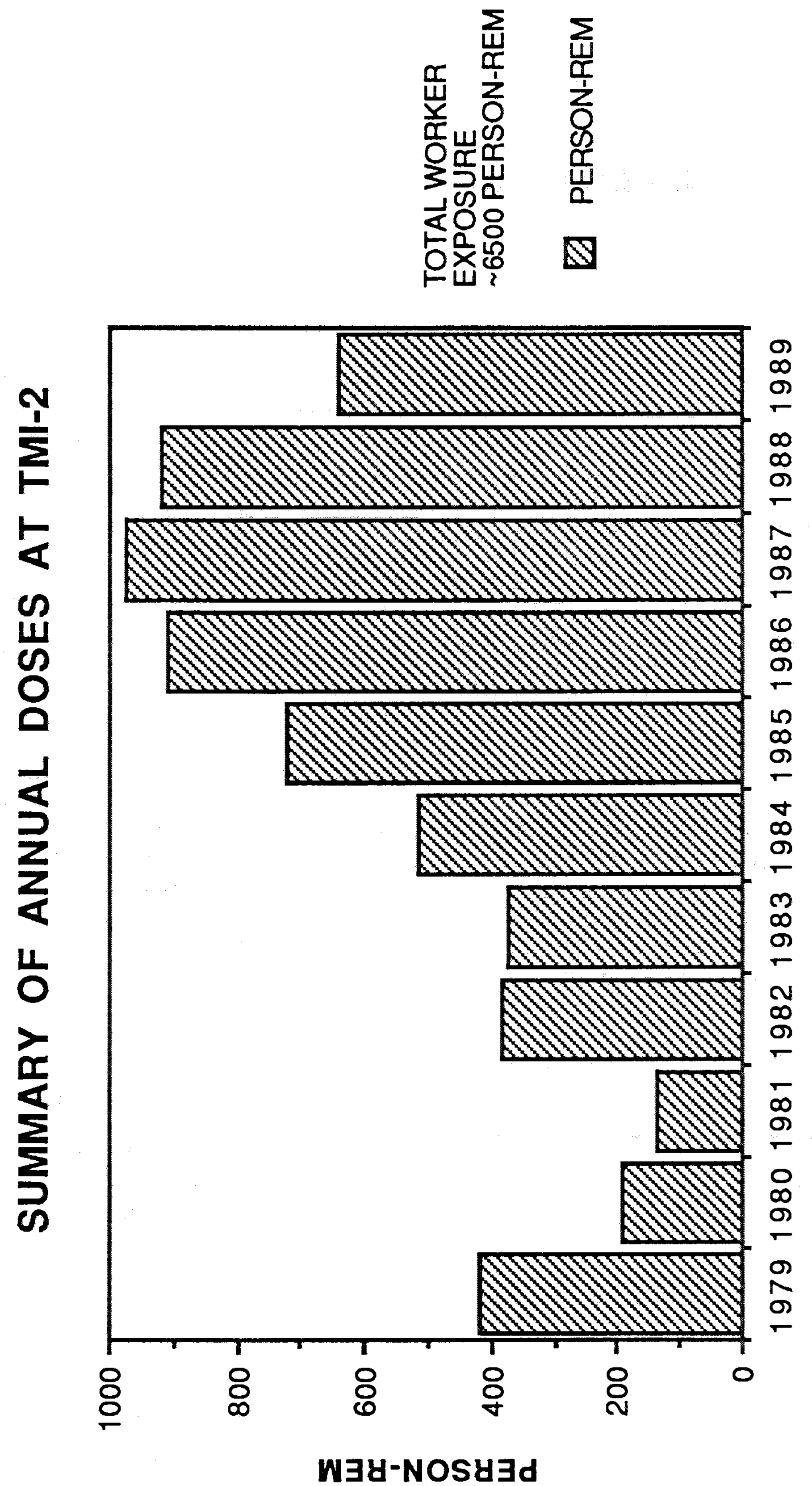

Figure 4-1. Summary of Annual Doses at TMI-2

This example is not unique. Other aspects of personnel protection that might be required to meet the challenges posed by the cleanup had been identified prior to 1979 and deferred. These included:

- Autonomous radiological controls program management
- Enhanced radiation exposure management record keeping and tracking
- Remote and robotic tools
- Heat stress protection.

The TMI-2 project management's decision to address these areas and to develop a strategy for managing safe cleanup operations was a key element of the cleanup. The radiation management program developed for TMI-2, and subsequently implemented at GPU's other nuclear power plants, provided the opportunity for the industry to see radiation protection principles applied in a new way. The program used four of its key elements very effectively:

- **Task Planning**—Throughout the cleanup, the project managers relied on small teams for task-level strategic planning. The two committees that were dedicated most specifically to personnel protection were the Containment Assessment Task Force (sections 3.7 and 4.2.2) and the Dose Reduction Task Force (sections 4.3.3 and 7). By defining the boundaries and end point objectives for their respective efforts, these task forces kept the cleanup on track and contributed to maximum use of the available resources.
- **Mockups**—Although expensive, mockups proved to be the most effective tool for radiological protection. They were the best means identified to ensure that workers performed at their maximum efficiency.
- **Staging areas**—The organizing function of staging areas ensured rapid entry and exit, which allowed workers the maximum staytime in the containment.
- **Training**—Training programs had to be modified extensively, not only to ensure that all worst-case scenarios were covered by the trainers, but to ensure also that the trainees could perform in an array of hazardous conditions.

One area that would have benefited from additional attention was the emotional and psychological mindsets of the cleanup workers. Their real concerns—associated with the safety of their families, or "taking doses home"—were often not factored into management decisions. In hindsight, GPU management could acknowledge that it was naive to have assumed that the workers would find comfort in information presented by industry experts. Management should have realized that the cleanup would have special needs in the area of worker morale.

An example of this naivety became evident in the summer of 1982, when the Radiological Controls (RadCon) Department reduced the requirement for respirators in the containment to "exception only" status. In preparation for this change, RadCon management assembled survey data and technical experts' assessments to show the workers that the air in-containment was safe to breathe.

Management and RadCon technicians met with the workers to discuss their concerns and to try to allay their fears of incurring additional internal dose and taking it home. The Manager of RadCon volunteered to be the first person to enter the containment without respiratory protection. Some workers still resisted the change. The eventual success in getting workers off respirators when conditions permitted was the result of personal attention on the part of the HP technicians. Moreso than the findings of experts, trust among co-workers and confidence in RadCon were the keys to crossing this hurdle (Hildebrand 1989a).

When selecting equipment or tools for cleanup tasks, or deciding whether to develop new hardware, the management strategy was:

1. Use what existed on site;
2. Use what was available elsewhere; or
3. Develop a new piece of hardware as a last resort. (As an example of this approach, robots and some shielding designs had to be customized for use in the containment.)

Initially, each piece of equipment was tied to a specific challenge. Each selection required a close look, against a pre-determined set of criteria, to determine its efficacy within this unique cleanup program. Decisions centered around tradeoffs between efficiency and cost factors; i.e., minimum recontamination of plant areas, minimum risk to workers, and avoiding, where possible, the need for decontamination of previously clean equipment (e.g., tools brought in from the outside).

4.1.2 Approach

It would have been much simpler to write 500 pages on the TMI-2 personnel protection program than to write 50. So much occurred, so many useful tools and innovative approaches resulted from this effort. Fortunately, much of this information, principally those aspects that can be applied at operating facilities, is well-documented and readily available through EPRI, the GEND report series, and industry associations' conference proceedings.

The discussions in this section concentrate on those aspects of TMI-2 personnel protection not readily accessible elsewhere. For the most part, these topics include initial efforts and, later, the decision-making processes behind many of the tools and programs developed or modified for use at TMI-2.

The section is intended to provide an overview of how work was managed and workers were utilized within the TMI-2 personnel protection program, and to leave a clear understanding of how that program evolved and operated. The principles more than the products are the focus.

This section does not include much data on personnel dose, nor does it present a chronological narrative on the application of personnel protection techniques during the course of the TMI-2 cleanup. Dose data are site-specific and voluminous (during the height of cleanup activities, approximately 6,000 personal dosimeters per month were processed at TMI-2). A chronological narrative is not revealing because the personnel protection program at TMI-2 was reactive in that it was driven by an R&D intensive environment. Thus, there were large intervals during which work ran smoothly or hardware/software were being developed, tested, modified, retested, approved for use, etc. For a general chronological overview of some personnel protection issues, see the timeline figure in Section 7.1.

4.2 Initial Entries

For the first 15 months after the TMI-2 accident, recovery workers' time was spent on two principal activities:

- Data collection, new systems installation, and decontamination tasks in the AFHB
- Preparations for containment re-entry.

4.2.1 AFHB Entries

The entries into the AFHB commenced immediately after the accident. The primary radiological conditions of concern in these buildings were the result of cesium and strontium/yttrium radioisotopes (suspended in the atmosphere and entrained by the water that flooded these buildings) and noble gases.

The worker protection strategy was to have the initial work teams clean the AFHB so that the hundreds of other workers that would enter these areas later would incur minimal doses. At the same time, management had to balance the impacts of potential radiation exposure against other potential risks (e.g., heat stress, structural impediments, and possible physical harm) (TMI-2 TI&EP II:E 1982).

4.2.1.1 Dose Monitoring.
The Harshaw 2-chip thermoluminescent detector (TLD) personnel dosimeter badge, widely used at that time at nuclear power plants for personnel dose monitoring, was used for these entries. The Harshaw TLD enabled 30–35% of the high-energy strontium-90/yttrium-90 beta particles to penetrate the deep-dose chip. In normal situations, the deep-dose chip value was considered the gamma-ray dose and was subtracted from the opened-window-chip value to derive the beta dose.

TMI-2 managers noticed problems early on. The first was the inability to differentiate skin exposure from whole-body exposure. The over-response of the deep-dose chip to the high-energy beta radiation fields in the AFHB required a change in badge reading interpretation.

Another problem with the Harshaw TLD was its two-dimensional design. The dosimeter was designed to measure beta radiation coming directly at it. In some situations at TMI-2, the dosimeter self-shielded itself from beta particles, causing a 10–100 dose reduction factor that had to be accounted for. This self-shielding problem was overcome by having each entry team member wear several badges (Ibid I:I 1980). (See Section 4.4.1 for more discussion of the Harshaw TLD and the eventual development of a new dosimetry system.)

Badges were worn at chest level, on the forehead, thighs, and wrists. As radioisotopic activity was increasingly more confined to building cubicles and other enclosed areas, management had workers wear lead aprons to screen out beta radiation in front, and additional dosimeters were placed in the back pocket and on the back of each thigh to monitor potential exposures from the rear. The lead aprons were soon abandoned, as the added weight created excessive fatigue and stress problems compared to potential frontal exposure to beta particles.

4.2.1.2 Protective Clothing
For the AFHB entries, workers entering dry areas wore multiple layers of cotton protective clothing, booties, gloves, surgical caps, and cotton, shoulder-length hoods to protect the entire head. In wet environments, workers wore wet suits or rubber coats over the standard cotton protective clothing. Lead-lined gloves, like the aprons discussed above, were tried and abandoned. The gloves greatly impeded manual dexterity.

Management observed that some workers were incurring skin contaminations while undressing after exiting the AFHB. To lessen these occurrences, people were trained to assist in undressing workers at the stepoff pad. No additional exposures were incurred by adding this assistance (Ibid II:E 1982).

As time passed and entries into the containment itself became routine, this initial team of assistants developed, in a series of stages, into the coordinated personnel access facility (PAF) and contamination control corridor (C-Cubed) concept. These facilities provided assistance in dressing and undressing, entry staging, and in delicate equipment protection and decontamination. The facilities are described in Section 4.4.6.

4.2.1.3 Respirators
During the first six months after the accident, AFHB workers wore Scott Air Pack self-contained breathing apparatuses (SCBAs). Airborne activity gradually decreased enough to enable full-face respirators to be worn for general AFHB area entries.

By October 1979, routine entries into the corridors of the auxiliary building could commence without respiratory protection; workers were still required to wear SCBAs in areas with very high airborne radioactivity, and respirators in areas where smear samples exhibited >1E+05 dpm/100 cm2 or in cubicles exhibiting >0.3 mpc radioactivity concentrations or for any task considered to entail airborne radioactivity concerns (Ruhter 1980).

4.2.2 Containment Re-entry Program

Preparations to re-enter the TMI-2 containment after the accident required a multi-disciplined series of activities (see Figure 4-2). As part of its re-entry strategy, the project established the Containment Assessment Task Force (CATF) in May 1979, to gather data on the current plant conditions and to prepare for the re-entry.

The objectives and accomplishments of the initial containment re-entry program are well documented in *Three Mile Island Unit 2 Reactor Building Entry Program Summary* (Langenbach 1980). *Color Photographs of the Three Mile Island Unit 2 Reactor Containment Building: Volume 1—Entries 1,2,4,5,6* (Eidam and Horan 1981) reproduces the photographs taken during these entries. The radiological assessments and equipment tests that the CATF performed to foster entry program planning are condensed and provided in tables 4-1 and 4-2.

4.2.2.1 Re-entry Considerations
One of the primary goals of the first re-entry into the containment was prompted by a psychological incentive to "get the door open and put somebody in and start taking measurements." The technical incentives for early re-entry included:

- Performing radiological surveys (isotopic analysis, radiological mapping) needed to plan decontamination tasks
- Assessing the integrity of the containment, the nuclear steam supply system, and other hardware inside the building
- Determining whether various plant components would be required to function to preserve the integrity of the containment and the primary system
- Deciding whether people could make future entries (For example, the containment fans were operating under conditions that exceeded their design basis and the possibility of failure had to be assessed.)
- Determining whether emergency maintenance and construction work would be required and, if so, would be feasible.

At the Facility Decontamination Technology Workshop, sponsored by the DOE and EPRI in November 1979, members of the CATF discussed some of the challenges they faced at that time (TMI-2 TI&EP II 1982, personal citation later). One speaker described the erratic radiological conditions inside the TMI-2 reactor coolant system:

"...the cesium concentrations are decreasing in time, as you'd expect because of the dilution, yet the strontium concentrations are increasing; we're getting more

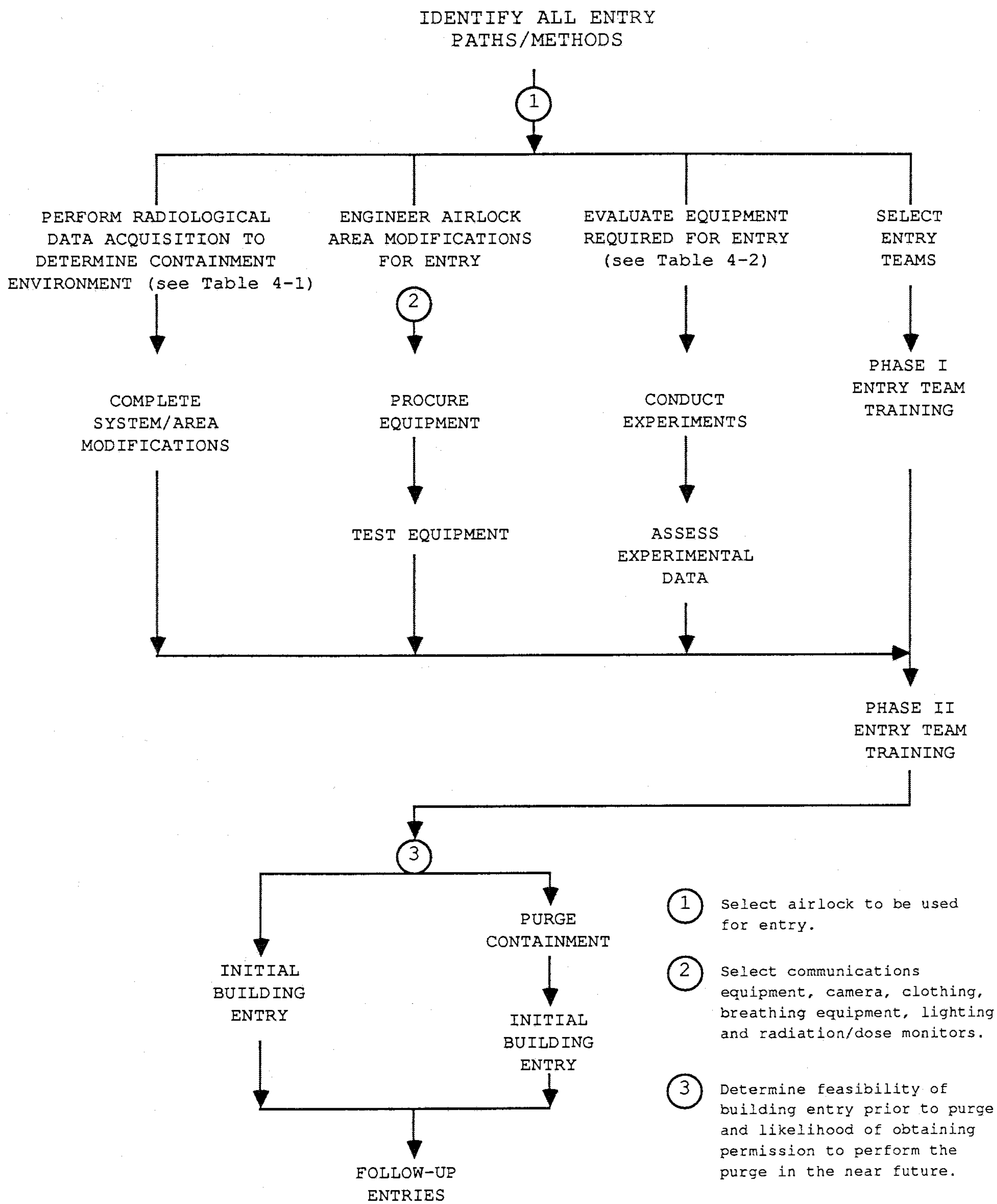

Figure 4-2. Containment Re-entry Strategy

Table 4-1. Pre-entry Data Acquisition

DATA ACQUISITION METHODS	PURPOSE	RESULTS
• WEEKLY CONTAINMENT AIRBORNE SAMPLES.	1. ANALYZED FOR PARTICULATES, GASES, IODINE, AND GROSS BETA	* PROGRAM SHOWED THE MAJOR ISOTOPE OF CONCERN REMAINING IN THE RB AFTER THE SHORT HALF-LIFE RADIOISOTOPES HAD DECAYED WAS KRYPTON 85.
• GAMMA RADIATION READINGS THROUGH THE EQUIPMENT HATCH, USING A Ge(Li) DETECTOR.	1. TO DETERMINE ISOTOPIC COMPOSITION AND MAGNITUDE OF PLATEOUT ON EL.305'.	* ESTIMATED PLATEOUT @6.3 TO 17.3µCi/sq.cm (FROM VERTICAL SURFACES OF HATCH). * DOSE RATE FROM PLATEOUT @177 TO 457 mr/hr. * MAJOR SOURCES WERE CESIUM AND LATHANUM. IODINE-131 EXISTED IN SIGNIFICANT AMOUNTS BUT SUBSEQUENTLY DECAYED.
• GAMMA RADIATION READINGS THROUGH THE INNER FLANGE OF A PENETRATION, USING A Ge(Li) DETECTOR AND TELETECTOR.	1. TO DETERMINE SUMP LEVEL AND SPECIFIC ACTIVITY OF THE CONTAM-INATION IN THE CONTAINMENT SUMP.	* MAX DOSE RATE MEASURED 31 R/hr, ESTIMATED DOSE AT WATER SURFACE 123 R/hr. * Ge(Li) SHOWED MAJOR SOURCE IN WATER WAS CESIUM-137 @ 366 µCi/cc
• GAMMA RADIATION READINGS THROUGH THE INNER METAL FLANGE OF PENETRATION R626, USING A NaI DETECTOR AND TELETECTOR.	1. TO DETERMINE GENERAL RADIATION READINGS AS WELL AS ISOTOPIC IDENTITY AND MAGNITUDE OF PLATEOUT ON EL. 347.	* DATA USED TO: (a) CALCULATE DOSE RATE AT EL. 347' TO BE @ 297 mr/hr. (b) ESTIMATE PLATEOUT ACTIVITY WHERE, Cs-134 WAS PRESENT IN THE HIGHEST CONCENTRATION AND Cs-137 WAS PRESENT IN HIGH CONCENTRATIONS. * NaI SCANS SHOWED CESIUM, BARIUM, AND LATHANUM PEAKS.

Table 4-1. Pre-entry Data Acquisition (cont.)

DATA ACQUISITION METHODS	PURPOSE	RESULTS
• RADIATION MAPPING OF No. 2 PERSONNEL AIR LOCK BY TAKING AIR SAMPLES AND PLACING PROBES INTO THE AIR LOCK.	1. TO DETERMINE LEVEL OF AIRBORNE ACTIVITY INSIDE AIR LOCK No. 2.	* SHOWED INNER AIR LOCK DOOR SEAL NOT PERFECTLY TIGHT, BECAUSE SOME ACTIVITY FROM CONT'T FOUND ITS WAY INTO THE AIR LOCK. * SHOWS LEVELS OF KRYPTON 85, IODINE 131, AND XENON 131M ABOVE THEIR RESTRICTED AREA MPC. (Kr-85 @2.0E-3μCi/cc; I-131 @1.5E-8μCi/cc; Xe-131M @8.0E-6μCi/cc)
• ANALYSIS OF HYDROGEN RECOMBINER INLET SPOOL PIECE BY ORNL.	1. TO DETERMINE PLATEOUT ON THE SPOOL-PIECE FROM SEVERAL DAYS OF FLOW THRU IT DURING THE FIRST THREE WEEKS AFTER THE ACCIDENT.	* DETERMINED PRINCIPAL SURFACE CONTAMINATION WAS Cs-134/137 AND Sr-89/90. NOTED UNPAINTED CARBON STEEL MAY BE DIFFICULT TO DECONTAMINATE.
• REMOTE TV CAMERA AND RADIATION SURVEY THROUGH PENETRATION R626	1. TO OBTAIN INITIAL VISUAL ASSESSMENT OF DAMAGE AND FIRST DIRECT RADIATION READINGS INSIDE THE CONTAINMENT.	* SHOWED NO DAMAGE, SOME DUST DIRT ON FLOOR, AND SOME CON-DENSATION & RAIN. * SWIPES SHOW MOSTLY Cs-134/137.ED * AIR SAMPLES CONFIRMED Kr-85 MAJOR ISOTOPE PRESENT IN AIR. * CONSISTENT GAMMA AND BETA MEASUREMENTS DIFFICULT PARTIALLY BECAUSE Kr-85 INTERFERED WITH INSTRUMENTS

Table 4-1. Pre-entry Data Acquisition (cont.)

DATA ACQUISITION METHODS	PURPOSE	RESULTS
• AIR LOCK ENTRY DETAILED SWIPE SURVEYS, RADIATION SURVEYS, AND Ge(Li) SCANS THRU THE INNER DOOR OF THE PERSONNEL AIR LOCK.	1. TO OBTAIN BETTER INFORMATION ON RADIATION LEVELS AND PLATEOUT SOURCE ON EL. 305' 2. TO AFFORD SOME VIEW THRU INNER DOOR VIEWPORT OF EL. 305'.	* NO SMEARABLE CONTAMINATION FOUND. * SEALS TO OUTER AIR LOCK NOT DAMAGED. * VISUAL INSPECTION THROUGH VIEW PORT SHOWED NO INDICATION OF STRUCTURAL DAMAGE, VIEW PORT GLASS DISCOLORED FROM RADIO-ACTIVITY.
• ADDITIONAL EXPERIMENTS WERE PERFORMED TO DETERMINE THE SUITABILITY OF PROTECTIVE CLOTHING, TELEMETERED DOSIMETRY, AND DOSE RATE INSTRUMENTS. • RB ATMOSPHERE WAS ANALYZED FOR HAZARDOUS TOXIC GASES.	1. TO ASSESS PERFORMANCE OF SEVERAL PIECES OF CANDIDATE EQUIPMENT. 2. TO ASSESS ADDITIONAL HAZARDS WHICH MIGHT EXIST IN THE CONTAINMENT ENVIRONMENT.	* SELECTION OF: (a) ZETEX TELEMETERED DOSIMETRY, RO-7, AND TELETECTOR AS THE DOSE RATE INSTRUMENTS. (b) VIKING DRY SUIT FOR PROTECTIVE CLOTHING. (c) BIO MARINE' BIO PAC 60 FOR RESPIRATOR PROTECTION, INITIALLY

Table 4-2. Re-entry Equipment Selection

EQUIPMENT TYPE	PRIMARY REQUIREMENTS	EQUIP CONSIDERED (* = CHOSEN)	RATIONALE
1. Lighting	1. Portability 2. Off-the-shelf		
• personal lamps	weight, beam intensity, time rating	• Rally Hardhat Throwaway Lights, Model #11759, rated at 2 h*	1. Meets requirements 2. Inexpensive & disposable
• floor lights	[same as above]	• Teledyne hand auxiliary flood lamp, rated at 5 h	1. Beam too dim
		• Ikelight Underwater Systems' Clear Modular Superlite Model #1095, rated at 2 h*	1. Beam adequate 2. Submersible design facilitates decon 3. Rechargeable
2. Respirators	1. Size & weight 2. Ample air supply 3. Positive pressure 4. Off-the-shelf		
• First entry (Kr-85 = 0.80 µCi/cc)	1. Self-contained breathing apparatus	• Bio Marine Bio-Pac 60	1. Meets requirements 2. NIOSH/NRC recall
		• MSA 401*	1. Entry team preference 2. 30-min air supply deemed adequate
• Second entry (Kr-85≤MPC)	1. High protection factor for particulates 2. Positive pressure	• Scott Air Pack positive pressure, air mask	1. Meets requirements

Table 4-2. Re-entry Equipment Selection (cont.)

EQUIPMENT TYPE	PRIMARY REQUIREMENTS	EQUIP CONSIDERED (* = CHOSEN)	RATIONALE
3. Protective Clothing	1. Whole body (gamma/beta) Extremeties (beta) 2. Water proof 3. Off-the-shelf		
• First entry (Kr-85 = 0.80µCi/cc)	• Positive pressure • Minimum density = 250 mg/sq. cm (beta protection)	• Viking underwater, heavy duty dry suit * (with rainsuit over cover to shield dry suit from direct exposure)	1. Meets requirements 2. Only option available
	• Beta protection for extremities	• Multiple layers of rubber gloves; fireman's boots	1. Meets requirements 2. Boots also anti-skid
• Subsequent 2 entries (Kr-85≤MPC)	• Whole body	• Paper overalls, cotton overalls, and fireman's suit*	1. Meets requirements
	• Extremities	• Feet - cloth shoe covers, 3-pair plastic shoe covers, fireman's boots*	1. Meets requirements 2. Plastic shoe covers disposable
		• Hands - cotton surgeon's gloves, 2-pairs neoprene gloves (all but cotton taped to layers of PCs), lineman's gloves.*	
		• Head - cotton surgeon's cap, [respiratory mask] cotton hood, rain suit hood.*	1. Meets requirements

Table 4-2. Re-entry Equipment Selection (cont.)

EQUIPMENT TYPE	PRIMARY REQUIREMENTS	EQUIP CONSIDERED (* = CHOSEN)	RATIONALE
4. Communications	1. Compatible with PCs 2. Maximize free movement 3. Off-the-shelf		
• First entry		• Existing plant's communication system	1. Requires cable link for each entry worker 2. Requires open personnel airlock inner door.
		• Motorola MX 350 portable radios with set-com elbow switches, cranial microphone, ear receiver; Micor Base/ Repeater; T1617M remote control console; antenna thru R-626 and clamped thru outer airlock door portal *	1. Meets requirements 2. Overcomes limitations of existing system
• Second entry		• Same as above but placed on speaking diaphram of mask NOTE: MX 350 transmitter power is 5 watts; battery life is 8 hours with 10% talk, 10% listen, and 80% idle; Micor transmitter power is 75 watts	1. Clearer communications between entry team & command center

Table 4-2. Re-entry Equipment Selection (cont.)

EQUIPMENT TYPE	PRIMARY REQUIREMENTS	EQUIP CONSIDERED (* = CHOSEN)	RATIONALE
5. Dosimetry	1. Accurate assessment of dose to entry team members.	• Telemetered dosimetry: a dosimeter/transmitter was carried by each entry team member. Each millirem of accumulated dose activated a coded transmitter in unit. Signal sent to central station, decoded, and recorded. Xetex S03*	1. Continuously monitered dose being received by team members.
		• Self-reading digital dosimeter: same as above but accumulated dose was displayed via LED readout and called out by the entry team member*	1. Provide verification for telemetered dosimeter.
		• Thermoluminescent dosimeters (TLDs) multiple*	1. To determine official record of dose received to whole body, and extremities that came in contact with radioactive surfaces.
6. Instrumentation	1. Determine sources and their relative intensities in the containment	• Eberline Model 6112 Teletector gamma dose rate meter.*	1. Ability to extend probe 4 m from person using detector enabling the identification of radiation fields ahead of entry team 2. Ability to detect beta radiation.
		• Eberline RO-7 high-range beta survey meter*	1. High-range beta measurements

strontium in the primary system, [which] impacts our protection problems from the standpoint that it changes isotopic mix, changes the beta field mix, and complicates things as time goes on" (P. Ruhter, Ibid I:I–2 1980).

Another speaker highlighted the limitations placed upon the entry team as a result of the in-containment conditions:

"There are several problems with making an extended tour, such as going up to the 347' elevation. The biggest problem is that we have a 40-foot [12.2-m] climb through an open stairwell. The area probably has sump water in the bottom of it. That will be a high dose rate area. The guys are going to be loaded down with an extra 80 pounds [36.3 kg] of weight. A 40-foot climb with just your street clothes on can be kind of tough sometimes, especially if you are trying to do it in a hurry" (E. Walker, Ibid II:G–8).

4.2.2.2 Pre-entry Purge

As shown earlier, Figure 4-2 outlines a series of steps that led to two re-entry options: re-entry without purge and re-entry with purge. All preparations for the containment re-entry commenced with the objective of enabling a team of workers to enter and collect information regardless of whether the atmospheric decontamination project (i.e., the containment venting) was approved by the NRC.

The actual steps involved in the containment purge task were straightforward and documented in *TMI-2 Reactor Building Purge—Kr-85 Venting* (Kripps 1981). Venting took place between June 28 and July 11, 1980. The technical and regulatory steps, political aspects, and alternatives considered are discussed in Section 3.7.

4.2.2.3 Initial Containment Re-entries

Training for the initial entry into the containment consisted of over 100 classroom hours and 50 hours hands-on inside the TMI-1 containment (with the lights out) and the TMI-2 auxiliary building. Each member of the entry team underwent extensive pre- and post-entry physical examinations.

The project team hoped to enter the containment in the spring of 1980. In April of that year, a planned entry was postponed because of manufacturer-related difficulties with the SCBA. On May 20, 1980, entry was attempted but failed when the containment inner air lock door could not be opened. At that point, the decision was made to wait until the krypton-85 venting had been performed.

On July 16, 1980, the inner door to the containment was successfully opened. Preliminary radiation measurements were taken just inside: gamma ranged from 300 mR/h over the access ramp to 700 mR/h adjacent to the elevator shaft.

On July 23, 1980, the first manned entry took place. It lasted 22 minutes and resulted in 29 photographs, six 100-cm^2 smears, general area beta/gamma surveys, and removal of a 20-L plastic bucket. General area surveys indicated gamma at 500–700 mR/h; beta at 1 Rad/h. Each individual received a whole body gamma exposure of approximately 220 mrem with no skin (beta) exposures. See Appendix J for a transcript of the conversation that took place between the entry team and the command center.

Survey results from the initial entry indicated that the radiological environment inside the building was comparable to that inside the auxiliary building, and, therefore, the extensive training and physical exams were no longer required. Training for the second entry involved model review and review of the task plan and equipment, along with standard radiation work permit (RWP) training.

On August 15, 1980, the second manned entry occurred. It lasted 23 minutes for two members of the team, and 38 minutes for the other two members. The results included removal of a radiation monitor, 67 photos, 12 100-cm^2 surface smears, two scrape samples, one 30- x 40-cm painted plate, two pieces of reflective insulation, a carbon steel funnel, and a sample of discolored glass. Two experiments were performed to: 1) determine the amount of loose contamination that could be removed using a maslin cloth swipe; and 2) measure the beta-to-gamma ratios at floor level and again at 1 m off the floor. General area surveys at El. 347' indicated gamma at 100–200 mR/h and beta at 250 mRad/h to 1 Rad/h.

4.3 Radiation Protection Management Program

The GPU radiation protection management program was well regarded by the NRC, INPO, and other organizations charged with oversight of licensees' radiation protection activities. Several reports discuss those aspects of the GPU program that can be applied at operating nuclear power plants, including special equipment, facilities, and management practices.

Relatively little, however, has been written about GPU's radiation protection management program during its developmental days, plagued with the "dissection" audits and employee disgruntlement that spawned the program's creation. So that is where this section begins—in 1979, three years after ALARA was formally defined, and four years before INPO would issue its good practices guidelines on radiation protection management.

4.3.1 Early Audits

In 1979, three radiation protection audits were performed at TMI-2. The BETA Corporation, the NUS Corporation, and an NRC Special Panel each reviewed the utility's existing program and listed weaknesses. If not in tone, the principal vulnerabilities identified by these three independent audits were remarkably similar in substance. Each response identified:

- A morale and attitude problem within the radiation safety organization. Personnel within the organization felt that they had neither the authority nor the management support to stop operations in the interest of radiological safety.
- A need for a formal TMI-2 radiation protection organizational structure.
- A need to upgrade the radiological controls technician and radiation worker training programs, as well as a general need to provide more technical depth within the radiation safety program.

The lists went on to identify areas such as beta dosimetry deficiencies, upgrades to the radiation exposure management system at the site, and other issues discussed elsewhere in this and other sections of the report. But the comments cited above are the major radiation protection programmatic concerns to which GPU responded immediately. In February 1980, GPU issued its "Management Plan for the TMI-2 Radiological Control Program" (Heward 1980), which outlined the corrective actions taken and planned in response to the 1979 audits. These activities are summarized below.

4.3.2 Radiological Controls Department

The radiation safety organization, formerly a subset of the TMI-2 Site Operations Department, became the Radiological Controls Department. The new group reported directly to corporate headquarters, and had the authority to place potential radiation safety concerns above operational task priorities. Further, the corporate policy of using operations personnel to perform radiological control tasks during outages was abandoned. Under new policy, only radiological control technicians or their foremen who had undergone onsite training in accordance with the program described in the Radiological Controls Program Plan would be used for Radiological Controls tasks at TMI-2.

Project management held meetings and issued policies regarding their commitment to radiation safety and to ensuring that the health and well-being of all workers, corporate-wide, was considered in all aspects of their business.

The responsibilities of the new TMI-2 Radiological Controls Department were defined in writing, and this approach grew to encompass the radiation safety responsibilities of each person employed or under contract at TMI-2. The Radiological Controls manager now had a direct line of access to the senior managers in charge of the cleanup program and to a Senior Vice President at corporate headquarters. Internally, Radiological Controls comprised five groups:

- **Radiological Technical Support**—Prepared the department's procedures and coordinated the ALARA program with other departments on site.
- **Radiological Field Operations**—Performed radiation monitoring.
- **Radiological Training**—Trained RadCon technicians and supervisors.
- **Dosimetry**—Implemented and administered dosimetry programs.
- **Radiological Support Services**—Calibrated and repaired dosimetry equipment, tested respirators, performed bioassays, operated the radiological laboratory, and provided radiation health services.

During subsequent reorganizations, Radiological Technical Support was renamed Radiological Engineering, and Dosimetry and Radiological Support Services were merged to become Radiological Health. These groups are mentioned later in this section.

4.3.3 Training

The training program developed to certify RadCon technicians and supervisors included 40 hours of classes followed by written and oral examinations. During the oral examination, the examinee was given a situation involving either: 1) the interpretation of radiological data; or 2) an unusual radiation safety situation (e.g., high airborne activity, liquid spills, contaminated injured personnel). The examinee then assessed the problem and stated what should be done to control the situation. Based on the response, an examiner provided additional details on the radiological consequences. Upon completion, the oral examination board critiqued the performance with the examinee present, and subsequently documented the results.

Under the revised training program, RadCon technicians and their supervisors also performed practical ability exercises to demonstrate their proficiency in routine survey and TMI-2 routine radiation safety situations. This training approach was also used, in July 1980, to develop guidelines for when mockups, walk-throughs, and detailed worker briefings would be required for special cleanup tasks.

Eventually, this early training program evolved into the enhanced practical factors course that became mandatory training for all TMI-2 workers before their authorization to work in the containment. Throughout the cleanup, special training requirements changed, but the approach and criteria for determining who needed what level of training to make an activity safe remained the same.

4.3.4 ALARA Strategy

The principles of ALARA had been a nuclear industry goal for over 30 years. They evolved from recommendations made in the 1950s by the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection (NCRP). ALARA became a regulatory requirement for U.S. facilities in 1971, and in 1976 the phrase "as low as is reasonably achievable" (ALARA) was defined and added to the *Code of Federal Regulations.*

The philosophy behind the ALARA concept is that all radiation exposure involves some risk of biological damage and it is therefore proper to develop cost-effective ways to reduce personnel radiation exposure. As required by 10 CFR Part 20, all nuclear power plant licensees must have a formal ALARA program that defines personnel responsibilities and describes how organizations are to collectively implement ALARA principles.

In June 1981, an ALARA checklist for the TMI-2 containment entry program was produced (Brasher 1981). It organized the listed items into three categories:

1. Items of primarily an operational nature with which RadCon had very little direct interface except in the context of external exposure accumulation and which were unlikely to generate RadCon comments. Examples included: whether photographs should be taken, consideration of physical obstructions, worker routes to and from the jobsite, communications and lighting requirements, special tool requirements.
2. Items that required joint decisions between the Operations personnel and RadCon. These included shielding and ventilation requirements.
3. Radiological control requirements for the jobs that fell within RadCon's areas of responsibility. Dosimetry and respiratory protection requirements, protective clothing, radiation survey, and air sampling tasks all fell within this category.

The checklist concluded with a statement that the degree of success in implementing the ALARA program was directly related to the ability to identify tasks to responsible organizations and to interface where necessary. This was the basic premise for the entire GPU Radiation Protection Program.

Implementation of this ALARA strategy, however, proved challenging during the early 1980s, when efforts intensified to characterize the in-containment radiological conditions and begin decontamination and dose reduction. As data on the conditions inside the containment became available, the staggering number of worker hours required to perform cleanup tasks became evident. In addition, decontamination operations had not proved effective enough in lowering radiation fields to support fuel removal.

These factors led to a decision that dose reduction in the containment should be considered at the program level. Resources required for source removal, shielding, and decontamination became a high priority. Coordination of this effort was via the Dose Reduction Task Force, which was formed in 1982 with the urging of the NRC and comprised managers and engineers from a variety of operational disciplines.

The resulting dose reduction program was quite successful. It resulted in more than 50% reduction in average worker dose. In 1981, the average worker dose in the containment was 109 mrem/person-hour; by 1984 that average was reduced to 52 mrem/person-hour (Merchant 1988). Section 7 contains a description of the dose reduction activities in the containment, including shielding and coatings removal activities.

At the program level, the TMI-2 ALARA strategy worked well; however, isolated instances of human error occurred. One of the most disappointing incidences—in the eyes of the TMI-2 Radiological Controls managers—took place January 14, 1985. On that day, three men (two RadCon technicians, one photographer) entered the seal injection valve room (SIVR) to obtain radiation measurements and a dried boron sample, and to photograph an instrument panel. The SIVR was known to be a high-radiation area, exhibiting up to 250 Rad/h beta. All three workers received skin exposures in excess of the TMI-2 quarterly administrative limit of 5 rem (the 10 CFR Part 20 limit is 7.5 rem). The NRC was notified.

In Special Inspection Report 50-320/85-03, the NRC attributed this incident to the Radcon technicians failing to perform an adequate radiation survey and to the entry team failing to adhere to procedures and to the radiation work permit (Barr 1985). GPU's response to the Notice of Violation provided additional insights. The response listed a number of specific deficiencies associated with this incident, but pointed to two overriding causes:

1. The successful dose reduction campaign made work in high-beta areas infrequent. This created a reduced sensitivity to and awareness of the special considerations involved in protecting workers from high-energy beta radiation sources.

2. Radiological Controls personnel became too involved in doing the job to adequately perform the HP overview function.

The incident was classified as an isolated breakdown in the implementation of administrative controls rather than a programmatic deficiency. In support of this determination, GPU noted that: 1) the radiological conditions present in the SIVR should have been well known to the RadCon personnel involved; 2) both RadCon personnel involved were well-trained, qualified professionals; and 3) the pre-entry survey data and radiation instruments used during the job were adequate to alert the RadCon personnel to the high-beta radiation in the SIVR (Standerfer 1985).

Two exposures in excess of regulatory limits occurred during the cleanup: once in August 1979 during work on valves in the auxiliary building and once in 1989 during defueling-related decontamination. Both were to skin or extremities only, and neither are expected to result in any significant health effects to the individuals involved. No whole body exposures in excess of limits occurred during the cleanup.

The overexposure incidents, whether exceeding corporate or federal limits, served as periodic reminders to refocus the project on the need for continual vigilance in preparing for and executing tasks in radiologically hazardous areas.

4.3.5 Radiation Exposure Management (REM) System

Another programmatic area that was identified for improvement during the early audits was enhanced radiation exposure management. Actually, the large amounts of dosimetry data, the importance of accurate exposure records, and increasing pressure to minimize occupational exposure at the TMI site pointed to the need for an automated radiation protection data management system as early as 1975, long before the accident. The initial system that was installed maintained exposure histories and helped generate required NRC reports.

The TMI-2 accident created a need for a more advanced system because of the large increase in both radiation workers and the number of tasks to be performed in radiologically contaminated areas. To accommodate these changes, the system was expanded, in 1980, to include real-time data processing. Like the evolution of the radiation training program, the REM system developed for the special needs of the cleanup program evolved into a sophisticated exposure tracking and reporting system that also served GPU's operating plants and the data analysis needs of the corporate headquarters.

The REM system maintained radiation exposure, training, and qualification records and tracked the activities associated with RWPs using a central mainframe computer. RadCon technicians and others with a need for access to this information gained access through remote terminals. The REM system provided real-time control of area access by checking RWP requirements against worker qualification data. Security was maintained via a hierarchy of access codes that allowed varying degrees of information access and alteration.

The REM system had substantial exposure analysis and reporting capabilities. Job category codes indicated the type of work being done and exposure tracking numbers identified specific systems, components, or activities. Exposure estimates made before the task and a system to relate an RWP to a specific maintenance job were also available. The REM system could thus be used to extract and sort the RWP and exposure records in several ways; e.g., spot data trends, compare actual doses to estimates, identify high exposure systems or components, compare TLD to pocket dosimeter measurements. The software also permitted very specific inquiries sorted by data field and could generate a number of required reports, including Regulatory Guide 1.16 reports, 10 CFR Part 20.407 reports, and NRC forms 4 and 5 reports.

4.4 Other Personnel Protection Methods

Initially, there were clear deficiencies in technical knowledge associated with the personnel protection equipment and tools that could be used to address TMI-2 cleanup needs. The principal needs that could not be met via existing technology were accurate beta dosimetry and the use of robots for initial data collection and decontamination tasks in containment. Another principal challenge was how to protect entry team members from radiation without creating undue physical burdens and the potential for heat stress.

Because the environment was changing every week and accurate data were not always available, recovery planners chose to be very conservative rather than risk worker health and safety during the initial entries. This resulted in sometimes overly complex planning and procedures, but achieved its goal of safety.

4.4.1 Beta Dosimetry

In 1979, the industry's experience base in monitoring and protecting workers against high-energy beta radiation was insufficient. Equipment for monitoring and detecting high levels of beta radiation was not available.

In August 1979, the beta dose rates in the TMI-2 AFHB first resulted in several skin exposures in excess of regulatory limits during an entry into a makeup purification (MUP) valve cubicle. This incident prompted considerable research at TMI-2 to establish personnel protection criteria and to develop monitoring equipment to adequately detect high beta radioactivity (Hildebrand 1985c).

The Harshaw 2-chip TLD badge, which was used early in the TMI-2 recovery program, had only a 270 mg/cm^2 filter over the penetrating detector, and could not accurately measure doses received from high-energy, high-level mixed beta/gamma radiation fields. It did not provide the accuracy needed for either skin dose or whole body dose associated with the postaccident environment (Rich 1981).

Since an improved dosimeter was needed at TMI-2 for continuing work in the auxiliary building and for upcoming entries and cleanup activities in the containment, the project team reviewed existing dosimetry systems in order to obtain the best available system as soon as possible (Rich 1981). The selection criteria included:

- Capability to monitor mixed beta/gamma energies
- Provisions for 10 CFR Part 20 reporting
- Built-in quality control
- Operation within ANSI N13.11 (draft) parameters
- Automated and operator-oriented.

After reviewing a number of available dosimetry systems, a personnel dosimetry system was developed that included a modified Panasonic 4-element dosimeter and an upgraded automated system to process the data from the approximately 6,000 dosimeters in use at TMI-2 each month.

The system was installed in February 1983. The dosimeter was a modification of the Panasonic model 802 design. It used two lithium borate and two calcium sulfate elements, each 14 mg/cm^2 thick. Filtration ranged from 14 to 1000 mg/cm^2. This corresponded to the standard 802 design with the exception of element number 2, the filtration capabilities of which were significantly reduced (from 350 to 75 mg/cm^2).

The interpretation algorithm used in this system was designed to enable use of a changing beta correction factor, based on the beta spectral data collected by the dosimeter. The assessments were between thallium-204 and strontium-90/yttrium-90, as these beta source spectra were believed to approximate in-containment conditions. Besides processing the dosimetry readings, the software also supported the quality assurance aspects of this dosimetry system, which included:

- Inspecting dosimeters on a routine basis
- Evaluating key TLD parameters daily

- Aligning and calibrating TLD readers
- Generating and comparing element correction factors.

Aside from computer component upgrades and adding the capability to collect glow curves, the original personnel dosimetry system remained as initially designed. It was certified by the National Voluntary Laboratory Accreditation Program for all monitoring categories (Schmitt 1988).

The DOE sponsored a separate research project to develop a dosimeter for TMI-2 cleanup survey tasks. The design features that were considered included:

- Automatic correction for differing beta energy spectra
- Ability to determine doses in fields of mixed beta and gamma radiation
- Implementation of passive beta dosimetry using widely available health physics resources
- Small, compact design for ease in handling and to reduce the effects of radiation fields that vary based upon dosimeter position.

The research produced an 8-element dosimeter, about the size of a pocket calculator, that satisfied all of the above criteria (EPRI 1984). However, it was never used at TMI-2 because the Panasonic was sufficient, the 8 elements on each dosimeter would have been uneconomical to process, and large-scale production capabilities were lacking.

In the latter part of 1986, some areas within the containment began exhibiting significant increases in beta radiation. To address this, the project team instituted three changes to the TMI-2 dosimetry program, effective January 1, 1987. The objective was to reduce radiation exposures to the lens of the eye of in-containment workers and to assess more accurately whole body radiation exposures. The three changes involved:

1. Increased eye protection to provide total shielding thickness of 700 mg/cm^2. This change enabled 100% effectiveness in shielding the lens of the eye from beta radiation.
2. Assessment of the whole body dose at 1000 mg/cm^2 by implementing a computer processing change to the Panasonic TLD system. This action enabled the dosimeter to report true body dose for all tissues.
3. Modification of the self-reading pocket dosimeters to accurately measure whole-body dose at the equivalent depth of 1000 mg/cm^2 below the surface of the skin. This modification increased the accuracy of real-time dose tracking (GPUN 1986).

4.4.2 ALARA Reviews

The TMI-2 ALARA program included a pre-work ALARA review for all tasks to be performed in radiation areas that met a pre-determined set of criteria. These ALARA reviews were the responsibility of the Radiological Engineering section of the Radiological Controls Department.

The objectives of the pre-work ALARA review were:

- Analyze the radiological conditions data
- Perform ALARA evaluations
- Recommend dose reduction options based on the cost/benefit ALARA evaluations.

The criteria for determining whether the ALARA review was warranted included:

- Any task anticipated to accumulate 5 person-rem or more of total exposure
- Any task for which the dose/dose rate to the skin or the extremities, or both, might be limiting without special radiological controls
- Any task in which the airborne concentration was expected to exceed by a factor of 1000 times the limits specified in 10 CFR Part 20, Appendix B, Table 1, Column 1 (i.e., respiratory protection of 1000 times was inadequate)
- Any task that could release radioactive material directly to the environment
- Work inside highly contaminated systems or components as identified by Radiological Engineering
- Reactor disassembly/defueling operations involving core alterations.

The steps for performing an ALARA evaluation included:

- Evaluate radiological conditions in each location of work to determine the sources and relative percentage contribution to the total area dose rate.

- Evaluate the area work occupancy in terms of the total jobhours and the schedule of work.

- Determine applicable dose reduction methods (i.e., shielding or source removal). Identify options and estimate the degree of reduction from each.

- Calculate the net positive benefit derived from each combination of options.

- Select options with the highest net positive benefit for prioritization and implementation.

The radiation ALARA review process is described in further detail in *Radiation Protection Management Program at TMI-2: Noteworthy Practices and Accomplishments* (Owen, Brady, and Owrutsky 1987). The outline above is presented as the basis for discussing two shielding aspects that illustrate the TMI-2 program: 1) shielding requirements for the head removal activities; and 2) software developed to expedite shielding evaluations.

4.4.2.1 Shielding for Reactor Head Removal
In December 1982, as plans for the reactor head removal accelerated, the supporting dose reduction measures that could be accomplished via shielding began to crystallize (Buchanan 1982). At that time, the project team identified the following needs and shielding equipment that would serve to reduce dose:

- **Reduce doses incurred during head removal preparations:** Because the leadscrews would be suspended in the head service structure and were expected to significantly raise the background radiation concentration, there was an additional need to shield the upper portion of the head service structure.

  Resolution: Purchase custom-made shielding that was suspended from the head service structure and installed using the stud detensioning equipment. This was expected to reduce radiation exposures by a factor of 10.

- **Reduce doses incurred after head removal:** After head removal and placement on the storage stand, the head would continue to present a source of radiation at El. 347' and to persons walking in that vicinity.

  Resolution: Purchase water shielding columns, which were fiberglass columns designed to be filled remotely and provide approximately 56 cm of water shielding or 2 tenth-value thickness for cesium-137. These columns were cylindrical but interlocked to avoid gaps. They were free-standing and stackable and available in 1.2- and 2.4-m heights. These would provide a circumferential shield of the storage stand to a height of 3.7 m. Once purchased and applied, the water columns leaked and were subsequently filled with sand.

- **Shield the personnel traffic areas near the personnel air lock**

  Resolution: Also use water shielding columns, placed around the northwest corner of the elevator shaft to reduce dose rates measured at 450 mR/h or greater at that time.

- **Provide shielding on the working platform and down in the refueling cavity over the reactor vessel annulus**

  Resolution: Purchase lead blankets.

- **Portable shielding for general use:** Flexibility was the primary concern, both in location and in shield thickness. Examples included: shielding of the incore instrument seal table, shielded booth for removing highly contaminated protective clothing in containment, interim radioactive waste storage in containment, tool staging area in containment, radioactive waste staging areas in the ante-room and equipment hatch area, and decontamination activities.

  Resolution: Purchase 1.8-m frames and special 1.8-m lead blankets that could be assembled in place or wheeled into location as required.

Although the shielding materials recommended for the TMI-2 program were expensive, this equipment was designed consistent with the TMI-2 objectives of keeping radiation exposures ALARA. Additionally, the purchase was considered justifiable from a cost/benefit standpoint.

4.4.2.2 Shielding Analysis Software
Radiological Engineering personnel used the ISOSHLD computer code for intricate dose calculations. This code was widely used by shielding analysts and is not described here. However, the number of shielding analyses required to support the cleanup was enormous. Many of these analyses had to consider several sources and shield geometries. The evaluation process was time-consuming and, because the ISOSHLD system was accessed via an IBM Time Sharing Operations (TSO) system, engineers had to wait for an available terminal. Also, the ISOSHLD II version was not particularly "user friendly."

These two limitations led to the development of two microprocessor-based shielding analysis programs that, although not directly associated with GPU or the TMI-2 cleanup, were used extensively by technical planners and Radiological Engineering to perform comparative assessments and thus remove the need for access to ISOSHLD except for preparing the final documentation for regulatory reporting.

The commercially available software was MicroShield (Grove Engineering) and RADCALC (Schneider Engineering). Although they differed in format and physical appearance, both programs freed the engineer to run shielding calculations on a microcomputer; both emulated ISOSHLD in libraries and algorithms; both were extremely easy to use; and both enabled the analyst to store calculation data for subsequent analyses.

4.4.3 Robotics

The concept of using robotics (i.e., robots and other remotely operated equipment) for the TMI-2 cleanup was introduced during the initial containment re-entry planning stages. At that time, the principal motivation for considering this option was the uncertainty as to whether the NRC would allow the containment purge.

The general consensus at that time was that the initial entries, due to their short duration and task objectives, would require the logic and mobility attributes that only a person possessed. However, as part of the pre-entry planning, TMI-2 engineers, with the cooperation of the DOE and national laboratory experts, had a number of commercially available robots demonstrated for consideration. Through these information collection efforts, re-entry planners were able to identify specific physical limitations associated with deployment of then-available robots for cleanup tasks.

Three principal limitations associated with these robots were identified:

- The tractor-type treads could not negotiate the personnel air lock.
- The cables were not remotely powered; i.e., a tether was required.
- Radio control of the robots was not possible; thus controller communications from outside the 1.2-m. thick concrete walls was difficult.

Remote equipment began to play a limited role in the TMI-2 cleanup in June 1981 (for limited data acquisition). As early as 1982, plans had been made for remote access to the containment basement. The primary incentive for this approach was the need to acquire more data on the radiological conditions. The remote vehicles available for use at that time are shown in Table 4-3.

Most of these robots had very limited capabilities and were relatively expensive. The two remote vehicles that were used at TMI-2 during this time were:

- **System In-service Inspection (SISI)**—Furnished by the DOE in 1982, it was a small, tracked, tethered vehicle that contained several CCTV cameras, a radiation detector, and a small manipulator arm. SISI was used to take radiation surveys and smear samples for assessing the contamination on the floors in makeup and purification cubicles "A" and "B". The smears were really wipes; i.e., sticky paper dabbing rather than wiping. SISI also performed video inspections of the area. The principal limitation was the height of the vehicle (0.3 m above the floor). Also, the tracked drive presented a problem, as the belts would come off and require retrieval by pulling the tether that transmitted control power and video signals.
- **Remote Controlled Mobile Manipulator (RCMM a.k.a. BIG AL a.k.a. FRED)**—Purchased in December 1982, FRED was to remotely flush areas in the plant. This was the first direct tie between use of remote equipment to perform cleanup tasks and the TMI-2 ALARA program's objective of minimizing occupational radiation exposure (Giefer 1987). First use of FRED was in the auxiliary building makeup pump room. A high-pressure water nozzle was mounted at the end of the manipulator arm. A support camera with pan/tilt was mounted in a stand to provide additional viewing capability. Directional flush was not completely successful by total remote control, but the results of this effort provided a wealth of knowledge concerning the approaches and understandings required to implement a remote device at TMI-2. These included: 1) adequacy of viewing required to attain a feeling of presence of the operator; 2) type and duration of training and mockup required; 3) adequacy of the mockup; 4) attitudes of the operators and support personnel; and 5) adequacy and reliability of the equipment. This limited experience established a basis for the foundation that would lead to future successful efforts with remote equipment at TMI-2.

Table 4-3. Early Candidate Robots

NAME	MANUFACTURER	DIMENSIONS	ATTRIBUTES
Big Al a.k.a. FRED	Hodges Robotics International Corp.	1.0 m long x 0.7 m wide	6-wheeled, 4-drive wheels, weight 160 kg, 360° rotation, manipulator arm cap 140 kg
PaR-1 Manipulator Vehicle a.k.a. HERMAN	Programmed & Remote Systems Corp.	0.9 m long x 0.75 m wide	2-neoprene tracks, could turn around on own center, provided mobile, maneuverable, and stable mounting for power electro-mechanical manipulators
Wheelbarrow MD7	Morfax Ltd., England	1.2 m long x 0.7 m wide	Tracked, can be powered by batteries or tether, developed for bomb squad use, could be fitted for manipulator arm
Centipede	MBA Associates	—	Designed to remotely transport a manipulator and TV system
MF3 Manipulator Vehicle	Blocker-Motors, Germany	—	Track-driven, intended for use with a manipulator
Oscar	France	0.3 m long x 0.3 m wide	Tracked or direct-wheel drive, had mounting plant for viewing & sampling equipment.

By 1984, the project's use of remote equipment and robotics was well underway. The custom-designed remote reconnaissance vehicle (RRV) was delivered to the TMI-2 site. The remote reconnaissance vehicle and others are also discussed in sections 5 and 7. The RRV's development and use is described in several EPRI reports (Geifer, Hine, and Pavelek 1985; Owen, Brady, and Owrutsky 1987; Schwartz 1989)

Also during this time, the remote controlled transport vehicle (RCTV a.k.a. LOUIE) was loaned to the project by the DOE. LOUIE made its debut in the 1950s, at the Hanford Engineering Development Laboratory (HEDL) (EPRI 1984b). It was a tracked vehicle containing a manipulator arm (454-kg lifting strength) mounted to a telescoping column. Its power and controls are transmitted through a tether. LOUIE was used to perform a radiation survey and measure the radiation profile from the "A" and "B" makeup demineralizer tanks. New approaches that were tested during this task included: 1) evaluation of the effectiveness of a sonar range finder as an accurate means to remotely measure the distance from an object such as a tank; and 2) evaluation of the ability to retrieve a complex remote device from a contaminated area and, with minimal effort, completely decontaminate the device to releasable levels.

This was a successful effort, due in large part to the lessons learned during the earlier deployments. Use of the sonar device allowed an accurate radiation profile not previously possible with remote devices. The task was performed one day in June 1985, and by the following morning, LOUIE was completely decontaminated and released to the noncontaminated remote equipment evaluation area.

This section has briefly described how robotics technology was introduced into the cleanup program for personnel protection. Additional robotics descriptions are presented throughout this report, in the context of their associated cleanup discipline (e.g., disassembly and defueling, decontamination, data acquisition).

4.4.4 Heat Stress

The activities required to cleanup the TMI-2 containment, coupled with the highly contaminated in-containment environment, required use of several layers of protective clothing to minimize worker contamination. Project management recognized that this protective clothing requirement would increase the risks of heat stress-related illness. Therefore, early in the cleanup program, a heat stress protection program was initiated. Here, again, the project team followed its basic strategy of: 1) use what's available on site; 2) use what's available elsewhere; and, as a last resort, 3) develop a tool to meet the need. The report *Personal Cooling in Nuclear Power Stations* (EPRI 1983) provides an excellent overview of R&D in this area.

At first, the heat stress protection approach involved pre-entry medical screening, pre-entry training on the symptoms and hazards of heat-related ailments, and imposition of in-containment work time limits. Table 4-4 provides these limits (Ritthamel 1980). These initiatives only slightly improved the situation. Time limits controlled heat stress but severely hindered productivity. In addition, increased rotation of workers added travel time through high radiation areas.

Another approach that was tried involved attempts to reduce protective clothing requirements. However, the activities required for the cleanup program afforded little room for leniency in this area.

In 1981, the project team initiated screening and acclimation for in-containment workers (Pastor 1981). The screening consisted of pre-entry mockups in full dress and monitoring oral temperature and heat rate recovery post-mockup. The advantage of this screening was that it enabled managers to match individuals to particular tasks. The principal disadvantage was that pre-entry lead times were three to four weeks. The acclimation effort involved workers, in full protective clothing, walking around in 290 K (63°F) areas for increasing time periods (15 < 30 < 45 min) over a 14-day exercise term.

Each of these early initiatives fell short of its mark—each reduced the risk of heat-induced illness, but at the sacrifice of worker versatility and productivity. To overcome this phenomenon, body cooling systems were investigated. Table 4-5 shows the results of that study, and traces the results that led to selection of the EPRI-Pennsylvania State University (PSU) short frozen water garment (SFWG). The minor problems associated with deployment of the SFWG included:

- **Freezing and Storage**—Onsite freezers were not large enough to store several SFWGs or to freeze ice packs in advance of need. A chest-type freezer was purchased and equipped with a center rail to support this project. Liquid nitrogen was used to freeze ice packs quickly during peak demand periods.
- **Laundry**—The TMI-2 primary laundry facility cleaned contaminated clothing. To prevent potential cross-contamination of clothing, a washing machine was purchased for the SFWGs.

Table 4-4. Work Time Limits

WORK TIME LIMITS*

TEMPERATURE/WORK TIME LIMIT

PROTECTIVE GEAR	≤300 K	300-305 K	305-310 K	310-316 K	316-322 K
None		No limit	No limit	2 h	45 min
Coveralls, gloves, neg. press. resp.	4 h				
Coveralls, gloves, hood, boots, MSA resp.		2 h	1.5 h	1 h	30 min
Wetsuits, anti-Cs, neg. press. resp.	1.5 h**				
Wetsuits, Anti-Cs, MSA resp.	1 h	45 min	30 min	15 min	15 min
Supplied airsuit, anti-Cs, supplied air system	4 h	4 h	4 h	2 h	1 h

*Sources: Ritthamel 1980; Pastor 1981

** Required completion of acclimation program.

Table 4-5. Body Cooling Systems

Type/Princ.	Name (Model/Man)	TMI-2 Applic. Test	Results of Evaluation
Body cool.device; pumps cool air close to body; convection & increased sweat evap.	Vortex tube & manifold	1981 small-scale decon experiment	• Protected workers from heat stress • Vortex tube could be decontaminated • Required too much service air • Umbilical design of hoses restricted worker mobility
Body cool. garment pump water through tight garment; ice/chemical packs collect body heat	Cool vest Cool head	none	• High cost • Systems (garment/pump) couldn't be decontaminated • Pump required regular maintenance
Body Cool. garment ice packets placed in vest (SWG) or jumpsuit (LWG) under PCs	Short FWG (PSU/EPRI) Long FWG (PSU/EPRI)	Early 1981. None - not used at TMI.	• not expensive • protects workers from heat stress • easy to maintain • minor concerns re-storage, laundering, timing, degradation (see text for details)

- **Durability**—Over time, the SFWGs, fabricated of rip-stop nylon, began to unravel. Cotton twill was tested and found to be more durable and to cost much less than the nylon. Also, approximately 5–10% of the ice packets cracked and leaked during usage. Although this did not appreciably reduce the effectiveness of the SFWG, the packets had to be replaced. Use of thicker plastic (8 mil) might have resolved the problem.

- **Decontamination**—Slight contamination of the SFWGs could be cleaned easily; however, SFWGs that became extensively contaminated had to be disposed of. Very careful undressing after containment entries alleviated this concern.

- **Entry Timing**—Close coordination between worker dressing and entry into the containment was required to minimize the amount of ice melting before beginning actual cleanup tasks. By changing the dressing sequence and improving communications between the dressing area and the entry command center, this concern was lessened.

The final aspect of the TMI-2 heat stress control program that bears discussion is engineering controls. Early on, engineering controls were considered along with the acclimation and stay-time initiatives, but the variety of work locations and magnitude of required environmental changes caused project management to dismiss engineering controls as an impractical option to combat heat stress at that time.

The engineering controls option was re-visited shortly after deployment of the ice vests. Principally, the objectives were to reduce the use of ice vests, and thus the need for upkeep and maintenance on a large scale. The developer of the SFWG also developed a method for performing heat stress calculations to establish parameters for use of body cooling garments. The methodology was used at TMI-2 to demonstrate that the need for body cooling garments could be significantly reduced if an ambient temperature of <290 K (63°F) could be maintained.

Consequently, a building chiller system was installed to augment the containment air cooling equipment. Once deployed, the chiller system enabled operators to maintain the ambient temperature <290 K, thus reducing the need for SFWGs to an exception-only basis. This also increased worker stay-times and decreased entry preparation time (EPRI 1983b).

4.4.5 Respiratory Protection

It became abruptly clear after the accident that the station was ill-prepared for the respiratory protection demands which had arisen. The respiratory protection program in place at TMI-2 was designed mainly for planned maintenance under controlled situations. While the in-place program was very capable of handling anticipated plant demands (including accidents such as fires and radioactive spills), it was not geared to handle the respiratory protection requirements of large-scale or long-term cleanup from a radiological accident.

4.4.5.1 Immediate Problems

In the immediate aftermath of the accident, problems with respiratory protection arose quickly, including:

- Insufficient number of SCBAs—only 35 SCBAs with 70 spare cylinders existed on site

- Inadequate recharge capability—one 2.4E-03 m^3/s compressor on site.

With a SCBA usage rate of 10 or more units per hour, a much larger inventory was needed, along with facilities to decontaminate, service, and recharge cylinders. In the initial days following the accident additional SCBAs and cylinders were procured and cylinder charging assistance was provided by several local fire departments.

A skid-mounted, 8E-03 m^3/s high-pressure air compressor and SCBA charging station were placed on a flatbed truck equipped with a quick-disconnect power supply. The unit was mobile to be able to avoid the discharge plume from the plant (Gee 1989).

Other early problems included:

- Shortage of charcoal canisters for respirator masks, which were being used at a much greater rate than the SCBAs.

- Inadequate number of respirator-qualified personnel. This created an immediate need to medically evaluate, fit-test, and train many personnel.

Additional charcoal canisters were obtained from neighboring utilities and with the prompt response of one manufacturer. Shortages in respirator-qualified personnel were further magnified by additional problems like the inaccessibility of site training facilities and the respirator quantitative fit-test booth. Several local physicians helped to conduct the necessary physicals for the respirator qualification, and a school bus was rented and used as a mobile training facility. The inaccessibility of the respirator fit-test booth was overcome by changing the method of testing. The adopted method was qualitative, and used either stannic chloride (irritant) smoke or isoamyl acetate (banana oil).

4.4.5.2 Respirators

Of the numerous types of respirators used at TMI-2, the two most useful devices from the standpoint of worker comfort, productivity, and dose reduction were powered-air purifying respirators (PAPRs) and supplied air hoods. Advantages of the PAPRs—which were used almost exclusively for cleanup and recovery operations at TMI-2—were that they limited dead air space and lens fogging while providing a cooling effect on the face. Advantages of the supplied air hoods were that they were much more comfortable to wear and very useful in mitigating heat stress because the exhausted air was directed down the wearer's torso.

Because of the variety of airborne contaminants, both accident- and decontamination-induced, virtually every type of respirator that was available commercially was included in the TMI-2 equipment inventory.

4.4.5.3 Respirator Cleaning and Storage

For the first two years following the accident a mobile, contractor-operated, two-trailer facility was used to maintain and reprocess the respirators. The facility consisted of a processing trailer, a support trailer, and a facility ventilation unit. In 1981, the cleaning rate was averaging 50 per day (8 h).

Respirator cleaning comprised only a small percentage (35%) of total service provided by the facility. In addition to cleaning, the facility personnel also:

- Performed receipt and release radiological surveys on all equipment processed
- Received, decontaminated, tested, and prepared all ancillary respiratory equipment; e.g., air lines, air manifolds, air cylinders, powered-air-pack power units
- Performed all routine maintenance on respirators, where qualified
- Sleeved all air line hoses, belts, and powered-air-pack blowers to minimize contamination, and prepared equipment for use
- Performed a daily survey of the working inventory of respiratory equipment at all control points for both TMI units.
- Performed monthly checks of emergency SCBAs and emergency respirators
- Maintained all maintenance and inventory records for all respiratory equipment.

These additional functions accounted for approximately 65% of the contractor work load (Peterson 1981).

While the design capacity of the facility was quite adequate to handle projected respirator usage, the testing, repair, staging, and equipment preparation functions of the facility overtaxed the available space. The recommendation to build a utility-owned-and-operated facility was made in December 1981. The recommendation was based on economics and the need for efficient turnaround. The two basic options considered were the expansion of the existing trailer facility and the construction of a utility-owned-and-operated facility. Criteria used in determining the suitability of the options were:

- Decontamination factors
- Throughput
- Radiological controls
- Capacity
- Functions performed
- Waste generation
- Capital and operating cost
- Overall operating efficiency.

Based on the evaluation results, a utility-owned-and-operated facility was constructed in 1983. It had the ability to decontaminate respirators with very high levels of contamination and could process and maintain 100 PAPRs per day. For further information regarding the more detailed aspects of the evaluation of the respirator protection program at TMI-2, see "Respiratory Protection, Lessons Learned at TMI" (Gee 1989). Further information on the engineering evaluation of the laundry and respirator facility operation are presented in the excellent report: *Evaluation of Laundry and Respiratory Cleaning Operations at Three Mile Island* (General Dynamics 1981).

4.4.6 PAF And C-Cubed Concept

The TMI-2 personnel access facility (PAF) evolved based on activities from quite early in the cleanup program, by way of the entry support group. That group was responsible for all aspects of the initial series of containment entries.

Once the data acquisition and decontamination projects got underway in the containment, the number of workers and the variety of in-containment tasks multiplied, as did the requirements for protective clothing, respiratory protection, and support equipment combinations. These new and increasing demands, coupled with a need to coordinate suit-up timing more closely with entry schedules (to reduce heat stress-related risks), led to the decision to develop a dedicated, centralized area for proper and efficient dressing and equipping of workers before entries.

The resulting PAF, located in the turbine building, was the initiation point for the organized process of entering the containment. It included a check-in area, changing area benches and lockers, showers, an office, a protective clothing supply room, equipment storage areas, and work benches.

One principal difference between this PAF and the traditional staging and dressing areas at nuclear power plants was the role of the PAF staff in assisting entry team workers. Because the PAF staff knew the dress requirements for each RWP scheduled on any given day, they assembled sets of properly sized clothing and actively assisted entry teams in suiting up. The PAF staff watched for errors in procedures such as dressing sequence (e.g., donning an ice vest too far in advance of entry) and taping.

While the entry team dressed, the PAF supervisor checked the entry schedule with the coordination center. If the entry was to be delayed, that information was given to the PAF staff, who, in turn, delayed final dressing and taping so that the entry team workers could wait more comfortably.

Upon receiving directions from the coordination center, final dressing was completed, and the entry team exited the PAF and proceeded to the RWP control point. Each worker logged onto the REM system and then the team proceeded to the contamination control corridor (C-Cubed), located a few feet away and just outside the containment airlock. The entry team picked up any special tools or equipment associated with the day's RWP, then entered the containment.

The C-Cubed's most important function within the TMI-2 personnel protection program was in providing a means to prevent workers from contaminating themselves while undressing upon exit. This was accomplished by the C-Cubed staff, who provided hands-on assistance, as necessary, and ensured that prescribed clothing-removal techniques were followed. The C-Cubed staff also vacuumed exiting workers to control the spread of "hot particles."

The C-Cubed staff were tasked with recording digital dosimeter readings, retrieving TLDs, and providing medical response inside the containment. They served as the caretakers for all equipment (i.e., radios, respirators, dosimeters) associated with containment entries.

This PAF and C-Cubed concept was very successful in protecting workers and streamlining the ingress/egress process for the TMI-2 cleanup project. GPU later used this approach, on a smaller scale, to coordinate entries for a steam generator repair outage in TMI-1.

In February 1989, with the cleanup activities winding down and the number of in-containment workers diminishing proportionally, Radiological Engineering decided to move the dress out area from the PAF to the TMI-2 auxiliary building. The new dressing area, which was staffed by PAF workers, was more convenient and saved time. It also reduced the risk associated with previously laundered but slightly contaminated outer PC garments "dropping" contamination in walkways as the work teams walked from the PAF to the radiological control areas.

This move was made possible by a change in the type of PCs worn by TMI-2 workers. The change, which was prompted by a desire to reduce low-level radioactive waste volumes, became effective September 1988. It called for use of disposable, polypropylene suits as undergarments in lieu of wearing two sets of PCs. These undergarments, referred to as "paper suits," were put on by workers after they removed their street clothes in the PAF. The work team then walked to the new dressing area, and put on the outer layer of protective clothing.

An additional advantage noted by Radiological Engineering was that the paper suits greatly reduced the number of incidents of skin contamination at TMI-2.

4.4.7 Communications Equipment

Throughout the cleanup program, optimal audio-visual capabilities were a high priority for ensuring that the

ALARA objectives were met and for minimizing the potential risks posed by other occupational hazards.

4.4.7.1 Video Equipment

Video equipment was used for two basic support functions: 1) surveillance and task management; and 2) inspection and data acquisition. The latter is discussed in the Section 5; surveillance and task management support is discussed below.

Eight RCA 2000 series cameras were mounted in the containment. Each had remote pan, tilt, and zoom features. As a combined system, these cameras survey 75% of the building area subject to cleanup activities.

The video surveillance of the building was directed through the coordination center, which was located in the turbine building. Engineers in the coordination center relied on this surveillance capability to:

- Monitor in-containment work teams and ensure that cleanup activities were performed safely
- Observe and video-record tasks for future reference
- Watch for fires in contaminated or inaccessible areas
- Provide task support (in combination with radio contact) by observing work team progress and communicating detailed procedures, recording data, resolving unanticipated questions, and recording the activities for subsequent debriefing and future training sessions.

4.4.7.2 Audio Equipment

Radio communications was an extremely important part of the worker protection program. Supervisors and task-support engineers relied on these systems to communicate remotely with workers inside the containment.

A number of radio systems and components were tried at TMI-2. These are summarized in Table 4-6. The lessons learned from the extensive use of radio systems are discussed below.

There were two principal areas in which GPU identified and overcame potential problems associated with radio equipment. The first was that these components were difficult to decontaminate. GPU's program stressed protection of this equipment and relied on the trained PAF and C-Cubed staff to test and wrap the equipment before use, and then remove the contaminated wrapping after use.

The second potential problem was associated with component reliability. To address this, a performance monitoring and tracking program was established for the radio equipment. This program:

- Provided component usage records that indicated frequency of replacement of parts and identified failure-prone components
- Tracked transmission and reception problems from the coordination center
- Included periodic analyses of performance problems data
- Identified chronic problems that were then referred to the manufacturer for component upgrade or modification
- Provided a means for scheduling routine replacement of problem components while awaiting manufacturer upgrades.

The objective and strategy used for this program resembled the basic elements of the system life assurance methodology used to track hardware reliability at operating power plants. The magnitude of the program (i.e., the complexity) was the principal difference.

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Table 4-6. Candidate Radio System Components Used at TMI-2

Equipment Description	Attributes	Limitation	Use at TMI-2
1. Portable Base-repeater in containment wired through penetration	• Unlimited number of listen-only headsets can be accommodated	• 8 workers per base station can utilize full-duplex feature	• basic system design for audio communications
2. Belt-paks with FM receiver and transmitter; cable to throat microphone from transmitter; cable to ear speakers from receiver	• clear sounds • light weight • vendor modified neck strap to fix slippage problem	• microphone is prone to slipping; can increase risk of skin contamination when re-adjusting	• very common usage
3. Cranial Microphone	• more consistent signal than that produced by 2.	• less comfortable than 2. • sound less clear than 2.	• limited
4. Inductively Coupled Microphone	• mounted outside respirator faceplate	• requires careful adjustment of voice activator or noise-muting circuitry to remove inhalation sounds from workers	• limited
5. Boom Microphones	• comfortable • clear transmission	• incompatible with most respirators	• tried with forced-air respirators; unsatisfactory results
6. Earpiece incorporating speakers and mini-microphone; voice energy transmitted through ear (structural)		• no duplexing	• tested only
7. Integrated headset transmitter-receiver; earphones contain transmitter/receiver	• alternative to belt-pak and earpiece options	• heavy (28 oz.) • have a tendency to slip	• common usage

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5 DATA ACQUISITION AND ANALYSIS

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DATA ACQUISITION AND ANALYSIS

5.1 Overview

Information about plant conditions was the single most important factor in planning cleanup operations. A great deal of data was available, but, for three years after the accident, it was not systematically tracked, analyzed, or organized. Important data were often not available early enough to support design work. Consequently, assumed or inferred conditions were based on analytical models or partially complete data, which in turn led to overly conservative or overly optimistic design and schedule assumptions.

Planners, engineers, and analysts had to continually ask: Is it necessary to go after the data at all? How accurate is the information already at hand? What kind and how much data are needed to do the job? What are the resources required to obtain the data? What will benefit the nuclear industry? What are the consequences of overly conservative assumptions? What are the consequences of not knowing?

The role that data acquisition and analysis played at TMI-2 can only be understood in the context of the time—postaccident conditions were unknown, unpredicted, and unprecedented. Throughout most of the cleanup, there existed a natural reluctance to believe the worst-case scenarios for reactor core damage. Even when accurate predictions of core conditions could be made, engineers and management were reluctant to accept the bad news until it was seen on a video screen. A picture equalled far more than a thousand words.

In time, knowledge about plant conditions increased as the quality of information improved:

- **Conditions unknown (1979–80)**—Immediately after the accident, many areas (including the containment) were inaccessible, instrumentation was unreliable or suspect, and radiation readings were so high in some areas that little discrimination could be made regarding source term.
- **Conditions generally understood (1981–83)**—A general knowledge of conditions in the plant was obtained as gross decontamination progressed. Some samples and videotapes were taken, instrumentation was repaired or replaced, and some characterization of isolated areas was performed (including limited reactor vessel inspections).
- **Conditions well characterized (1984–1990)**—As decontamination and dose reduction reduced radiation fields, a much more specific knowledge of plant conditions was developed, along with the ability to provide characterization support for defueling and to characterize the plant in preparation for long-term storage. A relatively complete picture of reactor vessel conditions did not emerge until 1987; at the end of the cleanup, some uncertainty still remained regarding details of the core damage.

Management recognized the vital importance of data from the beginning; however, its timely acquisition and use were affected by several different factors. The two primary factors were the need to support ongoing cleanup operations and the need to extract information of scientific value. The importance of cleanup operations caused the project management to give priority to starting a task as soon as possible to attain production-level work, especially in defueling the reactor vessel. This could be self-defeating if the existing data led to under- or overestimating conditions.

For the research-oriented scientist, TMI-2 presented a full-scale test of existing accident models and an invaluable opportunity to study the effects of a nuclear accident. Management policy throughout the cleanup was that research work could not significantly interfere with cleanup work. Nevertheless, research work often furthered the progress of the cleanup by providing information crucial to planning cleanup operations. This information might not have been available otherwise. For example, the research-oriented Core Stratification Sampling Program characterized a large mass of resolidified fuel debris impeding defueling progress and also provided the means of breaking up the mass (see Section 5.4.3).

When large numbers of outside contractors and support personnel arrived immediately after the accident, they found a scene of general disarray in terms of pre-accident information on plant conditions. No rapidly accessible information center existed to collect, evaluate, or distribute pre-accident information about the plant (e.g., drawings, technical specifications, operating procedures). The technical advisors and staff had considerable difficulty in obtaining exact information related to their areas of expertise (Brooksbank and King 1979).

For several years afterward, although an enormous amount of data was gathered, a similar situation persisted. Work was conducted by different groups, and the data gathered to support the work were held within that group, with each performing its own data management function. In addition, each group had its own data requirements and own method of managing information. In the rush to stabilize and control the plant, data had only short-term value to support decontamination or the rapid installation of recovery equipment.

Through the 1980 GEND agreement, the national research establishment filled the role of a project data management and analysis organization (see Section 5.2). However, the role was limited by the objectives of the GEND agreement and the difficulty in retrieving much of the existing data. Since DOE and EPRI had R&D funds available during the early years, they were able to accomplish several of their proposed projects, which provided both operational support and important data on conditions; e.g., the containment entry program and the Gross Decontamination Experiment.

In September 1982, the TMI-2 project team was reorganized to integrate GPU and contractor personnel. Project management took this opportunity to create a central data management and analysis group (see Section 2.2.3). This group essentially took over the onsite role of the DOE in maintaining an accessible information base for the project. (Not all daily plant operational data were centralized in this group.)

The data management and analysis group was conceived to serve as a library of information, but its value was quickly seen to include systematically analyzing, interpreting, and distributing data. The creation of this group focused project attention on the value of obtaining data, as did the creation of the Technical Assistance and Advisory Group and the Safety Advisory Board. Still, the debate continued over when and how much data were needed to support cleanup operations.

Centralizing the data management and analysis function was a primary lesson of the cleanup. A related lesson was the importance of controlling how data were gathered. When possible, the same group would both collect and analyze the data. This greatly improved the way the data were reduced, interpreted, and reported. Flexibility was also increased because it was easier to respond to or take advantage of unexpected conditions. Finally, when samples were sent to offsite laboratories, it was vital to understand the specifics of the analysis techniques used; e.g., when radionuclide concentrations were reported per gram of sample, were the results a wet or a dry reference?

Data acquisition and analysis were intrinsic to every technical decision made at TMI-2. Consequently, this section will highlight the major issues and point to the relevant sections of the Technical History where the interactive roles of characterization and cleanup operations are discussed.

5.2 Organization and Plans for Characterization

The general objectives motivating and affecting characterization work were:

1. **Support Cleanup Operations**—Of first concern was the need to obtain data to support personnel protection, defueling, decontamination, and waste management activities. This was in addition to the characterization work required by normal plant operating specifications (required in diminishing degrees as the cleanup progressed).

2. **Research and Development**—The opportunity and funding existed to extract invaluable information from this reactor vessel "full-scale" accident test case.

3. **Ensure Safety**—A large effort was also spent to measure parameters of various types to ensure that safe conditions existed; e.g., that no radiation was escaping to the environment or that no potential for a recriticality existed.

The balance between the three was sometimes difficult to strike because no one could be certain if the data gained from a particular characterization task would provide information immediately useful to the TMI-2 project team, useful in the long run to the nuclear power industry, or of no real use at all. Figure 5-1 illustrates how the first two objectives were implemented during the cleanup.

5.2.1 TMI-2 Project Data Acquisition Plans

A detailed long-term data acquisition plan that could support cleanup operations was difficult to formulate. Changing or newly discovered conditions required flexibility. Conflict with cleanup work and changing program strategies hindered any long-range plan, as the first plan below illustrates:

- **1980 Summary Plan: Data Acquisition Entries** (BNI 1980)—This plan supported the initial cleanup strategy of decontamination, fuel removal, and plant requalification. Phase I of the plan secured urgently needed input for detailed planning, including mapping radiation and contamination fields, making photographic and video records, collecting and removing small objects, and gathering paint samples from floors and walls and water samples from the basement. This work was basically performed during the early containment entry program. Phase II, never undertaken, emphasized examining specific equipment and materials necessary to establish requirements for repair, replacement, or requalification.

- **1982 Data Management and Analysis Plan** (BNI 1982)—This plan was written shortly after the integration of the TMI-2 project team. It reflected the existence of the newly created data management and analysis group to centralize data management. It did not dictate what data were to be obtained but instead identified the method to be used; i.e., the sample package, which served as the vehicle through which data would be requested, approved, obtained, and analyzed. It was a flexible plan that could support both operations and R&D work.

The second plan showed that characterization had gained recognition as an important support function to be integrated with operations.

Getting the characterization work actually performed required more than a plan. Decontamination, waste management, and defueling operations consistently preempted much of the time, materials, and workforce required for thorough and methodical characterization. The possible needs of tomorrow rarely outweighed the work that had to be done today.

5.2.2 R&D Objectives

The GPU, EPRI, NRC, and DOE (GEND) R&D agreement was signed in 1980 to obtain information about the accident sequence and its effects (see Section 2.7). As a result, the DOE established the Technical Integration Office on site. An R&D characterization plan was developed that sought extensive data about many areas of the plant (Eidam, et al. 1982). Its scope was broad, in part, because the plan was conceived before information about data acquisition costs and interactions with the cleanup program were known.

The general approach of the R&D characterization plan was related to understanding plant accident response and recovery for the nuclear industry. Of special interest were the adequacy of regulatory guide assumptions, the resolution of unresolved safety issues, severe accident model validation and/or improvement studies, and modifications to standards design, operation, and qualification. In addition, information was sought about radioactive waste processing and disposal, defueling and decontamination technology, and requalification technology.

5.2.3 Sample Packages

The concept of a sample package document was developed because a formal method was needed to have data acquisition activities integrated with the high priority cleanup operations underway. Sample package documents were used to describe the characterization activity to be performed, the implementation requirements, estimated person-rem exposure, disposition of the data obtained, and how the results would contribute to the cleanup.

The review and approval process included safety and criticality evaluations, a review of the effects on permanent plant systems, and a review of dose assessments, feasibility studies, and funding availability. Following approval, the sample package was scheduled into the ongoing work.

Full-scale mockups of various components were assembled in the turbine building as a regular part of training for data gathering. The mockups were used for checking out tools, checking that ALARA objectives could be met, and ensuring the efficiency of the workers during the actual data gathering. A task in containment was usually performed by a team of four craftsmen, one engineer, and a health physics technician. The containment work was directed by a supervisor in the coordination center via a radio-communications system and closed-circuit television (Patterson, Estabrook, and Wilson 1988).

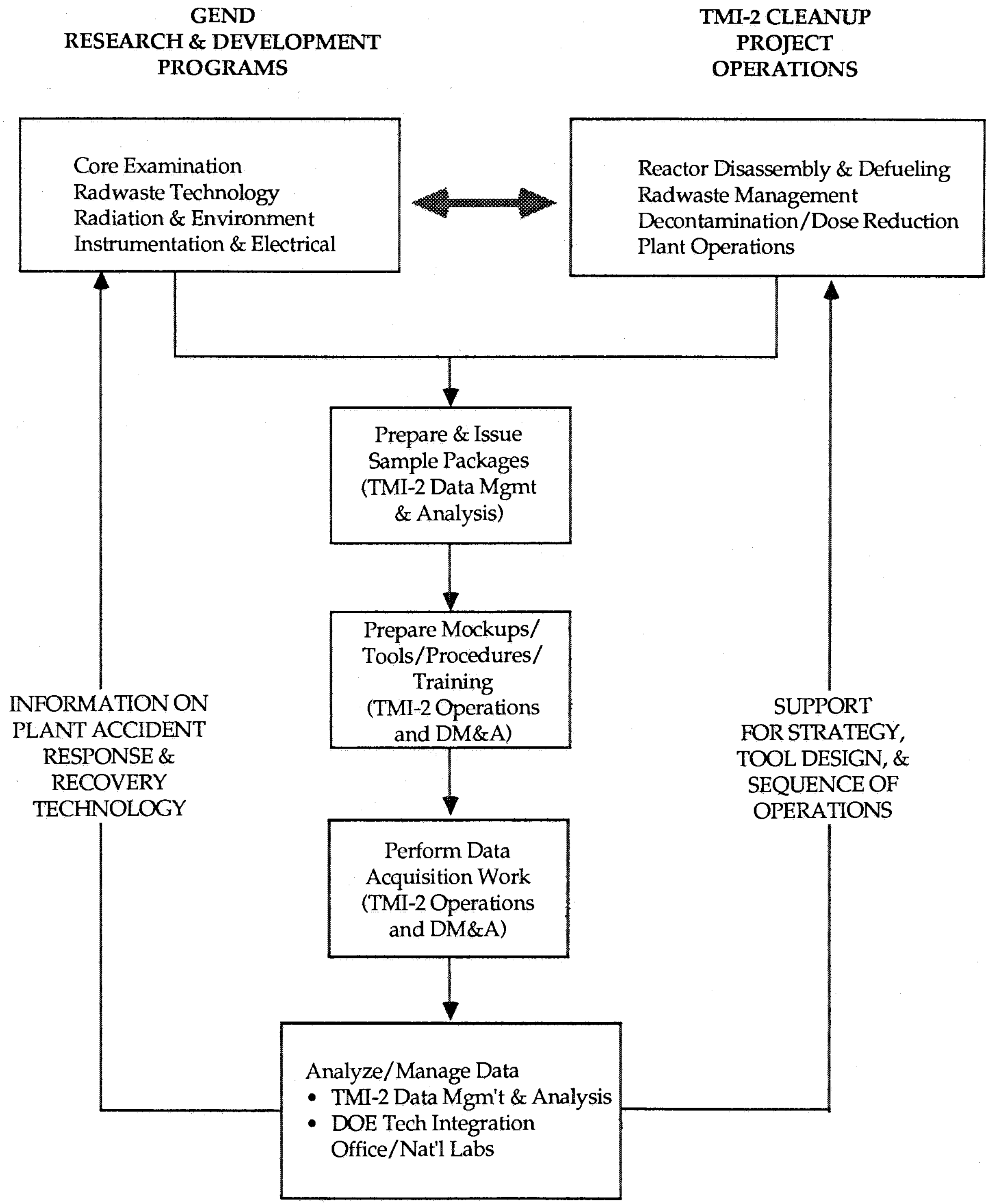

Figure 5-1. Data Requirements/Implementation

5.3 Synopses of Major Projects

A selective listing that reflects the major characterization activities and their relationship to cleanup operations appears below:

- **Containment Characterization from Outside the Building**—Containment radiation monitors, water sample lines, and existing penetrations were used to gather data about conditions in the building before the first entry. Samples allowed fairly accurate analyses of the water in the basement and the reactor coolant system, and a video camera inserted through a wall penetration provided a limited picture of the dark and dripping wet interior of the building. However, radiation levels were estimated to be several times higher than they actually were because in-plant radiation monitors gave false or misleading information, and the wall penetration used did not provide enough range to thoroughly survey the building. Section 4.2.2 provides more information on the pre-entry containment assessment program.

- **Containment Entry Program**—The entry program was the first effort to characterize general conditions by personnel operating within the containment. The first entry took place in July 1980, followed by entries at the rate of approximately one per month until November 1981, when preparations began for the Gross Decontamination Experiment, at which point the rate greatly increased. All entries were extensively planned for and had specific area or equipment characterization goals (GPUN 1981–82). See Section 3.7 for a description of the programmatic events leading up to the first entry and Section 4.2.2 for a description of pre-entry planning and personnel protection measures.

- **Gross Decontamination Experiment**—This series of experiments was conducted in the containment in early 1982, and required months of preparation and pre-decontamination characterization. The results showed that various decontamination techniques could be effective, but that recontamination would be a strong factor that could counteract much of the success. Accurate results were often difficult to obtain because of a lack of comprehensive data about pre- and post-decontamination radiological conditions (Mason, et al. 1983; Lazo 1988). See Section 7 for an overview of the Decon Experiment and its role in decontamination activities.

- **Dose Reduction Program**—The establishment of a Dose Reduction Task Force in 1982 led to a major source identification and dose reduction program to support defueling. Sections 4.3.4 and 7 discuss this program.

- **Instrumentation and Electrical Program**—This long-term program was primarily R&D, but the testing of components both inside and when removed from the containment identified a number of equipment installation problems and instrument response characteristics that had led to misleading information and equipment failures during and after the accident; e.g., the radiation monitors in the building (Mayo, et al. 1986).

- **Polar Crane Refurbishment**—A major effort was undertaken to characterize and refurbish the containment polar crane, which was essential for reactor head removal and defueling. The project is described in Section 8.4.1 (Graber and Lefkowitz 1984).

- **Reactor Vessel Characterization**—This work was the most fundamental and important part of all cleanup work from the accident onward. It began with analyses of computer codes and accident scenarios and continued with water sample analysis, video examinations, radiation and instrumentation readings, gamma scanning of an incore detector, debris sampling, topographical mapping by sonar, core stratification drilling, and removal of samples of the reactor vessel itself. Several tasks that produced turning points in knowledge of reactor vessel conditions are discussed in Section 5.4. Section 8 describes how the evolving knowledge of conditions was related to defueling planning and operations.

- **Ex-vessel Fuel Characterization**—Some fuel debris was known to have escaped from the reactor vessel during and after the accident. How much and where were questions of great concern. The combined use of flowpath analyses; visual observation; detection of gamma rays, neutrons, and alpha particles; and sample and analysis led to the identification and quantification of ex-vessel fuel (Kobayashi, Distenfeld, and Ferguson 1989; GPUN 1990). The spectroscopy methods developed for use in the field were especially innovative. Ex-vessel defueling methods are addressed in Section 8.7. The small quantity of fuel left in the plant at the end of the cleanup was identified in a special nuclear materials (SNM) accountability program discussed in Section 5.5.4.

- **Containment Basement**—The high radiation fields, sediment, construction debris, water, lack of lighting, and floor layout made it very difficult to gain a clear picture of conditions. Once 2.5 million liters of highly contaminated water had been removed, there was not a great urgency in addressing the area, although it remained a challenge for many years. All characterization work was performed remotely, either by workers on the floor above lowering measuring instruments and devices to sample sediment or concrete, or by the remote reconnaissance vehicle (RRV), which was developed for work in the basement (Geifer, Hine, and Pavelek 1985; Owen, Brady, and Owrutsky 1987; Ferguson 1988; Schwartz 1989). See Section 4.4.3 for a discussion of the role of the RRV in lieu of personnel entry and Section 7 for discussions of the robotic characterization and decontamination of the basement.

- **Radioactive Waste Characterization**—Extensive characterization, much of it innovative, was required during the processing and disposal of the enormous quantities of radioactive waste produced by the accident or generated during the cleanup. The waste included not only dry activated waste (DAW), but also the resins and processing vessels used to decontaminate the high-activity water that had been in plant systems and the containment basement. Hydrogen gas generated by radiolytic decomposition of residual water in processing vessels was of particular concern. Section 6.4.4 describes the characterization of EPICOR II and SDS processing vessels.

- **Sample Shipping Characterization**—The normally routine shipment of samples to offsite laboratories became a significant challenge after the accident. Detailed planning and specially designed packaging were required. Because many shipments were fissile or exceeded Type A criteria, an NRC-certified package was required. Approximately 400 separate sample shipments were made from TMI-2. These and the onsite laboratory capabilities of TMI-2 are discussed in Section 5.5.5 and other EPRI reports (Urland and Babel 1990, Deltete and Hahn 1990).

- **Water Characterization**—The large quantities of contaminated water in the plant required constant and timely characterization. Loss of water clarity in the reactor vessel had a tremendous impact on defueling progress and commanded months of attention to identify and treat the microorganisms and suspended particulates causing the problem. See Section 6.2.3 for a discussion of this challenge.

- **Makeup and Purification (MUP) Demineralizer Resin Characterization**—This project was begun with urgency because of concern that the two MUP vessels contained a large quantity of core debris. Extensive characterization work, some of it robotic, preceded the elution of cesium from the resins and the removal by sluicing of most of the degraded resins. Section 7 describes both the characterization and decontamination of the MUP vessels.

More details on the conduct of many of these projects is provided in *The TMI-2 Data Acquisition and Analysis Experience* (Urland and Babel 1990).

5.4 Reactor Vessel Conditions

A generally accurate and complete picture of reactor vessel conditions was not available until 1987—specific conditions in some regions were not seen until defueling was almost complete in 1989. Section 8 interweaves the story of how the final picture emerged as it related to defueling plans and operations. However, a comparison of the known and hypothesized conditions in Figures 5-2 through 5-7 conveys both the optimistic thinking and uncertainty prevalent at various times.

The figures are simplifications and do not fully represent the complex analyses behind estimates of conditions. For example, Figures 5-2 and 5-3 are based on GEND-007 (Croucher 1981), which served as the basic planning document for designing the initial defueling system. GEND-007 presented a complex set of potential damage parameters, with conditions stated in terms of minimum, reference, and maximum damage scenarios (see Table 8-1). Throughout every stage of the cleanup, considerable debate existed about the actual core conditions. Yet the fact remains that these drawings were the most commonly available portraits of conditions and represented the general expectations of defueling planners.

The following sections discuss the major data acquisition efforts that changed one view of the reactor core into its successor. Appendix B describes the actual end state conditions following the TMI-2 accident.

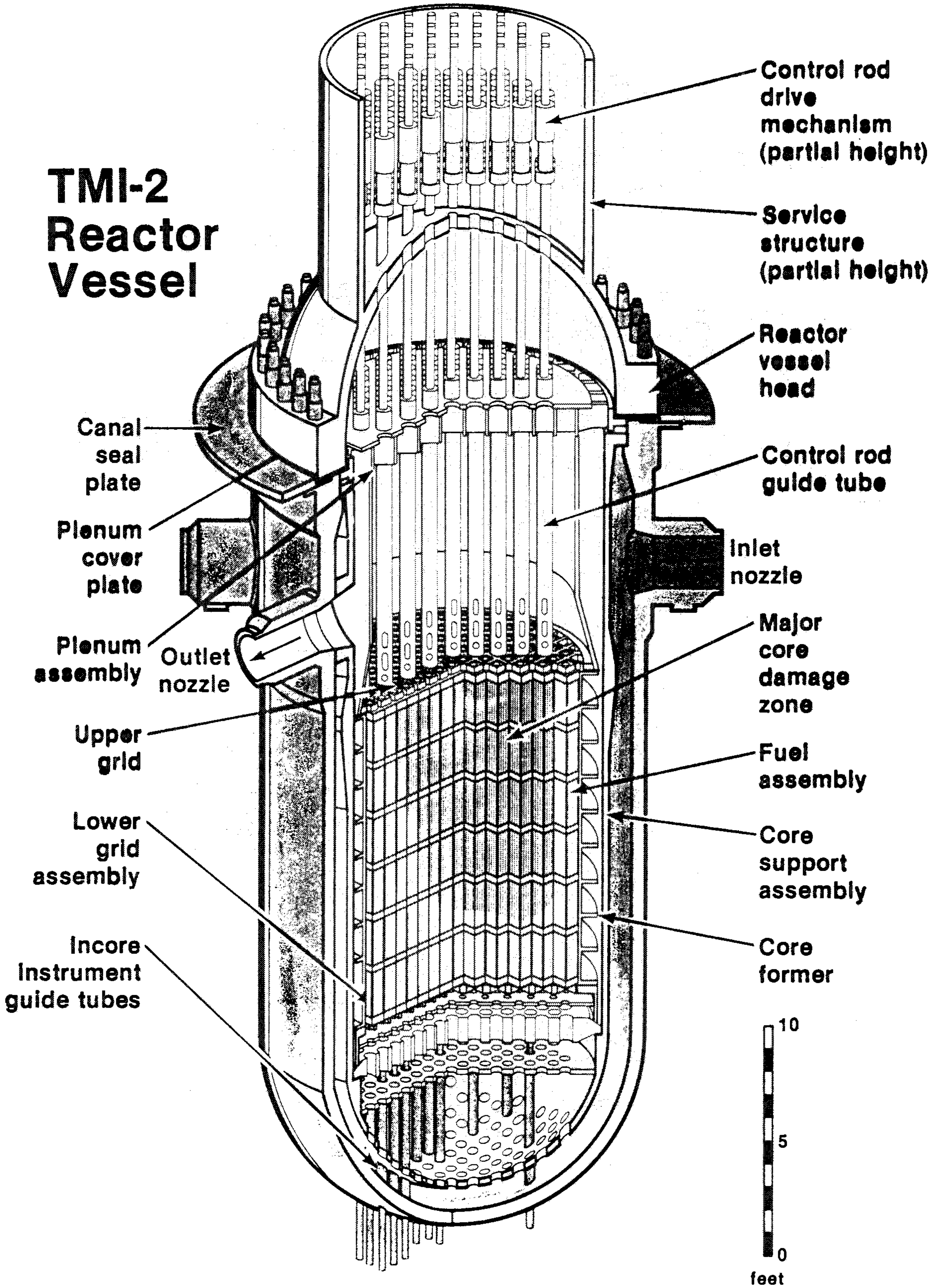

Figure 5-2. Initial Reactor Vessel Damage Projections: 1979–1982

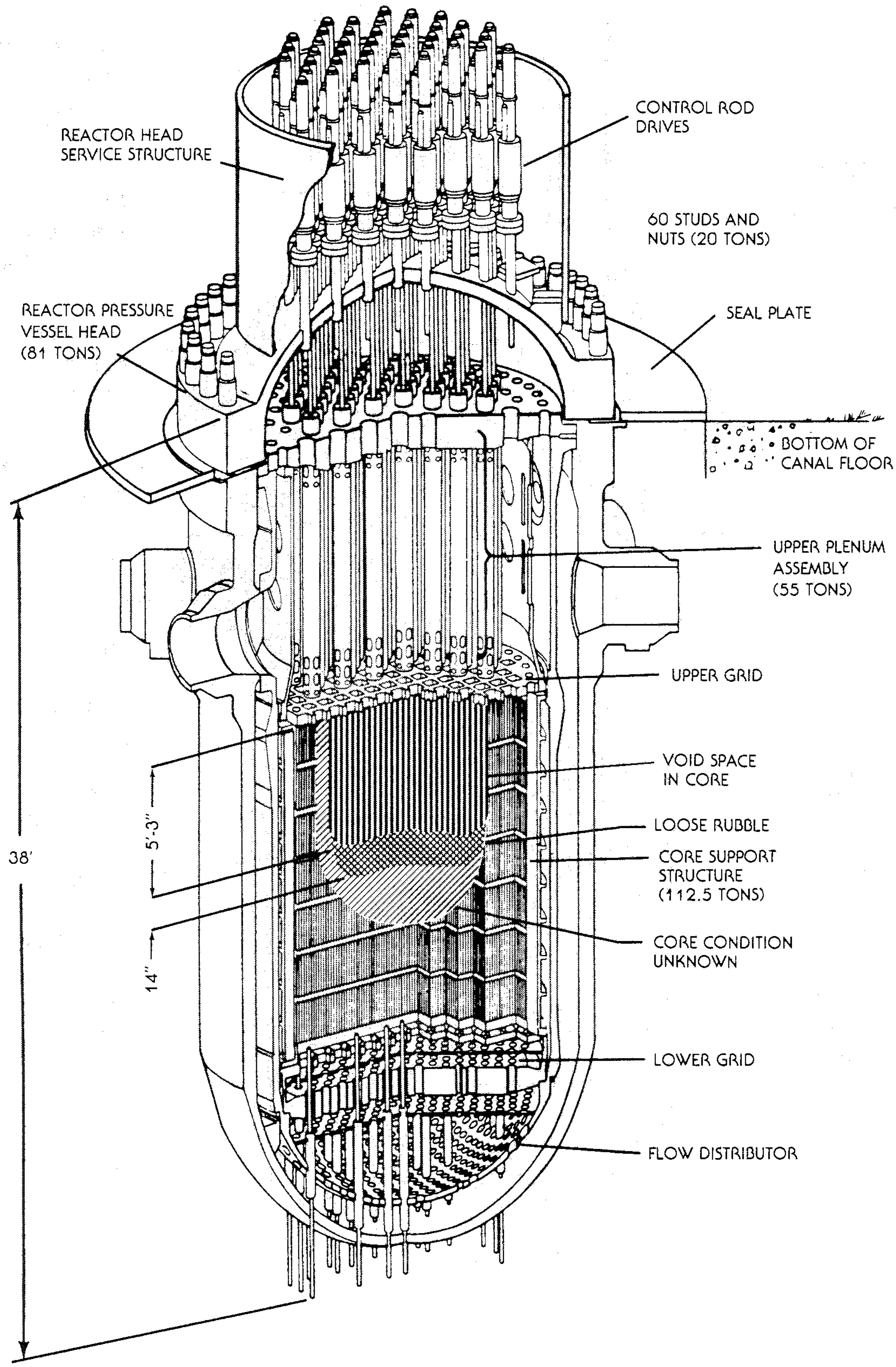

Figure 5-3. Known Post-Quick Look Core Conditions: 1982–1985

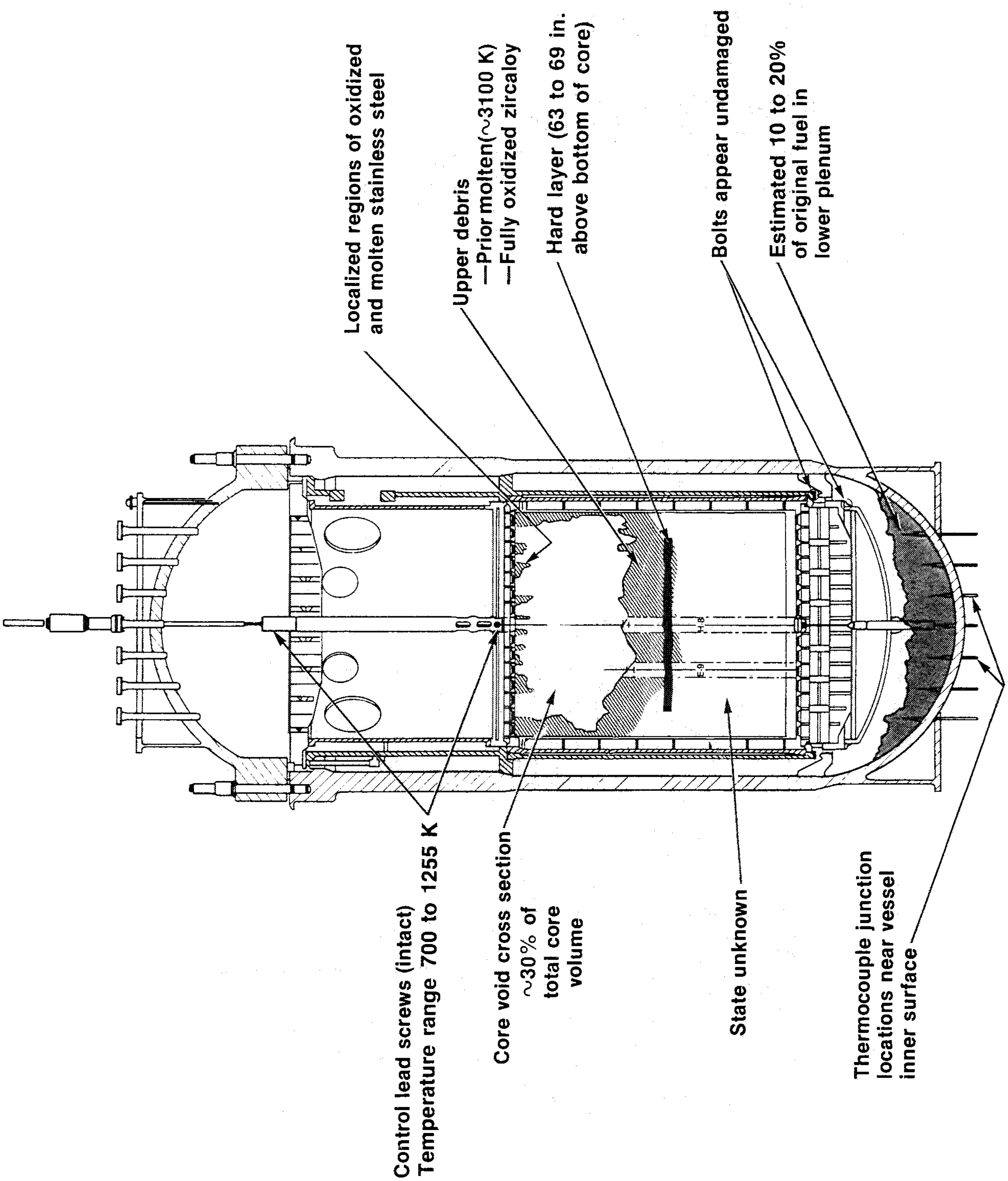

Figure 5-4. Known Core Conditions at the Start of Defueling: 1985–1986

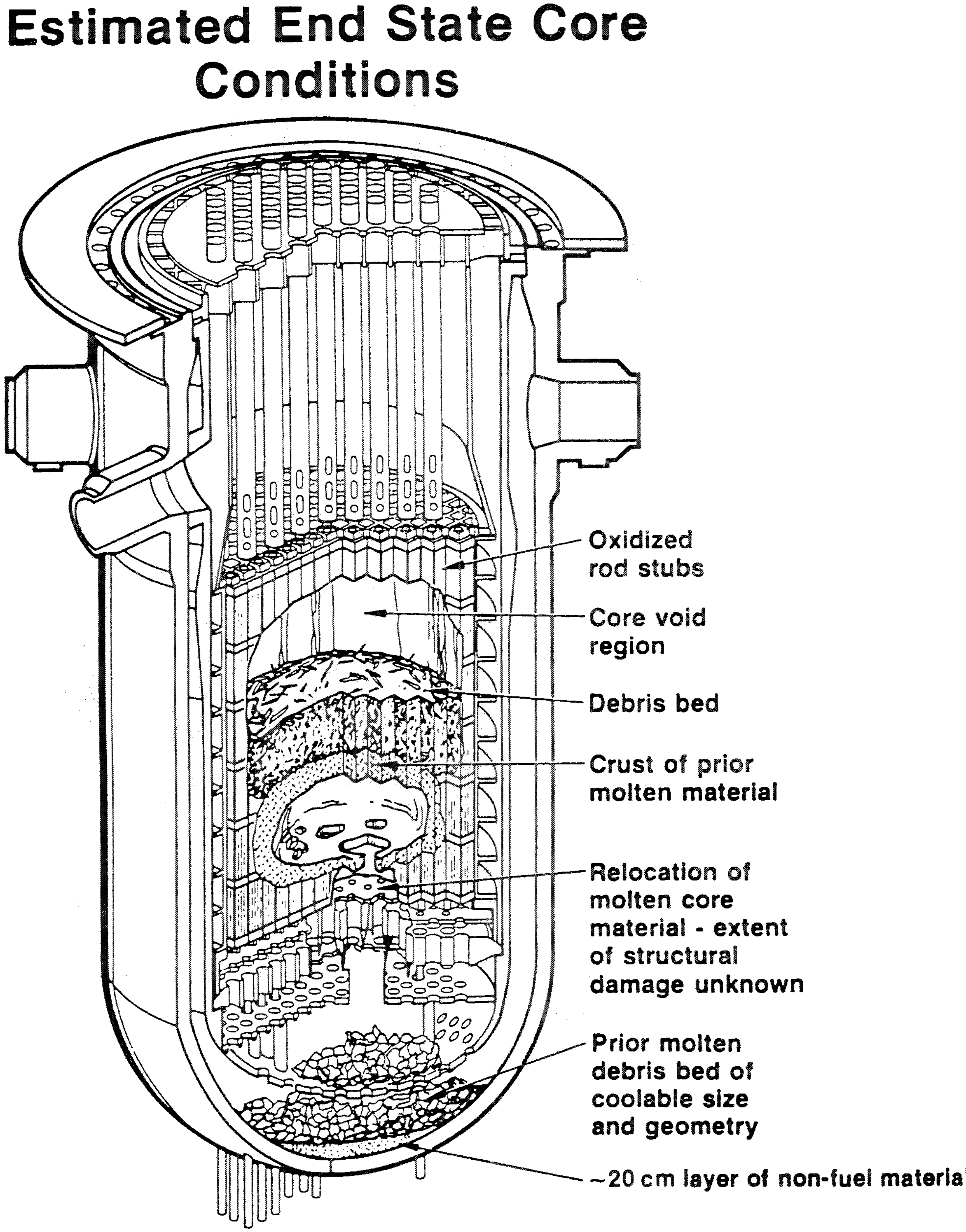

Figure 5-5. Hypothesized End State Conditions before Core Boring: 1986

Hypothesized End-State Condition of the TMI-2 Reactor Core

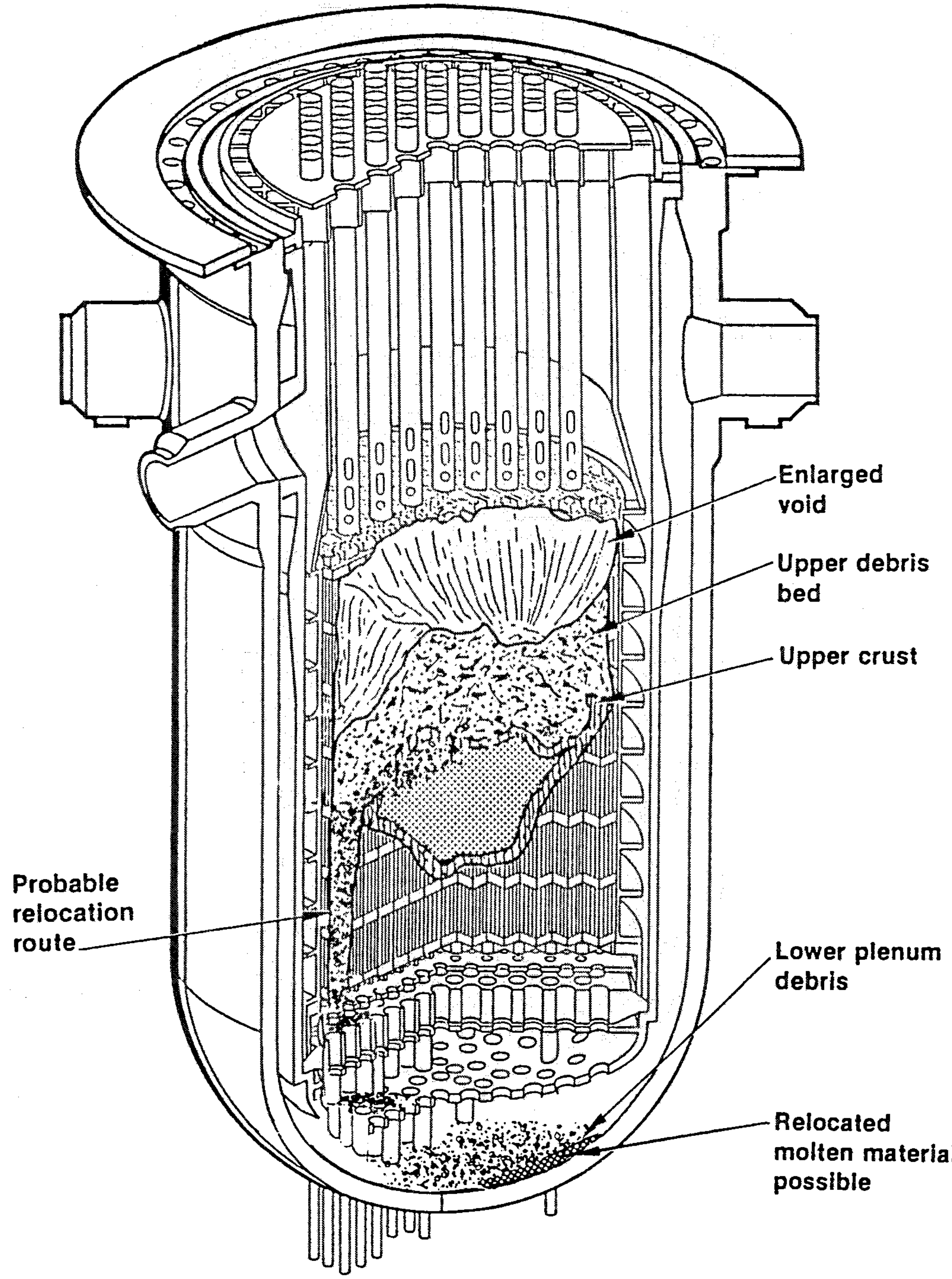

Figure 5-6. Hypothesized End State Conditions after Core Boring: 1986–1987

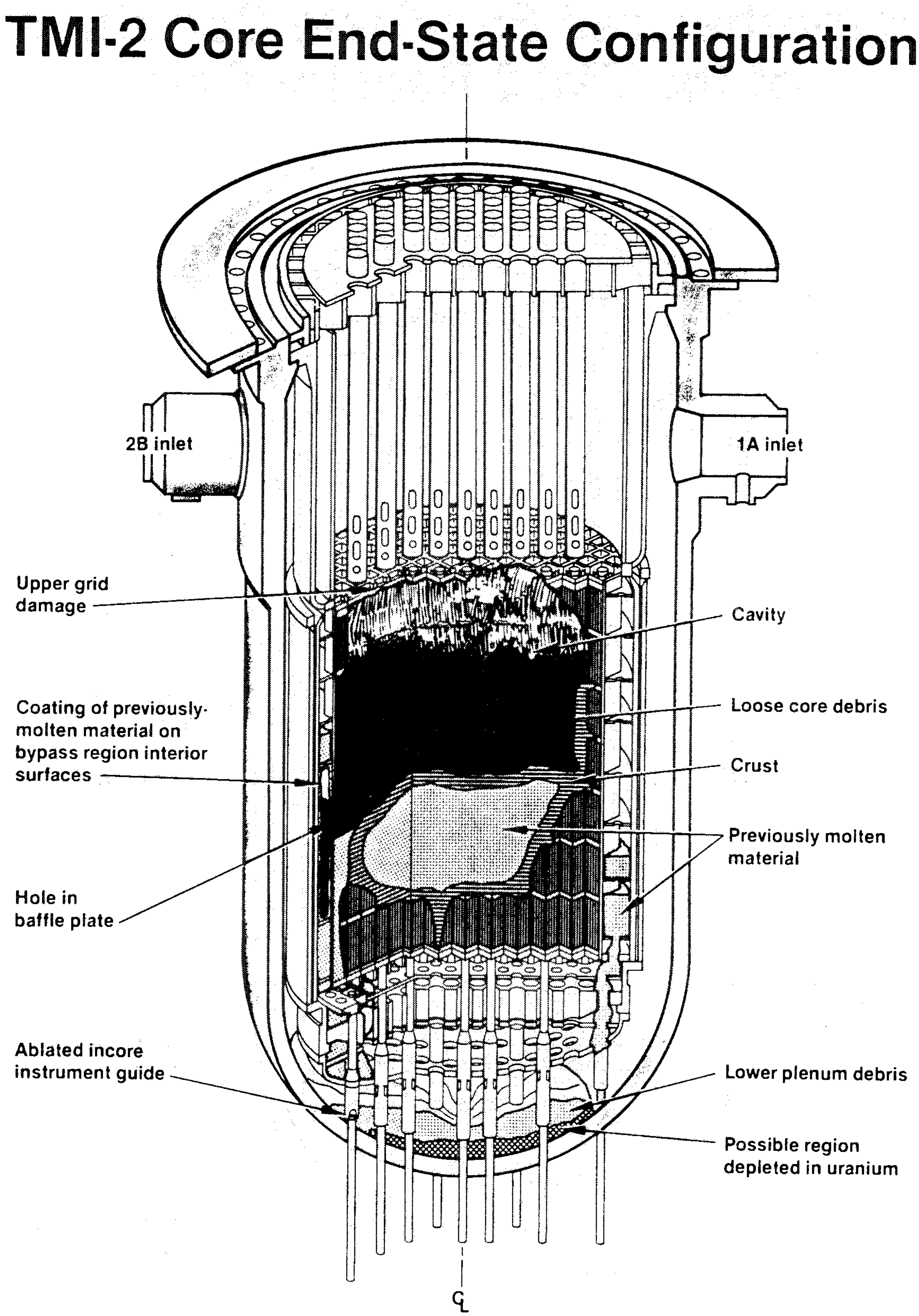

Figure 5-7. Known End State Conditions in the TMI-2 Reactor Vessel: 1987–1990

5.4.1 Quick Look

Since recovery planning began, the project team had recognized the importance of examining conditions in the reactor vessel before removing the head (AGNS, et al. 1980). Numerous access routes and approaches were investigated to:

- Determine operational conditions to be encountered during head removal and defueling
- Obtain information to help benchmark the range of damage estimates then existing
- Provide information on in-vessel conditions to the technical community.

Several inspection areas were considered; i.e., plenum cover, internal structure of control rod drive mechanisms (CRDMs), fuel assembly upper structures/core region, interval vent valves, and plenum-to-core support flange. The CRDM nozzles were chosen as the primary access routes for video inspection equipment because of their size and location (Calloway 1981).

The inspection plan developed in 1981 was for the initial penetration into the reactor vessel to be made through vent valve thermocouple nozzles. Five of these nozzles were to be opened, four for use by a purge system and one for the reactor vessel primary water level indicator. Up to three CRDMs were then to be removed by normal or abnormal procedures to permit the insertion of a video camera and lighting and sampling equipment. The CRDMs to be removed were on the outer periphery of the control rod drive matrix. A sonar system was considered instead of video because it would be an advantage in murky water; however, the real-time benefits of visual observation were judged more important. An inflatable plastic bag was conceived to fit over the camera lens and provide clear viewing to at least 5 cm (Carter, et al. 1982).

The plan was workable but had many prerequisites, chief among which was the requirement to remove the missile shields above the reactor vessel refueling canal in order to provide more than the existing 6 m of clearance above the service structure. Removing the 36,000-kg missile shields required the use of the containment polar crane. The problem was that the crane was not scheduled for refurbishment until 1983, and so the pre-head lift examination could not be scheduled until then. In fact, the polar crane was not available until early 1984 (see Section 8.4.1.1).

Such a lengthy delay in so fundamental a characterization task was unacceptable. Consequently, an expedited method of obtaining first-hand, visual data about the core was sought—Quick Look.

Quick Look afforded camera access to the reactor vessel by the simple expedient of removing only the leadscrew inside the CRDM and inserting a 3.2-cm dia., off-the-shelf camera down the 3.8-cm dia. shaft to the core region 12 m below. In addition, a method was devised to disassemble/cut the leadscrews as they were withdrawn by a hoist/trolley. The trolley was located above gaps between the missile shields and replaced the polar crane in lifting the components. This approach had been used at other plants to confirm whether vessel internals had broken bolts (TMI-2 TAAG 1982).

Instead of large-scale plans for a pre-head lift inspection, Quick Look used industry experience and proceeded one step at a time to address obstacles (Cole 1985). Mockups were used to demonstrate feasibility and to practice, and a successful demonstration was made on the reactor vessel in TMI-1, which had not yet resumed operation.

On July 21 and August 6 and 12, 1982, the Quick Look examinations were conducted. Among observers during the inspections, there were many exclamations of surprise, disbelief, and confirmation, which illustrate the diversity of views about core conditions and the effect that hard data can produce. Figure 5-8 illustrates the operation.

Three leadscrews were withdrawn to provide access for the camera and a steel probe (only two of the three locations were successfully accessed). The results were the discovery that a 1.2-m deep void existed in the top of the 3.7-m high core, and that approximately 36 cm of debris lay on top of a "hard stop" reached by a steel probe (Fricke 1982; BNC 1983). Photo 5-1 shows the Quick Look team on top of the reactor vessel service structure and photos 5-2 and 5-3 show views inside the vessel.

Because visibility was very limited, the complete outline of the void could not be determined. The fact that access through the third inspection hole was not successful led to the belief that the only damage was in the center of the core and that all peripherals were intact. Another year passed before clear, comprehensive videos were taken and a sonar device created an accurate map of the core topography.

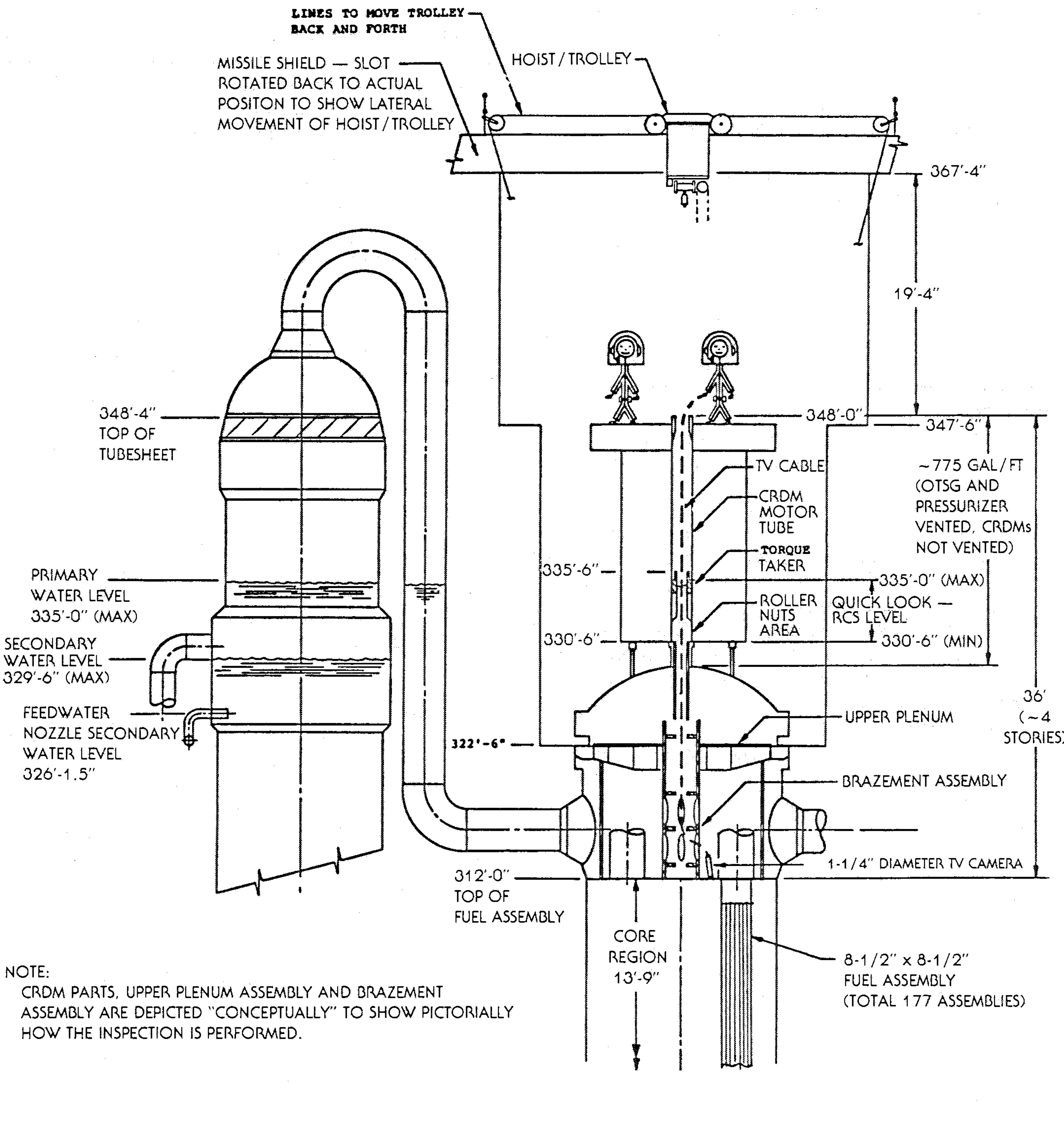

Figure 5-8. Conceptual Arrangement for TV "Quick Look" via Leadscrew Hole

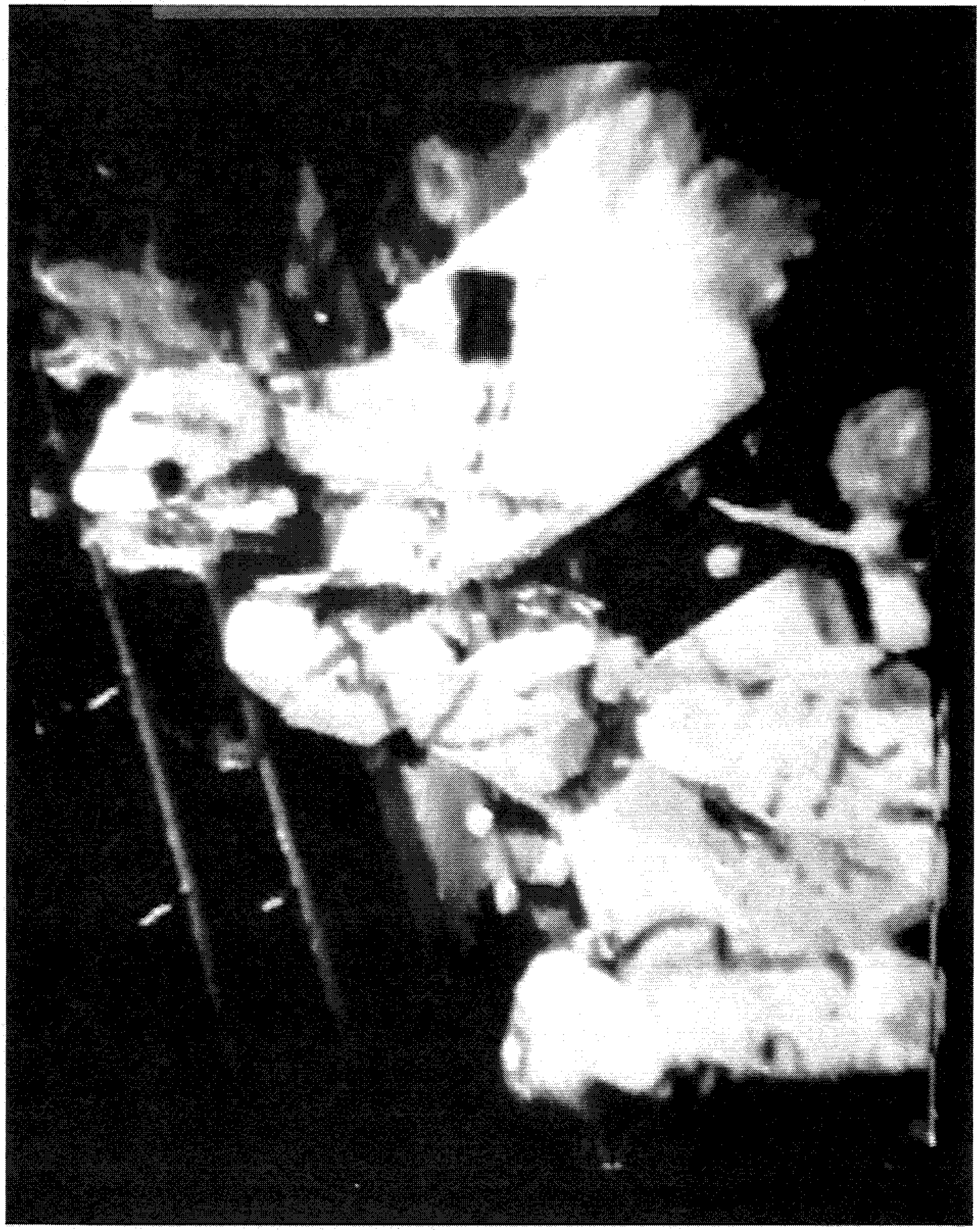

Photo 5-1. Quick Look Team on top of the Reactor Vessel Service Structure

Photo 5-2. View of Debris Bed from Quick Look

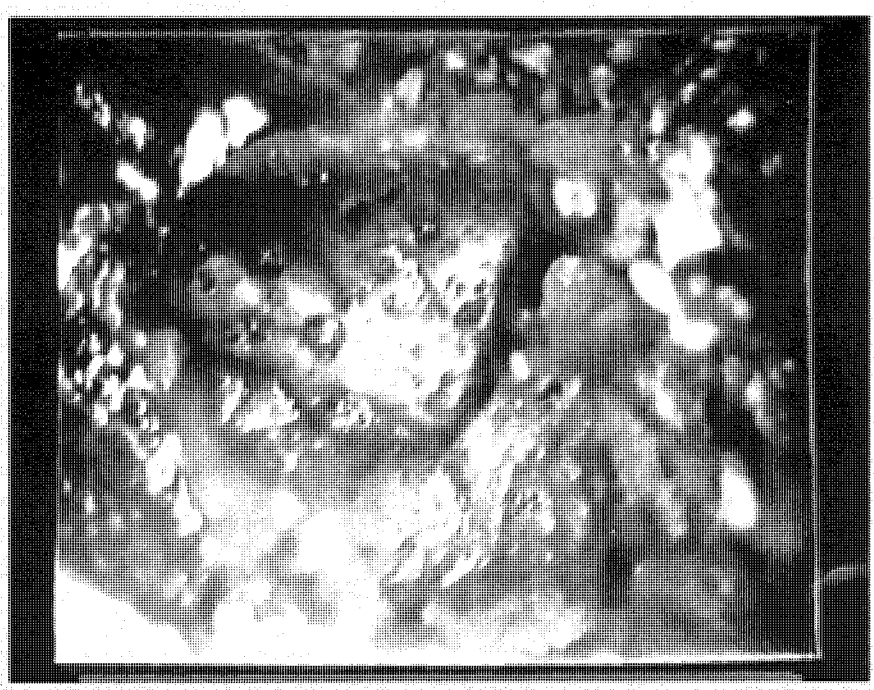

Photo 5-3. View of Debris Bed from Quick Look

Furthermore, the Quick Look results were inconclusive because the observable extent of damage to the upper core fit within one of GEND-007's worst-case predictions of core damage. Without further exploration of other core regions, the data had no real impact on defueling strategy (see Section 8.2.3).

Nonetheless, Quick Look was a major step forward for the cleanup organization. The cost in resources and time was high—27 containment entries (243 personhours spent in the containment) to prepare for and execute the operation. However, the inspection provided solid evidence that the core had been severely damaged, and it showed that the upper reactor vessel internals were essentially intact. Also of importance, it showed that challenging technical work could be conducted in the reactor vessel and it focused the project's attention on the primary task of planning to remove the core debris.

5.4.2 Lower Head Inspection

The success of Quick Look and other video inspections indicated that a camera could also be used to examine the lower regions of the reactor vessel. Although little or no core debris was expected there, a camera examination had been recommended in 1980 (AGNS, et al. 1980). Access could have been gained by cutting a hole in the primary system inlet piping and snaking a CCTV camera to the bottom of the vessel. This was rejected because of the technical difficulties and the high radiation fields in personnel work areas.

The first positive evidence that a mass of debris existed in the lower head came in 1983, an indirect result of work to obtain a radiological profile of the reactor vessel. In this effort, solid-state track recorders (SSTRs) were used in the cavity between the reactor vessel and the biological shield (Gold, et al. 1985; Baratta and Bandini 1985). SSTRs were selected because of their sensitivity and selectivity for neutron detection under adverse conditions, with an excellent signal-to-background ratio.

This characterization effort eventually resulted in an initial lower bound estimate of approximately 1800 kg of debris (equal to approximately four fuel assemblies). The results indicated that significant quantities of core debris lay at the bottom of the reactor vessel, provided quantitative evidence supporting a core relocation, and confirmed and explained the high count rate of the source range monitors (SRMs) in the cavity.[1] However, the conclusion that significant quantities of debris actually existed in the lower head was not fully understood or accepted until verified by later visual inspection.

The estimate was limited to a lower bound because only SSTRs were used. The use of gamma-ray spectrometry in conjunction with the SSTRs would have complemented the neutron dosimetry data. This was proposed, but by the time plans were ready to be implemented, the decay of cerium-144 (which would have provided the signal-to-background measurements) had progressed too far to be of value. This reemphasizes the need to give high priority to timely data gathering (Gold 1987).

An opportunity to examine the region visually came in February 1985, after the upper internals (plenum) had been jacked several centimeters up from the core support flange in preparation for removal (see Section 8.5.2). Raising the plenum created an access path into the annulus region between the reactor vessel wall and the thermal shield, and thence down to the periphery of the lower head. The first inspection was made with a very simple camera/cable system. When that inspection revealed very large quantities of debris, more sophisticated viewing systems were devised that were able to look within the periphery of the lower core support assembly as well.

In addition, samples were taken and water displacement tests conducted in the lower head. A gamma survey of the lower head was conducted by inserting a miniature ion chamber into the vessel from below through the center calibration tube of an incore assembly. Only one of 17 such probes was successful; the remaining 16 incore tubes were either blocked or damaged (Rainish 1985). With all this, the project team was not able to obtain a complete picture of conditions in the region—including those at the bottom surface of the head—until 1989, when the debris was removed and damage to the incore nozzles and tears in the vessel lining were observed.

With the discovery and mapping of between 9,000 and 18,000 kg of core debris in this region, planners had to rethink the defueling strategy for the lower core support assembly and lower head region of the reactor vessel.

5.4.3 Core Stratification Sampling Program

In July 1986, a unique characterization program was undertaken that was to greatly improve knowledge of conditions in the formerly "unknown" regions of the

[1]The SRMs, once correlated to the quantity of debris in the lower head, were later used to track additional material added to the region by the defueling operations conducted above it (Rainish and Fricke 1988).

core. The technique employed was also to play a crucial role later in defueling operations.

The Core Stratification Sampling Program was sponsored by the DOE to obtain samples of the "hard layer" below the upper debris bed and from the regions below it. The motivations for the data acquisition program were a mixture of R&D and operations support:

- Extract representative sample materials from the core region that were of suitable quality and quantity to support research needs. It was especially important to the research community that samples be taken before the original postaccident conditions of the core were altered by defueling operations. One potential defueling method that used a shredder would have destroyed much of the postaccident structures (see Section 8.2.7).

- Provide supporting data on conditions in the lower core support assembly and lower head to answer questions about the extent of material relocation.

- Provide the recovery team with data that would help in defueling operations (Martin 1986).

One existing model of postaccident core conditions (represented in Figure 5-5) indicated that one or more void spaces might exist below the hard crust. Other models predicted a resolidified mass of once-molten material and/or relatively intact fuel assemblies. To explore this region, a method had to be developed that would work in an extremely hostile and uncertain environment, would be relatively simple, and would work the first time.

These requirements led to equipment that was effective under relatively equivalent conditions; i.e., in the mining and geology industries. A commercial drilling machine was selected—a core boring machine of the type used to explore for oil. It was to be the only equipment used in the cleanup that was capable of applying great mechanical force within the reactor vessel. This machine provided an important alternative technique for several later defueling operations.

The core boring machine was tested, modified, and retested. Special drill bits were fabricated. To obtain feedback data on the material while it was being drilled, a computer-based control and data acquisition system was installed.

The machine was able to drill 1.8-m core samples approximately 6.4 cm in diameter. Ten holes were drilled during a month-long operation in the summer of 1986. Three locations were over access holes in the lower core support assembly, which permitted core samples of the lower head region; a lower head core bore was attempted in two of these locations.

The samples were then sent to INEL for analysis. (Surprisingly, some of the material drilled through was so friable it could not be captured inside the core bore casings for extraction.) Figure 8-20 depicts the core boring machine on top of the work platform above the reactor vessel; Figure 5-9 shows one cross-sectional picture of the core that emerged as a result of the sampling program.

The information of most immediate value came from the video inspection of the insides of the access holes and the lower core support assembly as the drill casing was removed. The computer feedback from the drill head also gave some idea of the density and resistance of the material:

- A region of once-molten material estimated to be approximately 3.4 m^3 (approximately 10% of the core volume) existed in the lower central region of the core. It was a solid structure approximately 1.5-m thick in the center and 0.3- to 0.6-m thick at the core periphery, shaped like a funnel extending toward the bottom of the vessel. Fuel assembly rod stubs existed from the bottom of the core to the solidified area.

- The central regions of the lower core support assembly, which was apparently undamaged, did not contain much fuel.

- Major amounts of material were seen in the lower core support assembly periphery, indicating that the migration path had probably been in the periphery (Tolman, et al. 1987).

These data changed the hypothesized accident sequence and added questions that were still unanswered at the end of the cleanup. On the practical side, it did permit a fairly accurate picture of postaccident conditions to be drawn—at least from the point of view of defueling planners. They finally had a good indication of most general conditions to be faced in the future. The picture showed that the most substantial defueling challenges lay ahead.

After the core boring machine was removed from above the reactor vessel, defueling operators tried to break apart the resolidified mass with various long-handled chisels and scoops. When these proved unsuccessful, the core boring machine was reinstalled to continue what it had proven capable of doing—drilling through the mass.

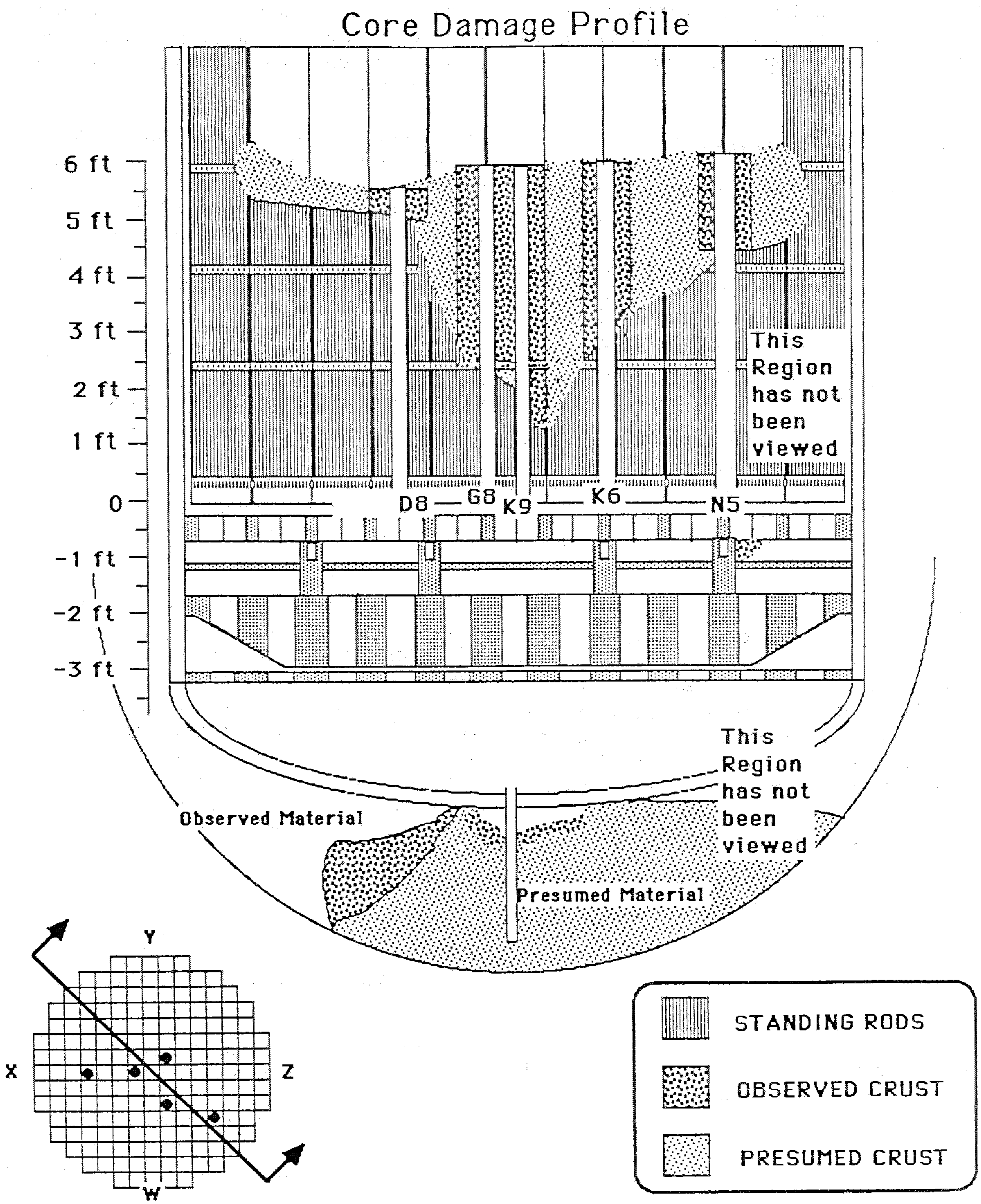

Figure 5-9. Cross-Section of Core with Drill Holes

See Section 8.6.1.2 for a discussion of the "swiss cheesing" operation performed on the resolidified mass and Section 8.6.2 regarding how the machine was used to help disassemble the lower core support assembly.

5.4.4 Core Former Region Inspection

The area behind the core former baffle plates was the last remaining unexplored region in the reactor vessel (see Figure 5-10). Since no evidence of damage to the plates had been observed up to the time when partially intact fuel assemblies were removed, the general assumption was that little or no significant quantity of debris was present behind the plates. This was another example of optimism in the face of repeated evidence that conditions were worse than expected.

In 1987, as the stub ends of the fuel assemblies were being removed, the first notice of damage to the baffle plates was seen. On the east side, near assembly R-7, a large hole and several smaller ones were observed. Fuel debris could be seen in the hole and so reason now existed to suspect it had travelled elsewhere

To explore the region behind the plates, several examinations were made with a fiberscope and a video probe in 1987. These were lowered through flowholes in the top of the core former. Numerous blockages were encountered and high turbidity restricted the field of view. Enough of the region was seen that, when the visual data were analyzed along with a radiation profile, the general outlines of the debris formations could be mapped. Approximately 4,000 kg of core debris were estimated to exist behind the baffle plates.

This information completed the general picture of reactor vessel conditions needed to plan the major defueling operations.

5.5 Other Issues

Several other issues related to knowledge of plant conditions and ensuring a safe cleanup are discussed below.

5.5.1 Preventing Recriticality in the Reactor

The potential of a recriticality at TMI-2 was an issue in all planning and was a consideration in every technical decision that affected core debris or a potential location of core debris. It was a concern because no one could define the postaccident core geometry. And if/when it was known, then no one knew the concentrations and enrichments; consequently, the worst-case geometries could not be ruled out.

The method of preventing recriticality was to rely upon bounding conservatisms. In the reactor vessel, this meant poisoning the core debris with a soluble boron concentration great enough to ensure subcriticality under all possible postaccident- or defueling-related conditions—a very conservative approach. The decision to use a high concentration of boron was to have many impacts on defueling and water processing operations.

Before the accident, the boron concentration in the reactor coolant system was approximately 1050 ppm (see Section 3.2.4). Within several days of the accident, the minimum concentration was raised to approximately 3000 and then to 3500 ppm, where it was maintained until 1984 (Stratton 1985). After the accident, there was no evidence of a recriticality, even at boron concentrations that may have been as low as 2300 ppm.

The 3500 ppm concentration was derived from criticality analyses that focused on best-estimate and worst-case geometric configurations (Westfall, et al. 1979; Barr, et al. 1979; Worsham, et al. 1982; Thomas 1982). Consideration was given to geometrical arrangements in the core with combinations of damaged and undamaged fuel at different enrichments from the core average to 2.96% for the batch of fuel, as well as out-of-core accumulations. The postaccident analyses accounted for potential damaged-fuel-and-moderator configurations more reactive than the original fuel assembly lattice and for a probable reduction in control rod poison effectiveness (Knief undated).

In 1983, the 3500 ppm number was reverified as part of planning for defueling operations, which would disturb and reconfigure the core. Three approaches were considered:

- Continuing use of an infinite poison, which would maintain subcriticality under all core configurations
- Systematically reviewing planned activities, defining credible core configurations, and determining poison requirements for maintaining these configurations subcritical
- Using design and procedural measures to preclude fuel configurations that were potentially more reactive than those previously analyzed (Rider 1983).

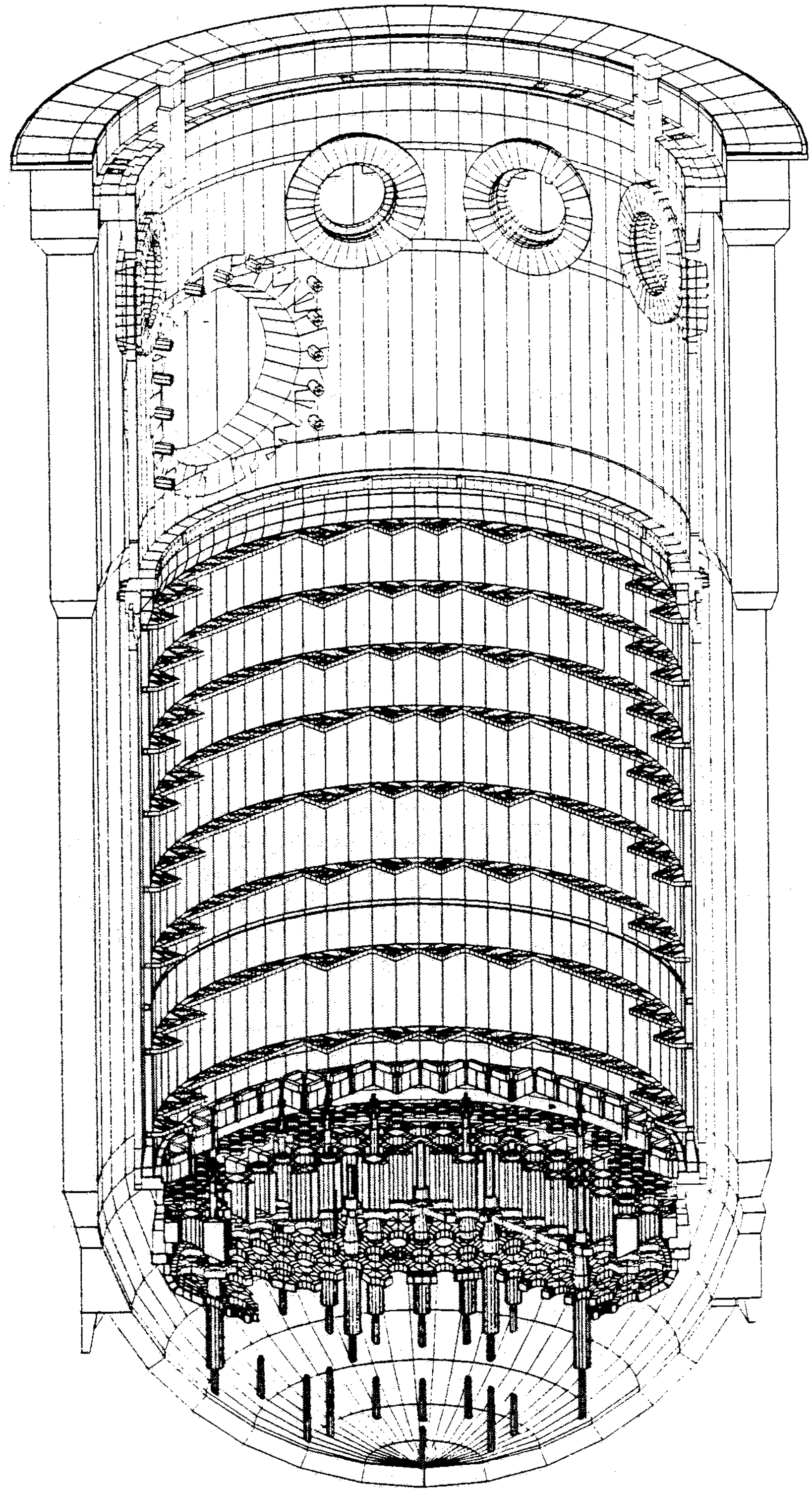

Figure 5-10. Reactor Vessel with Baffle Plates Removed and LCSA Cut Out

A policy decision was made in 1984 that the boron poison should be increased to a concentration that would protect against all conceivable conditions during defueling (Stratton 1985). The minimum concentration of 4350 ppm boron was chosen with the aid of a degraded core model that was deemed to be conservative enough to cover all situations.

This number was also chosen before a video examination of the lower reactor head region confirmed that 10 to 20% of the core had relocated there. This new information required a modification to the degraded core model and additional analyses, but no change in boron concentration to ensure subcriticality. The calculated lower limit of 4350 ppm boron was then translated into an operating range of approximately 5000 ppm following a thorough evaluation of the potential for boron dilution.

Concerns about boron dilution arose from the possibility that unborated or underborated water might unintentionally be injected into the reactor vessel. This could provide an environment in some or all of the vessel where a recriticality could occur. The underborated water could come from several sources; e.g., demineralized water used in the plant or "slugs" of water trapped in pipes since the accident. Three alternatives were considered for guarding against this:

- Block the inlets to the vessel with artificial barriers; e.g., block the two hot legs, four cold legs, and two core flood nozzles at the reactor inlet
- Isolate known sources of underborated water; e.g., the demineralizer water tank
- Isolate the reactor vessel with existing components.

Affecting the choice were the regulatory commitment to provide double isolation against boron dilution, the need to preserve the makeup path to the vessel, and the need to allow other plant operations to continue unimpeded.

After lengthy study, project management chose to isolate the reactor with existing components (GPUN 1985). This required extensive work to identify flowpaths and provide double-isolation with existing barriers; e.g., close valves, remove spool pieces, separate water volumes by height. Once identified, these barriers were verified regularly. Mitigation procedures following any detection of boron dilution were also established.

The practical implications of the decision to use high boron concentrations increased the difficulties of many defueling operations. For example, in the use of the plasma arc torch, the high boron concentration required additional design features in order to add unborated water in the immediate area of arc. Many analyses were required to prove that it was safe to have a small volume of unborated water injected near fuel debris. In addition, the reservoir of unborated water that could accidently drain into the vessel had to be limited to less than 11 liters by modifications to the design of the torch coolant system (GPUN 1987; GPUN 1988a). Furthermore, electrical-type cutting processes were made very difficult by the high conductivity of boron.

For information about the evaluation that established the post-defueling critically safe fuel mass limit of 140 kg, see the *TMI-2 Defueling Completion Report* (GPUN 1990).

5.5.2 Pyrophoricity

The *TMI-2 Programmatic Environmental Impact Statement* (US NRC 1981) had suggested that pyrophoric materials might be present in the core debris and could be a safety concern during defueling. The principal element of interest at TMI-2 was zirconium, along with its pertinent compounds and alloys.

By 1984, laboratory tests had been conducted on zirconium-bearing core debris and ignition experiments performed on nonradioactive simulated core debris. No pyrophoric potential was discovered (Baston, et al. 1984). This result, in combination with literature studies and computer-generated accident analyses, allowed the project team to proceed with defueling plans although data continued to be reviewed to ensure that pyrophoricity was not a problem.

5.5.3 Reactor Vessel Integrity

The structural integrity of reactor coolant system components was evaluated in 1980 (B&W 1980). Because the reactor vessel lower head was assumed to have been submerged during the accident and not exposed to high temperatures, it was not expected to have suffered any damage. In February 1985, a video examination of the region showed that the basic assumptions were incorrect. This raised issues of debris bed coolability, melt progression, and the potential for bottom head failure. Once the issue was raised, it persisted in the form of repeated requests to prove that the lower head had not been breached and that a significant quantity of fuel did not lie below it on the floor of the containment basement.

5.5.3.1 Lower Head Integrity Studies

An early study indicated that, although the lower head was intact, temperatures may have been high enough to cause local deformations. Since the incore nozzles were the weakest component and may have suffered damage, the study recommended limits on forces that could be applied to them (Nitti, et al. 1985).

Following this study, a video inspection revealed that 10 to 20% of the core had relocated to the lower head. This inspired an aggressive and varied characterization campaign during 1985. It included a neutron flux profile, video inspections, wire probing of the instrument penetration tubes, gamma-scanning of the lower internals, probing of the debris bed with high-velocity water, and the retrieval of debris samples.

The conclusion reached by all these efforts was that the incore tube penetration nozzles might have been damaged; however, the molten debris had refrozen and may have plugged some of these nozzles. Little or no thermal damage to the vessel liner was predicted (Cronenburg, Behling, and Broughton 1986; EG&G 1987; GPUN 1988b). As a result of these studies, the project team took procedural measures to minimize any chance that a heavy load could fall on the region during defueling operations.

The actual damage—which did not threaten the vessel's integrity—consisted of a series of cracks in the 0.95-cm stainless steel lining of the vessel and damage to many incore nozzles (see photo 5-4). Damage to approximately 25% of the incore nozzles was observed as the lower head was defueled; the tears in the liner were seen and measured in July 1989. Although a temporary limitation was placed on defueling operations, no significant changes occurred.

A task force evaluation of the cracks made the following conclusions and recommendations:

- The cracks most likely penetrated the cladding (liner) in some areas. They were not necessarily associated with the nozzles.

- Samples and laboratory analyses would be used to determine the cause; but the cracks or "hot tears" were most likely caused by the high temperatures and thermal stresses during the accident.

- There need be no impact on defueling, although caution when working around the nozzles was necessary because the probability of a leak developing could not be precluded.

- Additional cracks may be found (Potts 1990).

The cracks were more fully evaluated in the sampling program that followed the completion of defueling (see Section 5.5.3.2).

5.5.3.2 TMI Vessel Investigation Project

A major research and development effort was conducted in the reactor vessel after the completion of defueling. This was the TMI Vessel Investigation Project, sponsored by the NRC and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD).

The project had its origin in the spring of 1987, when GPU, EPRI, DOE, and the NRC Office of Research (the GEND group) met to discuss TMI-2 research activities that could be useful but were not planned. Their recommendations were presented to the NRC Commissioners in April 1987.

The group emphasized the fact that, with the end of the cleanup nearing, plans were being developed to layup systems for long-term storage. In addition, much of the involvement of outside agencies was coming to an end. Thus, a unique opportunity to gather impossible-to-recreate data might be loSt. Information about the kinds of thermal and metallurgical stresses the reactor vessel experienced during the accident would be of great value in understanding severe accidents and source term phenomena. The questions were: How close did the vessel came to failure? Why did it come that close? Why did it not come closer?

As a result of this urging, the NRC Commissioners authorized the NRC Office of Research to take the lead in organizing a $7-million reactor investigation project. The project eventually included nuclear agencies from 10 nations of the OECD (USA, Belgium, West Germany, Finland, France, Italy, Japan, Sweden, Switzerland, and United Kingdom). EPRI assisted with both funds and technical support.

The objective of the project was to extract samples of the inner surface of the lower head of the reactor vessel. The vessel wall was 13.7-cm thick, with a 0.95-cm thick stainless steel liner and a 12.7-cm thick carbon steel shell. Interest in the project increased even more when the tears were discovered in the stainless steel liner in 1989.

To take the samples, a technique of metal disintegration machining (MDM) was selected. This slow but precise technique was selected based on its successful use at

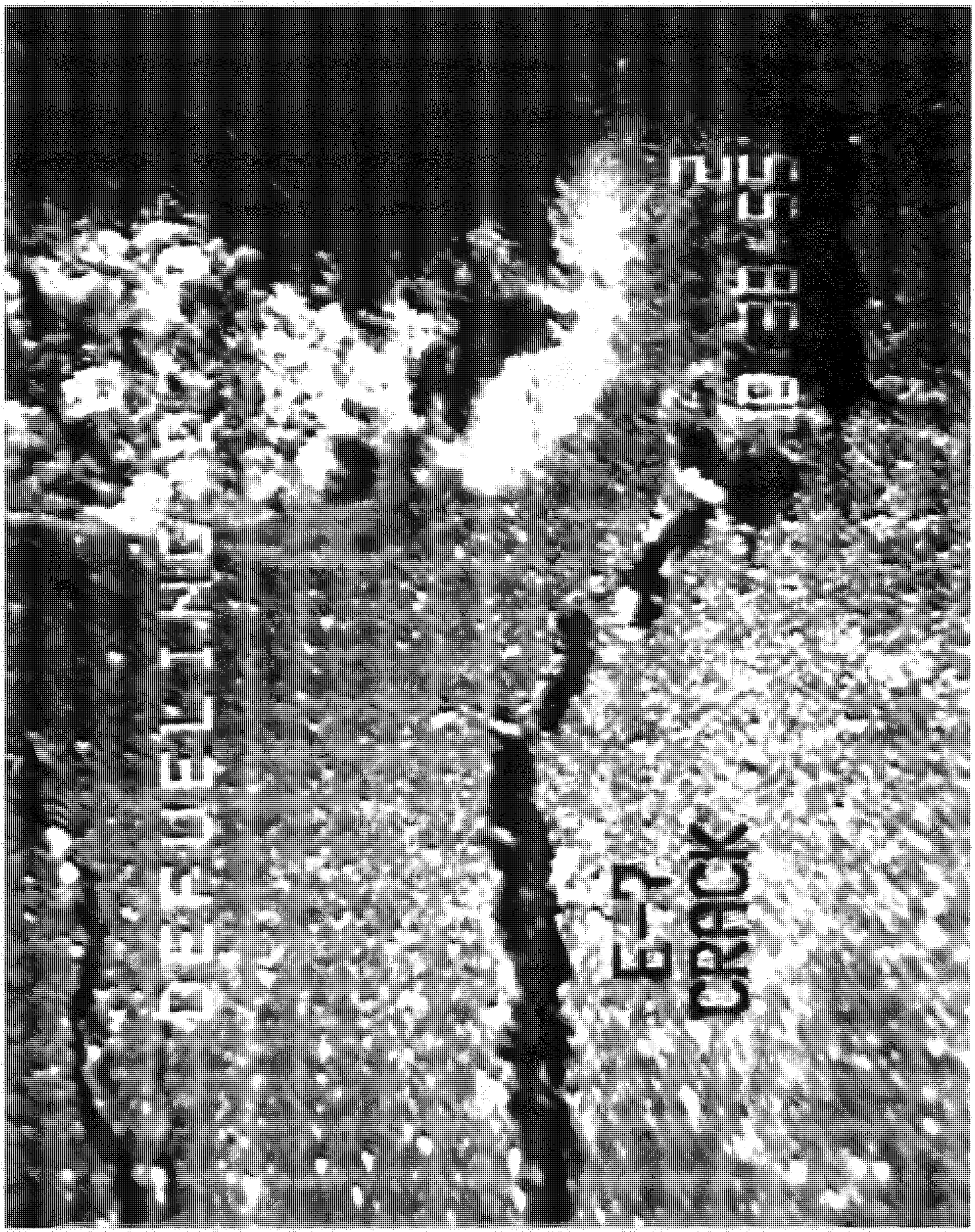

Photo 5-4. A Crack in the Lining of the Lower Head of Reactor Vessel Near Incore Nozzle E-7

other nuclear sites. The technique used a spark erosion principle to disintegrate metal by sending a series of electrical arcs from a cutting electrode to the base metal. It had previously been evaluated as a potential cutting method for disassembling the lower core support assembly during defueling but was rejected because of the large quantity of nonconductive material that might be encountered. Its use on the lower head was acceptable because any local nonconductive material was to be removed before cutting.

In February 1990, the samples were taken from incore penetrations and other areas. The lower head was cleaned of fuel debris and vacuumed to remove loose debris. Two kinds of samples were taken:

- **Incore Penetration Samples**—Any incore instrument string remaining after defueling was first cut and removed, and then an abrasive saw was used (as necessary) to cut the incore nozzle so that only 5 to 10 cm. protruded above the vessel surface. An expansion seal plug was inserted in the incore pipe to seal both it and the 0.12- to 0.24-mm annulus between the outside diameter of the pipe (the plug also retained the pipe after the retaining weld was removed). The MDM head was then lowered around the incore and a wedge-shaped piece, including the upper incore nozzle and weld, was extracted.

- **Other Samples**—Wedge-shaped samples were also extracted from the curved surface of the vessel wall using only the MDM head to cut the sample.

The program was expanded to include several incore nozzles like R-7 and several damaged incore guide tubes, which were removed earlier for lower head access. Analysis of the results was conducted in the late winter/ early spring of 1990.

5.5.4 Special Nuclear Materials

Corresponding to the plans for defueling and fuel shipment was the issue of special nuclear materials (SNM) accountability. The challenge was to meet the law (10 CFR Part 70) requiring accountability for all plutonium, U-233, and U-235 in the original TMI-2 plant inventory. The law was intended to prevent diversion of SNM for weapons. Any changes to the law for TMI-2 could have involved both the scientific and political communities. However, considering the condition of the TMI-2 core, an accountability program that met the letter of the law would have had many impacts in terms of personnel exposure, technology, and time.

There were two aspects of SNM accountability program at TMI-2: 1) accountability for fuel shipped from the site; 2) accountability for residual fuel remaining in the plant after the cleanup was complete. The solution to the first led into the second.

The SNM program was first required for fuel shipping since the law requires an accounting of uranium and plutonium as part of the paperwork for shipping. In addition, it is a reportable item on an annual basis to the NRC. GPU and the DOE had to resolve the issue of how the core debris could be legally shipped considering its condition.

The choices were to count the items as they were loaded, weigh the items, gamma-scan the debris canisters at TMI-2, gamma-scan them at INEL, or perform neutron interrogation at INEL. The first solution selected was to use shipping weight, but this would only provide an approximate inventory (probably between 120–130% of the actual fuel weight because of the addition of core structural material).

As a result, GPU, DOE, and NRC agreed on a novel approach that was acceptable to the NRC and yet not too burdensome to the project team. At the end of defueling, the project team would conduct a post-defueling survey to determine the quantity of SNM remaining in the plant. The total quantity shipped to the DOE for storage would then be stated as the difference between that existing at the time of the accident and that shipped off site for analysis, decayed away, or remaining in the plant (Porter 1985).

To conduct the post-defueling survey for SNM, neutron interrogation, gamma-ray spectroscopy, gross alpha scanning, and visual inspection and sampling techniques were used (Haghighi 1989; GPUN 1990).

5.5.5 Characterization for Decontamination

The decontamination work at TMI-2, especially in the containment, required innovative techniques to characterize sources while at the same time minimizing personnel exposure. One way to accomplish this was to define and treat surface sources that were important contributors to collective dose.

Fast Sort Technique. The challenge in the containment was especially formidable because of the high ceilings (12–13 m) and the complex overhead space, which was filled with conduits, pipes, cable trays, ventilation ducts, and steel structures. Two related approaches were considered to prioritize the work in order to provide the maximum reduction in personnel exposure:

- **Directional measurements with a high angular resolution detector**—A gamma camera developed at Chalk River Laboratory was evaluated for this approach. It was a heavily shielded, 35-mm camera body with a "pinhole" lens. Intense point sources were easily imaged by this high-resolution system; unfortunately, it lost contrast when superimposed on fields produced by distributed contamination. Field tests suggested it was not the fastest or most convenient method for locating key sources.
- **Directional measurements with low angular resolution detectors**—A commercial, tungsten-shielded, Geiger-Mueller detector was tested—the Eberline HP 220A probe. Initially, the probe had too little angular resolution, which resulted in overlapping fields of view and the counting of the same source more than once. However, with the addition of a conical collimator to narrow the angle of resolution and shielding, the instrument proved to be very capable.

The Eberline HP 220A probe provided rapid and accurate directional measurements in contaminated spaces. With this fast sorting technique, worker exposure was reduced, surfaces could be prioritized for detailed characterization, and exposure reduction efforts could be better planned (EPRI 1986; Distenfeld, Brosey, and Igarashi 1989).

A variation of this HP 220A probe was developed to measure radionuclide penetration in the containment basement concrete. Referred to as "Diver", the probe was encased in extra shielding and mounted in a waterproof, thin-walled stainless steel box. It was deployed on a suspension cable and orientation pole and lowered against either a basement wall or through water onto the floor.

Beta Surveys. To address the challenge of rapidly surveying the beta fields frequent at TMI-2, another commercial probe was modified—the Eberline R07. Commonly used to measure mixed beta/gamma fields, the R07 was modified to increase its field of view for beta particles. Thus, a worker entering a contaminated area had only to point the probe in two opposite directions to quickly reveal the beta/gamma general area exposure rates.

TLD Pseudo Cores. The containment dose reduction effort underway in 1983 (see Section 7) identified and reduced many major dose contributors. To prioritize the remaining sources, a TLD pseudo core was developed to take beta radiation measurements from the building floors in lieu of a less cost-effective and non-ALARA concrete sampling program. A TLD was placed facing downward on top of a 2.5-cm high plastic ring so that the measurements would mimic those of a concrete core.

Sixty pseudo cores placed on El. 347' revealed that the floor (and coatings) was a major and unacceptable contributor to the general area dose rate. Used on many other surfaces, the pseudo cores allowed dose reduction work to be accurately analyzed and prioritized, with minimum exposure to the personnel obtaining the measurements (Vallem, Distenfeld, and Peterson 1989).

5.5.6 Onsite Analysis Facilities

In contrast to the relatively standard types of investigative techniques used in normal nuclear plant operations, hundreds of diverse methods were required at TMI-2, ranging from customized computer programs to sophisticated radiochemical analyses to biological research.

For several years after the accident, laboratory facilities at TMI-2 were very limited; however, immediate analyses of chemical samples were needed. The turnaround time to analyze important samples was frustratingly slow. The project team had to rapidly construct or obtain new facilities to address specific analysis needs. Offsite facilities had to be relied on to provide vital supplementary work that could not be performed on site and complex analyses requiring the capabilities of a large, sophisticated laboratory (Urland and Babel 1990).

Several onsite laboratory facilities were of importance:

- **Nuclear Sample Sink**—The original nuclear sample sink for the TMI-2 reactor coolant system was in a shared facility in the Unit 1 auxiliary building. Concerns over the spread of accident-related contamination, high radiation readings (which affected Unit 1 work), and an administrative separation of the two units led to the construction of the TMI-2 temporary sample sink in the fuel handling building in 1979 (see Section 3.5.1). This was made into a permanent facility in 1984, when all TMI-2 sample lines were routed to it.

- **Plant Chemistry Laboratory Facilities**—The Unit 1 chemistry laboratory performed radiochemical analyses for Unit 2 before the accident. In 1979, the TMI-2 chemistry laboratory was only equipped to handle secondary-side, nonradioactive samples. Consequently, a chemistry laboratory facility consisting of several mobile trailers was set up in 1979. In 1980, the facility was upgraded and a supplementary laboratory was installed in the fuel handling building to provide rapid analysis of submerged demineralizer system water in 1982.

- **Mobile Radiochemistry Laboratory (MRL)**—To address the problem of insufficient onsite laboratory facilities, MRL was offered to the project through the DOE (see Section 3.5.2). The facility greatly increased the number of samples analyzed on site and reduced the turnaround time. MRL consisted of two mobile trailers—one was a counting lab and office, the other was a general radiochemistry lab. The ability to analyze TRU and 10 CFR Part 61 radionuclides plus high-activity samples (up to 5 R/h) allowed the other TMI-2 analysis facilities to concentrate on analyzing routine samples. A high-priority sample could be analyzed in two weeks. MRL was on site from 1982–1987 (Burton 1981).

- **Gamma Spectroscopy Facilities**—Although most gamma spectroscopy measurements were performed in situ, a counting trailer was developed on site after the accident and provided support throughout the cleanup. Likewise, the NRC temporarily provided a portable van equipped for performing measurements for the first few months after the accident.

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WASTE MANAGEMENT

6.1 Overview

From the day of the accident until the end of the cleanup, the project team was forced to manage radioactive wastes on a scale and of a type without precedent in the nuclear power industry. For several years after accident, everything seemed related to waste management. By far the most demanding waste management challenge was the highly contaminated water and its related solid wastes. Where to store it? How to clean it? How to dispose of it?

Cleaning up produced enormous quantities of contaminated equipment and trash, hundreds of processing vessels loaded with varying concentrations of radionuclides, and over 7.5 million liters of mildly radioactive water. In addition to managing this waste within a normal context that demanded worker and public protection, the shortest possible schedule, and financial restraint, the project also had to face a difficult political climate. This climate included changing federal and state regulations regarding waste disposal, ambivalence in state and local governments regarding TMI-2 waste, and a sometimes hostile local populace.

Most radioactive trash and solid decontamination wastes were handled in ways similar to those used at other nuclear power plants, but on a much larger scale. The control of radioactive gases was only a substantial issue until the containment was vented of approximately 46,000 curies of krypton-85 in 1980 (see Section 3). Managing the contaminated water was not so straightforward.

The water was initially distributed in the containment basement, reactor coolant system, auxiliary building sumps and tanks, and over the lower elevation floor of the auxiliary building. It and the associated high radiation fields prevented system maintenance and hindered cleanup work. The existing plant systems were unable to process any of this water, which contained cesium-137 concentrations initially ranging from 1 to over 100 μCi/ml. Consequently, in the first few years, a major portion of the project's resources were spent on water management. Figure 6-1 depicts the magnitude and general history of this process.

Institutional difficulties greatly complicated the matter. As far as the public was concerned, the mention of any more radioactivity reaching the environment was reason to intervene. The most significant public intervention was a suit filed by the City of Lancaster to prevent the release of TMI-2 water to the Susquehanna River, even if this water met all regulatory restrictions. As a result of an out-of-court agreement signed by the NRC and Lancaster in early 1980, the discharge of "accident-generated" water was not permitted. Over 7.5 million liters of processed water eventually fell into this category (see Section 6.2.4). Thus, the cleanup work was destined to be water-bound—a consequence that, although not prohibitive, ensured that water management at TMI-2 would require operations managers to constantly juggle volumes of water to support or permit other cleanup activities.

Arranging to dispose of the captured radioactive wastes created by the accident or generated during the cleanup required extensive negotiations between the project managers, DOE, NRC, and state-regulated disposal site operators:

- Until the publication of 10 CFR Part 61 (Licensing Requirements for Land Disposal of Radioactive Waste) and Part 20.311 (Transfer for Disposal and Manifest) in 1982, the regulations affecting shipment and disposal of radioactive waste were not fully defined regarding waste concentrations and forms. It took several more years for the exact methods for complying with these regulations to evolve.
- Shortly after the accident, the Richland, WA and Barnwell, SC commercial low-level burial sites were closed to any TMI-2 accident-related wastes because

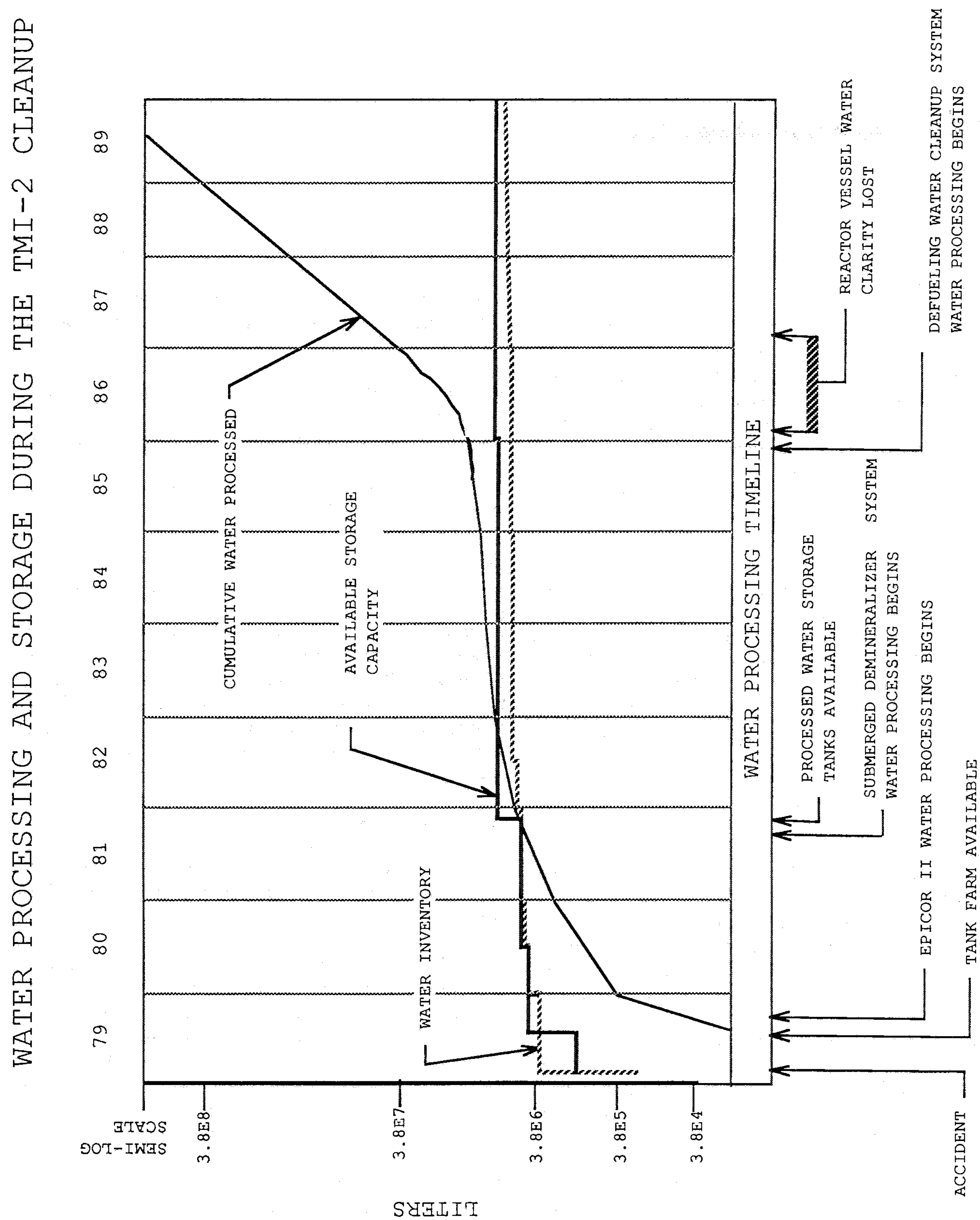


Figure 6-1. Water Processing and Storage during the TMI-2 Cleanup

of concerns about the volumes that might be produced. Later in 1979, the site in Richland was reopened; the Barnwell site did not accept TMI-2 accident-related wastes until 1987.

- Wastes that exceeded commercial burial ground criteria for forms or content were even more difficult to dispose of and required extraordinary measures by the DOE and NRC. Eventually, the DOE agreed to take the waste for R&D work and temporary storage to ensure that TMI did not become a de facto long-term waste repository (Snyder 1981).

The volume of waste removed from the TMI-2 site is depicted in Figure 6-2.

In spite of the institutional obstacles, several technical factors made waste management at TMI-2 considerably less difficult than it might have been. These factors had an important influence on the decision-making process and existed because TMI-2 had operated commercially for only three months. They included two empty spent fuel pools, an unused steam generator chemical cleaning building, and a relatively limited quantity of fission or activation products (see Section 2.1.3).

In addition to using the existing uncontaminated structures, many new facilities were built. Several others were planned but never constructed. Figure 6-3 shows the locations of both conceived and constructed TMI-2 waste management facilities. Figure 6-4 shows the overall cleanup of TMI-2 from a waste management perspective.

6.2 Water Processing and Disposal

The emergency stabilization measures of the first few months were successful and the water was contained in tanks, systems, and sumps. Initially, the most important issues were tracking the water volumes and minimizing any increase because of the dire shortage of spare tankage. When these were accomplished, the challenges were how to collect the fission products in the water for safe handling, storage, reuse, and disposal. Table 6-1 shows the series of water management campaigns to achieve this goal.

These "campaigns" can also be viewed in four phases:

- First—The project team acted to stabilize the situation by isolating and controlling the water wherever it existed; transferring and cleaning it when possible; and analyzing its characteristics to increasing levels of accuracy. Many of the actions taken during this phase are described in Section 3. This phase lasted until the project team felt that control had been established.
- Second—A large-scale cleanup was needed to process the water and capture the radionuclides. Over a 2-year period, the project team constructed two new water processing systems (EPICOR II and the submerged demineralizer system), over 3.7 million liters of new tankage, and 1,120 m^3 of new solid waste storage space.
- Third—The water had to be maintained in an acceptable condition for reuse or disposal. This meant continued operation of the two new systems plus constructing a third new system (the defueling water cleanup system) to process the water used during defueling.
- Fourth—The processed water had to be disposed of. For this, an evaporation system was installed to begin operation at the end of the cleanup.

The water processing challenges facing the project team and the how they were met are shown in Table 6-2. Operationally, this resulted in radioactive water being processed as summarized in Figure 6-5.

6.2.1 Processing Lower-Activity Water

The multiple volumes of radioactive water in the auxiliary building were a major obstacle to complete control of the plant. Access was needed to the systems and equipment in that building to ensure control of the reactor coolant system and to establish a base for entering the containment. The project team's response was to immediately build a new, three-stage, ion-exchange processing system to supersede the existing two-stage version brought on site right after the accident (EPICOR I) and described in Section 3.6.2.1.

The resulting system—EPICOR II—was an excellent example of what can be accomplished in a short time under evolving circumstances. It grew out of the urgency of a crisis and evolved into an important system that provided relatively trouble-free and consistent operation throughout the cleanup.

Disposing of the EPICOR II processing vessels was difficult, but the eventual solution—development of a high integrity container (HIC) overpack that could be buried at Richland—was a major accomplishment. In addition,

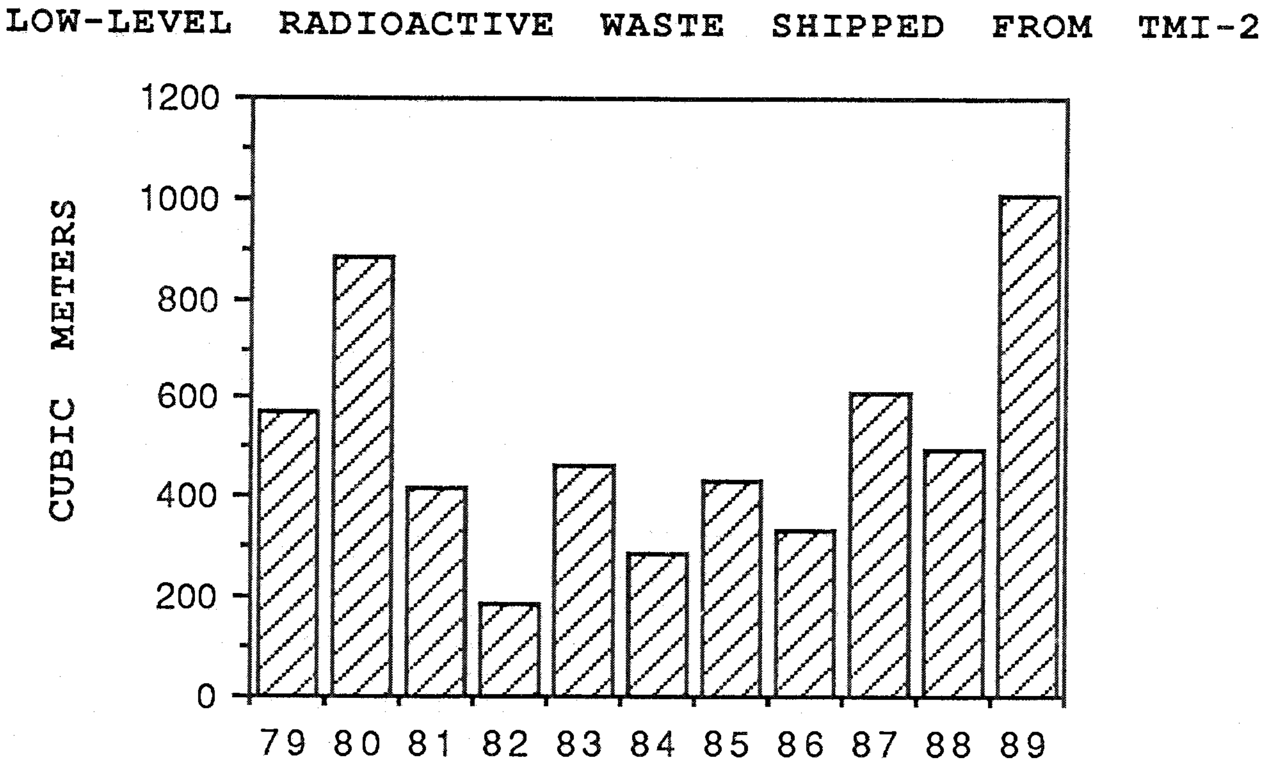


Figure 6-2. Low-Level Radioactive Waste Shipped Offsite During the Cleanup

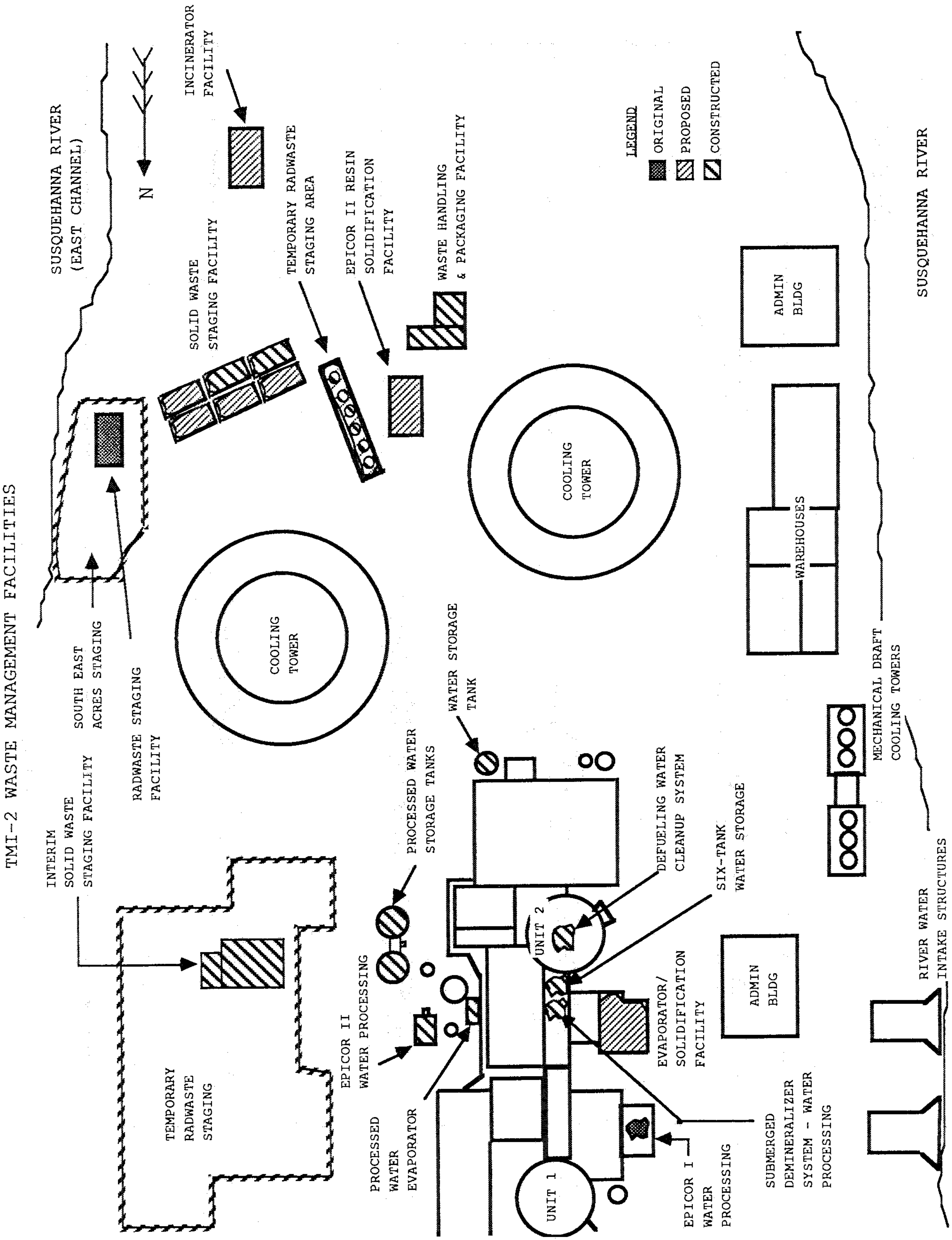


Figure 6-3. TMI-2 Waste Management Facilities

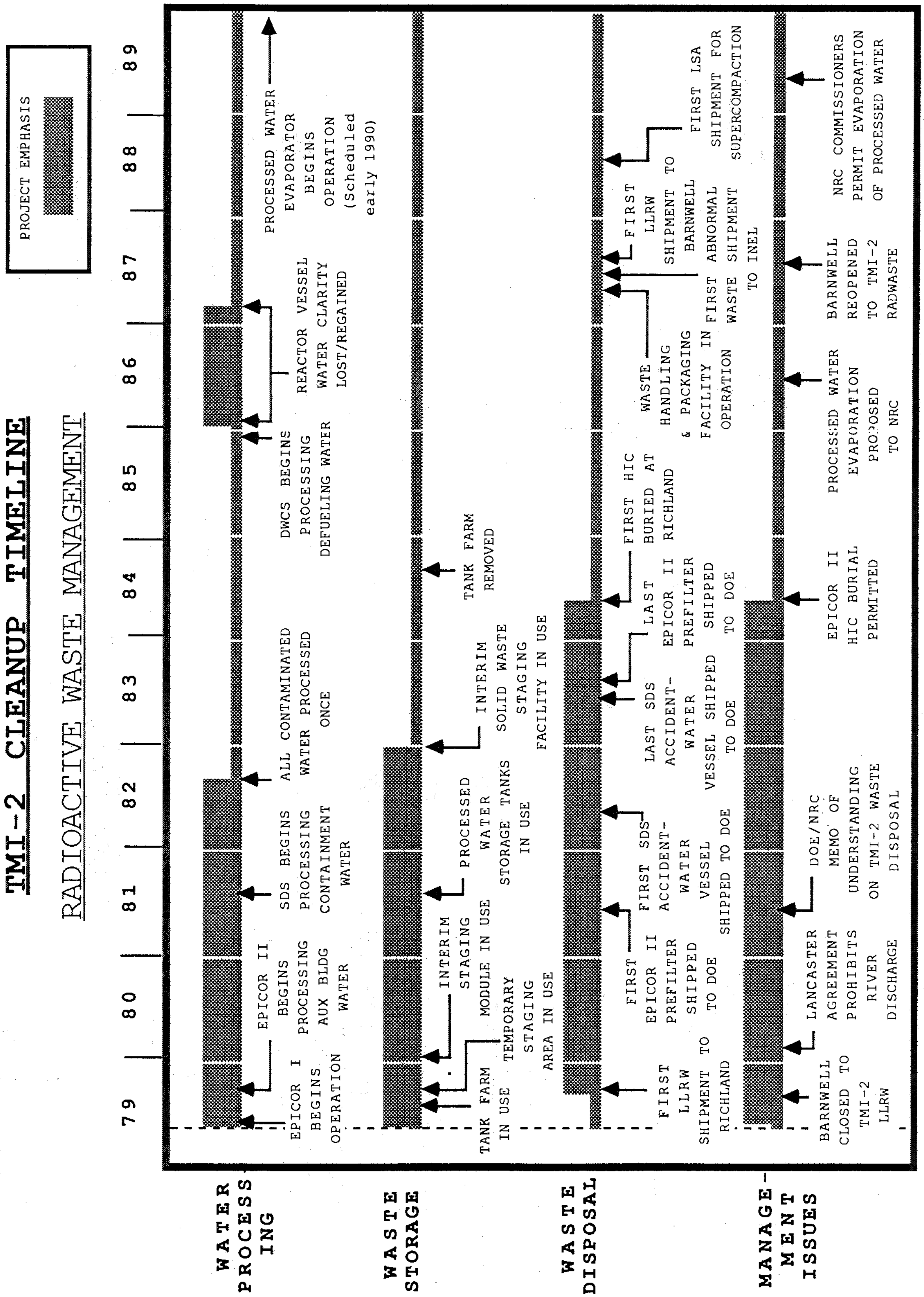


Figure 6-4. TMI-2 Cleanup Timeline: Radioactive Waste Management

Table 6-1. TMI-2 Water Management Campaigns

PERIOD	EMPHASIS
1979	Contain contaminated water; process and release nonaccident-generated water
1980	Process auxiliary building water
1981-82	Process containment basement water
1982-85	Process reactor coolant
1985-90	Reprocess defueling water
1990-	Dispose of processed water by evaporation

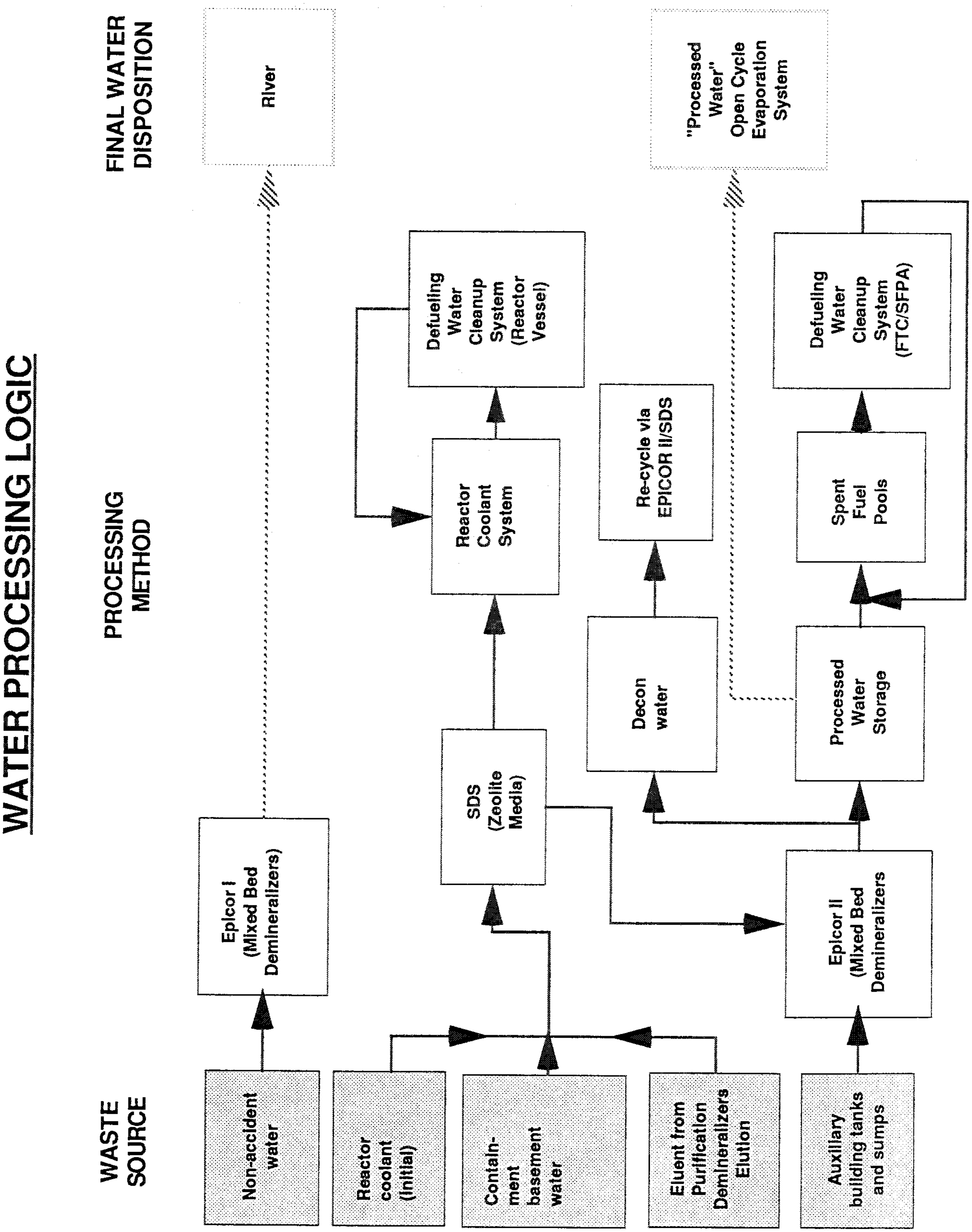


Figure 6-5. Water Processing Logic

Table 6-2. Liquid Waste Processing Challenges

NEED	OBSTACLE	SELECTED	REJECTED
Gain control of auxiliary building • Prepare for containment entry • Capture curies • Ensure safety	Water in auxiliary building tanks, sumps, floor	Mixed bed three-stage ion exchange system (EPICOR II)	Two-stage ion exchange system (EPICOR I)
Establish acceptable working conditions in containment • Capture curies • Ensure safety • Reduce radiation fields	Highly contaminated water in containment basement	High-capacity/low-flow zeolite demineralizer (submerged demineralizer system - SDS)	• Closed-cycle evaporator • Solidification • EPICOR II alone • Onsite storage • Processing w/converted shipping cask • Fluid bed dryers/ calcinators
Maintain and control reactor coolant system • Conduct core examination • Prepare to defuel • Capture curies	Water in reactor coolant system following accident	High-capacity/low-flow zeolite demineralizer (SDS)	Reconfigure EPICOR II
Maintain acceptable defueling conditions • Visibility for defueling • ALARA for defuelers	Water in reactor coolant system after initial processing	High-flow zeolite demineralizers and filters (defueling water cleanup system)	Reconfigure SDS
Dispose of processed water • Ensure safety • Prevent TMI-2 from becoming a long-term storage facility	Lancaster Agreement of 1980 prevented discharge of water related to accident; public opposition	Open-cycle evaporation of 2 million gallons of processed water held in storage	• Solidification • Discharge to river

a processing vessel made of Ferralium became available on the commercial market and was qualified as a HIC at TMI-2. Radioactive accident water in the nuclear power industry can now be processed without many of the disposal concerns that plagued the TMI-2 cleanup.

The design and location of the EPICOR II system were dictated by the available experience with ion exchange technology and the availability of the unused steam generator chemical cleaning building located on the east side of, but not attached to, the Unit 2 auxiliary building. The facility contained two large tanks, unused floor space, a sump, and a seismic bathtub foundation, which would minimize the consequences of spills or releases. Various mixtures of organic and inorganic resin beds were used in a series of three processing vessels—the initial version of EPICOR II used two 1.2- by 1.2-m vessels followed by a 1.8-by 1.8-m vessel. The *TMI-2 Waste Management Experience* (Deltete and Hahn 1990) provides details on the system and volumes of water processed.

A shielded transfer bell and flatbed truck were used to transport expended vessels to a storage area. Expended vessels were temporarily stored in radwaste staging areas not far from the building (see Section 6.4). The liquid effluent from the system was essentially nonradioactive except for tritium, which was unaffected by the ion exchange process. The general design of a 1.2- by 1.2-m vessel is shown in Figure 6-6. Photos 6-1 and 6-2 show views of the system and building.

As noted, the system was based on existing technology; however, there were several unique aspects (Hofstetter 1987):

- Layered resins—The layering provided selective removal of different nuclides and also allowed multiple removal; i.e., by adjusting the mix, removal could be made from water from different sources.
- Proprietary resin mixes—This aspect slowed some development and modifications of the system because of restricted access to formulas. Also, the constantly changing recipes allowed for little preplanning.
- Contractor operation—This helped by freeing cleanup staff to perform other essential work. The contractors left the site in 1982. After that, the project team assumed responsibility for the system operation and its engineers were able to specify the resin mixes.

Installation work started on EPICOR II in early April 1979, as soon as it was apparent that the EPICOR I system would not be adequate for processing most auxiliary building water. The construction was slowed by the high dose rates in the auxiliary building and by the demands on the construction crews to work on other tasks. Working around these obstacles, EPICOR II was completed and ready for startup testing by the end of May 1979.

On May 25, 1979, the NRC issued a statement requiring an Environmental Assessment of the system and qualification of the system operators by the NRC before the system would be allowed to process water. The NRC also prohibited the discharge of water related to the accident from the site. This prohibition remained in effect until a suit by the City of Lancaster and resulting out-of-court settlement eliminated any prospect of discharging the processed water during the cleanup.

By July 1979, the concentrations of radionuclides in the auxiliary building water had changed because of decay; the short-lived iodine isotopes were no longer major constituents. Accordingly, the resin mix in the demineralizer beds was changed to optimize the performance of the system for the long-lived isotopes of cesium and strontium, which were predominant (Rusche 1979).

The NRC finished its Environmental Assessment in October and officially modified the TMI-2 license to permit the operation of EPICOR II (US NRC 1981). Permission to operate involved one added stipulation: the expended resins were to be solidified before shipment. This was something that the project team was not equipped to do. The efforts to comply with the solidification order and the eventual development of a high integrity container alternative are described in sections 6.4 and 6.5.

The system went into operation on October 22, 1979. By December 1980, EPICOR II completed its first major task by processing 2,140,000 liters of water existing in various auxiliary and fuel handling building tanks and sumps. Over 36,000 curies of cesium-137 and 2,000 curies of strontium-90/yttrium-90 were removed.

The EPICOR II system continued to prove its value throughout the cleanup. It was modified twice after processing the initial auxiliary building water.

- The first modification (1981-1987) allowed it to act in a polishing capacity to support processing containment basement water and reactor coolant (see Section 6.2.2).

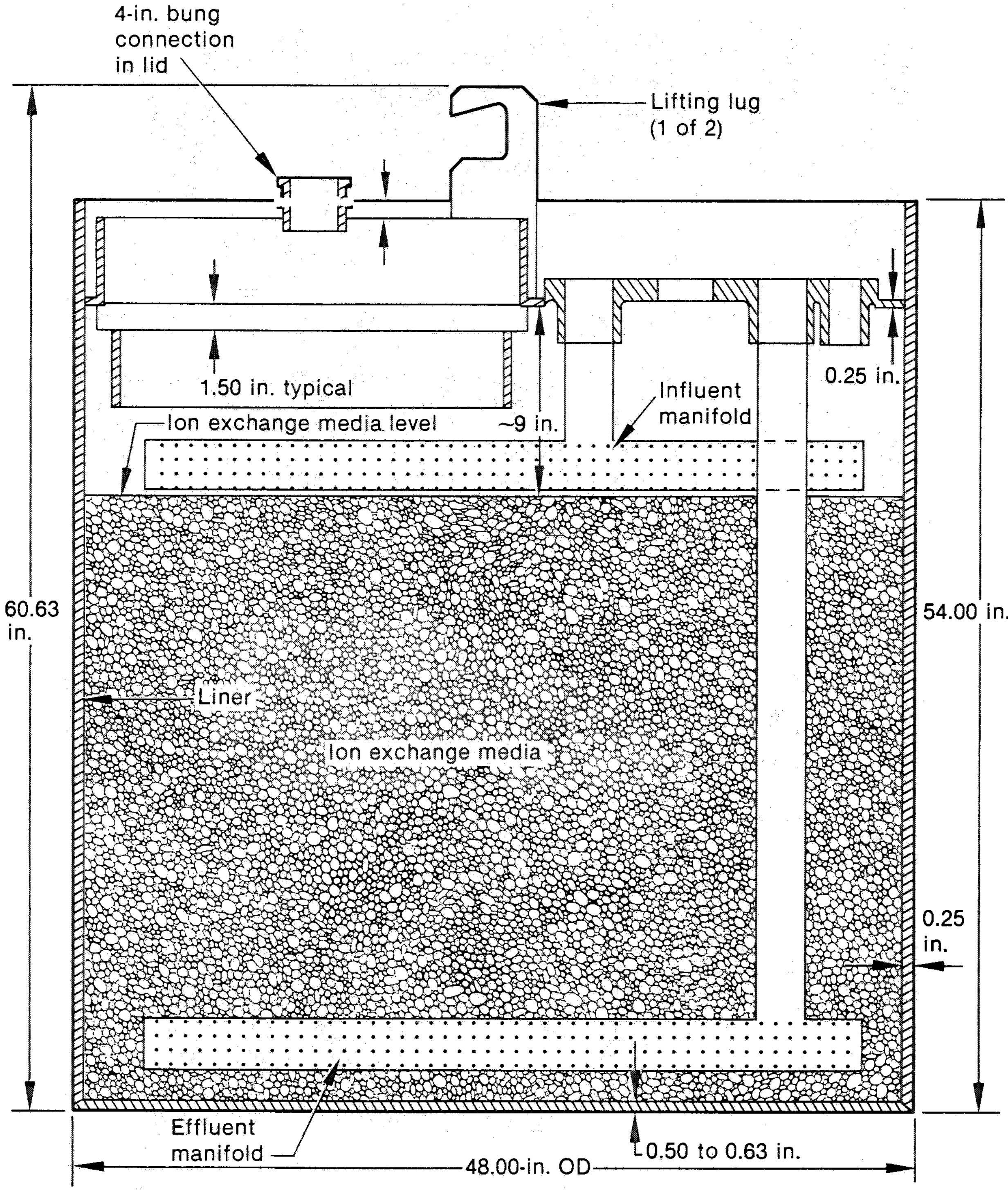


Figure 6-6. Cross-sectional View of Typical EPICOR II Vessel

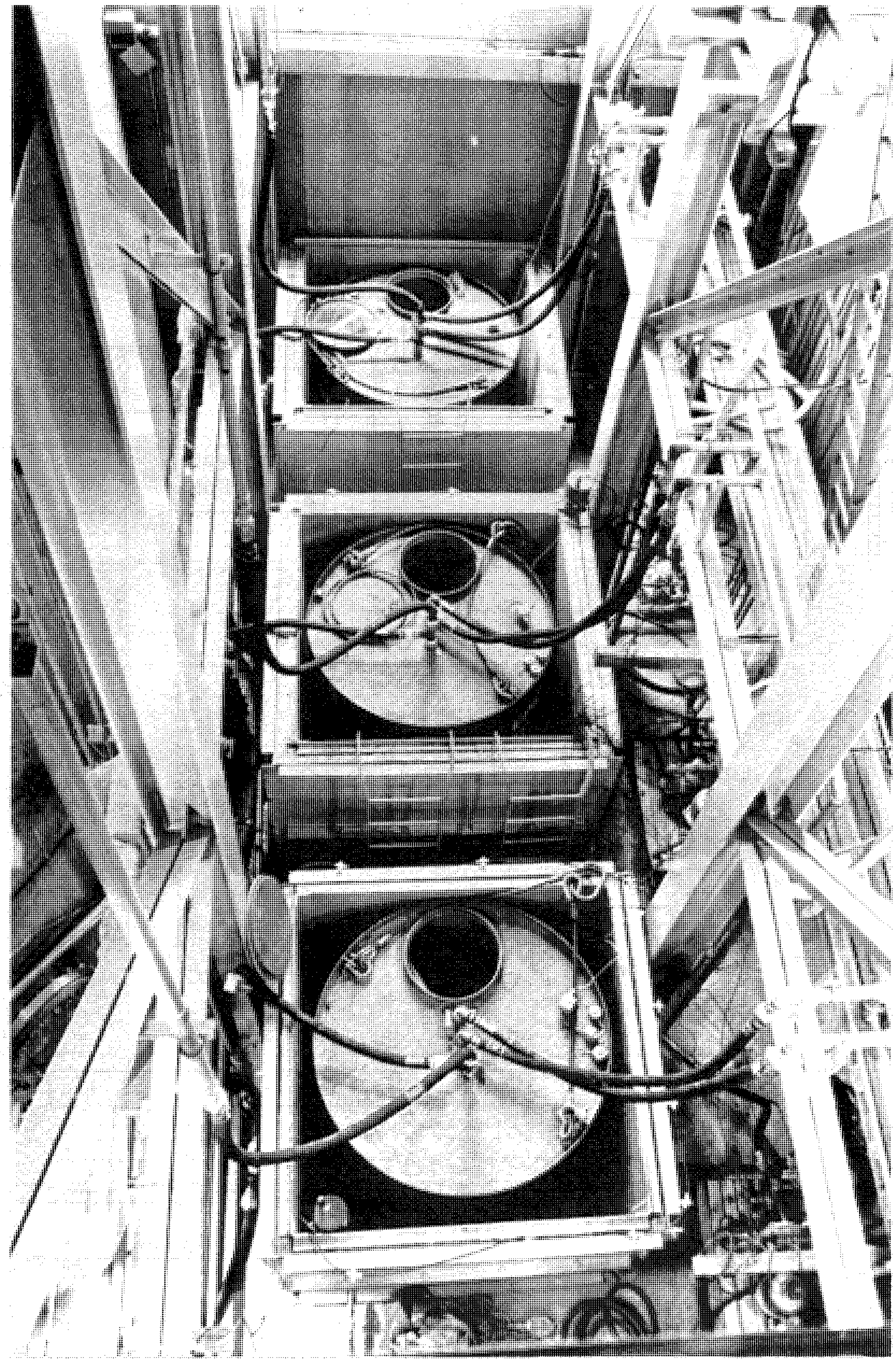


Photo 6-1. EPICOR II Vessels in Place

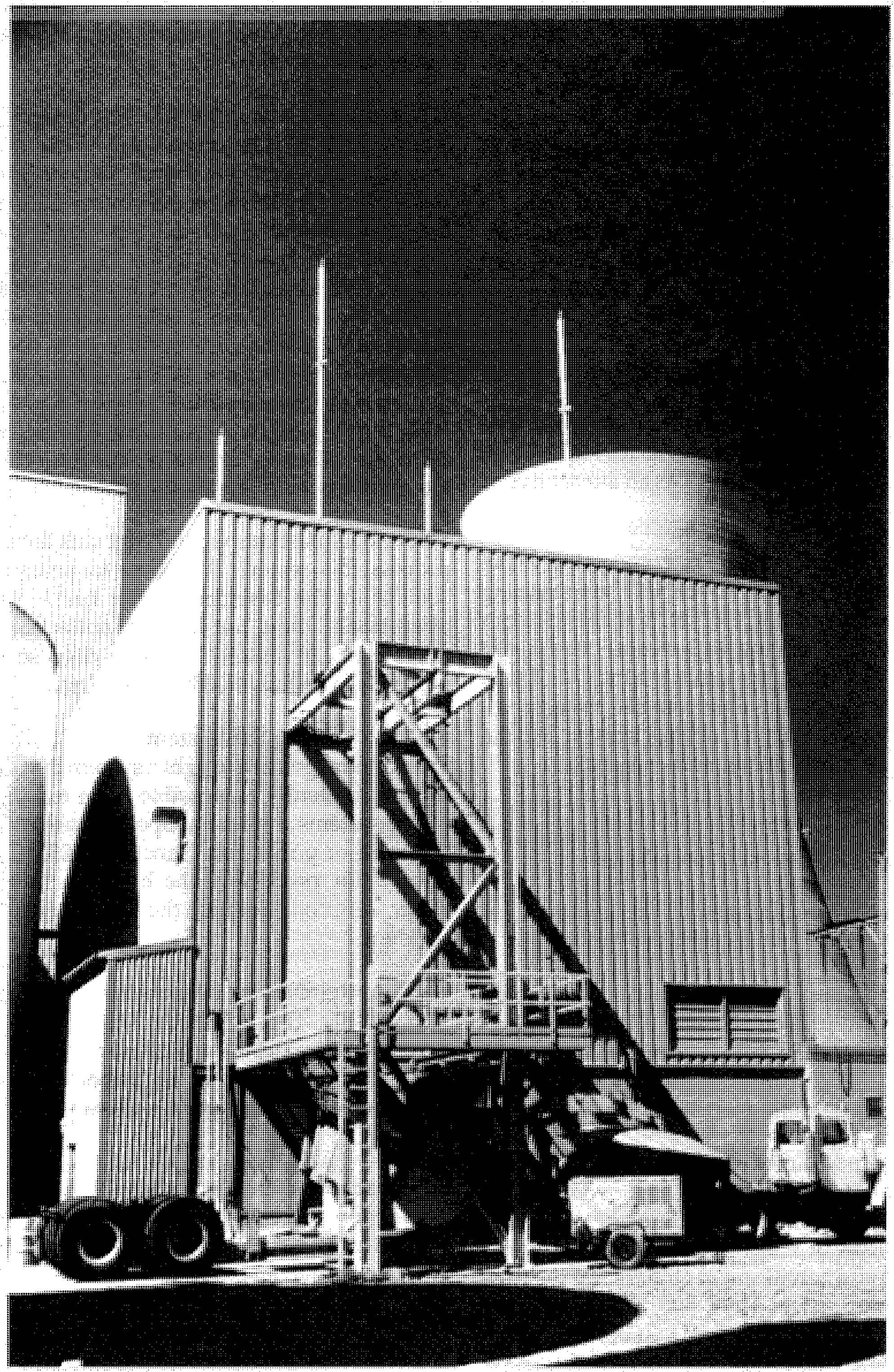


Photo 6-2. EPICOR II/Chemical Cleaning Building

- The second modification allowed it to act as the final water processing system for the cleanup. In April 1987, this version started up. It returned to the original concept of using the first vessel position as a prefilter; however, since there was no other system in operation to process higher-activity water outside the RCS, the first EPICOR II vessel contained a bed of sand zeolites designed to remove cesium and strontium. In addition, this vessel was the first use of a Ferralium in-line processing HIC. The remaining two vessels were 1.8- by 1.8-m. This system arrangement was able to process all types of water in the plant, including reactor coolant, if necessary.

Regarding this last EPICOR II configuration:

- By this time, the State of Washington had recognized Ferralium alloy as acceptable material for HICs. They also accepted the concept of HICs as an alternative to solidification.

- Inherent in acceptance of a HIC as a processing vessel was the project team's ability to meet the "free standing water" requirements of 10 CFR Part 61 and the State of Washington. Onsite testing with zeolite media and the existing dewatering system for EPICOR II was only marginally acceptable; therefore, the project team bought and qualified an air drying system for the HICs for zeolites, bead resins, and charcoal.

6.2.2 Processing Higher-Activity Water

The water processing challenges in the containment were far more severe than in the auxiliary building:

- Reactor coolant samples showed radionuclide concentrations of greater than 500 μCi/ml after the accident, and 100 μCi/ml in 1981 (primarily cesium-137 and strontium-90/yttrium-90)—too high to permit maintenance and defueling work.

- The basement was thought to contain many hundred thousand liters of water, much of it reactor coolant. In fact, the basement contained a volume of water that grew as the result of leakage from the river water cooling system and containment chillers. By the time the leak was stopped in 1981, 500,000 curies were suspended in 2.5 million liters of water. This water was thought to increase radiation fields on the upper elevations and inhibit any sustained level of work. Furthermore, although a quality liner existed (with vacuum-tested weld joints), the containment basement was not designed for indefinite water storage. Consequently, the potential of leaks to the environment had to be considered.

A new system was required. The design and construction of EPICOR II had taken only three months because it was based on an ion exchange technology already used at the plant. Processing the far more radioactive water in the containment required a new ion exchange system that, although based on technology in the noncommercial fuel reprocessing world, was not typical of power plant systems. Off-the-shelf equipment and technology were not sufficient and vendors were not prepared for the novel requirements. When the resulting submerged demineralizer system (SDS) was finished, it worked very well, but it took two years and $11 million dollars to complete. During this time, the project also pursued alternative processing methods

The initial questions were: What was the best system to process this water? Whether to process the reactor coolant or the basement water first? What was the operational limit for isotope collection in the SDS demineralizer vessels? How to dispose of the fission products once they had been collected?

6.2.2.1 Selecting the System

Project management sought a system that would operate as independently as possible from existing waste and offgas systems and would operate with minimal worker exposure or public risk (Sanchez 1983). Three primary options for processing the higher-activity radioactive water were considered in the first months following the accident:

- Natural circulation evaporator—Located in spent fuel pool "B". This approach was eliminated because it appeared to have the highest overall cost and many doubts existed about the technical feasibility of installing and running the system within a reasonable time.

- Forced-circulation evaporator/crystallizer—Located in a room in the fuel handling building or in a separate facility. This evaporator was developed as a backup/alternative to an ion exchange system. The evaporator was thought necessary to process the organics, oils, or decontamination solutions that would foul the ion exchange system. The conceptual development and eventual elimination of this evaporator are discussed in Section 6.2.5.2.

- Ion-exchange system—Located underwater in spent fuel pool "B". Although a limited experience base existed within the industry, the ion exchange system or SDS was selected, in large part because it was evaluated as being the most cost effective; i.e., promised to generate the least amount of solid waste (Hovey 1980).

The project team also considered and rejected several other options for dealing with the higher-activity radioactive water:

- Fluid bed dryers and calciners—Ruled out because of a lack of technical experience and because they concentrated radioactivity in the waste product to an unmanageable level.
- EPICOR II alone—Proposed and rejected because it was first designed for lower-activity radioactive water—a use for which it was still required. Also, increased personnel exposures were anticipated with EPICOR II and costly modifications would be required to cover the shielding, shipping, and burial requirements of its larger vessels.
- Direct solidification—A solidification system was designed to be installed in the fuel handling building; however, it would have generated an enormous number of drums of solidified waste (greater than 33,000 if 3.8 million liters were processed) and would have required four years to complete the task. Also, it involved substantial radiation exposure to workers.
- Long-term onsite storage—One or more large, shielded tanks to hold the unprocessed water were considered; however, the cost to build one tank was excessive (greater than $13 million for one 1-million-liter tank plus shielding) and two years would be required for construction. The potential for leaks during handling was a concern, and, after storage, the water would still require processing and disposal (GPU Nuclear 1986a).
- A DOE waste shipping cask—A 40-cm naval gun barrel was available for conversion from a transport cask to a processing vessel. It would have fit on a railcar and been loaded with zeolites for processing the water; however, since it would perform in essentially the same manner as the SDS, it provided no technical diversity. Also, it was not NRC-licensed and would not be available any earlier than projections for completing the SDS (Wilson 1980). The general value of the system for emergency radioactive water processing was thought to be good and the project management therefore recommended that such a system be developed as part of a national emergency response organization.

The general design of the SDS was based on experience and technology at the defense atomic energy facilities and some commercial operations. The following points were made when the SDS was first selected (Williams 1979). The parenthetical statements are based on the experience gained during actual construction and use of the SDS:

- Demineralizer-type systems had been widely used for removing radioactivity from water streams and were relatively flexible, although experience was limited in processing higher-activity radioactive water. (A strong selling point; however, the limited experience affected its development because the processing design had to be refined.)
- The contractors for the system could perform most of the activities of design and skid-mounted construction off site, thus minimizing interference with work in progress at the plant and not requiring technical expertise from the project team. (This was a plus in terms of not diverting limited onsite resources from other ongoing work. However, the offsite vendor was not intimately familiar with the plant and the tie-ins required. There was tremendous pressure to deliver a workable product as fast as possible; yet no permanent tie-ins to plant systems were allowed—which complicated the design—and the entire system was to be disposed of within 18 months.)
- The system would be relatively simple to maintain and operate. Although the underwater location complicated some maintenance it also reduced personnel radiation exposures to less than the alternative methods. (A definite plus, especially since in fact little underwater maintenance was required and, once installed, SDS operated as designed nearly immediately. It operated well beyond its 18-month expected life; minimal dose rates were associated with its operation; and there were no significant radioactive releases from the system.)
- The system could be used to clean up the RCS volume in much the same way as a normal RCS cleanup system; e.g., through feed-and-bleed operations and allowing the ion exchange media to become boron saturated. (A very useful feature of the system that also provided flexibility for temporarily processing water during defueling.)

- Because the detailed engineering had not yet been done, unforeseen design problems could extend the schedule. (This occurred.)
- The cost effectiveness of the system depended on the quantity of radioactivity that could be loaded onto zeolites and the decontamination effectiveness of the zeolites—two aspects that had not been completely defined and could affect the relative value and cost of the system. (The system was very efficient in concentrating fission products for disposal, although disposal costs and options limited the loading.)
- In 1979, the evolving regulatory criteria regarding burial and the questionable availability of disposal sites made the final disposition of the ion exchange resins uncertain and potentially very expensive, especially if solidification or extended storage in a spent fuel pool was required. (The DOE eventually accepted 19 SDS vessels containing the highest curie loadings for R&D—see Section 6.5. The remaining vessels were commercially buried after several years of onsite storage. Disposal was expensive, but it is doubtful that any other system could have produced wastes whose disposal would have been less expensive. The system was shut down when EPICOR II could handle the activity in the remaining water and preparations were underway for the end of the cleanup, which required such actions as draining the fuel pool water and dismantling the SDS.)

6.2.2.2 Design and Startup

As discussed below, the startup of the SDS faced two distinct challenges: 1) design; and 2) institutional.

Design Challenges. A processing design for the SDS was selected by the DOE and the project team (see Figure 6-7). The system resembled EPICOR II in its use of disposable ion exchange vessels. In contrast, however, SDS ion exchange vessels were smaller and loaded with inorganic zeolites, which were selected because of their affinity for cesium and long-term stability under high radiation conditions. Each SDS vessel was able to process more water than an EPICOR II vessel before the zeolites were expended, leading to fewer vessels for disposal and less processing interruptions to change out vessels.

The SDS was located in and around the two fuel pools, with the major components in spent fuel pool "B". The components were grouped in four areas: filtration and staging equipment, ion exchangers, leakage containment system, and support systems. The locations of the components are shown in Figure 6-8 and Photo 6-3.

Figure 6-9 shows a cutaway of an SDS vessel. All processing operations were done in batches; i.e., the periodic processing of a staged amount of water through specific zeolite beds. Based on samples taken after each stage in the process, the "breakthrough" level of a vessel was tracked. When the zeolites were spent and could no longer remove a particular radionuclide or the desired curie loading was attained in the first vessel of a train, then the feed flow was stopped and the vessel was moved to a storage rack in the fuel pool. A vessel with lesser loading was then moved to the first spot in the train; i.e., the most highly loaded vessel was the first to process contaminated water.

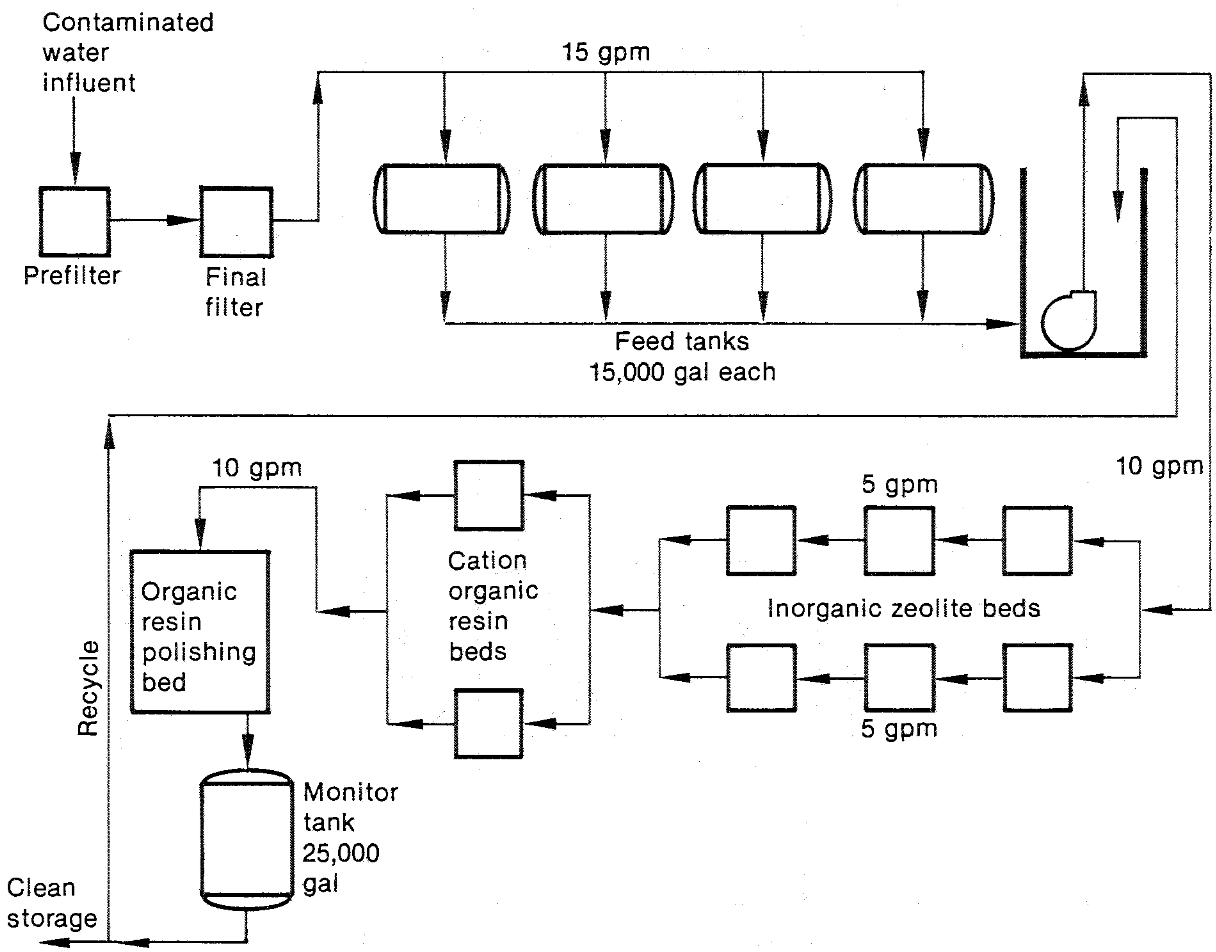
The initial design concept for the SDS was a satisfactory basis upon which to proceed; however, the design needed to be refined. The original processing design would have immobilized most cesium and strontium, but the effluent would still have contained enough activity in the form of cesium, strontium, antimony, and ruthenium to preclude undiluted storage. To address this, the project team and Oak Ridge National Laboratory (ORNL) performed additional evaluations, then:

- Modified the design and zeolite mix (now using synthetic zeolite Linde A-51 and the IE-96 zeolite-chabazite) (Campbell, et al. 1982)
- Replaced the original organic-based resin polisher in the SDS processing design with EPICOR II.

The final design is shown in Figure 6-10.

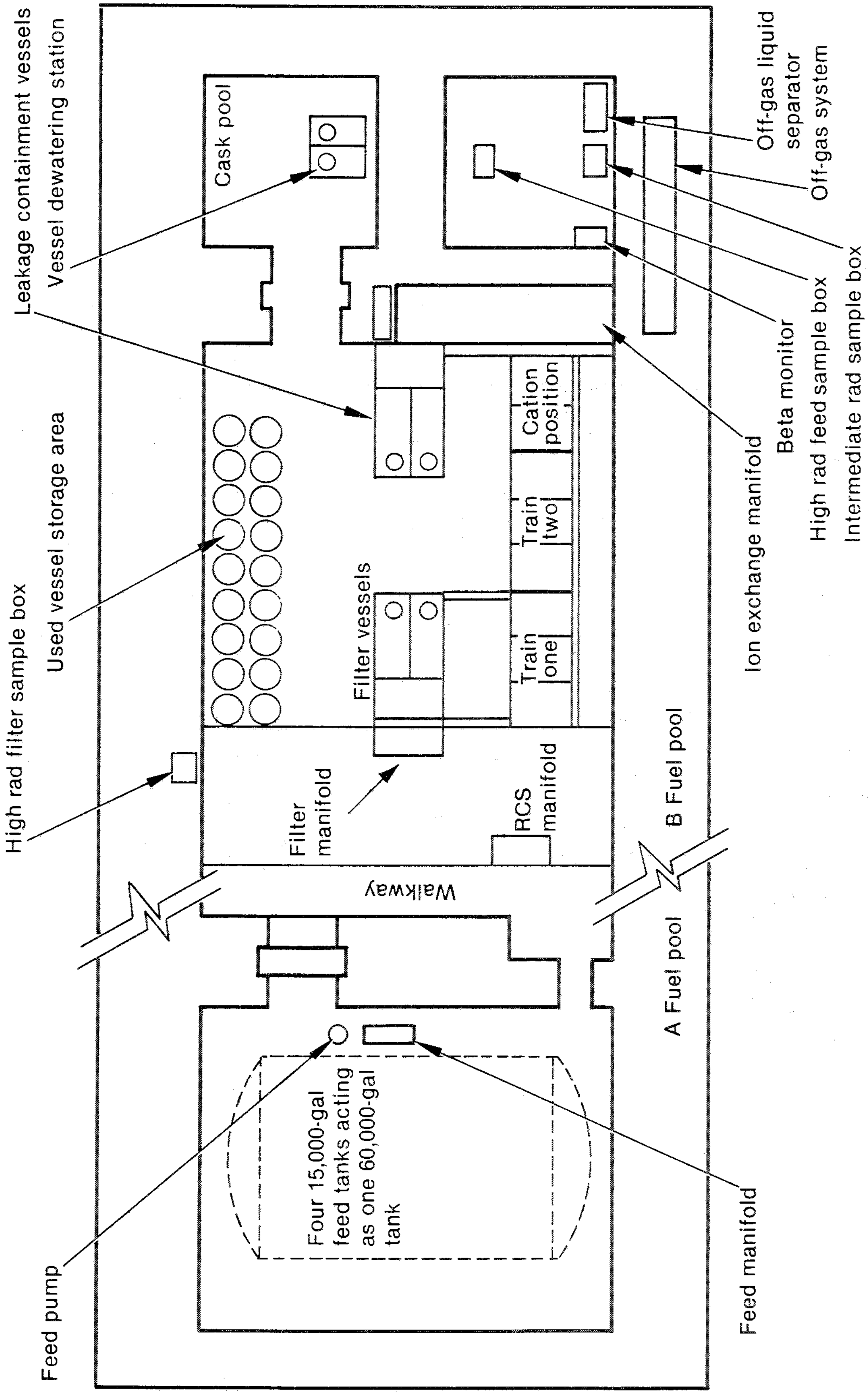
Improving the efficiency of the polisher was a major undertaking. Results from ORNL indicated that improving its efficiency would have generated unacceptably large amounts of low-activity solid waste. Also, the polishing unit would have required frequent changeouts, which would have affected the efficiency of the system and increased personnel exposures. As a result, in January 1980, the project team dropped the use of the SDS polisher, considered a new EPICOR-type polisher, and then proceeded to develop a method of using the existing EPICOR II system to polish the SDS effluent.

To determine what modifications to EPICOR II were needed, several engineers took a shielded test column on a pushcart into the containment in March 1981. The column contained a small volume of zeolites in a configuration designed to simulate the SDS operation; e.g., residence time, flow rates, geometry. Water was drawn from the basement using a roto-flex pump, passed through the zeolite bed, and 20 liters of processed effluent were obtained. One sample was sent to ORNL and



Original SDS flowsheet showing flowrates.

Figure 6-7. Original SDS Flowsheet



Arrangement of SDS components in and around the spent fuel pools.

Figure 6-8. Arrangement of SDS Components in and around Fuel Pools

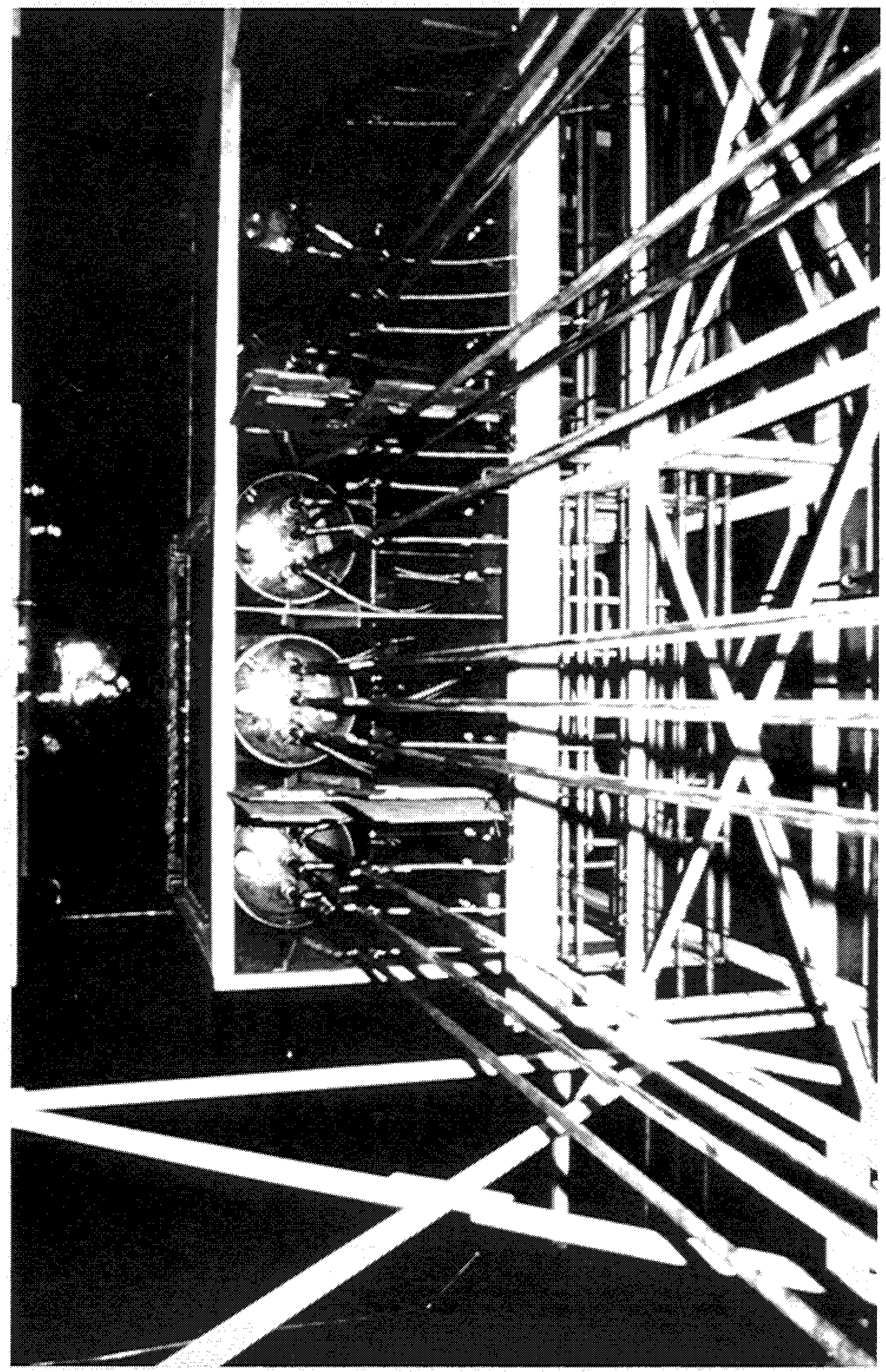


Photo 6-3. SDS in Spent Fuel Pool "B"

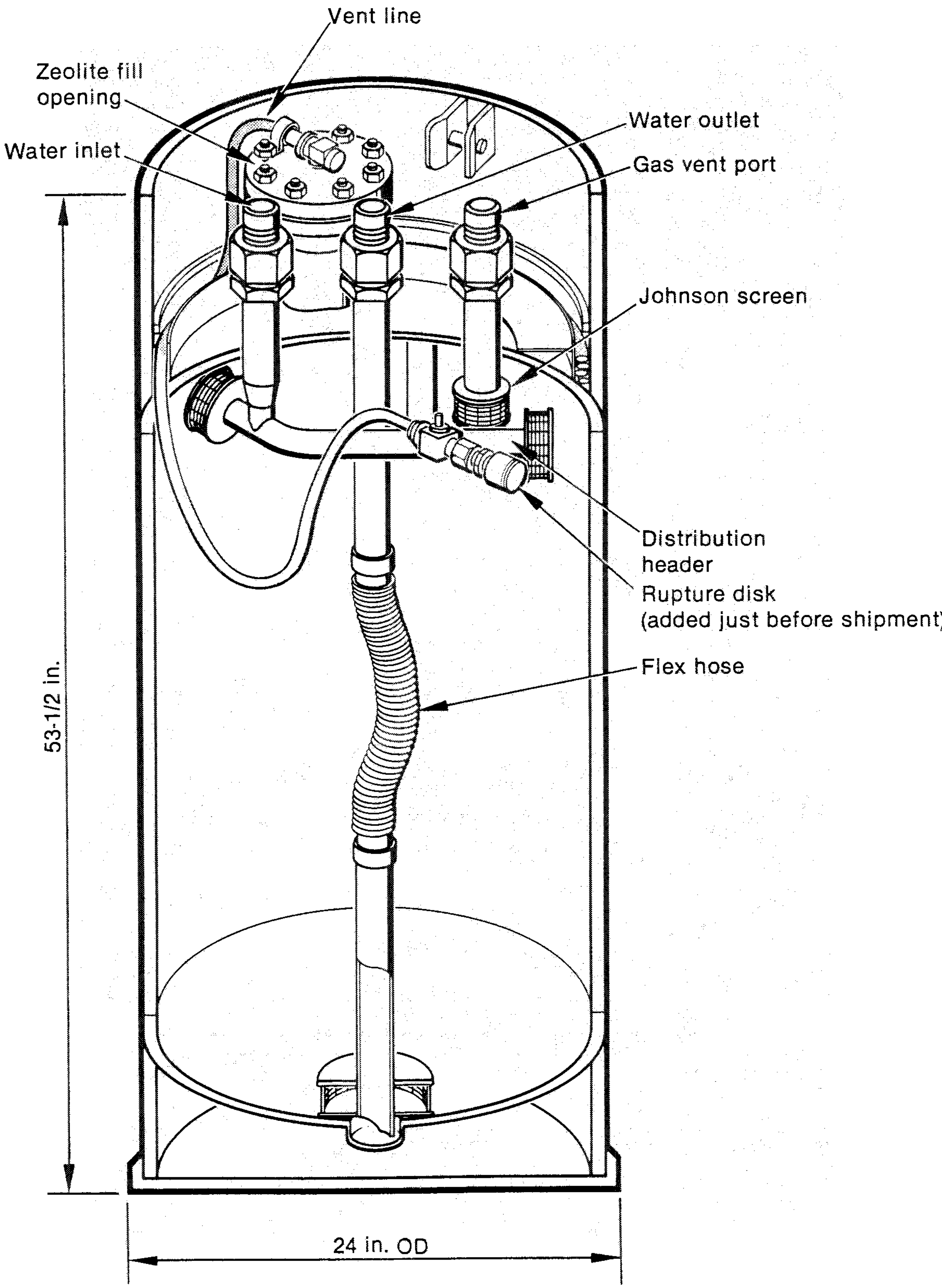
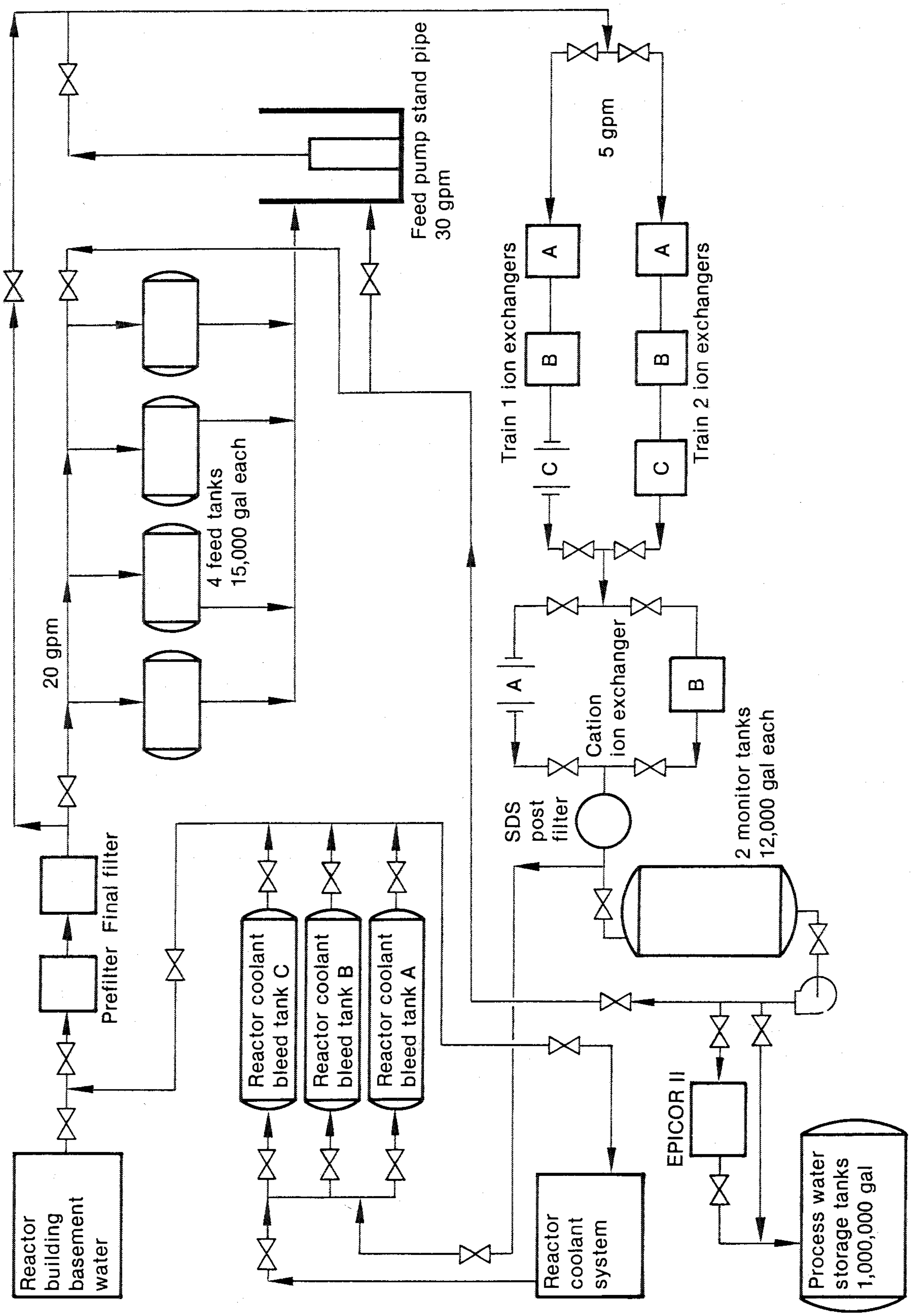


Figure 6-9. Cutaway View of an SDS Vessel



Final flowsheet for water processing through the SDS.

Figure 6-10. Final Flowsheet for SDS

one sample was analyzed on site to determine the right mix of organic and inorganic resins to use in the EPICOR II system for polishing.

Based on the results, the order and contents of the EPICOR II vessels were changed to remove levels of radioactivity <10E–03 μCi/ml. The first vessel was now a 1.8- by 1.8-m vessel containing only organic resins (instead of the previous 1.2- by 1.2-m prefilter). The second vessel position was a 1.8- by 1.8-m vessel and the third was a 1.2- by 1.2-m vessel; both contained a mixture of organic resins and inorganic media. (This version of EPICOR II remained in use until April 1987, to support SDS and to independently process various volumes of water in the auxiliary building.)

In May 1981, with construction of the SDS basically complete, the DOE completed an important study on the technical and financial implications related to the quantity of radioactivity that could be loaded onto each zeolite bed. The original processing design had limited each bed to 10,000 curies of cesium and strontium. The new DOE study concluded that the optimum loading for each vessel was actually 60,000 curies of cesium and 2,000 curies of strontium.

Although technically the loadings could be even higher, safety and economic considerations determined these levels. The increased loading reduced the expected number of expended vessels by a factor of six, meaning that approximately 20 vessels would be needed for processing the highest activity water in the containment basement and reactor coolant system (Sanchez 1983; US DOE 1981).

Institutional Challenges. At every step of the design process, the project management had to confront the issue of how to dispose of the zeolite resins once they were loaded with radioactivity. Several contingencies had to be considered. The actual disposal of the resins is described in Section 6.5, but it is important to note that, from the start, an NRC requirement to solidify the resins was a real possibility, with the potential of greatly affecting the schedule and cost of processing the higher-activity radioactive water.

In late May 1980, during construction of SDS, the NRC notified the project management of several concerns that pointed out the absence of clearly delineated criteria applicable to the cleanup:

- Until the Programmatic Environmental Impact Statement (PEIS) was completed, the NRC could not be certain it would approve the SDS; consequently, the project team was proceeding with construction at its own risk.
- There was no assurance that the resulting waste forms could be buried at commercial burial sites; consequently, the project management should not preclude the necessity of modifying the waste form (i.e., solidification) for long-term storage or burial.
- The water in the containment basement might begin to leak to the environment and so the project team should have contingency plans ready for shielded storage (Denton 1980).

Project management responded as follows:

- It proceeded with installation of the SDS because it realized the urgency of the situation and the length of time involved in constructing the system. In fact, the PEIS was issued in March 1981, several months before the SDS began operation.
- As for concerns about waste forms, an underwater storage rack was constructed in spent fuel pool "B" to store expended SDS vessels. In addition, the project team believed that enough temporary solid waste storage space existed on site if needed and, pending modifications to the burial regulations (the forthcoming disposal criteria in 10 CFR Part 61), this would provide enough flexibility to proceed.
- The emergency water storage volumes available in Unit 1 and 2 tanks and spent fuel pools were also sufficient, although their use was undesirable because of the potential for increased radiation exposures and the contamination of clean components (Arnold 1980; Greenwood and Kelly 1980).

To address the NRC's concern about the ultimate disposal of the zeolites, the DOE announced that a program had been initiated to demonstrate the feasibility of immobilizing the radioactivity on the zeolites in a vitrified form (Gates 1981). This was later expanded into a larger R&D program via the July 1981 Memorandum of Understanding between the DOE and NRC, which is discussed in Section 6.5.

6.2.2.3 Processing Containment Basement Water

Water in the containment basement was processed first since it, unlike water in the RCS, was not contained in pipes and components. The final technical step to removing the basement water involved finding a method of transferring water from the containment basement to the SDS. The existing sump pump was not considered reliable because its motor had been submerged for three years.

To solve the problem, a TMI-2 engineer constructed a floating pump (later known as "sump sucker") in his garage. It consisted of a commercially available well pump installed inside a Styrofoam float. In early September, this was lowered onto the water surface from El. 305' and deployed. (It worked satisfactorily until the water level neared the floor, then a second well pump was emplaced in the basement in a recess below the incore tubes. The water was pumped through a flexible hose that had been lowered into the basement, and then out through a penetration to connected piping.)

In July 1981, the system was ready to begin operation. To demonstrate its performance, the SDS was used to process some lower activity water (<1 μCi/ml) that had accumulated in the auxiliary building. During that summer, the SDS processed approximately 570,000 liters of water from the reactor coolant bleed tank (RCBT) to ensure that the system was prepared to handle the basement water.

Water from the containment basement was first processed on September 22, 1981. The containment basement water was pumped through filters to the storage tanks known as the "tank farm" in fuel pool "A" (see sections 3 and 6.3) and then on through the four ion-exchange vessels in Train 2 of the SDS (see Figure 6-9). The size of each batch ranged from approximately 38,000 to 190,000 liters, depending on processing conditions.

As examples of the processing methodology, the first 57,000-liter batch was processed between September 22 and 25, 1981; the average influent radioactivity was 94.2 μCi/ml and the average polished effluent was 15.9E–03 μCi/ml. The second batch contained 165,000 liters and was processed between September 26 and October 4. It had an average influent radioactivity of 115.9 μCi/ml; the average effluent contained 13.5E–03 μCi/ml. As a result of this processing, the vessel used in the first position for these two batches was loaded with 57,000 curies of cesium and 2,000 curies of strontium before being removed from service.

Containment basement water processing was essentially complete by early March 1982, although some took place later in the summer and then periodically to clean and recycle water from decontamination flushing and inleakage. Approximately 2.5 million liters were cleaned and stored after polishing by EPICOR II. The SDS removed 278,000 curies of cesium-137, 29,800 curies of cesium-134, and 11,600 curies of strontium-90/yttrium-90 from the basement water (Hitz 1986).

6.2.2.4 Processing Reactor Coolant

In May 1982, the SDS began processing the reactor coolant water. The processing method involved a "feed-and-bleed" technique in which 190,000-liter batches were bled from the reactor coolant system at the same time that 190,000 liters stored in the RCBTs were fed into the RCS. This ensured that the reactor coolant system was full and the core covered. Because of the lower activity concentrations in the reactor coolant, Train 1 of the SDS was used. The effluent was not polished by EPICOR II but instead was directed back to the RCBTs for future use in the cycle.

The most immediate value of this processing was to lower concentrations in the reactor vessel in order to support the Quick Look camera inspection of the reactor core in July 1982 (see Section 5.4.1). For this, the SDS processed five batches between May and July 1982, removing approximately 60,000 curies and reducing the concentrations in the coolant from approximately 15 μCi/ml of cesium-137 to 2–3 μCi/ml. After Quick Look, the reactor coolant was processed on a regular batch basis that reduced levels to less than 0.1 μCi/ml of cesium and 2.2 μCi/ml of strontium. Since this water was in continual contact with the core debris, regular processing by the SDS was necessary until a new system designed for processing water during defueling became operational in 1985 (see Section 6.2.3).

Alternative Processing Scheme. Before the SDS processed the water in the reactor coolant system, the project team had considered using EPICOR II for the same purpose. A study (Barton 1980) found that using EPICOR II in this manner could cut perhaps one year off the overall development-plus-processing schedule. The primary drawbacks were:

- The requirement to contaminate some clean equipment
- The considerably higher cost of the labor involved in a longer processing campaign; i.e., the result of a larger number of vessels to be used
- The larger number of vessels would result in higher disposal costs.

Only two demineralizer beds would have been used: the first containing only inorganic zeolites in a 0.28-m^3 vessel (identical to an SDS vessel) for higher curie loading; the second would have contained organic resins for additional isotope removal and as much chemistry control as possible. The major and fatal drawback to using EPICOR II to process RCS water was that EPICOR II would have removed the sodium and boron from the water. These chemicals would have had to have been re-added to the processed water in the storage tanks in order to provide suitable makeup chemistry to the reactor vessel—a very expensive undertaking.

The SDS remained in operation until July 1987, processing over 17 million liters of contaminated water and providing an essential element of TMI-2 waste management operations. The system was put in standby condition when other systems were able to handle the water processing. It was restarted briefly in 1988 to process flush water in the containment basement.

6.2.3 Processing Defueling Water

Decontaminating the reactor coolant after the accident was one thing, and the SDS performed that well. Keeping the water clean enough for defueling operations was another. In early 1982, as the SDS processed the reactor coolant, the project team began considering what it would take to process the millions of liters required to support defueling.

From a water management perspective, there were two essential ingredients necessary to support defueling: visibility and low concentrations of radionuclides. Success with SDS and EPICOR II gave confidence that a new water processing system could do this during defueling by filtering out suspended particles and removing radionuclides. The difficulties were the vastly different scale of processing required and the little known world of TMI-2 core damage. Instead of processing 4 million liters of water over the course of six months as SDS had, defueling would require that much every few weeks. Filtration requirements would exceed earlier volumes by tenfold. And the actual nature of the reactor coolant during defueling was impossible to predict.

The project team designed and built the defueling water cleanup system (DWCS) to meet these challenges. It was composed of two subsystems: one for the reactor vessel to process a 150,000-liter volume and the other for the refueling canal and fuel pool "A" to process a 1.4 million-liter volume. Fine particles of fuel and other particulates suspended in the water were removed via a sintered stainless steel filter. To remove radionuclides, 1.2- by 1.2-m zeolite ion exchangers were used.

In the course of designing, installing, and operating this system, the project team learned some hard lessons. In the end, the system worked—but not as designed and not without a severe impact on the schedule:

- The little known nature of the postaccident reactor core and the changing nature of defueling plans resulted in a cumbersome, overly conservative water processing system.
- Filtration proved to be a difficult science that did not follow strict laws—especially under changing conditions. Rigid schemes and general wisdom could not be relied upon and experts were necessary at every phase.
- Inadequate testing caused problems. The design specifications for off-the-shelf equipment did not always fit the reality of the situation.

6.2.3.1 Selecting the System

The conceptual design of the DWCS was closely intertwined with defueling plans and so went through similar tribulations as defueling plans changed (see Section 8 for a history of defueling strategies). This was further complicated by a lack of precise data about core conditions. Within this context, the project team had to address the two guiding issues:

- Visibility—Based on industry experience during refueling operations, the team knew that turbidity (which directly impacts visibility) would be a problem. Corrosion products usually comprised the chief source of suspended particles. This problem would be greatly accentuated by the potential 13,000 kg of fine, loose debris (<40-micron) in the vessel and the additional debris resulting from aggressive defueling (Croucher 1981). Particles stirred up during defueling were expected to exceed the turbidity goal of 1 nephelometric turbidity unit (NTU) (US NRC 1981).
- Personnel Exposure—Of even greater concern than visibility was radiation exposure to personnel, primarily caused by soluble cesium in the reactor coolant (Hofstetter 1986). Defueling operations had to be conducted at less than 15 mrem/h. The source term from the water was a key ingredient (along with

transit dose, area background, and defueling operations) and the radioactivity concentration in the water had to be kept below 0.1 μCi/ml to achieve the goal. SDS processing of RCS had reduced the concentration to approximately that level. However, the leaching of cesium-137 from the debris bed (assumed to be at a rate of 2 curies/day) was expected to raise the radionuclide concentrations above the targeted value without regular processing.

But how much water would have to be processed and filtered? The original concept for the system was based on 1981 defueling plans that called for a "wet" defueling scenario; i.e., one in which the entire fuel transfer canal was flooded and very large quantities of water were used (reactor vessel/contamination barrier: 300,000 liters.; refueling canal: 1.2 million liters; fuel pool "A": 530,000 liters). By the late summer of 1982, this defueling concept had been further defined to require:

- A water processing system for the fuel pool portion with a minimum flow rate of 2.5E–02 m^3/s (the fuel pool cleanup system preferably)

- An upgraded SDS reconfigured to four parallel lines of two columns each and a 2.5E–03 to 3.8E–03 m^3/s flow rate

- A backup system for the SDS (preferably a modified EPICOR II).

Consideration of a contamination barrier to control the spread of suspended material in the refueling canal was also recommended. For this, the two-part system shown in Figure 6-11 was conceived:

- For water directly associated with the reactor vessel, the SDS could be used for demineralization; a new high-flow particulate removal system could be used for hydro-vacuuming fuel debris and capturing suspended particles.

- In the large body of fuel pool water, the fuel pool cleanup system could perhaps work if modified in several ways; e.g., to use zeolites like the SDS (Burton 1982).

Considerable debate took place over whether SDS could be used for the task. In the end it was judged not acceptable because of the nature of debris to be filtered. Using SDS would have meant pumping fuel debris particles outside of containment, which would have required extensive shielding and raised perhaps unsolvable criticality concerns (Hofstetter 1987). Also, this dedicated use of SDS would have prevented it from supporting other cleanup work. (If a processing HIC had been available at the time, then EPICOR II could have handled all other water.)

Methods using hydrocyclones and centrifuges were investigated and discarded because the solids collected would still have required another intermediate operation to transfer them to a canister that could contain fuel—a dose-intensive operation with the potential for leakage.

Events in July 1982 emphasized the difficult conditions under which defueling would have to be conducted and thus the urgency of having an adequate water processing system. Then, with Quick Look, a camera provided views of the shattered remains of the upper TMI-2 reactor core. The examination also provided more specific data for use in designing the system. By the spring of 1983, the concept for a modified DWCS had been developed (Bell 1983).

A new design concept was developed that retained from the earlier concept the major advantage of separating two distinctly different volumes of water: 1) the fuel pool/transfer canal with a large volume and expected low concentration of contaminants; and 2) the reactor vessel with a relatively small volume and an expected high concentration of contaminants. The separation permitted the processing rates and components to be tailored to the sources, and the operations to be simplified (Figure 6-11). This plan for the DWCS was seen as technically sound because the source terms for the design had been established; however, it was very conservative (TMI-2 TAAG 1984).

The question of water volume again entered the picture in the spring of 1984. Plans for defueling the reactor had been modified to the "dry" method, which required that only the refueling canal adjacent to the transfer tubes to the fuel pool be flooded, and that a 1.8-m-high fixture be placed above the reactor vessel.

This change reduced the quantity of water to be processed by approximately 60% (Katonak 1984). It simplified matters for the designers of the DWCS by providing more usable space and reducing the quantity of water to be processed. The reduction in volume was not enough, however, to drastically simplify the design. The final conceptual design is shown in Figure 6-13.

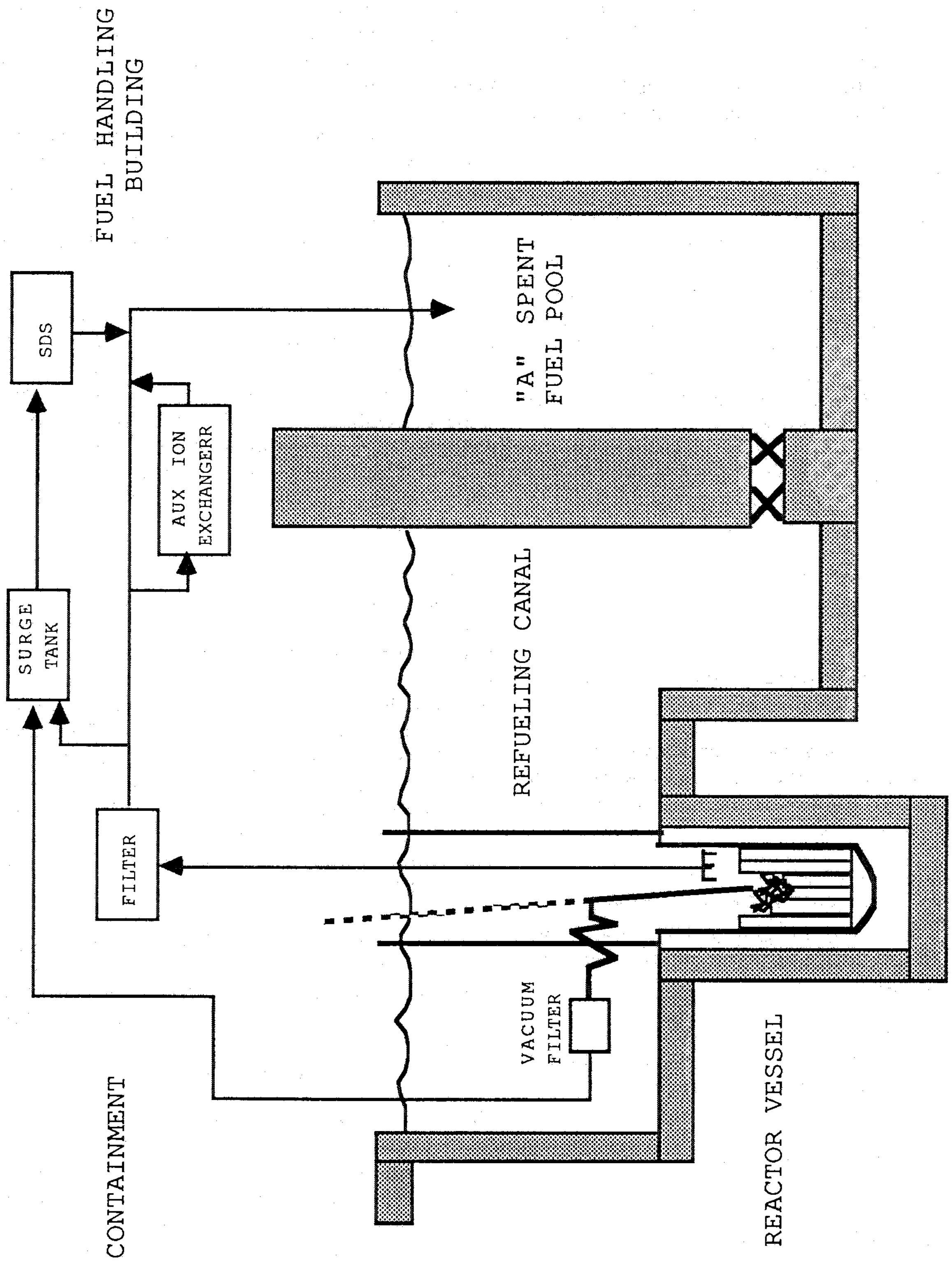


Figure 6-11. Early Conceptual DWCS Design

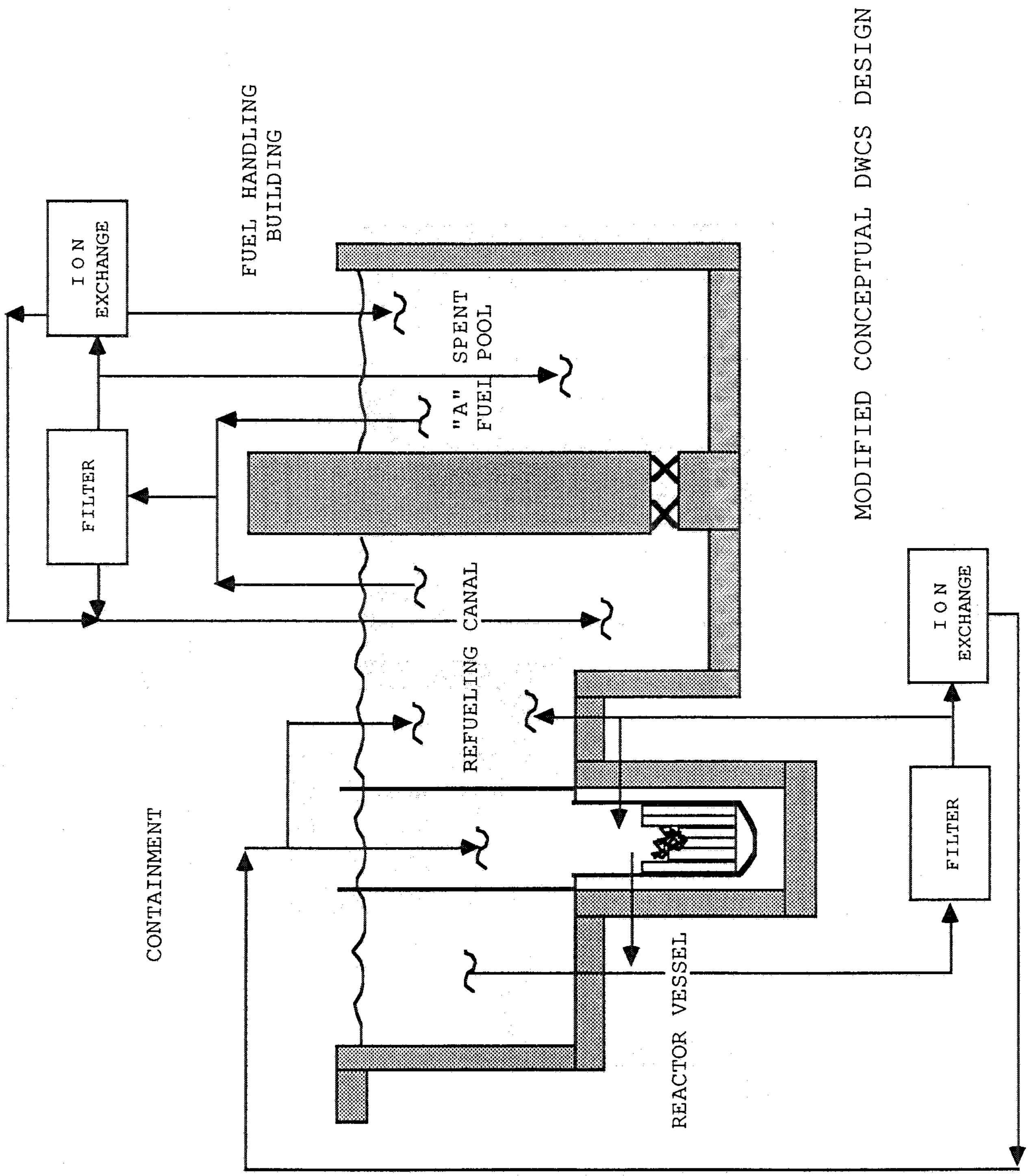


Figure 6-12. Modified Conceptual DWCS Design

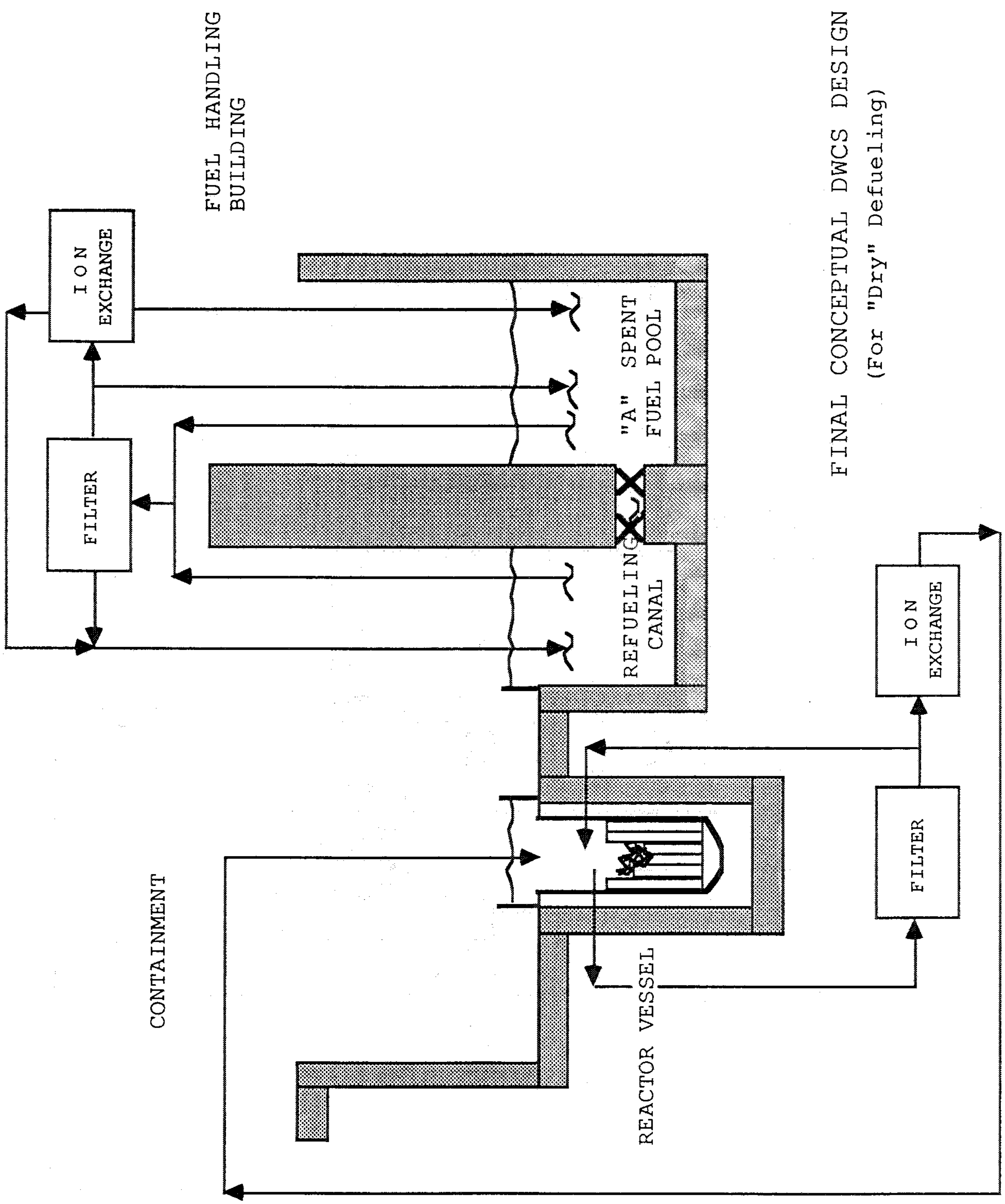


Figure 6-13. Final Conceptual DWCS Design

6.2.3.2 Design Challenges

To meet the challenges of maintaining visibility and removing radionuclides from the water, filters and ion exchange vessels were needed.

Filters. The project team put substantial effort into developing a filter; and yet, at one stage, the lack of adequate filtration was to cause one of the biggest technical headaches in the cleanup. The original design objectives of the filter were to:

- Remove fine debris between 0.5 and 800 microns
- Keep effluent turbidity at 1 NTU or less
- Maintain a minimum flow capacity of 6.3E–03 m^3/s
- Be unaffected by the chemical nature of the borated water
- Be safe against criticality of the filtered material during 30 years of storage
- Minimize the quantity of waste containers requiring disposal
- Operate in a straightforward manner that minimized complexity and promoted reliability (Storton 1985)
- Be compatible with the handling techniques developed for loading fuel debris (Bell 1983).

The project team decided to place the filtering system within the same canisters that would be used to load fuel debris (see Section 8). The filters, with a poison material designed in, were in a geometry safe from criticality and there was no need to qualify a second canister design. The canister was 380 cm long with a 35.6-cm nominal outer diameter.

Three types of filters were considered:

- Etched disc filter—Etched disc filtration was rejected because of the unavailability of 0.5-micron filter media; the difficulty adapting the discs to the canisters; and the 300-psi backwash pressure required, which exceeded the design pressure of the fuel canister (150 psi).
- Ultrafiltration—Ultrafiltration using an organic polymer membrane was eliminated because of limited loading capacity, potential effects of extended exposure to high-level radiation, and uninvestigated means of disposal (Rao 1984).
- Sintered metal filters—A sintered metal filter was selected by the canister vendor. The filter was a backwashable, tubular, porous 0.5-micron unit. During testing, a second 0.5-micron unit was developed and chosen for the DWCS. The advantage of this filter was that it eliminated backwashing, permitted essentially continuous filtration versus cyclical operation, and eliminated a complex filter control system. The pleated-sintered metal filter also offered a higher surface-to-volume ratio than the tubular form.

The concept of using a sintered metal filter was debated at some length for several reasons. At TMI-2, cartridge filters had not worked in either the SDS prefilters or in the original makeup and purification filters. Government and utility (Millstone) experience raised questions about the ability of such filters to process bulk media, especially without pre-coating. The project team persisted with sintered metal filters in light of the pressing need to minimize waste and because the filters would ease potential future reprocessing operations. Figure 6-14 shows the design of the filter canister.

In June 1984, initial tests of a single-element, sintered metal filter at the vendor's shop using a zirconium-oxide-based slurry indicated satisfactory performance. Seventeen single elements fit inside a filter canister. Water was pumped to the outside of the elements until either the filter canister was full or the differential pressure across the filter element reached a preset limit. The solids were then allowed to settle to the bottom before processing was resumed (Katonak 1985).

A single-element test was conducted at TMI-2 in late summer 1985 in spent fuel pool "B". The test was only partially successful (Katonak 1985); but based on the test and the need to support the imminent start of defueling, no changes were made in the filter design. With dramatic impact, these filters, as first installed, proved inadequate for handling the unexpected conditions in the reactor vessel.

Ion-Exchange Development. To keep the radionuclide level in the defueling water under control, the following criteria were developed:

- The concentration of cesium-137 had to be maintained in the range of 0.01 to 0.02 μCi/ml, which corresponded to 10 to 20 mR/h at 1.5 m above the water surface.
- The system, backed by redundancy features, had to process reactor vessel water at a continuous flow rate of 1.3E–03 m^3/s to maintain acceptable dose rates on the defueling platform above the vessel.

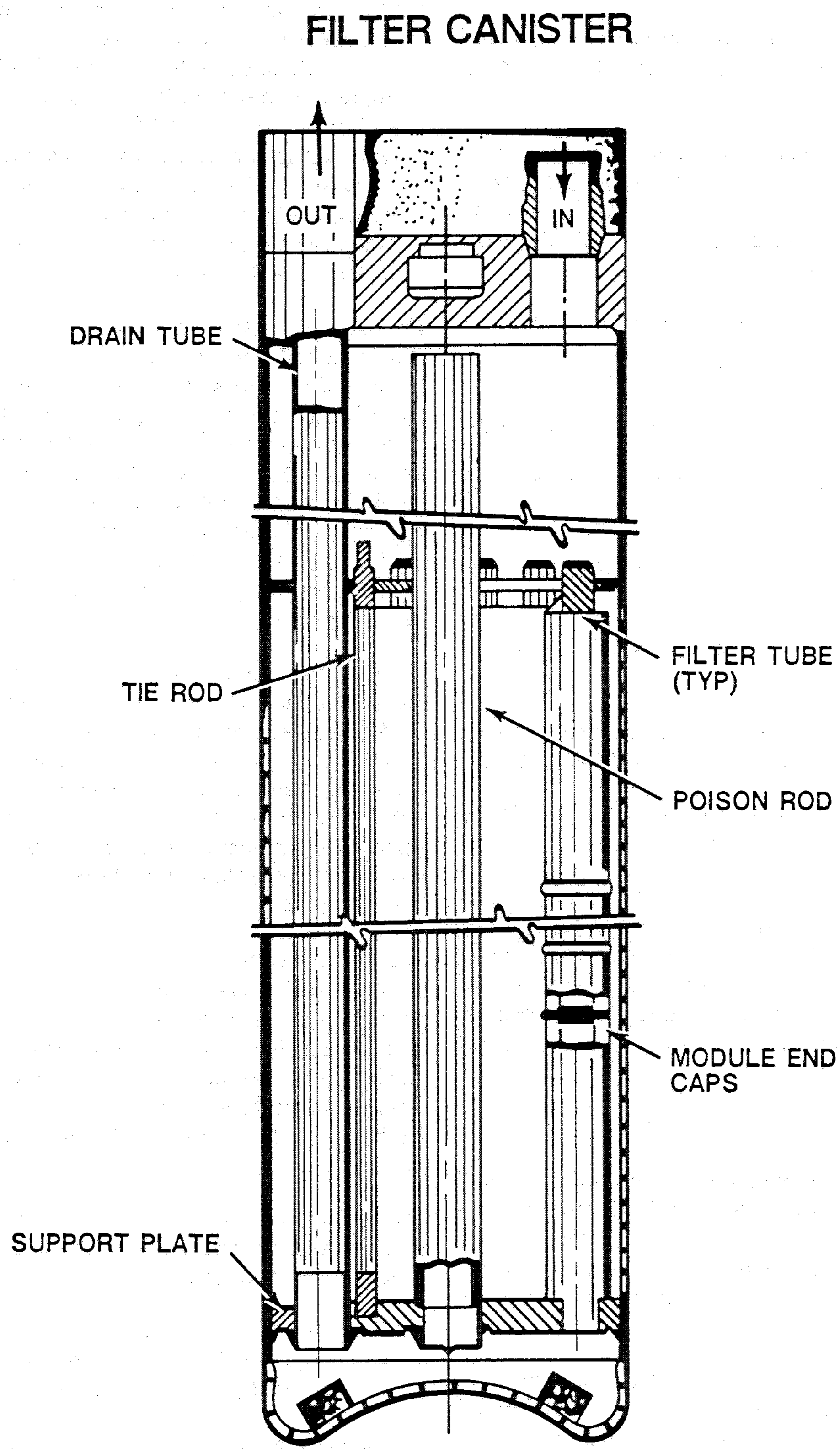


Figure 6-14. Filter Canister

- The expended zeolites had to be handled in a manner similar to that used for lower activity SDS and EPICOR II vessels (Bell 1983).

Because of the varied nature of the water in the two different subsystems of the DWCS, two different ion-exchange systems were used. Two 1.2- by 1.2-m ion exchange vessels were used in the smaller but more contaminated reactor vessel portion of the system. One 1.2- by 1.2-m ion exchange vessel was used in the refueling canal/fuel pool "A" portion.

The ion exchange vessels were loaded with approximately 0.8 m^3 of zeolite sorbents—a choice that was based on the successful use of the zeolite ion exchange system in the SDS. The 0.8-m^3 Ferralium alloy vessel had been marketed as a HIC and had been adapted for use at TMI-2. This permitted it to be qualified as an in-line processing HIC and thus commercially buried as a stabilized waste form.

6.2.3.3 Installation and Operation

The DWCS was installed and turned over for operation in stages. The components located in the deep end of the refueling canal were ready when the upper internals (i.e., plenum) were removed from the reactor vessel and placed there for storage in May 1985 (see Section 8). During the following months, the project team focused on completing the reactor vessel subsystem of the DWCS to support the start of fuel removal operations. Until this subsystem was ready, reactor coolant was processed through the SDS in a series of feed-and-bleed operations. Fuel pool "A" was flooded in October 1985 and a temporary system was used to filter that water until its DWCS subsystem could be completed.

In November 1985, visibility in fuel pool "A" declined and hydrogen peroxide was added to kill the microorganisms causing the problem. This was the first real indication of future problems.

Defueling started in late October 1985, and in November the reactor vessel DWCS subsystem began operation. Almost immediately the filters began to plug up. Instead of processing 22 million liters as designed, less than 750,000 liters passed through a filter canister before it plugged. The effluent was good (0.4 NTU) during a canister's few hours of useful life, but the canisters were too expensive to permit this rate of usage to continue. To allow the system to continue operation while modifications were made, the filter canisters were bypassed. Project engineers speculated that an unknown, probably amorphous and inorganic material was blocking the filter pores (Worku 1985).

A separate but related problem loomed in January 1986. A video survey in the reactor vessel showed a general cloudiness and a variety of organic, sometimes kelp-like shapes floating in the coolant and attached to surfaces. These microorganisms were eventually discovered to exist in almost all bodies of water in the plant.

Their presence was attributed to conditions found in normal nuclear plant systems and to the river water that had leaked into the containment basement, been processed, and then introduced into the reactor vessel during SDS operations. The sudden, rapid growth of the microorganisms was the result of aeration and light associated with in-vessel defueling operations, and, especially, the occasional leaks of hydraulic fluid from long-handled defueling tools. The carbon-based borated hydraulic fluid used in the tools not only served as a nutrient but contained microorganisms itself.

The project team now faced a double challenge inside the reactor vessel.

- First and most pressing was the biological contamination. The microorganisms obscured visibility and blocked the filters. Beyond the immediate effect of reducing visibility during defueling operations, the contamination threatened all aspects of the cleanup by raising difficult questions; e.g., was there microbial-induced corrosion (MIC) on vital surfaces like the welds between incore guide tubes and the reactor vessel lower head, and inside fuel canisters that had already been loaded and sealed? (There was not.)

- Secondarily, and more resistant to cure, was the inorganic contamination, primarily hydrated metallic oxides in colloidal suspension. These inorganic particles were less evident but were known to clog the filters. Also, when large quantities of the fine, inorganic particles were suspended in the coolant, visibility was only a few centimeters.

There were no quick fixes since the natures of the biological colony and inorganic particles were complex and changing. The only way to address the situation was to understand it, take some emergency steps, and devise some long-term maintenance strategies. Meanwhile, the colony of microorganisms grew exponentially.

In February 1986, the temporary reactor vessel filtration system (TRVFS) was installed in an attempt to stabilize the declining visibility and gain some control. The turbidity level was approaching 100 NTU and visibility had fallen to 2 to 5 cm. Fortunately, defueling operations were able to continue during this time—although not efficiently—because loose rubble could be loaded into canisters sight unseen. The TRVFS took and returned water to the reactor vessel at a maximum of 4.7E–03 m^3/s using a 1-kW pump and 3.8-cm dia. hose. The water was processed through what was essentially a swimming pool filter with diatomaceous earth media.

An emergency task force composed of project engineers and scientists from national laboratories (referred to as the DeBioContamination Task Force) was formed in February 1986 to analyze the problem and recommend solutions.

Since the microorganisms were surviving in the reactor coolant solution chemistry, the biocide had to be effective in that matrix. Several initial treatment methods were considered, even though their use might be detrimental to the coolant or their effectiveness suspect:

- Halogenated hydrocarbons—High halide concentrations in excess of the RCS technical specification limits would result.
- Ozone—Ozone would quickly decompose in the high radiation field.
- Ultraviolet light—The organisms had already demonstrated tolerance to radiation.
- Heating to 355 K (180°F)—Many microorganisms requiring high temperatures for development were identified in the system.
- Strong acids—Strong acids could potentially degrade core components.
- Metal biocides—The microorganisms had already shown a resistance to metals; e.g., silver concentrations higher than recommended treatment levels

Laboratory and pilot testing of these methods confirmed their ineffectiveness (Hofstetter 1988).

Between mid-April and mid-May 1986, defueling operations were suspended in order to implement what was referred to as the "bug kill". This involved a series of equipment changes and modifications to operations. The "kill" was to be effected by using a pieced-together water cleanup system; i.e., a portion of the DWCS, operating without a filter, passed water to a high-pressure pump (10,000-psig) and then through a hydrolance.

Exposure of the microorganisms to high pressure and then rapid depressurization as the water was returned to the vessel had been demonstrated to be effective in killing a high percentage of the colony. The hydrolance was also used to remove organic growth from reactor vessel internals and to mechanically scrub surfaces. An upgraded TRVFS cleaned the water of dead microorganisms. Drain-and-refill operations were conducted to reduce the biological contamination level through dilution.

The effort was not successful because the hydrolance was not able to generate a sufficient flow rate to keep ahead of the organic growth rate. On the positive side, the drain-and-refill technique and the TRVFS were able to prevent further deterioration and so visibility in the vessel ranged from a few centimeters to one-half meter.

At the same time as the hydrolance attack, the NRC approved the use of 200 ppm hydrogen peroxide as a biocide. This came after a careful evaluation of hydrogen peroxide's successful use in the spent fuel pool and its compatibility with water chemistry, processing systems, recombiners in the canisters, and waste disposal constraints.

The most significant concern had been with a potential two to tenfold increase in activity levels because hydrogen peroxide was thought to increase the release rate of cesium from the debris bed. The NRC accepted INEL test results estimating that the increase would have no significant impact on worker safety if normal radiological control practices were implemented (Travers 1986).

The hydrogen peroxide was successful in killing the microorganisms. With concurrent drain-and-refill and TRVFS operations, some measure of visibility was regained. In the summer of 1986, the Core Stratification Program (see Section 5.4.3) was conducted and the water was clear enough for limited surveys of the debris bed and the core bore holes. The microorganisms were dead and new growth could be controlled with maintenance; however, visibility was still not acceptable.

Pressure to find a solution mounted rapidly. The TMI-2 Water Clarity Group, consisting of senior scientists and engineers from across the country, was established in September 1986. Evaluations were underway on site and at several national laboratories. The TMI-2 Water Clarity Group developed several approaches, as depicted in Figure 6-15.

During the fall of 1986, a variety of filtration approaches were evaluated and tested, including deep-bed diatomaceous earth filtration, polypropylene bag filtration,

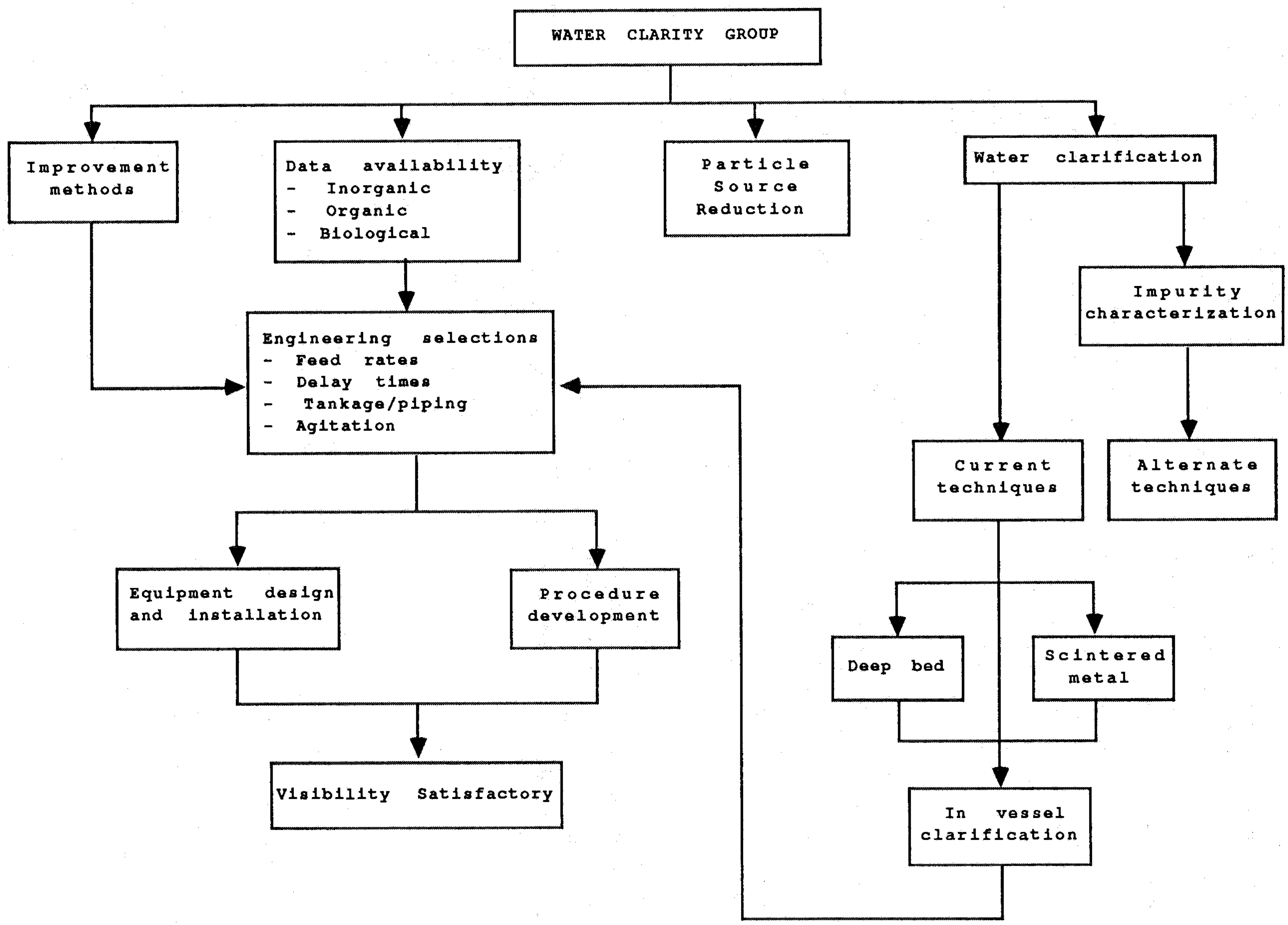


Figure 6-15. Visibility Improvement Logic

coagulants, and body feed and precoating of filters. Initially, some limited success with the DWCS was achieved by reinstalling reclaimed filter canisters that had been backflushed with hot boric acid. At the same time, water was sucked from above the debris bed and discharged at the top of the vessel.

To limit the growth of microorganisms, hydrogen peroxide was periodically added and an water-based hydraulic fluid with a 5% organic addition was substituted in the long-handled defueling tools. Although the canisters continued to clog, the combination gained enough visibility to see several meters down into the water from the work platform and between 15 cm and a meter at the debris bed level.

In December 1986, the NRC approved the use of coagulants in the reactor coolant. Laboratory tests had shown that a coagulant used with diatomaceous earth as a filtration medium could be effective in agglomerating the colloids to filterable sizes and in forming a cake on the 0.5-micron filters. This was verified in a full-scale test conducted with 470,000 liters of water from a reactor coolant bleed tank.

For implementation, the coagulants and diatomaceous earth were injected upstream of the DWCS filter canister. Two different coagulants were used; both were polymer-based cationic coagulants. Dimethyl-diallyl-ammonium-chloride was used from December 23–31, 1986, and from then on, melamine formaldehyde was used. The coolant was processed in a continuous mode.

The technique was so successful that by January 1987, good water clarity had been regained and turbidity levels were generally less than 1 NTU. The efficiency of DWCS filter canisters increased to the point that well over 38 million liters could be filtered through some canisters before they were expended. After the level of suspended solids had been greatly reduced, the coagulants and diatomaceous earth were only needed to deal with unusual increases in the vessel or when new canisters were installed in the system.

In May 1987, the TRVFS was disconnected and removed from the vessel. In the same month, a second train of the DWCS was put into operation—it had the same features as the first train and it permitted the system to operate more flexibly and with greater throughput if necessary. In June, the NRC approved a proposal to cross-connect the systems so that the water in fuel pool "A" could be processed with the use of coagulants and diatomaceous earth without the installation of new equipment (GPUN 1987).

In 1989, additional modifications were made to the DWCS to create the in-vessel filtration system (IVFS), which supported vacuuming and final cleanup of the vessel. This is discussed in Section 8.

6.2.4 Disposing of Processed Water

The means for processing the contaminated water existed in the form of EPICOR II, SDS, and DWCS; but the means for disposing of it had been left unresolved. Under normal circumstances, a substantial portion of the processed effluent would have been discharged to the river. If this option had been pursued at the end of the cleanup, it would have resulted in a river dilution factor of 220,000 or a radioisotopic concentration in the river at the nearest downstream user of less than 6% of the federal limit (Standerfer 1986). From human health and ecological perspectives, there would have been no impacts from doing so. However, the discharge of processed water was a political issue from the beginning.

The issue broke out in full force in the summer of 1979, when EPICOR II was about to begin operation. The City of Lancaster, which draws it water supply 32 km downriver, filed suit against the NRC to prevent any discharge of water.

During the last half of 1979, there was continual interaction within and outside of court to resolve the issue. Upon reviewing the future needs for water for operations such as decontamination and filling the spent fuel pools and refueling canal, project management concluded that the best course of action was to build a storage capacity to complement existing storage. A significant factor in this decision was that there were just too many other issues to resolve. Instead of diverting management attention to press the water disposal issue, a final resolution was deferred to a later time. Two 1.9 million-liter storage tanks were erected to store the water (see Section 6.3.2).

An out-of-court settlement was entered in the U.S. District Court in Washington, DC, on February 27, 1980. Essentially, the Agreement between the NRC and the City of Lancaster stated that TMI-2 could not discharge any accident-generated water into the Susquehanna River until the NRC completed its PEIS or other reviews related to river discharge. In a policy statement accompanying the PEIS in 1981, the NRC Commissioners reserved to themselves approval of any disposal method. Amendments to orders approving the use of EPICOR II and SDS similarity prohibited discharge. The prohibition was later incorporated into the TMI-2 license as a technical specification.

The definition of accident-generated water (AGW) was:

(a) Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings including the primary system as of October 16, 1979, with the exception of water which as a result of decontamination operations became commingled with nonaccident-generated water such that the commingled water had a tritium content of 0.025 µCi/ml or less before processing.

(b) Water that had a total activity of >1 µCi/ml prior to processing, except where such water was originally nonaccident water and became contaminated by use in cleanup.

(c) Water that contained >0.025 µCi/ml of tritium before processing (US NRC 1981).

There was an allowance for discharge; e.g., in the event of an extreme emergency. Some releases were made in accordance with this agreement, specifically water containing minute traces of radioactivity that originated from the industrial waste treatment system. However, none of the ion exchange systems employed at TMI-2 could remove tritium from the water. Hence, from an operational point of view, TMI-2 became water-bound, with operators juggling volumes of water to support operations.

By 1986, the total water accumulation was approximately 7.5 million liters. The tritium concentration was approximately 0.2 µCi/ml (versus approximately 1.0 after the accident). This reduction was a result of decay, evaporative losses, and dilution resulting from mixing. As the end of the cleanup was in sight, it was time to start planning for eventual disposal of processed water in line with the criterion to immobilize all radioactivity remaining at the completion of the cleanup. In January 1986, the NRC Commissioners requested that the project management propose a disposal method.

A 1979 study of disposal options (Negin and Smith 1979) was reviewed vis-a-vis the NRC's PEIS to determine if there were any new options to be considered. The conclusions were unchanged. There were only three feasible methods of disposal: one by discharge to the river; a second by evaporation; and a third by casting into concrete slabs (the third option would result in a 50% evaporation). A summary of the attributes of the three options is provided in Table 6-3.

In mid-1986, the project management proposed the evaporation option, even though it was initially evaluated as the most costly (Standerfer 1986). This decision was based on the fact that river discharge, the least expensive, would be extremely difficult to sell to the public (and hence the state), and that casting into concrete was not really a permanent solution. The NRC staff reviewed the proposal in a supplement to the PEIS and found it acceptable (US NRC 1987). Nevertheless, the NRC decided that public hearings, as requested by several intervener parties, were appropriate. These were held before the Atomic Safety and Licensing Board (ASLB) in the fall of 1988.

In February 1989, the ASLB issued a final initial decision recommending that evaporation be approved. The NRC Commissioners ordered the Board's finding to be immediately effective in April 1989. While interveners pursued various legal challenges, the evaporator was delivered and installed on site. Through late 1989 and early 1990, testing and modifications to the system were underway.

The processed water disposal system (evaporator) had two subsystems—one for evaporation and one for packaging. The evaporator subsystem contained four separate component that changed water from an aqueous to a vapor form. Two were evaporators; one was a dryer; and one a vaporizer. The concentrated solution (bottoms) was continuously recirculated through the concentrate tank. A portion of the recirculating concentrate was continuously drawn off to feed an auxiliary evaporator and concentrate tank for further concentration. The distillate from the auxiliary evaporator was returned to the main evaporator system; the bottoms from the auxiliary concentrate tank were sent to a dryer and pelletizer.

From the dryer, the dry solid waste was transferred to the packaging subsystem. The solids were discharged to a pellet mill and extruded into solid pellets. The pelletizer and drum filling station were in an enclosure that was maintained under negative pressure. The dried, pelletized bottoms were then packed into 200-liter drums and shipped for commercial burial (Cremeans 1990).

6.2.5 *Rejected Processing Alternatives*

As discussed in Section 6.2.2, several alternative methods of processing high-activity water were temporarily pursued.

Table 6-3. Processed Water Disposal Options

A. <u>FORCED EVAPORATION</u>	
Advantages	• Concentrates waste requiring LLW disposal • Insignificant offsite radiological consequences • Minimal SDS/DPRCOR II reprocessing
Disadvantage	• Inadequate, current LLW disposal allocation • Interim onsite storage may be required • Complicated logistics • Highest cost
B. <u>DIRECT SOLIDIFICATION - ONSITE LANDFILL DISPOSAL</u>	
Advantages	• Relatively simple to implement • Short time to complete • Second lowest cost • Decoupled from LLW disposal • Insignificant offsite radiological consequences
Disadvantage	• Requires separate submittal for NRC approval per 10 CFR 20.302 • Requires industrial landfill • Resources approval for leachate discharge • Retains onsite legacy
C. <u>CONTROLLED DISCHARGE TO THE RIVER</u>	
Advantages	• Technically simple • Lowest direct cost • Shortest time to complete • No onsite legacy • Minimal workforce required • Simple logistics • Insignificant offsite radiological consequences
Disadvantages	• PA Dept. of Environmental Resources approval required

6.2.5.1 Closed-Cycle Evaporation

Several years before the project team selected the open-cycle evaporation described above, a closed-cycle evaporator had been considered for several uses:

- Processing the containment basement water
- Processing oils and chemical decontamination wastes
- Processing water for decontamination flushing
- Processing water to fill the spent fuel pool.

No evaporator was ever used for these purposes, although extensive evaluations and design work took place. Evaporators were appealing because they are more tolerant of influent chemistry than are ion exchangers, and evaporators had been widely used in radwaste services throughout the nuclear power industry. Unfortunately, they were long lead-time components that were not quite as effective as ion exchangers in reducing the volume of the final waste form. They were also known to be more difficult to maintain and had higher associated exposure rates.

In 1979, the TMI-2 plant's existing boron recovery evaporator was a 9.5E–04 m^3/s boric acid concentrating and gas stripping system located in the auxiliary building (BNI 1982). This evaporator was unsatisfactory for projected decontamination chemical waste processing because substantial ALARA concerns existed regarding maintenance of the system if it was used with highly radioactive waste water (Lyman 1980; E/STG 1981).

Later, the project team considered using it to support the refill of the spent fuel pool/refueling canal volumes and to provide boron-free water for decontamination (Katonak 1983). This use was deemed to be too complex because of the requirements to tie it into existing systems and provide a new source of steam, which, because of the separation of Units 1 and 2, could not be supplied from Unit 1. The importance of simplicity and the disadvantages of modifying existing operating systems outweighed the potential savings associated with reclaiming boric acid.

The DOE had shipped a small (1.6E–03 m^3/s) electric evaporator to the site with the intention of using it in a proposed decontamination demonstration facility (see Section 7). The facility was never built and the skid-mounted, self-contained, pot-boiler-type evaporator was stored in a warehouse. This evaporator could have been used for either evaporating chemical effluents from the decon demonstration facility or, perhaps, supplying unborated water for decontamination (BNI 1982). Its small size and low capacity ruled against its use in any larger-scale radioactive waste processing.

6.2.5.2 Evaporation/Solidification Facility

In 1979 and 1980, an evaporator/solidification system was considered to back up the SDS and to process chemical decontamination solutions. This alternative was pursued for two years before it was dropped because of cost.

Section 6.2.2 describes the selection of the SDS for processing high-activity radioactive water. Eliminated by this selection were a natural-circulation, closed-cycle evaporator and a forced-circulation, closed-cycle evaporator. The forced-circulation evaporator was purchased as a contingency, however, because the reliability of the SDS was unproven and too much uncertainty surrounded future decontamination plans. For example, until disproved by samples taken in September 1979, the project team believed the containment basement water contained oils that would foul the SDS.

The location of the evaporator/solidification system was originally planned for the model room in the fuel handling building (adjacent to the truck bay on El. 305'). The location was changed in mid-1979 to a reinforced concrete building to be built adjacent to the diesel generator building on the west side of the plant. The model room would have been ideal from a materials handling point of view but suffered from being in a congested area where other recovery systems competed for space.

The cost for the proposed facility consequently went through a significant change. The initial estimate for a relatively simple evaporator/solidification system in the model room was approximately $5 million (Williams 1979). In the separate facility and as the design increased in complexity because of support systems and interferences, the estimate reached well over $24 million (E/STG 1981).

The evaporation/solidification facility would have occupied an area 3.7 by 18.9 and 20.4 meters high. The 1.9E–03 m^3/s evaporator/crystallizer process equipment could have operated in a range of modes producing either low or high concentrations of solids in the discharge slurry. If the process equipment operated as an evaporator, it functioned in the conventional evaporation mode. If operated at high solids discharge concentrations, it functioned in the crystallizer mode. The solidification system would have been capable of processing evaporator/crystallizer concentrates or spent ion exchange resins.

Besides the escalating estimate, three major factors worked against starting construction of the evaporator/solidification facility:

- Two-year construction schedule
- Inherent potential for higher personnel exposure during operation because of increased system maintenance
- Difficulty in handling the highly radioactive waste forms in an ALARA manner.

Most of the engineering work and funds were directed to the SDS to provide a short-term solution to the higher-activity water situation. Design work still continued on the evaporator/solidification facility because:

- Forced evaporation was seen as the most effective and economical method of processing chemical decontamination solutions.
- An evaporator would provide maximum operational flexibility, allowing the early decommissioning of the SDS and the consequent freeing of the fuel handling building for fuel storage.
- An evaporator could serve TMI-1 after the cleanup of TMI-2 and thus have long-term usefulness.

The corporate budgetary constraints of 1980–81 (see Section 2) ensured that the facility would not go beyond the design stages. In this atmosphere of limited resources, the project team conducted several reviews of the radioactive waste situation and reached the following conclusion: The ability to evaporate decontamination solutions, back up other water processing systems, and solidify various wastes would be useful, but was not urgent. In fact, the extent to which decon solutions might be used appeared far less than previously thought—so much less that evaporation might not be cost effective. And, everyone agreed, the facility was too expensive (RSFTG 1980; E/STG 1980; Menzel 1981).

The evaporator/solidification facility was cancelled. It had been a long-term solution whose schedule slipped so as not to divert funds from the SDS. In the end, it was a luxury that could not compete with existing and satisfactory solutions. Its absence required that special care and expense be taken to avoid chemicals that would be incompatible with the SDS or EPICOR II. An evaporator would have been useful, but not of critical importance.

6.3 Water Storage

For the months immediately following the accident, the problem of water storage was critical. A 680,000-liter storage capacity existed at TMI-2 before the accident; of that, only 190,000 liters of surge capacity were available in March 1979 (Kalman 1984). The growing water volume quickly outstripped the storage capacity during the accident. Even after the increase of radioactive water had been contained, the volume continued to grow as nonradioactive water was added from normal system makeup, cooling water leakage, pump seal drains, rain, condensation, and the like.

For the next three years, the water volume and storage capacity had to be tightly managed. At great expense and effort, enough new storage capacity was either converted from other uses or constructed to contain this liquid. Figure 6-1 depicts the growing volume of water (both processed and awaiting processing) and the available tankage.

Water storage was such a major undertaking because of the ban on discharging processed water to the Susquehanna River. To add to the difficulty, the project team segregated all water according to these criteria:

- Boron concentration for potential criticality control
- Whether the water was classified as accident-generated or not.

Another criterion—tritium concentration—was considered, but rejected because it would have added to the existing complications, taken resources, and required additional hardware.

By the end of the project, almost 9.5 million liters of storage capacity existed. The following sections discuss the most important tanks.

6.3.1 Fuel Pool Waste Storage System (Tank Farm)

The first emergency water storage facility was a six-tank, 416,000-liter group referred to as the "tank farm", located in spent fuel pool "A" and operational in July 1979. It gave the project team the necessary surge capacity to relieve concern and also provided vital staging for later water processing by SDS. It was the only capacity added to accommodate higher-activity water. Section 3.6.2.2 describes the installation and use of the tank farm.

The tanks were removed between May and October of 1984, to make space in the fuel pool for storing core debris canisters. Planning for, removing, and decontaminating the tanks were time-consuming operations that involved extensive cutting of pipes and support structures, arranging for radiological and airborne contamination protection, decontamination, and refurbishing and requalifying the spent fuel pool.

6.3.2 Processed Water Storage Tanks

The project team had to build more storage volume to hold processed water containing minute traces of radioactivity. Estimating how much volume to build was more art than science. No one could say with certainty in 1980 how much would be needed. A 3.8 million-liter capacity was a rough and generous estimate that proved to be adequate with enough excess capacity.

Whether to build one large tank or several small ones was also a question. A single large tank would have prevented water from being staged for different purposes during storage. Building two 1.4 million-liter tanks seemed the simplest and most efficient choice, and allowed some flexibility. The decision not to use tritium concentrations as a criterion for segregating water volumes also eased the potential complexity of needing several smaller tanks.

Construction of two outdoor tanks on the east side of the auxiliary building began in March 1980. The 1.4 million-liter, epoxy-coated carbon steel tanks were put into service in July 1981. Their primary function was to collect processed water from the EPICOR II and SDS feed-and-monitor tank system and store it for recirculation when needed (the capabilities of recirculating and sampling the water and returning it to the plant were not available until several years later).

The tanks were atmospherically vented via open-ended 30-cm roof vents. A pump recirculation system and an eductor system mixed the contents, which were limited such that a tank failure would not result in radionuclide concentrations, at the nearest drinking water intake, that would exceed 10 CFR Part 20 Appendix B limits. Normal operation of the system was a batch mode. After one storage tank had received a batch, it would be isolated and the contents recirculated and sampled. Based on the chemical and radiological results of the sample, the contents were stored, transferred for use, or routed to EPICOR II for further processing (GPUN 1986c).

6.3.3 Other Tanks

Because the plant was water-bound, plant operations personnel needed extra room to maneuver and/or temporarily stage water. Thus, existing tanks were converted to new purposes. For example, two 850,000-liter stainless steel condensate storage tanks existed on the south side of the plant, adjacent to the turbine building. These tanks were normally used to contain nonradioactive makeup water for the secondary system. In 1980, one of the tanks (COT-1A) was converted to store water containing low levels of radioactivity and boron. This water was used principally for decontamination flushes.

Some tanks, such as the two associated with the steam generator cleaning facility, had not been used for their original design function and so were used in conjunction with EPICOR II.

The size of several of the tanks in the plant limited the batch sizes used in processing water. For example, the reactor coolant bleed tanks were approximately 280,000 liters. So, factoring in a surge capacity, the average size of every batch of reactor coolant processed could only be approximately 190,000 liters. This limitation hindered the rate of processing liquid in the plant.

6.4 Solid Waste Staging and Preparation for Disposal

The TMI-2 plant site had little need for solid radioactive waste staging facilities before April 1979. As the cleanup gained momentum, various forms of solid waste rapidly accumulated, particularly trash and spent resin vessels from EPICOR I and II (and most importantly, the EPICOR II prefilters). The political situation made shipping such waste off site problematic. In addition, the regulatory transition regarding low-level waste disposal potentially affected the existing available commercial disposal sites at Richland and Barnwell. Consequently, the project team was forced to build onsite storage facilities for temporarily staging the waste.

The temporary staging designation of these areas (i.e., less than five years storage) is important because the facilities had to be used for waste that was being held pending shipment. Wastes staged here had to be in a configuration ready for shipment and disposal—an added headache. A permanent storage facility (greater than five years) would have required licensing changes and would have violated the policy of not letting TMI-2 become a waste disposal site (McGoey 1980).

6.4.1 Interim Response Facilities

Three interim staging projects were begun (See Figure 6-3 for locations). Two of the facilities were quickly adapted or constructed after the accident—the paint storage shed ("southeast acres") and the temporary radwaste staging area. These are discussed in Section 3.6.2.3.

The solid waste staging facility ("waste acres") was a more substantial engineering effort to provide stable staging space. This facility was built in modules for flexibility. Originally planned as a series of six modules containing 60 cells each, only two of the modules were actually built—each contained six rows of 10 cells. The flexibility was necessary to ensure that any sudden deprivation of disposal options could be handled. The facility was located in the Unit 2 desilting basin. Each rectangular concrete module was approximately 15 m wide by 27 m long by 6 m high.

The module base and walls were 1-m thick to ensure that surface readings remained below 5 mrem/h. The modules were located inside of the protective dike that surrounded the station and were elevated. The 2-m dia. by 4-m high cells in each module consisted of concrete-shielded, galvanized, corrugated-steel cylinders with welded steel base plates. A drain line from each cell led to a common sump. Each cell was covered by a concrete lid, 1-m thick. A mobile crane could load each cell with either one 1.8- by 1.8-m or two 1.2- by 1.2-m expended processing vessels (US NRC 1981). Figure 6-16 depicts the solid waste staging facility.

By late 1980, the project team calculated that still more temporary storage space would be needed for the expected rate of radioactive waste generated by both Units 1 and 2. Studies indicated that the storage capacity for low-level waste in 208-liter drums and low specific activity (LSA) boxes was rapidly diminishing. The answer was the interim solid waste staging facility (ISWSF) or "carport", which was ready for use in December 1982. It was sized for six months of Unit 1 and 2 waste (i.e., 810 208-liter drums, 90 LSA boxes, and 60 1.4-m3 vessels (RSFTG 1980; Negin 1984).

The pace of design and construction varied as a result of budget limitations, changing perceptions regarding the urgency, and the changing schedule of the hoped-for Unit 1 restart. During and after construction, various problems were noted that required remedial action; e.g., floor and wall surfaces had not been coated to allow decontamination and sidewalls were not planned except for the shielded storage areas and these did not go to the roof. Sidewalls complete to the roof were required to prevent rainwater from blowing in.

The building had a 44- by 20-m concrete pad protected by a roof and aluminum sidewalls. A partial-height concrete block wall enclosed an area where higher activity waste was stored. Six sumps collected any in-leakage. The facility acted as the primary DAW and LSA storage area for the entire TMI station.

6.4.2 Preparing Low-Level Waste

The higher-activity radioactive liquid and solid wastes posed the greatest challenges, but there was an enormous amount of lower-activity solid waste to be dealt with. Mostly it comprised trash from decontamination operations, solidified sediments and resins, and used tools and material. Controlling, cleaning, and/or preparing it for shipment were expensive and time-consuming operations.

Meeting the challenges of 10 CFR Part 61 required the project team to: 1) develop waste streams based on various cesium-137 to strontium-90/yttrium-90 ratios; 2) selectively package DAW; 3) strictly control liquid processing; and 4) submit an exemption request permitting the waste to exceed Part 61 limits on strontium-90/yttrium-90 concentrations for EPICOR II resins.

Further support for waste disposal came as the result of the qualification of a Ferralium 1.2- by 1.2-m HIC for burying DAW in the Richland site. This Ferralium HIC was later modified and qualified as a processing vessel and used in the processing stream of both EPICOR II and DWCS.

6.4.2.1 Volume Reduction of Lower-Activity Waste

The quantity of lower-activity radioactive waste was difficult to control because the cleanup had to proceed as quickly as possible. Consequently, the project team focused on controlling the final volume to be shipped. This was done by decontaminating and reusing equipment or material whenever possible, solidifying waste when necessary, and boxing or compacting the rest.

Designated decontamination areas were established in the plant in order to recycle as much equipment and material as possible (on El. 328' in the auxiliary building and El. 347' in the containment).

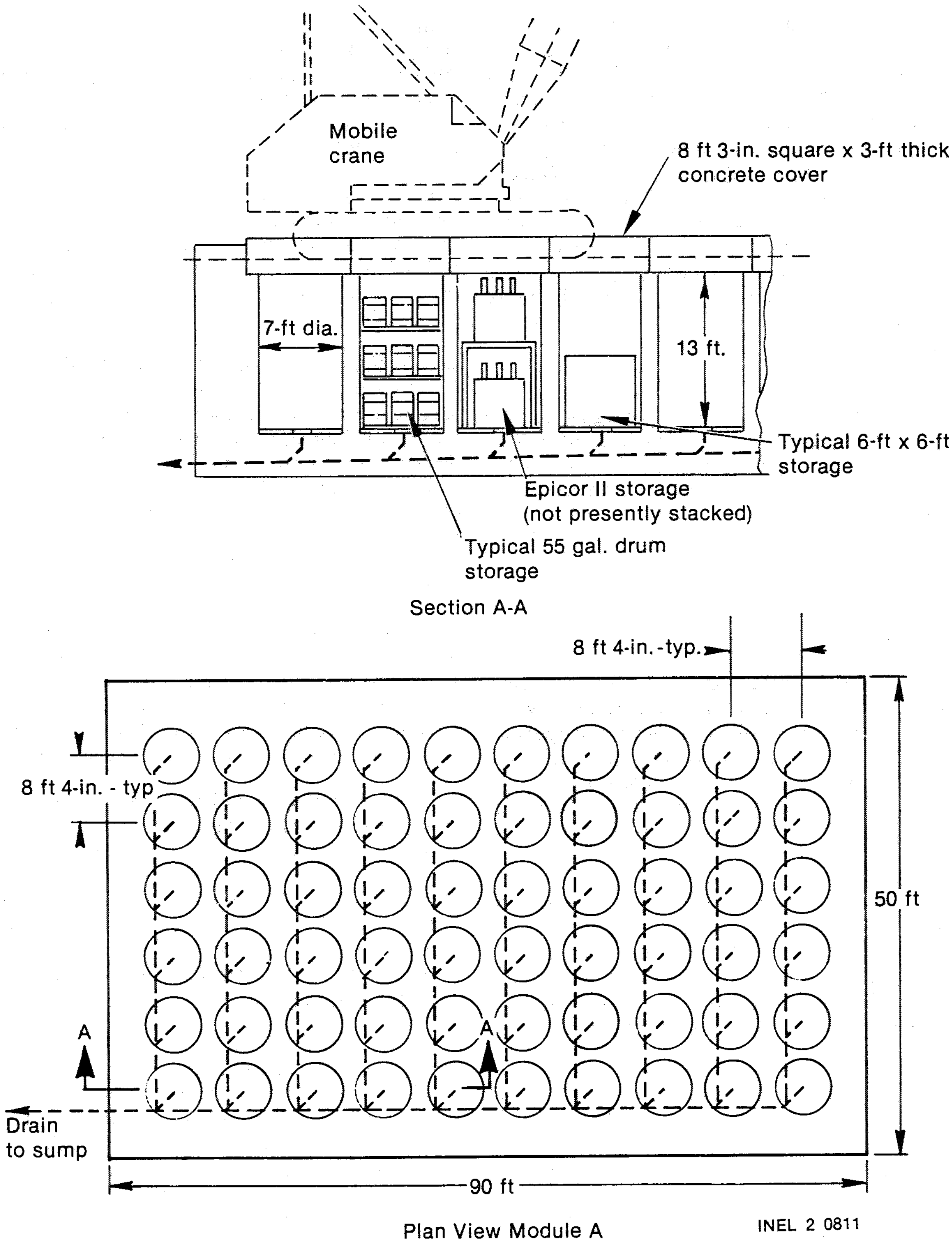


Figure 6-16. Solid Waste Staging Facility

A 208-liter drum compactor was located in the model room on El. 305' of the fuel handling building. The compactor operated at a pressure of 15,000 psi (later upgraded to 30,000 psi) (BNI 1982). Its use was limited to LSA waste and no trash over 500 mrem/h was compacted. Approximately two thirds of the DAW proved to be compactible. As the estimated and real volume of DAW increased, the reliability and production capacity of the compactor were not sufficient; so a new compactor (60,000 psi) was purchased and used in the newly constructed waste handling and packaging facility (see Section 6.4.2.3). By mid-1988, over 6,000 drums of compacted waste had been shipped for burial (Deltete and Hahn 1990).

In August 1988, DAW shipments were begun to an offsite super compactor. Selective packaging and the ability to have TMI-2's waste mixed with other GPU waste containing lower concentrations of strontium-90/yttrium-90 ensured that the end product did not exceed Class A burial limits. Waste processing and disposal costs involved with Class B and C wastes were reduced and less of the burial ground allocation was used.

6.4.2.2 Solidification of Lower-Activity Waste

Several types of waste generated by the accident or during the cleanup required solidification. Since no onsite solidification capability existed in 1979, the normal course of action when solidification was required was to contact a vendor, who would then deliver and operate a solidification system. To transfer the wastes from the original location to the solidification site required the modification of existing systems and, in some cases, the design and construction of new transfer systems.

The sediment in the containment basement posed a special problem because of its inaccessibility and the potential that it would dry out and become airborne. Design was begun on a sediment transfer system that took advantage of the remote reconnaissance vehicle (RRV) to gather up the mixture of dirt, riverwater sediment, concrete dust, and small amounts of fuel on the basement floor.

The sediment was pumped up to El. 305' of the containment and then to a spent resin storage tank for decanting (i.e., concentrating), sampling, and sluicing to solidification vessels (see Section 7). The simplified system was also adapted to pump sediment from the auxiliary building sump and sump tank. Solidification of these waste streams had to meet the stability requirements of 10 CFR Part 61.

Liquid wastes such as those in the concentrated waste storage tank (CWST) also required solidification. A 208-liter drum solidification unit was purchased and installed in the model room of the fuel handling building in 1986. It handled the 10 CFR Part 61 Class A wastes from the CWST.

Transfer and solidification of the spent resins in the makeup and purification demineralizer vessels was a more complex activity because of the concentrations of fuel and fission products and the agglomerated nature of the severely degraded resin beads. This operation is described in more detail in Section 7.

6.4.2.3 Waste Handling & Packaging Facility

As described above, insufficient space was available for staging and destaging lower-activity waste and containers, and little equipment was available to dismantle, decontaminate, or temporarily store the tools, fixtures, and large equipment needed for large-scale cleanup operations. (See Section 7 for a discussion of a proposed DOE decontamination demonstration facility and Section 8 on the proposed containment recovery service building.)

An evaluation of a special facility to perform waste management functions was made in mid-1983 (Lengyl 1983). In early 1985, specific recommendations were made for a facility to handle and package solid, lower-activity radioactive waste (Levin 1985). By February 1987, the 650-m2, $1.4 million-facility was operational.

The waste handling & packaging facility (WHPF) contained essentially the same equipment as the El. 328' facility in the auxiliary building but was more logically laid out and on a larger scale. The WHPF was justified by cost savings resulting from the commercial release of decontaminated material, improved packaging efficiency for noncompacted material in boxes, and the improved packaging efficiency for compacted material in drums (Deltete and Hahn 1990).

In terms of volume reduction, the new facility was a big improvement; e.g., packaging efficiency was improved by 25–30% and significant quantities of metal and other items were released for commercial scrap or reuse on site. Sixty-nine percent of the items brought into the WHPF in the first six months were successfully decontaminated, as compared to 31% by the decontamination facility on El. 328' of the auxiliary building (EPRI 1988).

After initial use of the facility for separation, decontamination, and volume reduction, its role partially changed. With the use of the super compactor vendor, the primary functions of the WHPF became to separate and decontaminate waste for release as clean and to package LSA boxes efficiently for shipment to the super compactor.

6.4.3 Rejected Processing Alternatives

Several alternatives for processing higher-activity solid waste were studied and designed. They were not implemented for a variety of financial and technical reasons.

Several volume reduction systems had been considered; specifically, an extruder/evaporator, an incinerator, and a super compactor (Delete 1985). Upon evaluation, the idea of volume reduction for higher-activity waste was abandoned because:

- No one volume reduction technique could effectively process all of the different physical forms of the waste.
- The installation costs for the viable systems were between $4 and $30 million.
- The procurement and installation lead-times and licensing issues were considerable and would have detracted from other commitments.
- The total volume of such waste was too small to justify the cost savings of onsite processing.

6.4.3.1 EPICOR II Resin Solidification

As described in Section 6.2.1, disposing of the EPICOR II prefilter resins was a difficult issue for both the project team and the NRC. The solution dictated by the NRC when it approved the operation of EPICOR II had been that the resins must be solidified to ensure safe shipment. A similar ruling on SDS resins was expected. This solution was in response to general industry problems with leaking vessels discovered at commercial burial grounds and public concern; it was also part of a general evolution in thought. On the subject of shipping radioactive resins, the NRC had been debating the virtues of solidification versus dewatering since before the accident. All new reactors were required to have a solidification system; older reactors were responsible for compliance with the intent of the requirement on a case-by-case basis.

The design of a major solidification facility was therefore undertaken by the project team, but the facility was never built. Project management was reluctant to solidify the resins because:

- Substantial technical problems existed with moving the highly radioactive resins from their existing vessels. Included in this was the potentially high personnel exposure for transferring resins.
- The technology for solidification was not well understood (solidification had been used for evaporator bottoms but not for resin beads). And the criteria and methods for achieving the required "homogeneous monolith" were not clear.

Consequently, while designing a solidification facility, the project management also investigated the option of using a high integrity container. The NRC eventually accepted this as a legitimate means of disposal.

In July 1979, project management anticipated that a solidification order would accompany permission to operate EPICOR II and so performed a value/impact study (DeVine 1979). The study concluded that the radiological and safety hazards associated with handling the large quantities and high radioactivity of unique TMI-2 resins were significant. The hazards overshadowed the slight value gained by solidifying the resins to decrease the risk of transportation accidents. In addition, the financial cost and delay caused by building a solidification system were substantial. The only potential value would have been the public's perception of improved safety during shipment.

Nevertheless, the October 1979 Order from the NRC specified the expeditious construction of a solidification facility. Conceptual plans were begun immediately. Several aspects were studied, including in-vessel versus ex-vessel solidification, what type of binder to use, and whether to build a separate facility or tie into the planned evaporator-crystallizer/solidification facility (see Section 6.2.5).

Because ex-vessel solidification involved sluicing—which raised ALARA concerns—modification to the existing EPICOR II vessels to permit in-vessel mixing was preferred. However, during this time, the auxiliary building water processing campaign was well underway and so many of the EPICOR II prefilters had already been used. The prefilters were the vessels of most immediate concern because of their higher activity levels. Project management did not want to delay the water processing while awaiting redesigned vessels that would permit in-vessel solidification. Consequently, planning had to include:

- Modifications to the vessel design for future in-vessel solidification, and
- A method for transferring resins from existing, used vessels to new vessels.
- In addition, a portable cement solidification system was studied for the treatment of second- and third-stage EPICOR II vessels, which did not contain as much radioactivity.

The concept of a separate solidification facility raised concerns about the complex nature of transferring resins, occupational exposures, and the possible plugging of piping and instrumentation lines. The separate sluicing facility studied would have optimistically taken a year or more to design, construct, and place in operation, and would have cost between $4 and $6 million. A study tentatively located the facility in the desilting basin near the temporary radwaste staging area (Miller, et al. 1980).

Separating out the planned evaporator-crystallizer/solidification facility's solidification system so it could also be used with EPICOR II resins was rejected because of technical problems related to pumping evaporator concentrates and the impact on construction costs (Hovey 1980).

Technical difficulties surrounding the solidification medium were also of concern. The desire for a proven system design with quick availability led to the selection of cement as the solidification agent. Urea-formaldehyde systems did not perform in a decontamination solution demonstration at the site; an asphalt system had operational experience in Europe but the cost of adapting it would be too high; and DOW binders had no full-scale operating experience. Cement appeared to be the logical choice, but a general lack of in-depth knowledge surrounded the performance of organic resins with cement. Evidence existed that the cement/organic resin matrix could swell and break apart if process parameters were not rigorously controlled.

The lack of specific criteria in this instance reflected the general regulatory changes in progress regarding radioactive waste disposal. In the midst of this, a way out of the dilemma appeared: on January 29, 1980, the NRC notified all power reactor licensees that, effective July 1, 1981, spent resins and filter media with greater than 1 μCi/cc of long-lived isotopes had to be stabilized by solidification. However, in lieu of solidification, an alternative such as a high-integrity container (HIC) could be proposed to the NRC and the states licensing the burial sites. This alternative—technically undefined at the time—was pursued by the cleanup project in parallel with the design of a resin solidification facility. (The actual use of a HIC to dispose of EPICOR II vessels is described in Section 6.5.1).

Studies and conceptual designs related to a resin solidification system continued through 1980. Management remained unwilling to commit limited funds for detailed engineering and construction until all design issues had been resolved and the acceptability of solidified EPICOR II resins for shallow-land burial had been approved (Hovey 1980). In January 1981, the NRC directed the project management to proceed immediately with either developing definite solidification plans or proposing alternative methods, which could include a request to be relieved of the requirement for solidification (Ahearne 1981).

The project team requested relief from the solidification requirement in February 1981. The request was based on four main reasons:

- To solidify the resins could rule out alternative options that might ultimately be selected by the NRC and the receiving organization as the preferred (or required) form for safe disposal.
- The operation of a solidification facility would inherently subject employees to radiation exposure that would likely be greater than from other alternatives—thus contradicting the principle of ALARA.
- The degree of confinement provided the spent EPICOR II ion exchange media in storage at TMI-2 was adequate to protect the health and safety of the public, making expeditious solidification unnecessary.
- Up to 25 lower-activity vessels could be shipped in the immediate future to a shallow burial site, and dewatered (not solidified) in accordance with existing NRC and burial site requirements in the same manner permitted for other licensees' LSA waste (Hovey 1981).

With a DOE agreement to take the EPICOR II prefilters in the offing (see Section 6.5), the NRC Commissioners approved deletion of the solidification requirement in March 1981.

In summary, the solidification of EPICOR II resins was pursued by the project team in response to an NRC ruling; however, the value of such a complex, expensive, and non-ALARA operation was debatable. As the regulatory and burial criteria were clarified (partly in response to this issue), alternative disposal methods became more appealing. The eventual agreement by the DOE to accept commercially non-disposable wastes from TMI-2 to research and develop high integrity containers satisfactorily resolved the issue and proved the wisdom of not solidifying the resins.

6.4.3.2 Incineration

As an alternative to compacting lower-activity solid waste, the project team investigated the use of an incinerator to reduce the volume. In 1981, the idea was studied and finally discarded because of political, financial, and regulatory uncertainties.

The possibility of installing an incinerator existed because the DOE already had a developmental project underway to install a controlled air incinerator at a commercial nuclear power plant. The cleanup project management hoped that the DOE would provide and license the incinerator on a commercial "turnkey" basis, with the project team operating the demonstration facility (RSFTG 1980). The idea was thought to have a good chance of success because many utilities were at least considering an incinerator for volume reduction.

In October 1981, the project team produced a study intended to provide the basis for a shared radwaste incineration project (BNC 1982). The incinerator facility was envisioned to be housed in a pre-engineered, metal frame building located south of and adjacent to the south dike, where it would pose the least interference with existing structures and not raise ALARA concerns. The building would be 43 m long by 15 m wide, with an eaves height of 6 m, and it would have a reinforced concrete foundation and internal shield walls for drummed ash storage.

The layout of the facility would have encompassed a waste receiving area, an incineration process area, an ash handling area, and an area for all support services, taking into account material and ventilation considerations. The controlled air incinerator system itself included an incinerator, wet offgas scrubbing components, dry offgas module, induced draft blowers, and an ash removal system. Nonradioactive support services equipment located adjacent to the building included a fuel oil storage tank, a demineralized water storage tank, a closed cooling water heat exchanger, an air-cooled condensing unit, and a power transformer. One of the two variations studied contained a cement solidification system; the other contained allowance for transport of the ash elsewhere for solidification.

Several technical issues were of concern:

- Controlling the waste forms and types would have been demanding. For example, burning PVCs and rubber would have released hydrochloric and sulphuric acids in the stack effluent. Preventing this would have required burdensome administrative screening or calcination of the offgas.
- Based on conversations with vendors, a significant number of unknowns existed relative to equipment supply and system operation and performance. The expected life of the refractory material had not been established, and one of the R&D aspects of the project would have been to measure the degree to which the material retained radiation.
- The incinerator refractory would probably not last as long as the facility itself, thus extensive disassembly would be required with potential radiological problems.

In spite of the technical concerns, the idea had appeal because it would be the first incinerator installed at a commercial power plant to handle beta-gamma contaminated wastes. The main obstacle was that the cost and cost-sharing aspects of the project could not be satisfactorily determined for several reasons:

- The amount of waste to be generated could not be accurately projected at the time.
- The location and costs of burial sites were uncertain.
- The extent of DOE financial participation might not be great enough to justify the cost to the utility.
- Building and operating the incinerator could entail extensive licensing problems—not so much with the federal or state authorities, but at the local level, with such things as public challenges to building permits.

The final blow came in December 1981. The DOE, based on a change in policy, decided to dedicate available funds to smaller, proof-of-concept projects rather than fund large-scale demonstration projects such as onsite incineration.

In consequence, project management deferred and then abandoned any planning for an incinerator. Compaction and administrative procedures remained the primary methods for waste volume reduction.

6.4.4 Preparing Processing Vessels

Preparation to ship the vessels used in water processing required special measures that revolved primarily around the issue of hydrogen generation.

6.4.4.1 EPICOR II Vessels

After processing the auxiliary building water in 1979 and 1980, the 50 1.2- by 1.2-m EPICOR II prefilter vessels were stored in the solid waste staging facility pending shipment. In preparation for shipment by the DOE, an analysis was needed to address concerns about potential hydrogen generation that might exceed the shipping limits and about the long-term stability of the resins, particularly the extent to which they would decompose and alter the pH of freestanding water. A low pH could lead to corrosion and the breaching of the carbon steel vessel.

To address these questions, the DOE contracted with a laboratory in May 1981 to characterize one prefilter. Before the vessel was shipped from TMI-2 to the laboratory, it was manually vented and, as predicted, a combustible gas was detected (Sheff 1981). The laboratory later confirmed the presence of hydrogen. Based on this information and the concern for safety, the 49 remaining vessels were required to be purged of hydrogen gas and the hydrogen generation rate determined for each vessel (Queen 1983).

The analysis of the prefilter indicated that the vessel contained 12% hydrogen by volume, and so DOE and the project team began developing a method of venting the remaining 49 vessels to below the flammable limit of 4% by volume to ensure safe handling. This work resulted in the development of a remote gas sampling tool and the construction of inerting facilities.

A remotely operated vent tool was devised to remove the vessel vent plug while maintaining a sealed environment around the module cell. Guided by cameras and mirrors, the tool was lowered from within a shielded, explosion-proof blockhouse onto an EPICOR II prefilter stored in a cell of the solid waste staging facility. A sampling and purging system worked in conjunction with the vent tool to safely sample, analyze, and purge the vessel of radiolytic gases through a 30-m stainless steel hose. A remote support facility containing the sampling and purging system was located in a mobile trailer, which was stationed on top of the solid waste staging facility.

The project team used the venting equipment to successfully purge and inert the 49 EPICOR II prefilter vessels in storage at TMI-2 before their shipment. The remote nature of the equipment precluded exposure, avoided hazards associated with critical quantities of hydrogen gas, and provided data to aid in the processing of highly loaded ion-exchange media (Queen 1983).

Later studies of hydrogen generation in EPICOR II vessels were sponsored by the NRC and resulted in changes to the certification of shipping casks. The NRC published an information notice requiring waste generators to demonstrate, by tests or measurements, that combustible mixtures of gases were not present in radioactive waste shipments; otherwise the waste must be vented within 10 days of shipping. A task force, formed by the Edison Electric Institute to evaluate these NRC requirements, developed a calculational method to quantify hydrogen gas generation in sealed containers. EPRI then demonstrated this calculational method using a desktop computer at TMI-2; it was accepted by the NRC (Flaherty 1986).

6.4.4.2 SDS Vessels

The SDS vessels used to process containment basement water and reactor coolant posed a similar problem regarding hydrogen generation and shipment. Like the 50 EPICOR II prefilters, 19 of the SDS vessels were destined for shipment to the DOE. The hydrogen issue was more formidable because these vessels had to be stored underwater on racks in fuel pool "B" and so were less accessible. In 1981, the project team knew there would be a problem with hydrogen gas and was tracking the generation rate. The issue was not really resolved until after the SDS began operation and several vessels were expended.

Three major technical solutions were evaluated to address the generation of radiolytic hydrogen and oxygen:

- Dry (heat and vacuum) the vessels before shipment
- Vent the system
- Add an adequately sized catalyst bed and pressure relief system.

Catalytic gas recombiners were selected in the belief that they could control, at nonflammable levels, the mixtures of H_2 and O_2 gases resulting from radiolytic decomposition of residual water. The catalyst recommended for gas recombination testing was commercially available palladium-coated porous alumina pellets.

The tests of a catalyst bed in the vessel demonstrated that recombination of the gases back into water would permit safe shipment of the sealed vessels. Catalysts were then loaded into an available screen assembly in each vessel except the first shipped to the DOE. The catalyst was not needed for this vessel because the gas generation rate was sufficiently low. Vessel pressure monitoring ensured that net gas generation had stopped and that hydrogen and oxygen concentrations were kept below flammable limits. As a result, over 99% of the hydrogen/oxygen generated was recombined (Henrie 1986).

6.5 Solid Radioactive Waste Disposal

For two years after the accident, disposing of solid radioactive waste seemed a discouragingly formidable challenge. The issue of shallow land burial of radioactive waste had been controversial before the TMI-2 accident. Concerns about leaking burial containers, deficiencies in shipping practices, and too much waste had prompted the Richland WA, Barnwell SC, and Beatty NV sites to scale down or temporarily close operations. The governors of these states did not want responsibility for all the Nation's radioactive waste; and they wanted a national policy on radioactive waste, stronger NRC enforcement action, and more guidance. This growing crisis led to the Low-Level Waste Policy Act of 1980, and 10 CFR Part 61 (Licensing Requirements for Land Disposal of Radioactive Waste) in 1982.

With the TMI-2 accident, the governors of Washington and South Carolina prohibited their sites from accepting accident-related waste. The Richland site was reopened to TMI-2 in the summer of 1979, when the governor was convinced that only waste typical of normal power plant waste (in terms of concentrations of radionuclides) would be shipped there. The governor of South Carolina remained concerned about the volume and TRU contaminants in the TMI-2 waste.

Although GPU could have challenged the Barnwell prohibition, it elected not to—in part because the Richland site was available and, considering transportation and disposal costs, about the same cost. Barnwell was reopened to TMI-2 accident-related wastes in 1987, when the new governor was convinced that the limits and classifications of wastes and the waste volume allocations were reasonable (Deltete and Hahn 1990).

The most difficult challenge associated with this national waste management picture was how to dispose of waste that exceeded Class C burial limits. Disposing of it was beyond the control of the project management alone because the wastes were not comparable to those generated at an operating nuclear power plant:

- The wastes contained a high concentration of fission products or small quantities of fuel materials
- The waste processing systems had not always been configured to produce wastes in the form and concentrations allowed for shallow land burial; e.g., the SDS vessels. (One of the most challenging aspects of 10 CFR Part 61 was the requirement that Class B and C wastes had to be either solidified or placed in a HIC.)

As part of the solution, the DOE and NRC signed a Memorandum of Understanding in July 1981 (Snyder 1981), to ensure that the TMI site did not become a long-term waste disposal facility. The agreement also took advantage of the chance to learn from the accident. The DOE agreed to evaluate each waste form to determine the R&D value, and, if of value, to accept the waste for research and later disposal. If the waste was not of research value or could not be made acceptable for commercial disposal, DOE would temporarily accept and store the waste from TMI-2 on a full cost-reimbursable basis. This agreement was the crucial ingredient in disposing of all TMI-2 radioactive waste that exceeded Class C limits.

The agreement identified six types of accident-generated solid radioactive wastes and potential means of disposal:

- EPICOR II wastes—For the highly loaded prefilters, the DOE proposed to develop a high-integrity container that might allow commercial land burial at Richland. Characterization work would also be performed on one or more of the vessels.
- SDS Wastes—For the 19 highly loaded SDS vessels, the DOE would conduct a waste immobilization R&D and testing program, including monitored retrievable burial.
- Reactor Fuel—Initially, the DOE planned to take samples for analysis characterization and research while the balance of the fuel debris remained on site in the spent fuel pool. Final disposition would await resolution of the national spent fuel storage issue. Because that issue was obviously going to take a long time to resolve, the DOE and NRC modified the Memorandum of Understanding in March 1982

(Snyder 1982), so that the DOE accepted the entire reactor fuel core. Part would be used for R&D; the remainder would be stored until ultimately disposed of under an agreement to be negotiated between DOE and GPU. (See Section 8 for more information of the disposal of reactor core debris.)

- Transuranic (TRU) Contaminated Waste—TRU waste that could not be qualified for commercial burial would be considered by DOE on a case-by-case basis for either generic R&D or cost-reimbursable storage or disposal.

- Makeup and Purification System Resins and Filters—If these were as contaminated as believed, then they would be treated the same as TRU wastes. (See Section 7 for a description of the cleanup of this system.)

- Other Solid Radioactive Wastes—These were less than Class C wastes associated with decontamination and maintenance (e.g., some ion exchange media, trash, sediment, clothing) and would be disposed of by GPU at commercial burial sites. They actually constituted about 98% of the non-fuel-related waste shipped off site.

The Memorandum of Understanding was supplemented by the Abnormal Waste Contract, signed by GPU and the DOE in July 1985 (GPU/US DOE 1985). This contract dictated the terms of disposal for any waste that was greater than Class C, not of R&D value, and not considered reactor core debris. GPU had originally considered sending up to 84 m^3 to INEL as "abnormal waste" (Negin and Urland 1984). However, the resulting estimated unit cost to GPU was going to be $475,000/$m^3$, including shipping, handling, monitoring, and interim storage at INEL in casks on a pad (Ayers 1984).

At such a high cost, it was often more economical to convert the waste to a form acceptable at commercial sites. In fact, dilution of the waste by as much as a 12:1 ratio was still economical enough to justify commercial burial in spite of the greatly increased volume (Urland 1986). As a result, only approximately 1.4 m^3 were actually planned for disposal via the contract with the DOE. (Much of it was actually made commercially disposable.)

For wastes that could be buried commercially as Class B or C, the project team had to choose the form into which the wastes was stabilized. Here, because of economics, a HIC was used if the waste form met the criteria. HICs played a large role in TMI-2 radioactive waste disposal because the technical, economic, and personnel exposure costs were usually lower. At times, however, the project team chose to solidify based on the waste stream form, regulations and guidance, and the burial site license for processing/packaging the waste to a form in compliance with the above.

The overall logic that the project team applied to disposing of the solid radioactive waste is shown in Figure 6-17.

6.5.1 Disposing of EPICOR II Prefilters

When the NRC waived the resin solidification requirement in March 1981, an intensive effort was begun to ship the EPICOR II prefilters to the DOE facility at INEL. It was a large-scale and ground-breaking job.

The DOE agreed to accept all 50 vessels for an R&D program to address the issue of resin disposal. The DOE believed such a program would benefit the government and nuclear industry and would serve as a basis for improved, more economical methods of treatment and disposal of low-activity commercial waste. Following the program, the vessels were to be disposed of in a method determined as a result of the program. This meant a HIC, and specifically an overpack-type of HIC to eliminate the personnel exposure-intensive job of sluicing the high-activity resins.

Two prefilter vessels were shipped to Battelle Columbus Laboratory (in May 1981 and August 1982) for the characterization work and later shipped on to INEL. In August 1982, shipment began of the remaining 48 prefilters directly to INEL. The final prefilter was shipped in July 1983 (Kalman 1984).

At INEL, four major activities were required to develop a HIC for disposing of the vessels:

- Developing the first-of-a-kind reinforced concrete HIC
- Locating/fabricating a suitable shipping cask
- Receiving regulatory approval for a disposal demonstration
- Conducting a disposal demonstration of a HIC containing a prefilter (McConnell 1985).

The approval process to license the HIC, which took approximately four years, was notable in that it involved the cooperation of several varied groups.

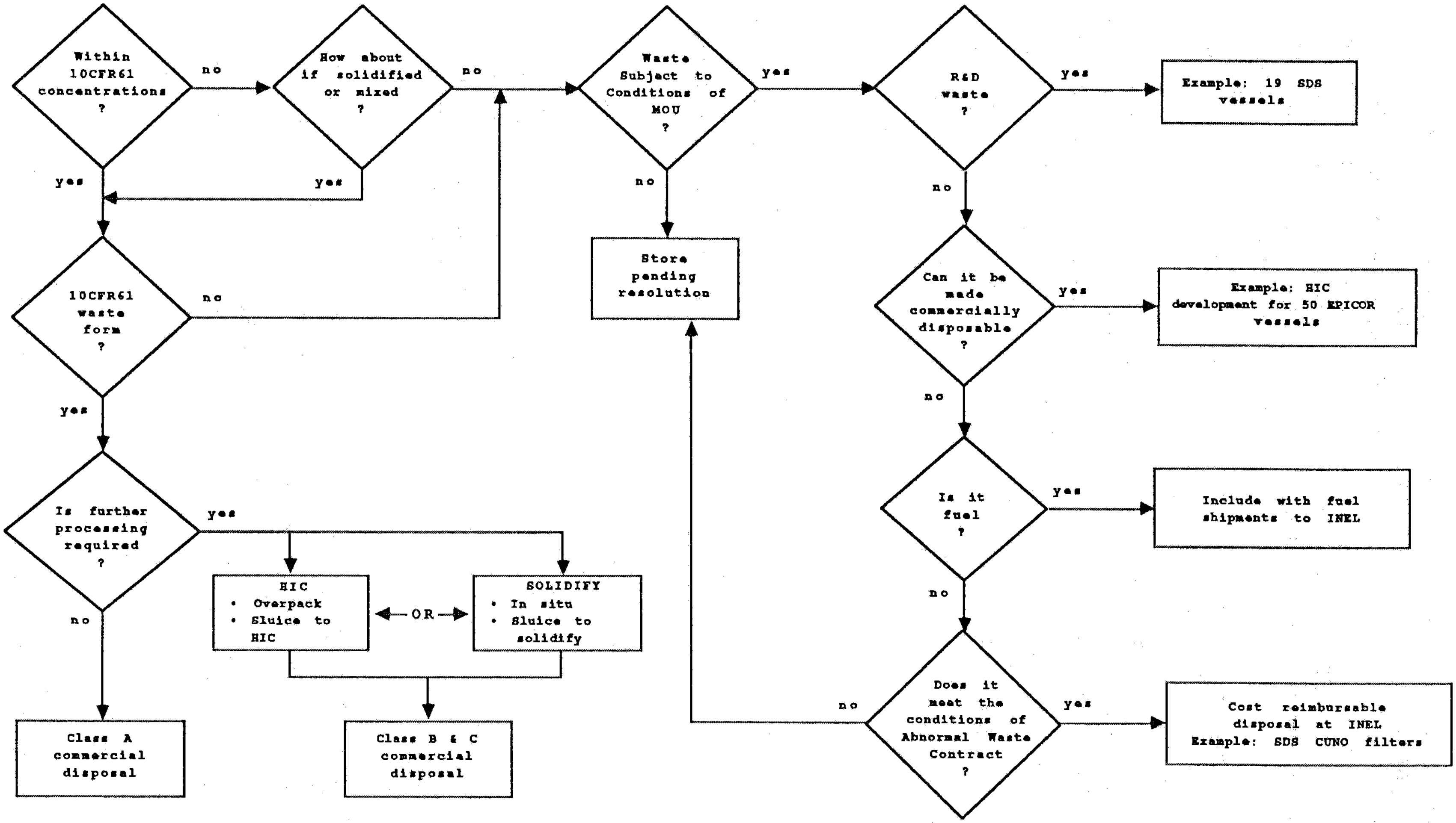


Figure 6-17. TMI-2 Waste Disposal Logic

The first attempt at functionally specifying a HIC had been conducted by the project team for EPICOR II vessels in 1980 (Wadsworth 1980). In the course of evaluating whether to lease a shipping cask or custom design one, the possibility of burying an EPICOR II vessel in a concrete HIC was examined. Two concepts were thought to be acceptable:

- Use 15 to 20 cm of concrete poured into a stainless steel shell surrounding the vessel. The vessel would be placed in the shell at either the site or the burial ground. A Type B cask would be required for shipment, which might limit the rate of shipment depending on the availability of the shipping cask.
- Encase the vessel in a concrete and steel structure that would serve for both transportation and burial, thus easing some of the transportation headaches (Such alternative shell materials as bentonite clay, fiberglass, or other organic resins would not qualify for the 300-year burial requirement.)

In early 1981, the DOE formally requested Sandia National Laboratories to develop the design requirements for a HIC. The concepts were expanded and eventually used in the licensing requirements for 10 CFR Part 61. The DOE selected a vendor to first design a HIC and then construct two prototype HICs of a type that could be loaded into a cask for shipment by truck from INEL to Richland.

The resulting HIC was a reinforced concrete cylindrical container. Figure 6-18 illustrates its configuration. Leakage was prevented by a corrosion-resistant steel liner that was coated inside and out with phenolic paint. The durability of the HIC was enhanced by the pH-adjusting amphoteric material placed on the inside bottom of the container. After loading, the HIC lid was sealed and bonded to the body using a bead of adhesive gel and flowable grout material.

A vent system allowed gas produced by radiolysis to escape. Without venting, the HIC had sufficient burst strength to contain the gas that may be generated within a 300-year lifetime (the hydrogen concentration in the vessels was below 5% initially because of shipping requirements). The concrete container attenuated radiation from the enclosed prefilter by a factor of approximately nine, which was not enough shielding to permit hands-on operation but enough to simplify handling procedures (McConnell 1985).

After an extensive comment and review cycle by the NRC, a Washington State Certification of Compliance was obtained for the HIC in March 1984. It was contingent upon the successful demonstration that preceded the full-scale burial campaign. Prefilter PF-18 was selected because it contained one of the highest curie contents (2025 curies of total activity). The HIC for this demonstration was fabricated in July 1982, and a series of tests were conducted. When the second HIC was delivered to INEL, further tests, including more drop tests, were conducted . The first HIC was buried at Richland in April 1984.

The prefilters, which had been stored at INEL's Test Area North Building, were prepared for shipment and placed in the HIC overpacks. From INEL, the HICs were transported to Hanford by truck in a CSNI 14-195 cask. Modifications were made based upon the experience gained in the disposal demonstration and the remaining EPICOR II prefilters were shipped from INEL to Richland for commercial burial. Four of the prefilters were stored at INEL for research pending future burial.

The DOE's decision to develop a concrete overpack HIC for the EPICOR II high-activity prefilter vessels was a pioneering effort. Until then, the concept of a HIC had not been clearly defined or recognized at Richland. The definition that resulted from developing the EPICOR II HIC was later used by the NRC in its waste form position paper for 10 CFR Part 61 implementation.

6.5.2 Disposing of SDS Wastes

High-Activity Vessels. As with the EPICOR II prefilters, disposing of the 19 highly loaded SDS vessels was an expensive and time-consuming enterprise. For the SDS vessels, a waste immobilization R&D program was established by the DOE to demonstrate two alternatives:

- Vitrification of SDS zeolites
- Monitored retrievable burial of SDS vessels in special concrete overpacks (Quinn 1984).

Of the 19 SDS vessels that the DOE agreed to accept, three were shipped to Pacific Northwest Laboratory in 1983 for use in vitrification experiments. The contaminated zeolites were removed from the vessels, glass formers were added, and the mixture was placed in special stainless steel canisters. A full-scale, in-canister melting process was then used to vitrify the material. In this process, the canister served as the container for the solidified (glass) final waste product (Bryan 1984).

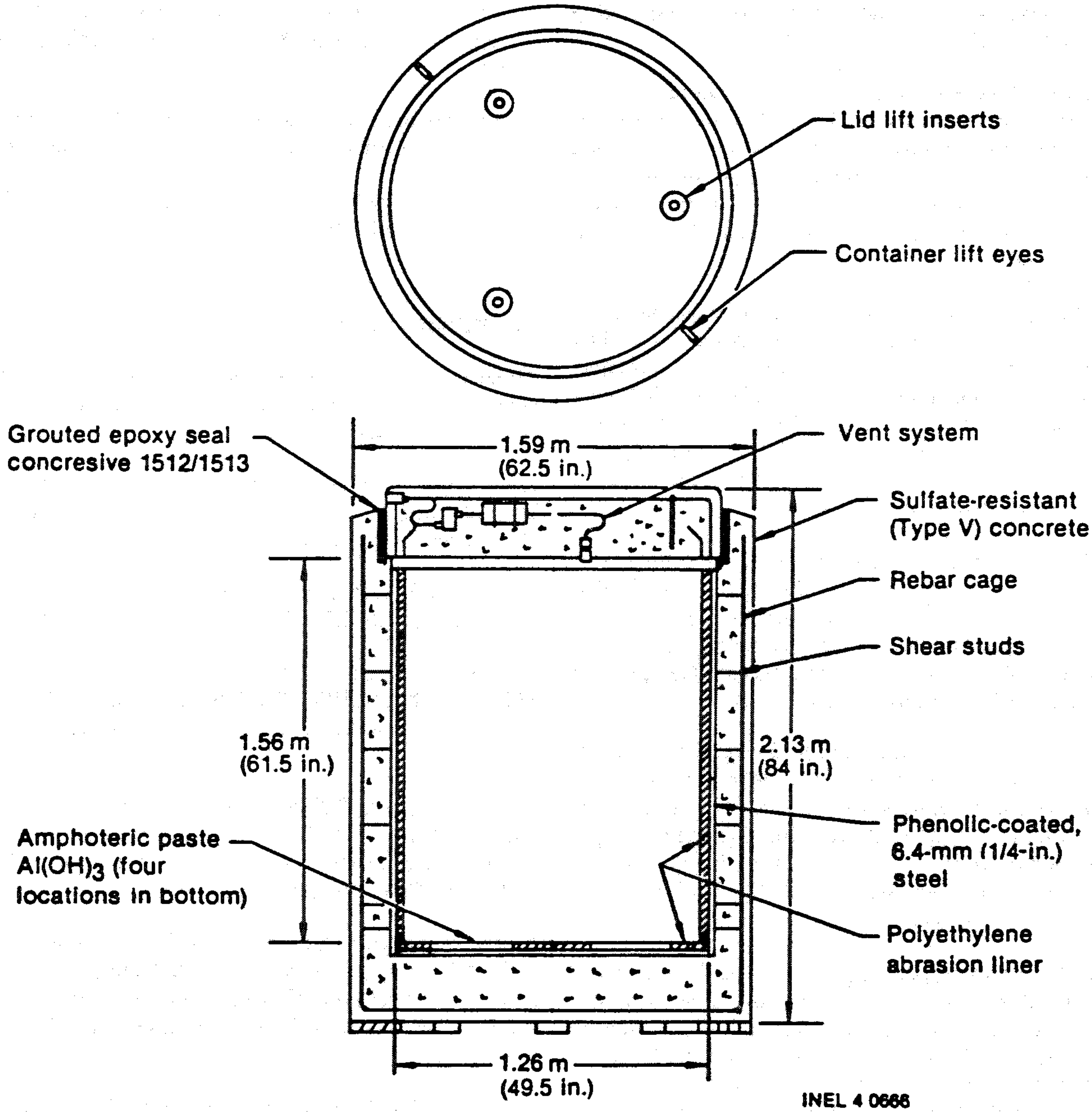


Figure 6-18. Design of EPCIOR II HIC

The other 16 vessels were sent to Rockwell Hanford for experiments demonstrating remote dry handling techniques and monitored burial in special concrete overpacks. The overpacks were buried at least 3 m underground in a trench. One of these SDS vessels and its overpack was instrumented for monitoring during long-term burial.

Other SDS Vessels. Disposing of the SDS vessels that were not buried at Rockwell Hanford was also complex. These were more lightly loaded because a decision had been made to meet 10 CFR Part 61 isotope concentration limitations whenever possible and because the DOE would not take any more than the original 19 (Negin and Urland 1984).

As a result, more SDS vessels were generated than if the system had operated as designed; i.e., instead of loading 10,000–15,000 curies in a vessel, a limit of approximately 1000 curies was established. After 1987, the SDS was removed from service and the DWCS and EPICOR II used instead—which produced a few highly loaded Ferralium processing HICs that were commercially/economically disposable.

The SDS vessels was expensive and problematic to dispose of because, although most of them met 10 CFR Part 61 isotope concentration limits, the contents were not stabilized. Consequently, the project team initially took action to make the vessels disposable by attempting to qualify the SDS vessels as HICs.

In November 1984, a study to certify the SDS vessels as HICs was submitted to the State of Washington (NucPac 1984). The study contained a stress and corrosion analysis, documentation of the dewatering experience and gas generation for the original 19 highly loaded vessels, and a point-by-point comparison with the NRC and state HIC qualification criteria.

The NRC, reviewing the request for Washington, asked for more information, particularly about vessel corrosion and the ability of the vessels to maintain their integrity for the required 300 years. Too much additional analysis would have been required to pursue this option (Deltete and Hahn 1990). When this option was determined to be impractical, the project team:

- Arranged with Battelle Columbus Laboratory (BCL) to transload acceptable SDS vessels into polyethelene HICs—19 vessels were sent to BCL, where they were loaded in the HICs and then transported to Barnwell for burial. (This option had only become available in 1987 when Barnwell reopened to TMI-2 wastes. It was advantageous to the TMI-2 project team because it minimized interference with fuel shipping operations—both SDS and fuel canisters were loaded into casks in the same location in the fuel handling building.)
- Shipped the three Cuno filters to INEL for storage under the terms of the Abnormal Waste Contract.
- Planned to load a few vessels onsite into polyethelene HICs or to solidify them for commercial burial.

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7 DECONTAMINATION

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DECONTAMINATION

7.1 Overview

With a plant as contaminated as TMI-2, questions had to be asked about the proper role of decontamination: What were its objectives? Was it a primary or support function? Would conventional decontamination methods suffice? What new techniques/methods would be needed?

The answers were based on knowledge of plant conditions, progress in defueling, and changing program strategies. After the first intensive efforts to gain control of major plant areas, decontamination became a support activity to defueling, and then an activity aimed at establishing stable and secure conditions for long-term storage. In everyday terms, most of the decontamination work was a hands-on job that required extreme care, preparation, and training. Robotics found a place on a task-specific basis, but there was no real substitute for a labor force with good morale employing conventional decontamination methods.

In the auxiliary and fuel handling building (AFHB) following the accident, the radiation readings ranged from 50 mR/h to 5 R/h, with local hot spots up to 125 R/h in some access areas and over 1000 R/h in the reactor coolant drain tank cubicles. Fuel handling building readings were 150-500 mR/h. Contamination in the lower elevations of the AFHB was transported there by water, initially from the containment sump pumps. Ventilation flows had acted as the primary transport mechanism to the upper elevations (second and third floors). Ten centimeters of water lay on some auxiliary building floors immediately after the accident (Tam 1984).

In the containment, the major sources of radiation were: the 2.5 million liters of highly contaminated water in the basement; the reactor coolant system; and general surface contamination. During the first re-entries in 1980, radiation readings averaged 425–450 mR/h on El. 305' and 225–250 mR/h on El. 347' (Daniels 1983). The contamination levels on the floors averaged between 1E+07 and 1E+08 dpm/100 cm^2; the contamination levels on the walls between 1E+04 and 1E+05 dpm/100 cm^2. In the basement, readings were 45 R/h at 1.5 m above the water. The major contamination resulted from activity released through the pilot-operated relief valve on the pressurizer to the reactor coolant drain tank 46-cm discharge line. Activity entrained in water droplets was then redistributed by means of the building air coolers (Peterson 1983).

In such a large-scale decontamination job, it was vital to examine all interrelated aspects of the work to avoid recontamination and adhere to the ALARA principle. In other words, the interrelationship of systems to areas to the logistics of worker movement had to be constantly evaluated. This was not done until later in the cleanup and consequently much duplicate work was performed.

Many assumptions from past years of decontamination experience became outmoded and new lessons had to be learned; e.g., floor coatings can be a major source of recontamination and surface removal may be the only recourse. Operational, disposal, and cost constraints on processing waste prevented the use of chemicals or the reflood of the containment basement for decontamination-via-leaching.

Limited resources, the limited effectiveness of gross decontamination techniques, and the time involved to fully decontaminate the containment caused the project management to stress dose reduction over decontamination. This was a major programmatic shift made in order to press on with the primary goal of defueling. Figure 7-1 depicts the general chronology of TMI-2 decontamination, dose reduction, and related efforts.

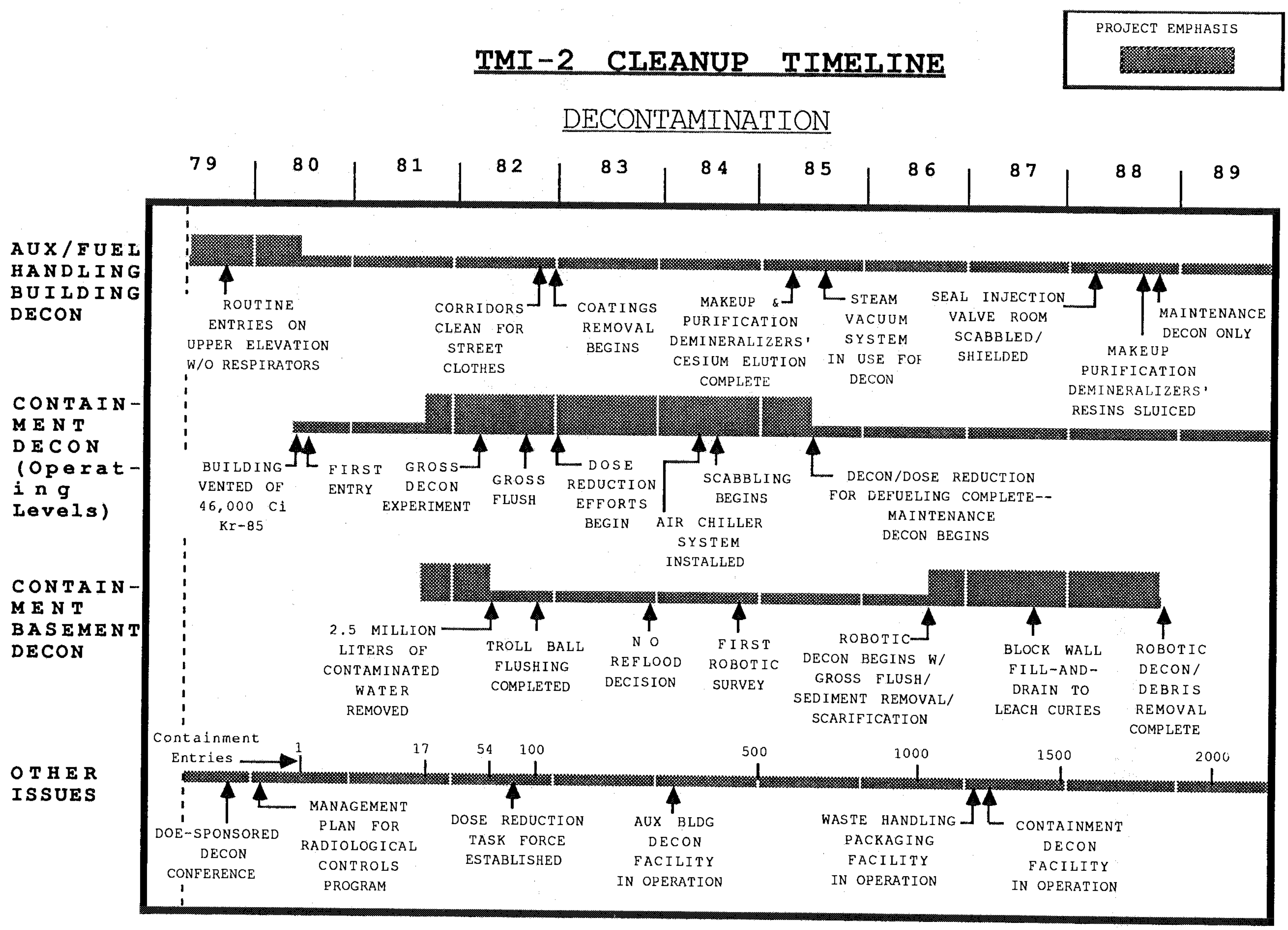

Figure 7-1. TMI-2 Cleanup Timeline: Decontamination

7.2 General Strategy

The strategy for decontamination at TMI-2 evolved from opposing approaches, both of which had the goal of ensuring a working environment that was safe and conformed to the ALARA concept:

- Decontaminate and then defuel—As with a construction project or plant outage, establish a relatively clean plant to support defueling and possible restart. This approach would have included the construction of several large decontamination support facilities.
- Decontaminate/reduce dose as necessary to support defueling—This approach assumed that limited time and resources were available and stressed the crucial importance of defueling. It was more in line with the research and development nature of the cleanup, which required emphasis on understanding and defueling the reactor.

The actual operations fell somewhere in between, with the first approach dominating the early years of the cleanup and the second approach the later.

7.2.1 AFHB Decontamination Plans

The first plan to clean the AFHB (Remark 1979) contained two phases:

1. Reduce smearable contamination levels to <100,000 dpm/100 cm^2 (generally eliminating the need for respiratory protection)
2. Reduce the levels to <1,000 dpm/100 cm^2 (eliminating the need for protective clothing).

Additional objectives were to minimize exposure from gamma and beta sources within pipes and components (by flushing), and to keep airborne contamination within 10 CFR Part 20 limits.

The appropriate decontamination techniques were determined in light of the radioactive waste generated, material compatibility, and person-rem exposure (e.g., abrasive scrubbing with Radiac wash, 2,000-psi hydrolaser, manual wiping). The cubicles could be isolated for later cleanup, but the open areas had to meet the criteria for general access. The final goal was to reestablish general area radiation levels at less than or equal to plant design values (Met Ed 1979).

A followup strategy for decontaminating the AFHB expanded this approach; however, it did so at a microscopic level of planning that proved impractical to implement (Hondorp 1983). The approach identified every action required to first characterize each of 134 cubicles; stabilize them (an interim decon/maintenance step to ensure safety from recriticality and to support complete decontamination when resources permitted); characterize their new conditions; and eventually decontaminate them to approximately pre-accident conditions.

In addition, the decontamination of 26 plant systems (subdivided into multiple pipe sections) that were contained in the cubicles had to be logically integrated with the cubicle decontamination sequence. The approach of complete decontamination of the AFHB also involved a complex study of related criticality issues (Davis 1985b).

This detailed approach was the subject of extensive debate because its detail, complexity, and large logistical requirements were at odds with a situation that required flexibility, had limited resources, and lacked a clearly defined decontamination objective (i.e., whether to support restart or long-term storage). In addition, some of the systems were needed for operations, which would have led to recontamination problems.

The approach was abandoned in favor of a strategy that focused resources on defueling, as discussed in Section 7.2.2.

7.2.2 Containment Decontamination Plans

From the beginning, containment decontamination plans were aimed at reducing exposure levels, then establishing and maintaining radiological conditions that would support defueling. The degree of decontamination this represented was subject to debate. A secondary goal—later abandoned because of changing program scope—was to decontaminate the reactor coolant system. In the initial plan, containment decontamination was to be accomplished in four stages:

1. Characterization of the containment to determine the current and evolving radiological and physical conditions
2. Gross decontamination to remove the bulk of the relatively loose and soluble contaminants (scheduled for approximately three months)

3. Hands-on decontamination to aggressively remove tightly adhering deposits and reduce radiation levels to those consistent with normal conditions (scheduled for approximately 12 months)

4. Maintenance decontamination to support defueling (Met Ed 1979).

A general change in approach began in 1981, and was formalized in 1984 (see Section 2.3.2). The approach changed to one of:

1. Gross decontamination/flush

2. Decontamination/dose reduction required to support defueling

3. Maintenance decontamination

4. Decontamination to establish a stable and secure condition.

The final strategy contained the end point criteria shown in Table 7-1 (GPUN 1988). These criteria were the general goals of final plant decontamination operations, but they were sometimes achieved in parallel with defueling when resources and priorities permitted.

In late 1988, the project management postponed most of the remaining plant decontamination until the completion of defueling because operations were found to be recontaminating some previously cleaned areas. This also focused project resources on the highest priority work. Only decontamination that directly supported defueling was performed.

7.3 AFHB Decontamination

Decontamination of the auxiliary and fuel handling building involved both surface removal and systems decontamination. The fuel handling building was not as contaminated and so most time and effort was spent recovering use of the auxiliary building.

7.3.1 AFHB Area Decontamination

As described in Section 3.6, the immediate efforts to decontaminate the AFHB were hectic, labor-intensive, and often repeated. The general pattern of early decontamination work was:

1. Rapid, labor-intensive, nonmethodical decontamination with little characterization of source terms or of potential methods of recontamination

2. Methodical decontamination that still overlooked the effects of ventilation systems on recontamination

3. More methodical decontamination with reduced ventilation-related recontaminations—but re-contaminations continued as the result of personnel movement and the failure to identify some sources.

In spite of frequent setbacks, routine entries in upper-level corridors and some cubicles could be made without respirators by late 1979. At the basement level (El. 282'), respirators were required in corridors for several more years. By the end of 1982, street clothes were all that were required for many upper-level areas.

Two distinctly different decontamination methods were required, although this was only distinguishable in retrospect (Pavelek 1988a):

- Decontamination of areas not directly exposed to reactor coolant water—These areas could be decontaminated by conventional methods (e.g., wiping, vacuuming, flushing).

- Decontamination of areas directly exposed to reactor coolant (particularly the lower elevations of the AFHB)—The contaminants in the water had remained on a surface for a longer period of time and were more resistant to removal. Conventional methods were only partially successful because of recontamination caused by contaminants embedded in the primer coat layers of the epoxy system migrating to the surface. The recontaminations were so routine that floors were scrubbed mechanically and wet vacuumed as often as three times per week. The final solution was to remove the coatings.

Understanding this source of recontamination had taken considerable time. Both tracking and fallout from overheads were investigated as sources and discarded —which left leaching from coatings as the most likely source. Acceptance of this fact did not come easily because surface removal was a decontamination technique of last resort. In addition, coatings were commonly regarded as impermeable, and, therefore, in theory, the contamination in the water could not have penetrated them (Schwartztrauber 1987). Forty years of industry experience, especially at PWRs, had never encountered soluble contamination on this scale. Consequently, both

Table 7-1. TMI-2 End Point Goals

Area Description	General Area Dose Rate (R/h)*
Containment	
Refueling Floor (El.347')***	< 0.03
Entry floor (El.305')***	< 0.07
Basement (El.282')	"As Is"
Auxiliary & Fuel Handling Building	
Corridors**	< 0.0025
Other Areas***	< 0.05 (majority)
Other Buildings	< 0.0025

* Conditions exclude hot spots and locked high radiation areas

** In general, <500 dpm/100 cm^2 below 2.1 m; <10,000 dpm/100 cm^2 in overhead wiring and pipes.

*** In general, <50,000 dpm/100 cm^2 below 2.1 m; <50,000 dpm/100 cm^2 in overhead wiring and pipes.

management and staff resisted acknowledging the source and the solution.

Tests, however, conclusively showed both the source and the solution. Full-scale coatings removal began in March 1983, first with a scarifier and then with a scabbler (Pavelek 1985). Scabbling was preferred to scarification because of its relative ease of operation, longer equipment life, and the increased rate of coatings removal.

A commercial steam vacuum system with modifications was also employed and proved effective in decontaminating overheads and cable trays, which contained too many complex surfaces to clean effectively using flushes or hand wiping. After its effectiveness was demonstrated, it became a standard tool for AFHB decontamination.

Once general access to the auxiliary building had been established, work on individual cubicles could proceed. Several were very contaminated and only limited human access was possible. For these, robotic devices were used for at least some of the work in the cubicles:

- Remotely Controlled Mobile Manipulator (RCMM) a.k.a. FRED—A small, commercially available, tether-controlled, six-wheeled vehicle, it was tested and primarily used for decontamination in the reactor coolant bleed tank cubicle and the makeup pump room. Although its capabilities did not allow it to be used for larger-scale decontamination, it was able to flush (with a 5,000-psi, 1.6E-03 m^3/s nozzle), position shielding, and remove some debris.
- Louie-2—This robot was designed on site specifically for the stresses associated with operating a three-piston pneumatic scabbler for concrete surface removal (see Photo 7-1). It was used in the seal injection valve room to remove a layer of highly contaminated grout. Although it vibrated excessively and required its own vent system to prevent recontaminations, it succeeded in scabbling the floors. Work crews then poured a new concrete pad that reduce readings from 20–60 R/h (exceeding 100 R/h in some areas) to 1 R/h.

7.3.2 AFHB Systems Decontamination

The initial strategy for decontaminating systems focused heavily on making systems "squeaky clean"; thus the project team pursued elaborate plans for chemical decontamination. No chemical decontamination of systems was ever conducted, although potential reagents were screened and tested (EPRI 1983; Scheidmiller and Simon 1985; Sjoblom 1986). Plans to defuel the reactor coolant system with chemicals are discussed in Section 8.7.

All planning efforts for chemical decontamination were terminated in 1984, with the rationale that any such application would only be a last resort for final "polishing" (DeVine and Negin 1984). Ultimately chemicals were rejected because of concerns about how to process and handle large volumes of chemical waste and because no sure way existed to dispose of the resulting solutions. Such reagents as sulfamic and citric acids tested as effective, but the problems with their use were too great.

7.3.2.1 General Systems Decontamination

After much planning and debate, the actual, limited decontamination of systems in the AFHB was a relatively straightforward, business-as-usual job. Contaminant control was achieved by: 1) securing contaminants in drained, sealed piping; 2) by flushing loose particles out of the systems with water flow; or 3) more aggressive methods determined on a case-by-case basis. While flushing may not have removed all internal debris, it reduced the concentration to a point where escape from a drained, sealed system was unlikely (Kelsey 1985).

Evaluations of potential fuel debris-bearing systems and general surveys determined that seven systems containing 76 individual, isolable pipe sections required flushing. Flushing was conducted primarily with borated water, although in two cases, sections of pipe had to be removed.

7.3.2.2 Flushes

The project team designed and built a flushing system on three transportable skids: a pump skid, filter skid, and tank or reservoir skid. The 100-psi pump used up to 1.3E-02 m3/s of reactor coolant-grade borated water to flush a pipe section to a drain or filter. The relative lack of installed drains and vents in the plant complicated the draining and increased the accumulation of contamination. Sixty-one flowpaths were flushed by the end of 1988; the remainder were to be flushed after defueling was completed.

7.3.2.3 Removal

Cutting out components was a drastic measure that ensured success, given proper training and team work. It was applied only to small pipe and to isolated, compact areas. Removal of the letdown block orifice was of interest.

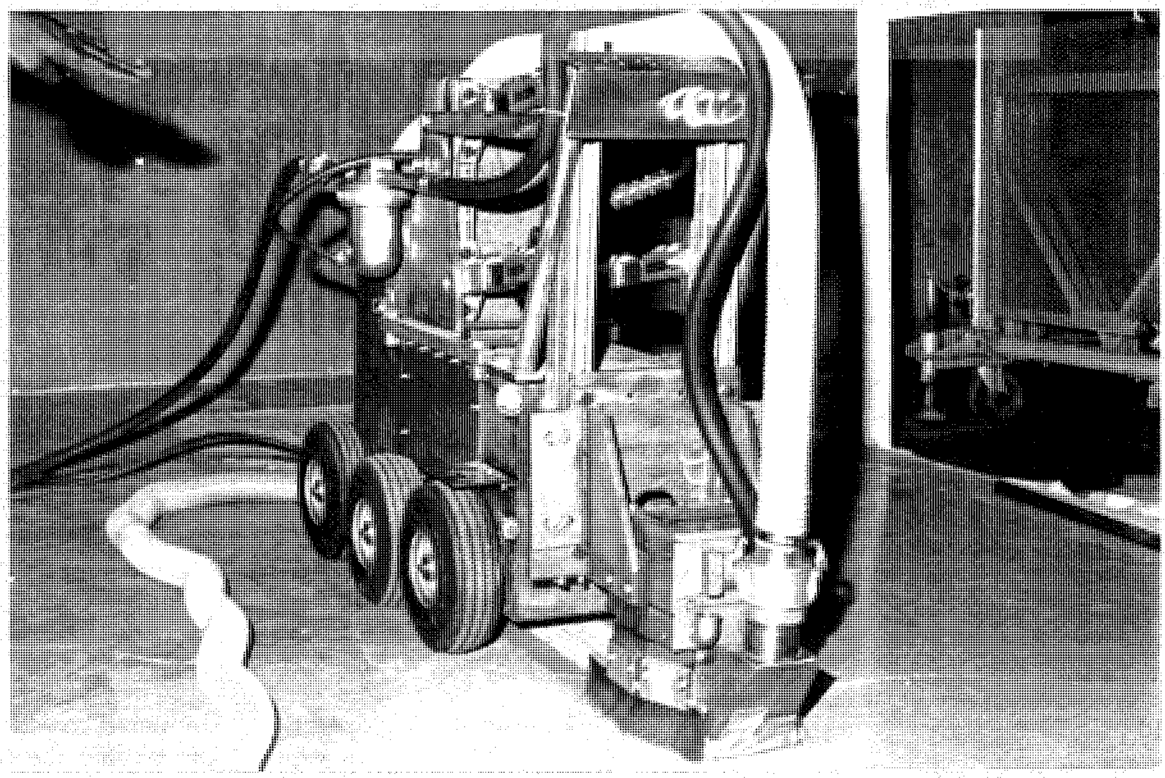

Photo 7-1. LOUIE-2

The letdown block orifice (MU-FE-1) was a 46-cm long section of 3.8-cm stainless steel pipe in the makeup and purification system. It reduced the reactor coolant letdown pressure by using a series of internal restricting orifices, which became clogged with debris during the accident.

Removing the block orifice required substantial effort. Approximately 100 gm of fuel debris and cladding were estimated to be deposited in the block orifice and associated piping (Daniel, et al. 1982). Its high readings, in combination with intruding pipes and structures, made robotic decontamination impractical and general surface decontamination non-ALARA.

Three solutions were considered:

- Chemical flush—Rejected because of projected high dose rates, expensive equipment, and lack of approved reagents
- Recirculating filter flush—Rejected because of high dose rates, expensive equipment, and uncertainty of success
- Removal and installation of new piping—Selected because it entailed less projected personnel exposure, guaranteed reduction in area dose, and provided a high probability of later dose reduction using turbulent water flushing.

A hydraulically operated, reciprocating hacksaw was used; it had a set of opposed jaws that allowed it to be attached to the pipe and cut without personnel present. In late 1986, the block orifice was drained, cut up, packaged, and removed for disposal, and a new spool piece was installed. The general area dose rates dropped from 1–4 R/h to 100–150 mR/h, allowing general surface area decontamination to begin (Murphy 1986).

7.3.2.4 Purification Demineralizers

The cleanup of the two makeup and purification (MUP) demineralizer vessels was unique because it was a low-profile, challenging project that proceeded with little fanfare over several years. The radioactivity contained in the vessels was substantial, but not in a location that interfered with cleanup and defueling operations.

The impetus for cleaning the vessels was that they contained the largest amount of highly concentrated radionuclides outside of the reactor vessel itself. They also contained an unknown quantity of fuel debris, which was a criticality concern. The project to clean them up had originally begun with urgency and commitments to the NRC for this reason. As data acquisition proceeded and much less than a critical mass of debris was found, the urgency subsided.

The two MUP demineralizer vessels were part of the pressurized water makeup and purification system, which was designed to provide chemistry control to the reactor coolant system. The 2.2-m^3 demineralizers each contained 1.4 m^3 of resins to control concentrations of potentially corrosive minerals and to reduce the buildup of radioactive deposits in the circulating system. In normal operation, expended demineralizer resins were removed by sluicing through installed piping to spent resin storage tanks. Figure 7-2 shows the layout of the demineralizer cubicles.

To clean up the vessels, three separate aspects had to be understood:

- The quantity of fuel had to be known to address the criticality potential during processing and handling
- The quantity of cesium had to be known so that potentially high dose rates during handling could be minimized
- The physical state of the resins as a result of overheating had to be known in order to establish a feasible removal process (the possibility that the resins had congealed would have made removal much more difficult).

Three approaches were considered for actually removing the resins:

- Remove the demineralizer vessels in shielded containers—Eliminated because the high dose rates in the cubicles would make access possible only by remote means. Pipes would have to be severed and capped, walls penetrated, shielding and transportation devices designed, fabricated, and tested. Personnel exposures would be high and building operations severely restricted because of the high potential for contamination.
- In-situ resin treatment to break down the resins into soluble products—Laboratory scale experiments indicated that hydrogen peroxide would react with the resins. This approach was considered viable but less attractive than the third approach.
- Sluice the resins from the vessels—Chemical tests showed that 90% of the cesium sorbed on the resin could be removed by elution, and virtually 100% of the cesium removed by elution could be recaptured by the zeolite ion exchange material in the submerged demineralizer system (SDS) (Bond, et al. 1986b).

A three-phase demineralizer cleanup project was devised: Phase I: rinse and elute the demineralizer contents without moving solids; Phase II: modify the existing spent resin removal system, and sluice and solidify normal resins for a system test; and Phase III: sluice, solidify, and dispose of the purification demineralizer contents.

The elution of the demineralizers was conducted between September 1984, and April 1985. The elution process removed approximately 790 curies of cesium-137 (68% of original inventory) from the "A" demineralizer vessel. From the "B" demineralizer, approximately 3,455 curies of cesium-137 (89% of inventory) were eluted (Hofstetter 1985).

The resin sluicing operation encountered considerably more difficulty than expected in moving the resins, which were agglomerated. Between October 1987 and September 1988, most of the resins were sluiced to a spent resin storage tank that had been modified to operate as a radioactive sediment separator (Cremeans and Mahla 1989). The "A" vessel was left essentially empty, while the "B" contained approximately 0.7 m3 of agglomerated resins.

7.4 Containment Decontamination

Containment decontamination was aimed at providing safe working conditions for defueling. The upper elevations (El. 305' and 347') were the arena for most of the effort; conditions in the isolated containment basement, however, also had to be addressed.

7.4.1 Gross Decontamination

Two major activities had to be performed before major decontamination of the containment could begin:

- Venting the containment atmosphere of approximately 46,000 curies of krypton-85—which took place between June 28–July 11, 1980.
- Human entry into the building for direct surveys of conditions—the first entry was made on July 23, 1980 (see sections 3 and 4).

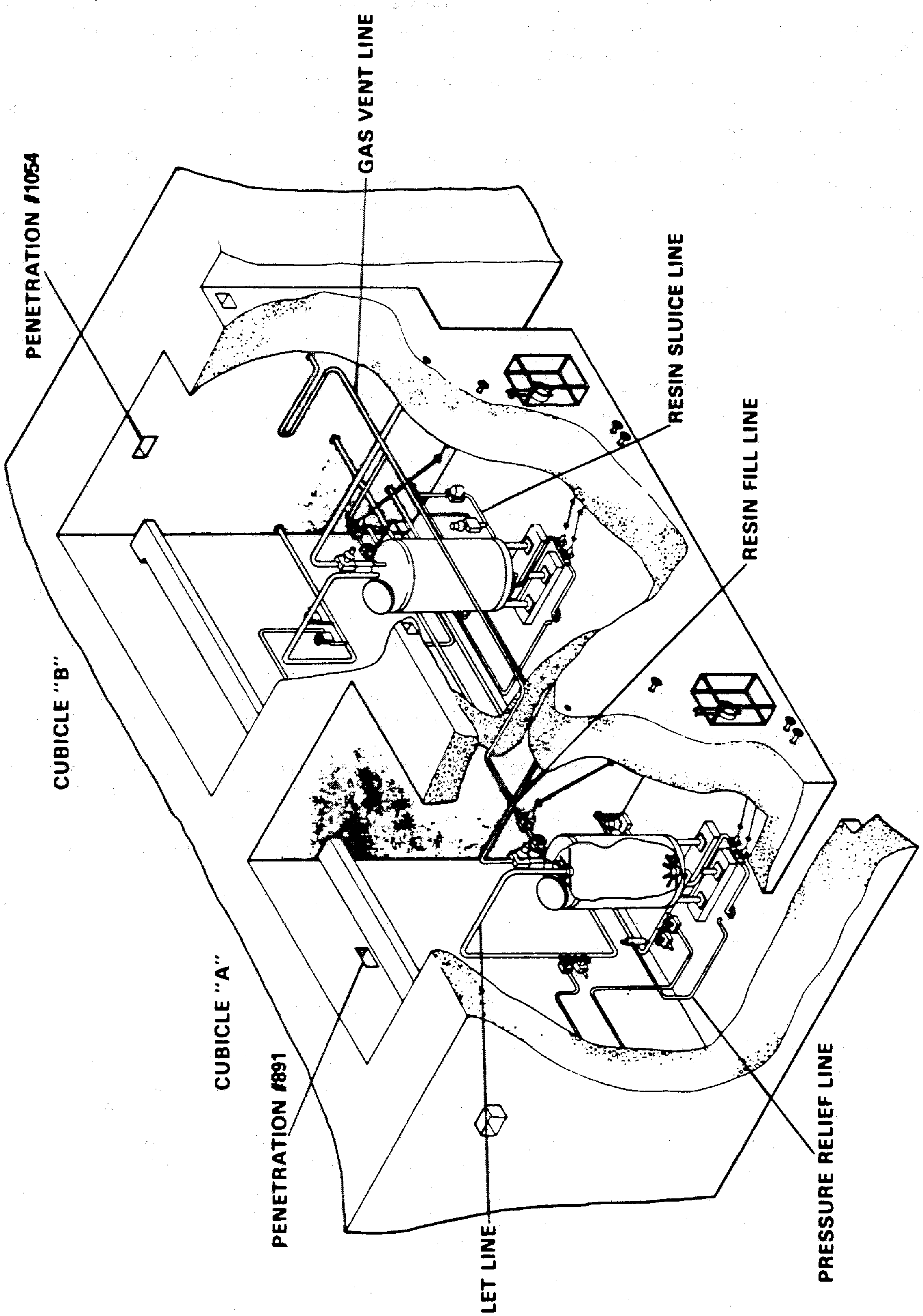

Figure 7-2.Makeup & Purification Demineralizer Cubicles

Brief consideration was given to accomplishing gross decontamination flush remotely; i.e., from outside the building. The installed containment spray system could have been used to spray large volumes of water and possibly detergents, chemicals, and steam on the interior. This approach was not pursued because: 1) lower-than-predicted radiation fields were found in the containment; 2) the effectiveness was uncertain; 3) a large volume of waste would be generated; and 4) damage to important equipment might result (BPC 1979).

The actual gross decontamination (or general flushing) of the containment began mid-1982, with decon crews flushing surfaces from the dome down. By this time, however, the project staff already had sufficient data to indicate that the flush would not be as effective as originally hoped. This information had come from the Gross Decontamination Experiment, completed in March 1982.

The Gross Decontamination Experiment had been funded by the DOE as part of its R&D charter to provide the industry with access to decontamination engineering and operational experience and to document the effectiveness, criteria, techniques, and radiation monitoring activities (see Section 5.2).

The Experiment was a five-month long series of activities, beginning with extensive data gathering and mapping of the building; installation of pumps, power, and equipment; and, finally, the application of several techniques (e.g., high- and low-pressure hot water flushes, spin jet flushes, strippable coatings, a mechanical scrubbing device, and wet vacuums). The specific objectives were:

- To evaluate safety, effectiveness, and efficiency of the decontamination equipment and methods, and to evaluate the support systems and organization for such a large effort.
- To reduce contamination on selected surfaces for personnel protection.

The results were often inconclusive because not all the data were permitted to be taken and what were taken were insufficient or not timely enough to accurately record the effects. Still, the Gross Decontamination Experiment showed that the techniques tested were effective to varying degrees in removing surface contamination. The major challenge would apparently be recontamination (Mason 1983; Lazo 1989).

This was borne out as the project team pursued the gross decontamination of the building during 1982. Smear samples showed considerable surface contamination reduction after decontamination, followed by recontamination to near pre-decontamination levels.

Extensive testing determined that one major source was airborne recontamination in the form of large, easily settled particles primarily originating in the building air handling systems, although some originated from small, resuspended particles. To reduce the primary sources, the air cooler fans were cut back to two fans running at slow speed; unneeded ducts were closed; and some top to bottom ducts and air coolers were flushed (Furio 1983; Tarpinian 1989). These efforts, combined with floor scabbling and recoating, eventually reduced recontamination as a concern (see Section 7.4.2).

One side benefit of the Gross Decontamination Experiment was gaining access to the polar crane in order to begin refurbishment to support the expedited defueling schedule (see Section 8.4). Another benefit was the great increase in time worked in the containment, accompanied by an increased confidence in the project team's ability to work in that environment. Until the Experiment, 16 containment entries had been made over the previous 15 months. In the five months of the Experiment, 45 entries were made (most over three months); from that time on, entries were made almost daily.

The Gross Decontamination Experiment also showed all the organizational inefficiencies involved in the complex review and approval process. It taught the project team about large-scale operations in the containment; how to move large numbers of workers in and out; about radioactive waste disposal; and about communications and procedures for conducting in-containment work.

7.4.2 Dose Reduction

In late 1982, plans were made to emphasize dose reduction rather than decontamination in the containment (BNI 1983). This was a response to the failure of gross decontamination operations to reduce radiation fields enough to support an accelerated defueling program. Even the processing of 2.5 million liters of highly contaminated water from the basement had not significantly affected radiation levels.

The average dose rates on the containment operating levels were: El. 305'– 350 mR/h and El. 347'– 150 mR/h.

On the polar crane, rates were 120 mR/h; and on the reactor head service structure 600 mR/h. For El. 305', this dose rate was only a 22% reduction from 1980, and most of that was the result of decay (17%) rather than decontamination (5%).

Given these facts, the project team focused on dose reduction because:

- A decision had been made to expedite defueling.
- At the existing exposure rates, the planned containment activities would entail an excessive expenditure of person-rem, which did not conform to the ALARA concept (e.g., the 2500–6000 workhours projected for reactor vessel head lift would require an estimated 750 to 1800 person-rem if the average area dose rate was 300 mR/h).
- The planned activities could not be conducted with the existing TMI-2 workforce without exceeding individual quarterly and annual dose rate limits.

The scope of dose reduction activities included better planning, decontamination, shielding, removal of source terms, reducing exposure time, and/or a combination of these in repetition.

Before and during these activities, the source terms in the building had to be accurately understood—a difficult task. Sources were masked by other sources. With multiple, relatively equal sources, the individual contribution was not always significant so it was difficult to calculate the benefit of performing an activity or how many person-rem would be required versus saved. In many cases, the sums of known sources did not equal the total by a wide margin.

The Dose Reduction Task Force was established in late October 1982 to identify the principal radiological sources in the containment. Since the programmatic goal was to provide an ALARA working environment for defueling, acceptable conditions varied; i.e., areas of high occupancy would have lower dose rates than those areas less often occupied (BNI 1983).

The task force recommended actions to minimize the dose received by workers on labor-intensive projects. In determining whether or not to take a specific dose reduction step, many factors were evaluated, including: 1) potential dose savings; 2) potential gain in productivity or efficiency; 3) total cost; 4) impact upon schedule and project resources; 5) regulatory guidance; and 6) comparison with alternatives. Section 5.5 describes some of characterization techniques developed to identify sources in this environment.

Two major dose reduction efforts took place on El. 305' and 347' of the containment:

- Early 1983—Based on the recommendations of the Dose Reduction Task Force, shielding was installed at several locations; the air coolers, ducts, incore instrument service area, and stairwells were flushed; and entry and transit routes were changed to new ones through lower dose fields. As a result, general area dose rates were reduced from 350 mR/h to 200 mR/h on El. 305'; and from 150 to 100 mR/hr on El. 347'—a good start but not sufficient to support defueling (Tarpinian 1984).
- Late 1984 to mid-1985—To support defueling preparations and operations, additional activities were undertaken: the air coolers, incore instrument service area, hatch covers, ducts, and cable trays were shielded; the walls of the refueling canal were decontaminated; and extensive scabbling operations were conducted. The use of new transit routes sometimes eliminated the need to shield or decontaminate difficult areas. As a result of this effort, the exposure rate at the defueling work area was reduced from 50–70 mR/h to less than 10 mR/h (Igarashi 1985).

Two specific operations were of note:

- Air cooler shielding—The decontamination versus dose reduction debate was epitomized in the two-year question of what to do about the air coolers on El. 305'. During the accident, the coolers had been the major flowpath for dispersing contamination on upper elevations; as a result, they were very contaminated and a significant contributor to nearby dose rates. In early 1983, they were partially flushed, which helped address the airborne recontamination problem but did nothing to lower the 120 mR/h fields in their vicinity. Effective decontamination would be person-rem intensive and expensive. The coolers continued to contribute 45–80% of the dose to the defueling transit route until lead blankets were installed in 1985—lowering the fields to 30 mR/h.
- Coatings removal—Analysis of concrete samples taken from the containment showed that dose could be reduced and contamination prevented by coatings removal. A major scabbling campaign was then conducted between October 1984 and June 1985 (see

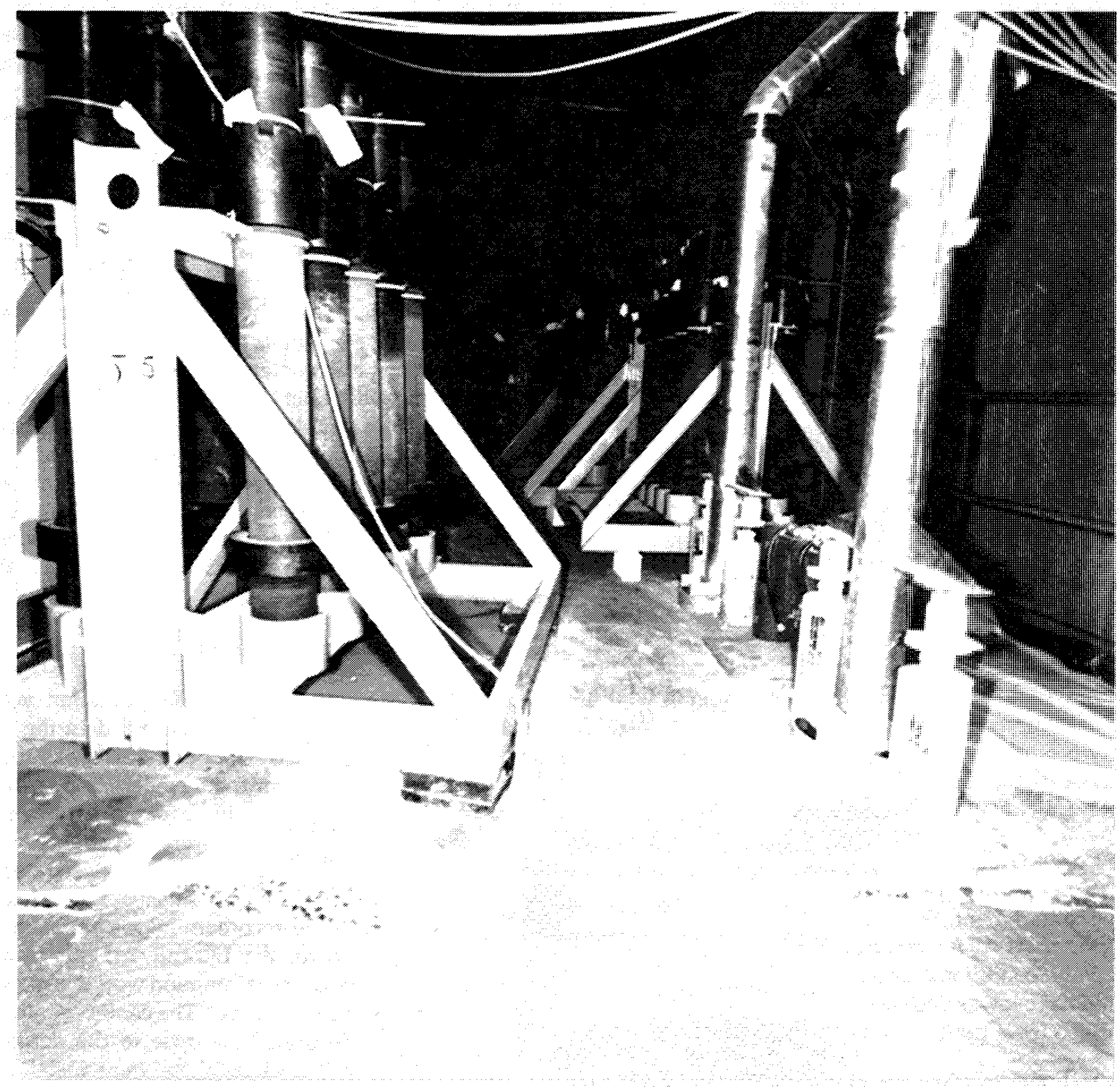

Photo 7-2. Scabbled Floor on El. 347'

Photo 7-2). Approximately 930 m^2 of containment floor coatings were removed and the floors recoated. General area gamma exposure rates were reduced by 85%, with beta radiation not detectable in the majority of the areas after scabbling. Contamination levels after scabbling, vacuuming, and scrubbing averaged less than 5,000 dpm/100 cm^2 (Pavelek 1985).

7.4.3 Containment Basement

The containment basement posed a unique challenge. Although addressed differently from the areas requiring human access, it played an important, interrelated role in containment decontamination/dose reduction and end point status.

For over two years after the accident, it contained approximately 2.5 million liters of highly radioactive water. Submersion of the basement concrete to a depth of 2.6 m during this time had resulted in the absorption of contaminated water containing radionuclides (primarily cesium-137 and strontium-90) into the surface concrete of floors and walls. The extent and penetration of this contamination was not well understood, but the resulting radiation fields ranged from 2 to 1000 R/h and contributed to area dose rates on El. 305' and 347'. The area was clearly not accessible to workers.

The challenges to working in the basement were more than just the high radiation fields. In addition, 15 to 50 cm of water covered approximately 2.5 cm of sediment (concrete dust, river silt, construction dirt) . The water was left there so that the sediment would not dry out and become an airborne problem. Planks, ladders, and scaffolding lay on the floor, and pipes were suspended from above. The concrete enclosures around the D-rings, reactor coolant drain tank, and stairwell/elevator limited or sealed off access to numerous areas. Since the basement was without lighting, the entire area was a dark labyrinth (see Figure 7-3).

7.4.3.1 Containment Basement Strategy

Until robotic technology was established as a viable decontamination method, decontamination efforts focused on the upper elevations and on shielding the radiation shine from the basement to the work areas above. During the first few years, basement surveys were taken using TLD strings suspended through penetrations and the seismic gap at the perimeter of the building. Likewise, a flushing ball was lowered through the gap and the basement walls sprayed in 1982. The effects of the flushing were minimal and the project team delayed any further action until more data were available and the overall cleanup strategy established.

A strategy plan for basement cleanup proposed to thoroughly characterize the basement, reduce the dose, and establish long-term stable conditions (Davis 1984). Decontamination was to be conducted with remote vehicles, although workers could be used where practical. The objective was to establish a condition where:

- Systems could be decontaminated if basement access was required
- Further decontamination work could be deferred until a final decision on the plant's disposition was made
- Surveillance and inspections could be performed
- The possibility of an environmental release was minimized.

The general area exposure rates at the end of this scenario were to be less than 2 R/h, with hot spots up to 20 R/h. This approach to decontamination required that a "beachhead" be established below the El. 305' equipment hatch, which was the largest open area in the basement, was directly accessible from El. 305', and was free of large obstacles to remote vehicle movement. Limited human access would be possible and other operations could be launched.

This approach changed when post-defueling plans were finalized. Because there was no real need for human entry to the basement and the drain on resources to establish access would have been high, no entry was planned.

The final decontamination strategy for accessible areas of the basement was to reduce general area exposures by: 1) remote sediment removal, 2) gross flushing, 3) surface layer removal; and 4) by doing something about the block wall surrounding the stairwell and elevator shaft, if practicable—it contained an estimated 12,000 curies and up to 1000 R/h radiation fields, and was a potentially major source of airborne recontamination (GPUN 1986a).

7.4.3.2 Basement Decontamination Operations

Robotic technology was the obvious and only choice for operating in this environment. Two robotic devices were developed to work in the basement:

- Remote Reconnaissance Vehicle (RRV) a.k.a. Rover—Developed from principles learned with an earlier robot (RCMM), RRV was a tether-controlled,

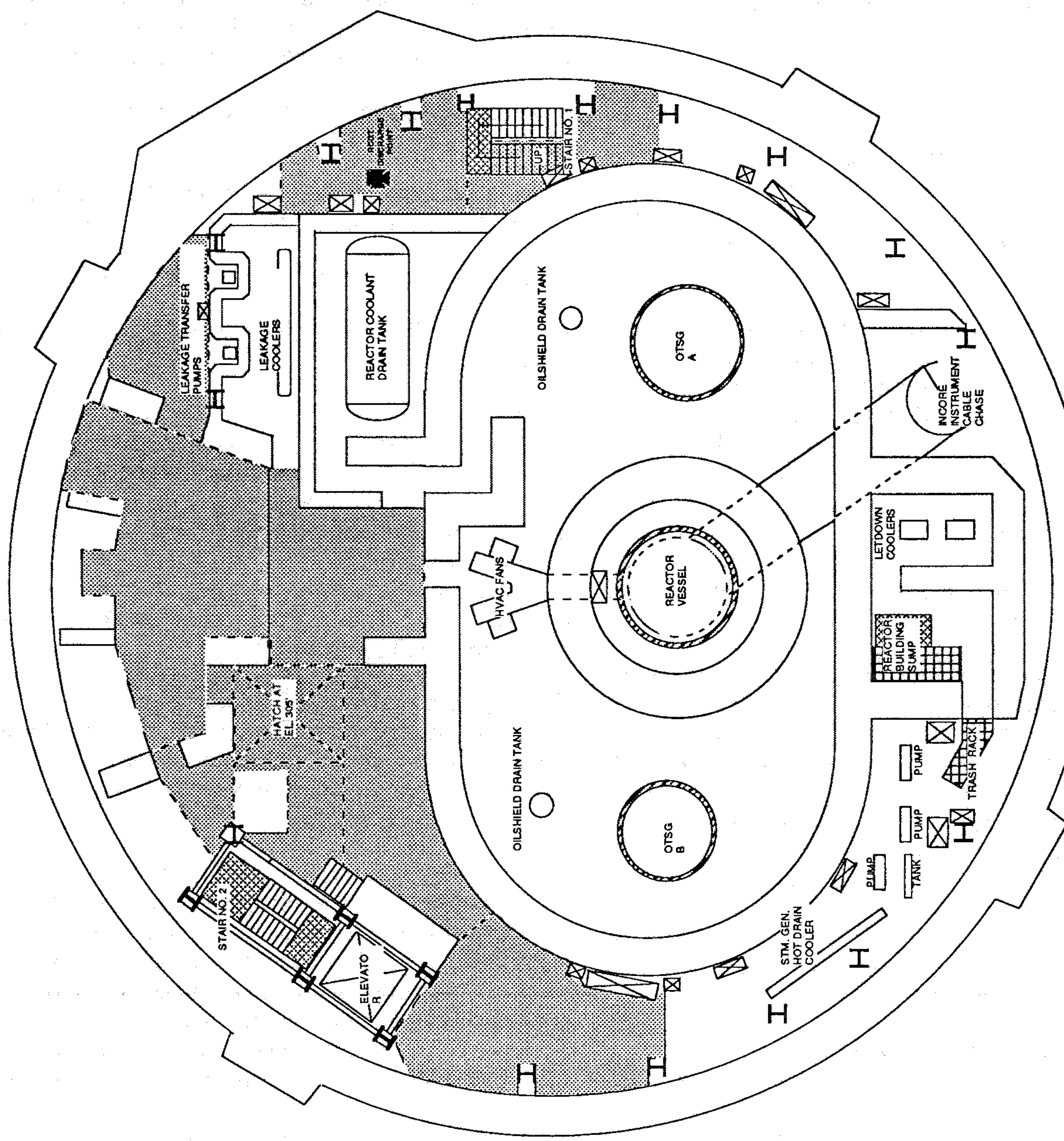

Figure 7-3. Containment Basement Floor Plan Showing Areas Accessible to Remotely Operated Vehicles

six-wheeled work platform. It weighed 567 kg and could handle a 567-kg payload, which consisted of cameras, survey equipment, and a variety of decontamination equipment (e.g., spray nozzles, heavy-duty arms). Three RRV vehicles were eventually fabricated—one was used only for training/mockups. RRV was first equipped for characterization work in the basement (see Section 5.3), then adapted with a variety of tools for decontamination. (See Photo 7-3 for a picture of RRV with suction device.)

- Remote Working Vehicle (RWV) a.k.a. Workhorse —This large vehicle was to be able to perform such major operations as dismantling and demolition. It was a large, tethered, four-wheeled vehicle able to extend its working arm with a wide range of tooling up to 7 m. RWV was never used because of the project management's unwillingness to risk an expensive, one-of-a-kind vehicle and a change in project direction away from more decontamination work in the basement (see Photo 7-4).

Four types of decontamination operations were conducted in the containment basement, primarily using the RRV:

- Flushing—Several flushes of the basement were conducted, beginning with a general but ineffective flush by troll ball (lowered from El. 305') in 1982. In late 1986, the RRV was mounted with a 7000-psi spray nozzle to perform a general flush of accessible areas.

- Concrete scarification—Following the flush, an ultra-high-pressure (35,000 psi) water scarification device was attached to the RRV and used to remove concrete/curies from the "bathtub" ring left after the accident water had been removed.

- Sediment removal—An estimated 450 kg of sediment existed on accessible basement floor areas. One system was considered to flush the sediment into the containment sump and then pump it to a settling tank (Davis 1983). Instead, the RRV, adapted with a suction device and connections to a tank in the auxiliary building, was selected based on its proven reliability (see Photo 7-3). To process the sediment, settling, filtration, centrifugation, and evaporation were considered—settling was selected because of the costs and lead times associated with the other methods (Negin and Urland 1983). The spent resin tanks in the auxiliary building were modified to perform the settling function (Cremeans and Mahla 1989).

- Block wall fill-and-drain—Section 7.4.3.3 describes this attempt to reduce the quantity of radionuclides that had leached from the accident water into the porous concrete block wall surrounding the basement stairwell/elevator.

7.4.3.3 Stairwell Block Wall

A number of options were considered for addressing the curies and high radiation fields associated with the block wall enclosing the stairwell and elevator shaft (Crawford 1987):

- Do nothing—This was the first choice because it had no project impact and still permitted limited human access. It would leave a legacy of curies and hinder future human access.

- Shield the block wall—Either a shotcrete or sand column enclosure could be installed by a robot. This approach would not have removed the curies, would have increased the radioactive waste and impeded future cleanup, and would have been difficult to install.

- Dismantle/remove the block wall—The remotely controlled vehicle Workhorse could be used to dismantle the wall and reduce the blocks to rubble for slurry or dry packaging. This approach was rejected because the engineering was difficult and the Workhorse was an expensive prototype without backup.

- Reflood the basement—Refill the basement with water to 2.6 m, essentially reversing the original process of contamination. It was rejected for the reasons discussed in Section 7.4.3.4.

- Fill-and-drain the block wall—Use RRV to remotely fill the block wall hollow core center with water, which would leach out curies as it drained out through the concrete. The water would then be processed. This was the second choice to doing nothing and the one that was tested in situ.

The cost was relatively low for a fill-and-drain test; it was technically easy, with minimum radioactive waste; it was compatible with defueling; and it promised to be effective in reducing the number of curies in the wall. The drawback was that if the test was successful, a lot of additional engineering would be needed to fill-and-drain the entire enclosed stairwell.

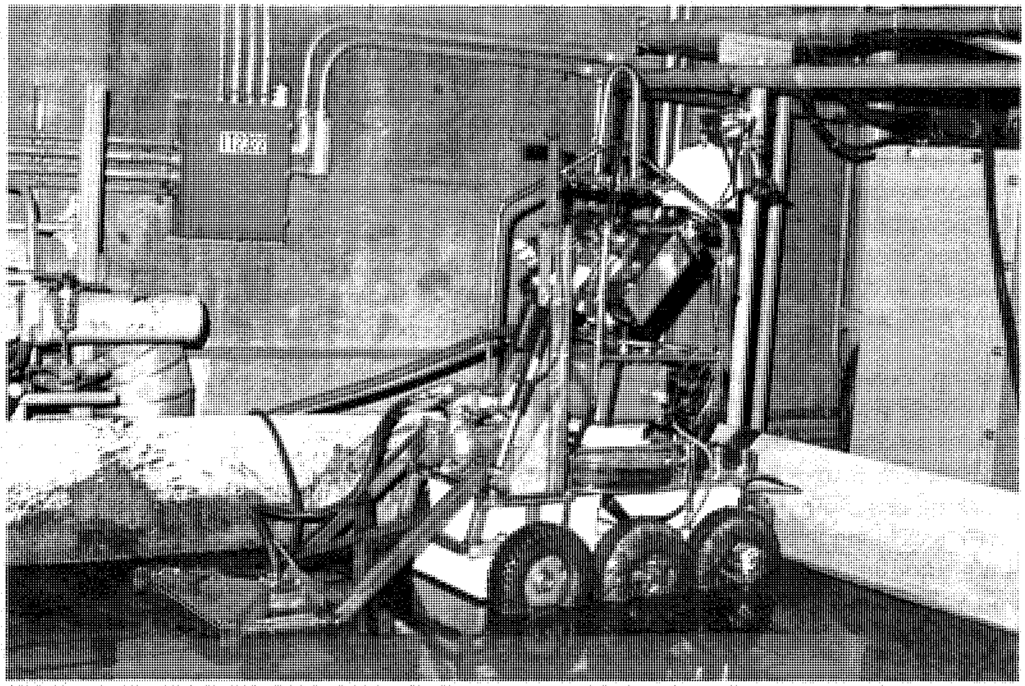

Photo 7-3. RRV with Sediment Pickup Device

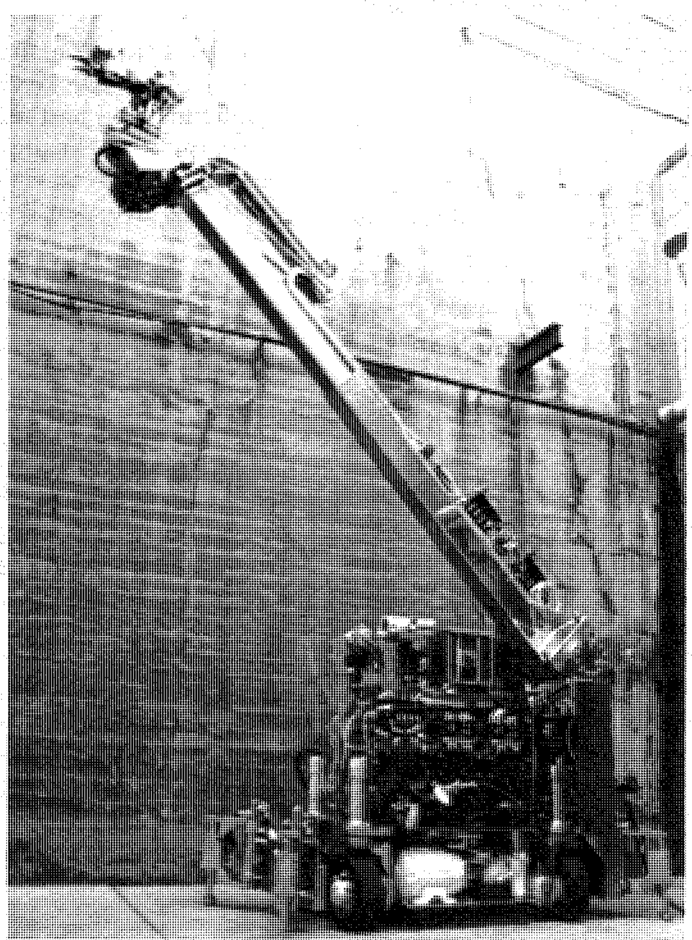

Photo 7-4. Remote Work Vehicle — Workhorse

Only tests were conducted and these showed a 33% reduction in radionuclides in the areas treated (Ferguson and Underhill 1988). Potentially, this methods might have been used effectively. However, defueling had the highest priority for resources—in effect, the "do nothing" option won out.

7.4.3.4 Basement Reflood

The reflood option had a long history at TMI-2. Since 1982, the argument had been forwarded that the basement should be reflooded to shield the upper elevations or to leach out curies, or both. The proposals recommended both limited reflooding (i.e., with 380,000 liters) and flooding to 2.6 m. After much debate, the logic of reflood was rejected, primarily because of the long-term demands on the water processing system.

Some of the project team's opposition to reflood was based on emotional as well as technical issues. The project team had already processed the basement water once in a major campaign—why do it again? How would the public and political community react to a reflood after a milestone processing campaign had already accomplished the objective once?

The reflood plan called for the water to remain in the basement for approximately seven years, being processed to remove the radionuclides that would leach from the concrete into the water. The water would be either borated (if begun before defueling was complete) or untreated (if after). The advantages were that this method of decontamination was passive, used a known technology, involved low personnel exposures, and removed curies from the entire basement, not just the accessible areas. The overwhelming disadvantages were that it was the most expensive because of water processing costs, required the longest time to complete, and had potentially negative licensing implications.

In 1984, a general policy decision on reflood was made: the method was not to be used unless new data resulted in a shift in the ALARA cost-benefit analysis (DeVine and Negin 1984). The new data came in 1986, when leaching tests showed a potentially large reduction in enclosed stairwell block wall activity in one to two years of leaching (Babel 1986). In terms of long-term planning for post-defueling conditions, this apparently simple and ALARA method of basement decontamination had to be reconsidered.

The new data, in part, led to the restudy of block wall decontamination options described in Section 7.4.3.3. The telling blow was a new analysis of leaching data that reestimated the time (six to eight years) required to perform meaningful leaching.

7.5 Decontamination Support Facilities

Several facilities were set up on site to support operations by either cleaning tools and equipment, or preparing trash for disposal as low-level radioactive waste. Facilities such as the laundry and the respiratory cleaning facilities and PAF and C-Cubed areas are described in Section 4. The waste handling and packaging facility (WHPF), completed in 1987, was a major boon to the decontamination efforts (69% of items were releasable after decontamination) and is described in Section 6.

7.5.1 Temporary Facilities

Temporary facilities were established in the auxiliary and containment buildings:

- El. 328' Facility (Auxiliary Building)—It was located in a radiologically protected area. Although only 30% of the items decontaminated were releasable, many of the items processed here originated in the containment and were only intended for gross decontamination before reuse (e.g., defueling tools). A vibratory finisher, ultra-sonic unit, high-pressure freon unit, and an electro-polisher were the primary equipment used in decontamination.
- El. 347' Facility (Containment)—This facility was a 3- by 7.3-m enclosure with 2.4-m high walls. It was used primarily to flush and wipe down defueling tools and equipment.

7.5.2 Rejected Alternatives

Early in the cleanup, several large support facilities had been envisioned:

- Containment Recovery Service Building—This complex was to be built adjacent to the equipment hatch to support the decontaminate-then-defuel approach. It would have contained a personnel access facility, high-level radioactive waste staging area, and a contaminated dry cleaning area (see Section 8.4.3.1 for more description). It was never built though much planning was devoted to it. The anticipated scale of operations, including defueling, that the facility would have supported was much larger than ever implemented; e.g., the personnel access facility in this complex was to be able to process over 100 in-containment workers per shift—in fact, the actual number of workers in the building at one time during decontamination was 15; four per shift was the average during defueling.

- DOE Decontamination Demonstration Facility (DDF)—This large building was partially designed in 1980 and never built. The DOE would have provided equipment to be "plugged in" and the project team would have provided housing, services, and operators. The purpose of the DDF was to demonstrate techniques for small item decontamination using TMI-2 as a test arena toward the commercialization of portable equipment. Techniques were to have included a freon bath, electropolisher, and vibratory finisher. Support facilities would have included a freon purifier, electrolyte purifier, evaporator, and waste solidifier It was cancelled because of financial limitations (TMI-2 TI&EP 1980).

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8 DEFUELING

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DEFUELING

8.1 Overview

The question of how to remove the damaged reactor core lay at the heart of the cleanup. In the early years of the cleanup, stabilization, waste management, and decontamination took precedence—but fuel removal was always the primary objective. The project could not be considered complete until the fuel was safely packaged and shipped from the site.

Despite the importance, five years followed the accident before the defueling strategy and initial equipment design were finalized, and another five years were required to complete the actual work. Without doubt, the pace was affected by factors such as funding, an NRC investigation of allegations related to the polar crane refurbishment, and an initial strategy of extensive decontamination before defueling. However, the real obstacle was the complex and unprecedented damage to the reactor core. Not only was the damage severe, it was not well understood.

When defueling started in 1985, the picture of the core available to planners and engineers resembled Figure 5-4. A persistent optimism existed that the damage beyond the known areas was not as bad as some estimates would have it. In fact, it was worse. Figure 5-7—which was not available until 1987—shows the actual postaccident end state of the TMI-2 reactor vessel. Appendix B describes the condition of the debris in each of the regions of the vessel.

The history of defueling was interwoven with how and when this information was obtained. The driving force in defueling was data. Major redirections or replanning of defueling operations were often forced by new information about core conditions. For this reason, one of the primary lessons to be derived from the defueling experience at TMI-2 was that defueling should not and could not be completely engineered at the start. Instead, such a novel situation required: 1) engineering to gather data first and to take initial steps; 2) gathering more data; 3) engineering to defuel; and 4) then repeating the steps as new areas/information were encountered.

Aside from determining core conditions, the key questions facing planners were: Should defueling be performed primarily with manual or robotic equipment? Should a cautious approach to removing the core debris be invoked in tool design or could higher production techniques (probably less conservative) be employed? How much water was required for shielding and how should it be processed? How to ensure subcriticality during defueling? How was the entire defueling concept from removal to shipment to final disposition to be integrated?

The concept had to be integrated in order to coordinate very disparate functions: 1) load debris into canisters; 2) transfer, store, and load canisters for shipment; and 3) transport canisters to Idaho, transfer to storage, and store for a postulated 30 years. A simplified picture of the TMI-2 portion of this system is shown in Figure 8-1.

The evolving knowledge of conditions in the reactor vessel led to frequent changes in design requirements for tools and equipment. These eventually ranged from simple vise grips, scoops, and airlift equipment to a modified oil drilling rig to state-of-the-art plasma arc technology. Along the way to deploying these tools, the project team pursued many parallel paths—often ending up with unbuilt designs and unused equipment—but the flexibility associated with multiple approaches was necessary to ensure progress.

Progress was frustratingly slow at times and difficult to forecast because defueling each debris region required a new learning curve. Progress was also difficult to quantify because, while weight removed per month could

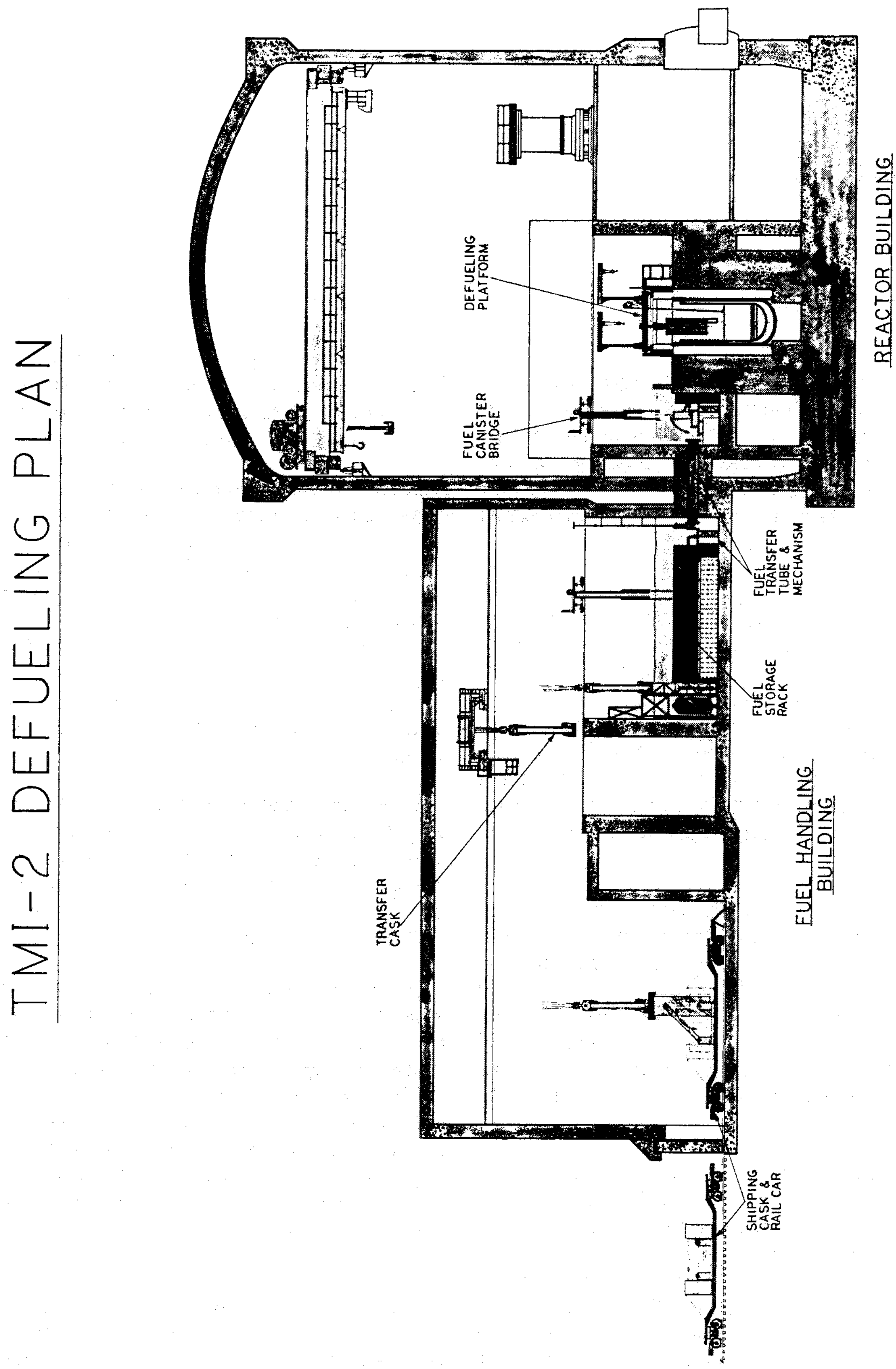


Figure 8-1. TMI-2 Defueling Plan

indicate excellent progress, a time would always come when months passed with little apparent change in conditions in spite of extensive and vital work. The year-long loss of visibility in the reactor vessel at the start of defueling operations was a harbinger of coming challenges. After that, defueling was not only behind schedule, but the scope kept expanding and the operations increased in difficulty.

Given all this in a situation ripe for second-guessing, the job was completed. Figures 8-2a and 8-2b illustrate the overall chronology of the TMI-2 cleanup from a defueling perspective.

8.2 Development of Full Removal Strategy

Many different defueling concepts and strategies were conceived, but in essence they were variations of either normal plant defueling practices or more sophisticated robotic approaches.

The development of the defueling strategy was played out against the background development of an overall TMI-2 cleanup strategy. This is described in more depth in Section 2.3; in brief, it was a shift from a decontamination-then-defuel approach to one emphasizing defueling, with supporting decontamination. The initial decision to proceed with an "early core removal schedule" was made in 1981.

8.2.1 Background

The fact that the core was severely damaged was widely recognized in 1979; however, the extent of the damage was not known and conditions in the containment building prevented manned entry.

Five independent groups made predictions of damage to the core in 1979 and 1980. They were:

- Nuclear Safety Analysis Center (NSAC)
- NRC Special Inquiry Group
- Los Alamos Scientific Laboratory
- Battelle Columbus Laboratories
- Westinghouse-Nuclear Energy Systems.

These groups predicted the core damage based on one of the following approaches:

- Reconstruction of the thermal-hydraulic behavior of the system
- Analyses of the amount of hydrogen produced
- Evaluations of the amount of fission products released.

A compilation of these estimates was performed by the DOE and published in May 1981, in *Three Mile Island Unit-2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling* (Croucher 1981), but it was more commonly referred to by its document number—GEND-007. GEND-007 served as the basis for all early defueling planning and design work.

GEND-007 predicted three potential damage scenarios: Minimum, Reference, and Maximum. The actual condition of the core was generally equal to or greater than the maximum damage estimate, although the fact that it exceeded the estimate was not established until 1985. Table 8-1 compares the GEND-007 predictions to the actual damage to the core. It can be seen that the relationship of actual to anticipated damage in the regions near the top of the core was ambiguous, with the plenum damage similar to the Reference prediction. Viewed from the first restricted-view video images produced in 1982 (Quick Look), the rubble bed and upper core damage appeared consistent with the Reference predictions.

The true extent of damage was not understood until after video inspections and sonar mapping of the rubble bed in 1983, video inspections of the lower head region in 1985, the core sample drilling program in 1986, and video inspections behind the core former walls in 1987. The uncertainty about the scope of the defueling tasks and the unfounded hope that damage was minimal were the major shortcomings of all defueling planning.

8.2.2 Initial Planning (Through Spring 1982)

The first plans were to decontaminate much of the containment to permit general access to the building and to create a near-normal refueling environment before beginning defueling (Met Ed 1979; BPC 1979 and 1980). Therefore, most planning and engineering efforts went into decontamination and waste management tasks: venting the radioactive krypton out of the containment,

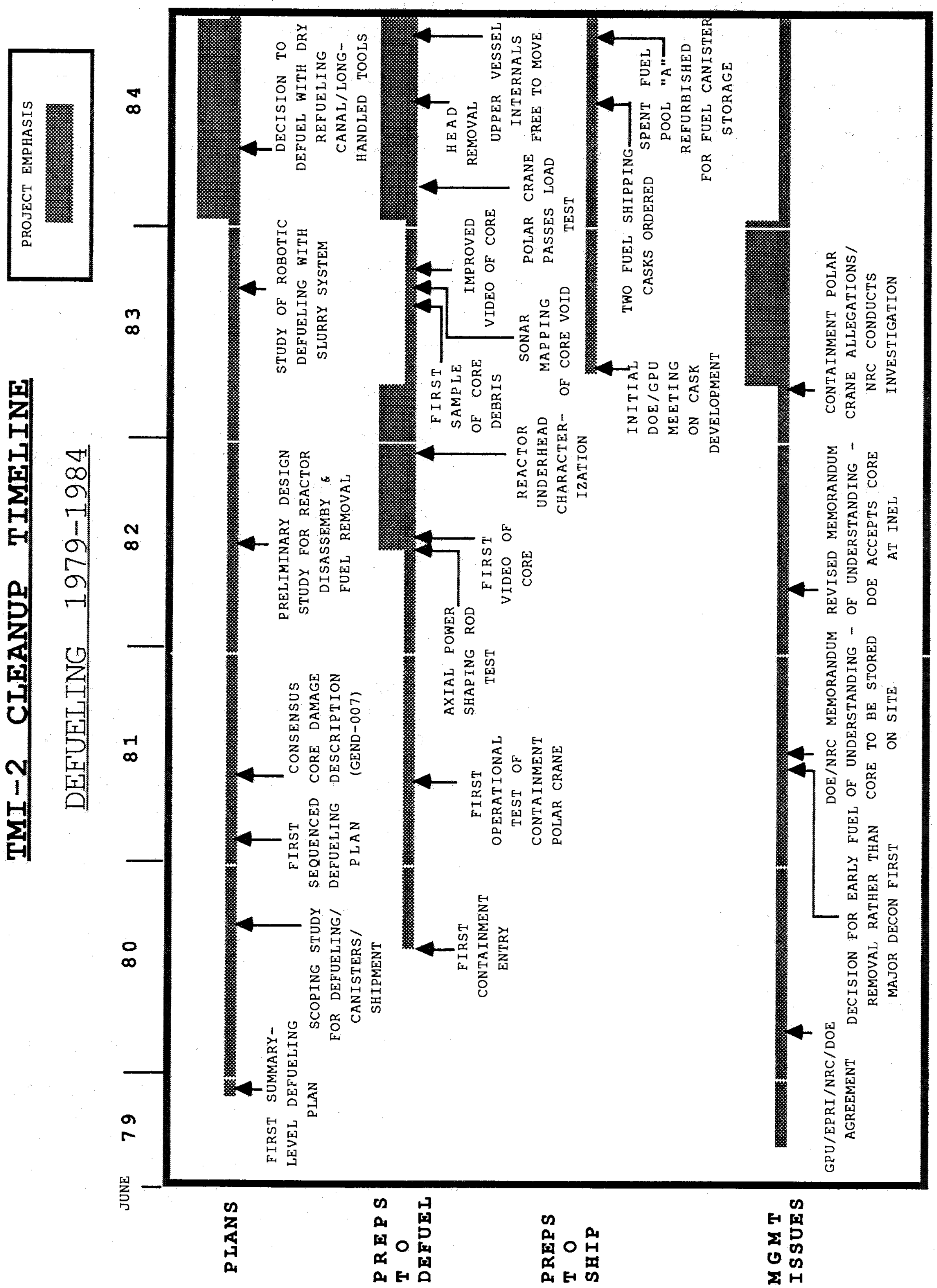


Figure 8-2a. TMI-2 Cleanup Timeline: Defueling 1979–1984

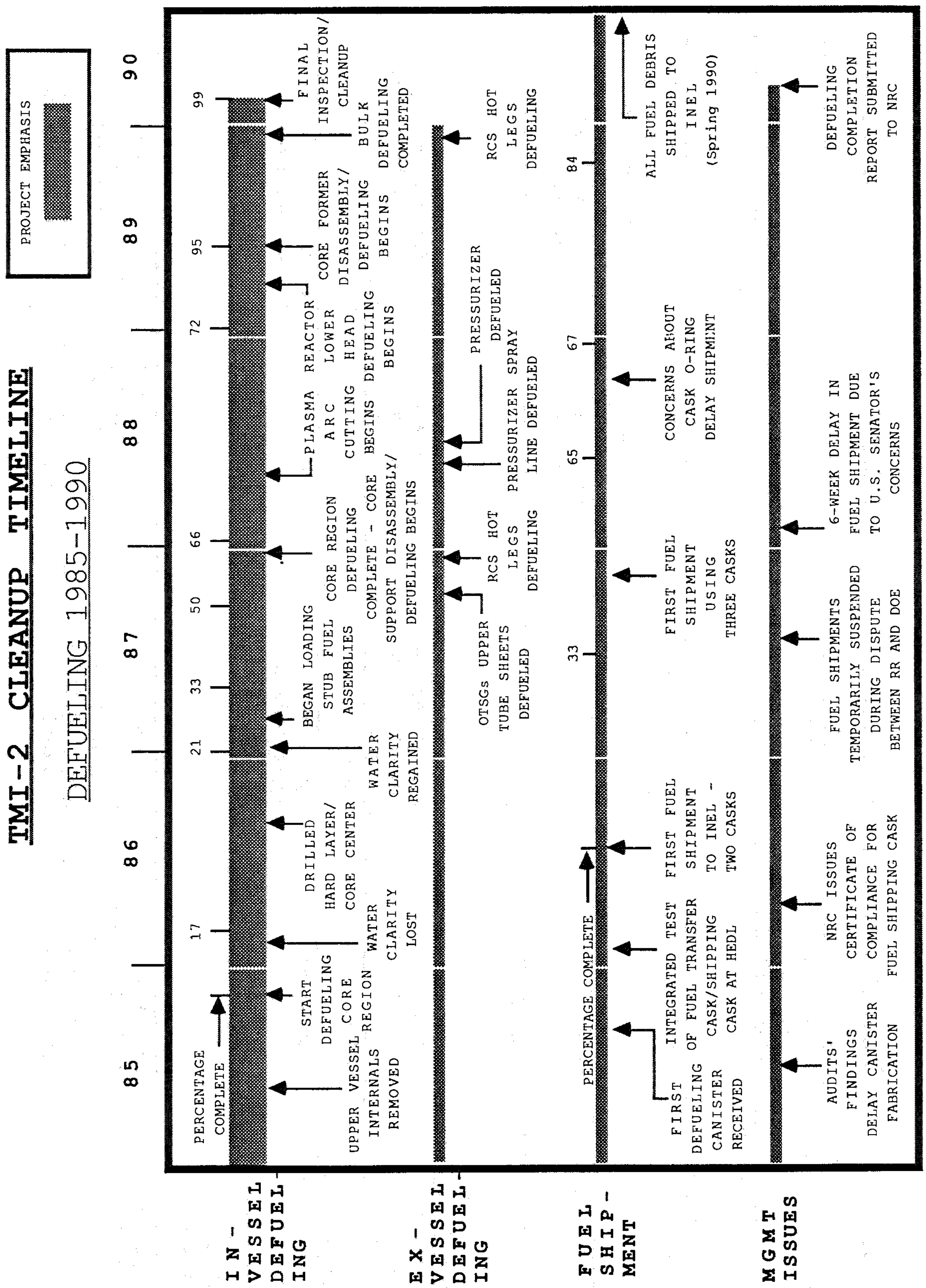


Figure 8-2b. TMI-2 Cleanup Timeline: Defueling 1985–1990

Table 8-1. Summary of GEND-007 Damage Estimate versus Actual

Condition	Minimum	Reference	Maximum	Actual
Failed Rods (%)	>90	~100	100	~100
Fuel Temp (K)	>2000 (gross average in damaged area)	~2600 (peak)	~2900 (peak)	>2900 (peak)
Cladding oxidized in active fuel region (%)	40	50	60	60
Liquified fuel	Locally possible	Present in several central core areas	Present over most of core radius, perhaps downward to ~1 m above core bottom	Present over most of core radius, downward to 0.2 m above core bottom
Molten fuel	None	None	Possible in few localized central core areas	Approximately 50% of core melted
Core slumping	Probable	Yes	Yes	Yes
Fuel rod fragmentation, debris bed formation	Yes	Yes	Yes	Yes
Peripheral fuel assemblies	A few not breached, some embrittled	Few, if any, not breached; most near core top embrittled	All failed and embrittled, many w/liquified fuel	All failed, embrittled and most w/liquified fuel
Control rods & spacer grids	Molten	Melted	Melted	Melted
Instrument strings	Most intact	Most in central region failed, peripheral tubes intact	All failed	All failed - many failed beneath core region
Embrittlement level (m above bottom of core at centerline)	1.8	1.4	0.9	0.2
Upper plenum assemblies	No distortion, melting, or fusing to other stainless steel components	Some distortion/ melting possible - may be fused to upper end fittings	Melting over central lower region; major slumping possible	Some distortion and local melting - fused w/end fittings
Relocated fuel	Not predicted	Not predicted	Not predicted	~30% core relocated from original core

processing the containment sump water, and decontaminating the auxiliary building. The Programmatic Environmental Impact Statement (US NRC 1981) schedule optimistically showed that decontamination would dominate the project and that defueling would take only a short period of time near the end. The PEIS did not, however, estimate damage to the reactor core.

In parallel, a scoping study was performed that described the defueling options for TMI-2 (Anderson 1980). It was the first formal assessment of the potential defueling options. The study considered all aspects of defueling: fuel removal, packaging, storage, transportation, and reprocessing. Figure 8-3 shows the key decision areas identified.

This study also identified two potential technical paths to be followed. The first technical path assumed that a large fraction of the fuel had fused together in pieces too large to fit into fuel element-sized shipping canisters. The second path assumed a wide spectrum of core conditions and a multi-pathed approach to defueling. The first option was never given much credence.

Based on the second approach, a reactor disassembly and defueling sequence and general plan were developed (BNI 1981). The sequence identified all the tasks needed to defuel the reactor, the interrelationships of the tasks, and the order in which the tasks were to be performed.

A significant concern was that GPU might go bankrupt while defueling the reactor (see Section 2.4.3). This concern forced project management to consider whether it should even remove the reactor vessel head before all the financial, engineering, personnel, and training resources were available to complete the effort (or to reinstall the reactor head, if necessary). One result was that the project team used a sequential plan to design a set of tools and procedures to address each potential condition or contingency before removing the reactor head.

Five system concepts for defueling were identified using GEND-007 as a base (GPUN 1982b):

- Dual Telescoping Tube/Manipulator System—This concept was a modification of the normal TMI-2 defueling approach. An overhead X-Y bridge, (possibly the existing defueling bridge suitably modified) would be operated from the defueling machine rails on El. 347'. The bridge would have two telescoping tubes. One of these tubes was to position a light-duty (25-kg capacity) manipulator while the other was to position a heavy-duty (180-kg capacity) manipulator (see Figure 8-4). These manipulators were to be remotely controlled and a closed circuit video system was to help position the tool. These tubes would work within an open top contamination control tank. The fuel transfer canal and the tank would both be filled with water for shielding. A water lock was provided to permit the movement of fuel assemblies or debris canisters from the tank to the canal. The tank had a vacuum system installed and either manipulator could be used to position the suction head.
- Manual Defueling Cylinder System—This concept consisted of a rotating cylinder supported from the reactor vessel flange and projecting down into the vessel. The bottom of the cylinder had a floor with a tool-mounting table that could move across the cylinder diameter (see Figure 8-5). Tools would be attached to this table and would operate on the core below the cylinder floor. Operators would work from a platform above the canal and would move tools and canisters manually using hoists, cranes, and manipulators. A seal would keep contamination in the reactor vessel unless tools were being installed on the mounting table. During that time, the water cleanup systems would ensure that water flowed from the fuel transfer canal into the reactor vessel to minimize cross contamination.
- Indirect Defueling Cylinder System—This concept was similar to the Manual Defueling Cylinder System except that an X-Y bridge with a telescoping tube replaced all the manual operations from the movable work platform.
- Flexible Membrane Defueling System—This concept consisted of an X-Y bridge with a single telescoping tube set, a conical flexible membrane contamination barrier, a 0.9- to 1.2-m-high extension tank with cover plate and four support arms (see Figure 8-6). The flexible membrane provided the contamination control while permitting some movement of the defueling bridge. The four support arms each pivoted about a vertical axis and were located under the extension tank. They could swing out over the reactor vessel and hold manipulators, canisters, debris buckets, and tools. The end of the telescoping tube was a "torque cube" that would grasp and hold defueling tools.
- Dry Defueling System—This concept was similar to the method used to defuel naval reactors and sodium-cooled reactors. It would have consisted of lead-filled shield rings (a stationary outer ring, a large rotating

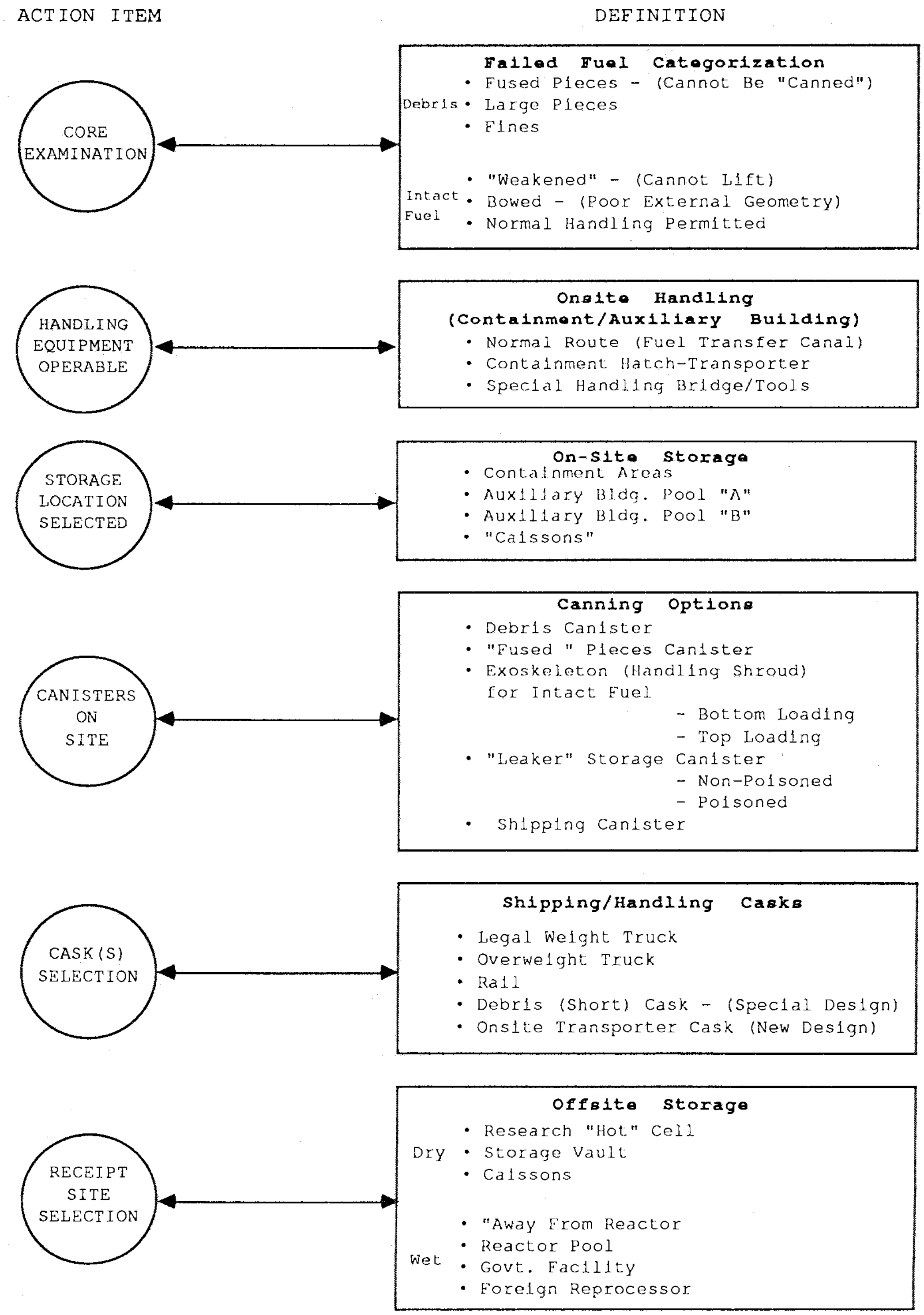


Figure 8-3. Key Decision Areas for TMI-2 Defueling

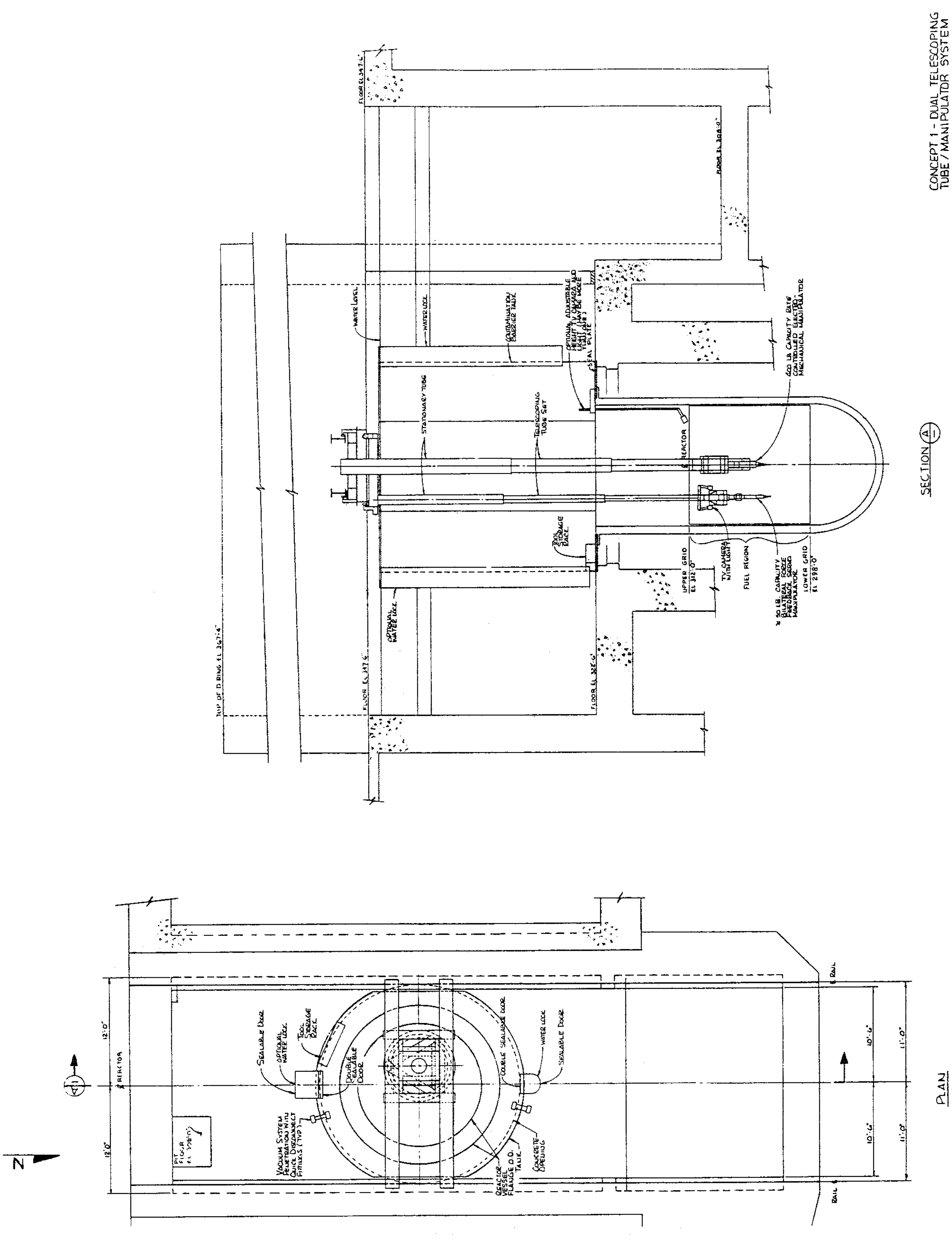


Figure 8-4. Dual Telescoping Tube/Manipulator System

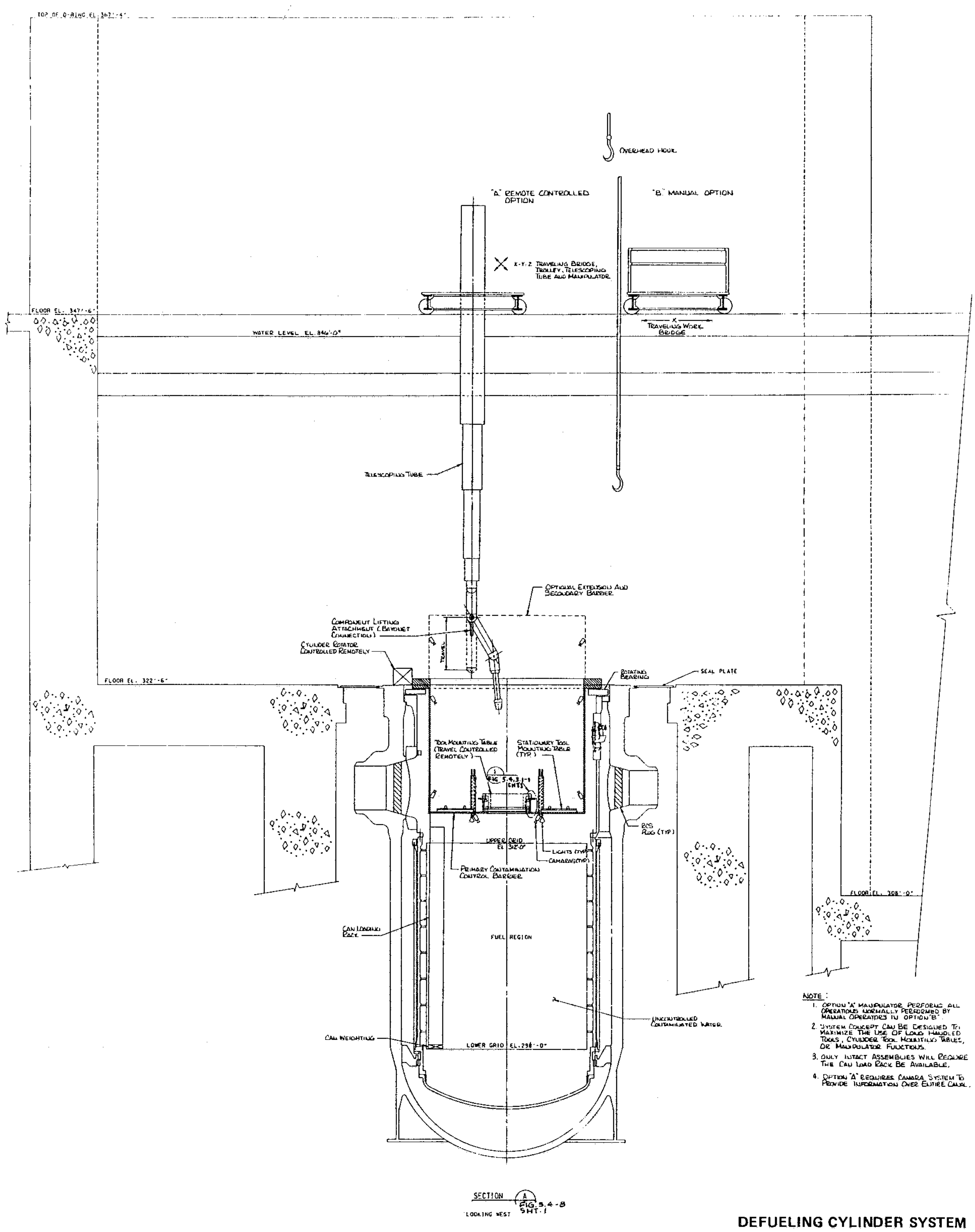


Figure 8-5. Manual Defueling Cylinder System

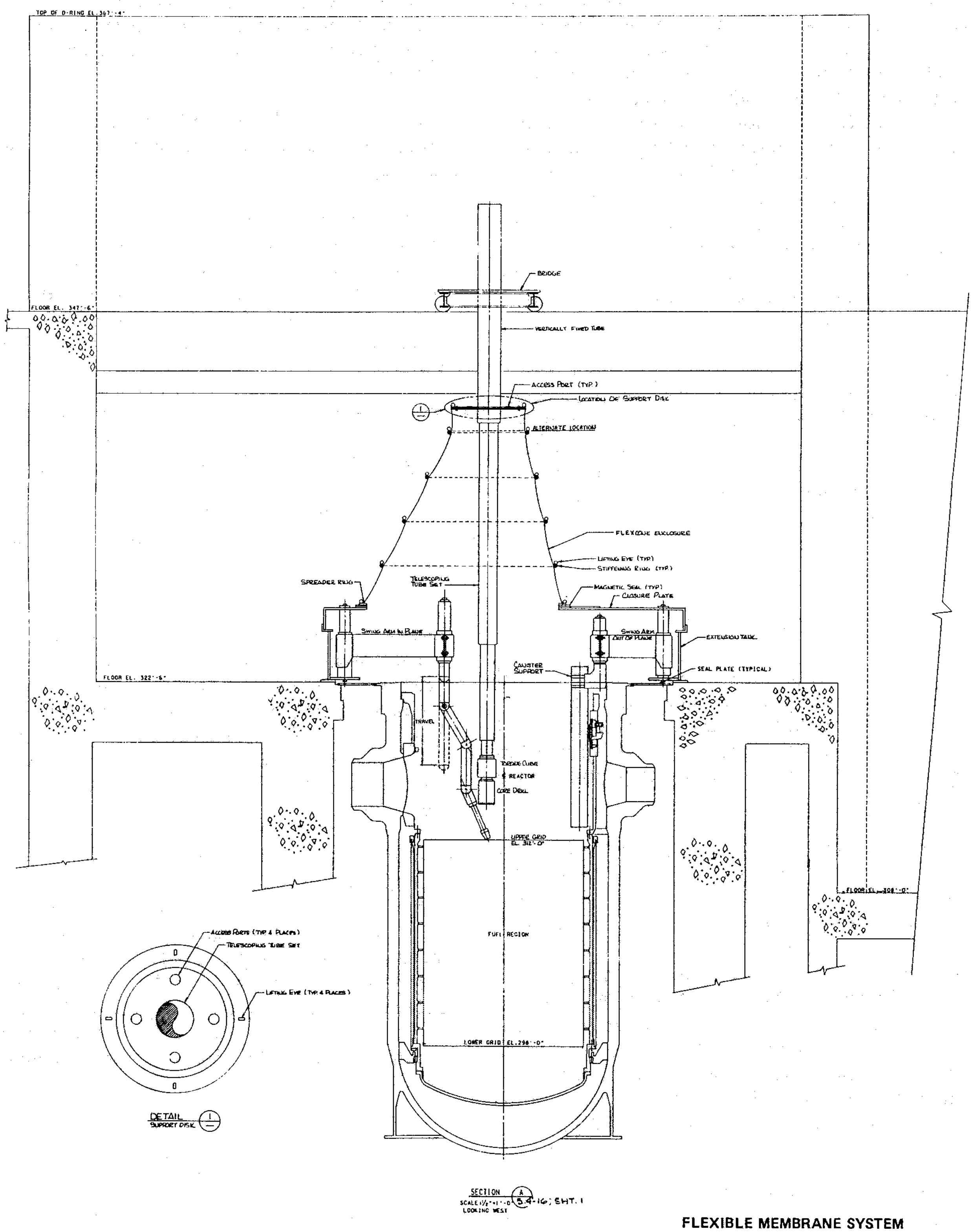


Figure 8-6. Flexible Membrane Defueling System

ring, and a small rotating ring) placed on the reactor vessel flange. The shield would have contained access ports that could be indexed to any location over the vessel. Core removal equipment would have been designed to interface with the access port and all canister transfer operations would have proceeded with bottom-loaded transfer casks positioned over the access port. The dry defueling system was rejected outright because: 1) a single-access port was too limiting to accommodate defueling operations. Canister transfer operations to and from the vessel would have been extremely time consuming; and 2) the system would have required a heavy transport cask to be handled inside the containment.

With the exception of the dry defueling concept, each of the proposed concepts assumed that defueling would require the fuel transfer canal to be flooded for shielding. Even the dry defueling system was assumed to have some tasks that required a flooded canal. All the systems used similar, although unspecified, tools to remove the fuel. The development of these tools was to be the next step in the planning for defueling.

8.2.3 Debating the Approach

Evaluation of the sequential plan and the proposed tooling concepts continued throughout 1981. Since debate on the merits was constrained by the lack of knowledge about actual conditions, two approaches emerged, both intended to focus limited resources in the most fruitful direction:

- Gather more data before committing to elaborate plans—This approach was grounded in the belief that, although the existing defueling plan was well conceived, its assumptions about core conditions might be incorrect and could lead to either inadequate or overly expensive preparations. Because the spectrum of possible core conditions was so broad, detailed plans and specifications should not be developed. This approach recommended shifting the project emphasis to that of gathering as much information about the core as possible, including a video examination and series of tests and analyses (TMI-2 TAAG 1982a).

- Proceed on the basis of GEND-007—While this approach did not dispute the value of new data, it stressed that the limited resources and time would best be spent on removing the reactor vessel head for direct access as soon as possible—while at the same time proceeding with tool development. GEND-007 was felt to adequately bound the core damage for tool development, at least to the extent necessary to start defueling.

The second approach won out after extensive debate, although some significant data acquisition work was undertaken:

- Incore Instrument Damage Assessment—Research performed in the spring of 1982 suggested that the thermocouple junctions had been reformed as a result of the accident (Wilde 1983). (The conclusion that once-molten material had found its way into the lower head was disputed and difficult to prove, although true.)

- Axial Power Shaping Rod (APSR) Test—In June 1982, all eight APSRs were energized and driven in as far as they would go. Results were widely interpreted as proof that the core was not badly damaged because four of the rods were inserted easily (Burton 1982). (In reality, the four had been destroyed during the accident.)

- Quick Look—In July 1982, a small video camera was inserted down through a leadscrew penetration in the reactor vessel head. The view showed that the damage in the upper core region was at least the Reference prediction made by GEND-007. Visibility was not clear enough to fully survey the damage, but the upper core center was obviously a 1.5-m deep void (Fricke 1982). Section 5.4.1 discusses this important operation.

Quick Look was a major undertaking for the project and provided a challenging sense of reality to the postulations about core damage; however, it required significant resources and did not change the approach to defueling because it was not comprehensive or conclusive. The project team could still assume that the outer fuel assemblies could be removed in a normal fashion and that only the center region of the core was damaged. A complete picture of the core void region would not emerge for another year.

8.2.4 Debris Defueling Working Group

The Quick Look showed that a large portion of the upper core material was rubble. Although this condition was anticipated in GEND-007, it had been expected that the rubble would be intermixed with intact fuel assemblies.

After Quick Look, it became evident that substantial quantities of material could be removed quickly using a vacuum or hydraulic dredge system.

With this in mind, the Debris Defueling Working Group was organized to draw on the experience of the industry and the DOE to recommend specific defueling techniques. The focus on techniques was intended to crystallize planning by focusing on near-term goals (Henrie 1984).

The task group developed a consensus defueling philosophy and sequence based on the assumed condition of the core shown in Table 8-2. Approximately 1% of the debris was assumed to have left the core region, primarily transported by fluidization.

The following defueling sequence was proposed after the removal of the reactor vessel head and upper internals (plenum):

1. Install a cylindrical barrier above the reactor vessel core to avoid contamination of the refueling canal as the water level was raised. Brackets, etc., for locating equipment on the inner wall of the barrier should be included as part of the barrier design. The method for moving canisters over or through the barrier must also be decided in advance and accommodated in the barrier design.

2. Use a well-developed "hydraulic vacuum dusting" tool and a particulate separation and removal system to hydraulically remove the loose fines (below 500 microns) from the exposed surfaces in the reactor vessel.

3. Use extended tongs to pick up exposed loose pieces of rubble that were longer than approximately 2.5 cm and place them in a debris bucket.

4. Use a well-developed, jet-boosted hydraulic vacuum nozzle and the particulate separation and removal system to hydraulically remove the exposed loose rubble and fine debris.

5. Fracture or cut the exposed agglomerated material into sizes less than approximately 30 cm. Relatively poor bonds were expected between the frozen material (mostly silver) and fuel debris. Further, a study of the core temperatures that occurred during the accident indicated that the agglomerate was probably not continuous and uniform. Therefore, the use of extended crowbars and jackhammers with spade-like tips might be preferable (at least for a first try) to the use of power cutting tools in reducing the agglomerate to manageable sizes. As manageable chunks were produced, they would be placed in canisters and removed.

6. Repeat Steps 2, 3, and 4, as necessary to remove loose particulate and rubble.

7. Grapple and remove the fuel assemblies in the outer rows that may be intact and not frozen in place. Working inward from the periphery of the core, loosen the remaining fuel assembly segments, grapple and lift them, place them in fuel canisters and remove them.

8. Repeat Steps 2, 3, and 4, as necessary.

9. Extend a long, slim (approximately 3.8-cm dia.) vacuum probe, through the holes in the lower core support assembly to the bottom of the reactor vessel and remove any loose fuel debris that might have settled there.

10. With what should be over 99% of the fuel material removed from the reactor, it might then be prudent to replace the reactor vessel head, operate the reactor coolant pumps to suspend and entrain loose fine particulate, then filter and demineralize the water by processing a side stream.

The Debris Defueling Working Group also developed a proposed vacuum system for defueling. It used a jet booster nozzle to pick up the debris and had a multi-staged solids separation and removal train. It used cyclone separators because of the low-pressure drop characteristics and a zeolite ion exchange system to remove soluble radioactive nuclides from the effluent water. The water was returned to the reactor vessel. Figure 8-7 is a proposed layout of the system.

Project management rejected the concept of operating the reactor coolant pumps to clean up the reactor coolant system because it would take more effort to repair the pumps than to decontaminate the system by more conventional techniques.

8.2.5 Alternative Defueling Concept

An alternative approach was developed based on a desire to expedite the reactor vessel head removal by developing a baseline defueling concept. In this approach, defueling plans were to focus only on the known conditions of the core. Since Quick Look had established

Table 8-2. Debris Defueling Group Assumed Core Conditions

(Mass Percentages)

Intact	59.9
Frozen	4.0
Rubble (2.5 cm to 800 microns)	31.7
Fines (800u to 4 microns)	4.3
Fine Fines (<4 microns)	0.1

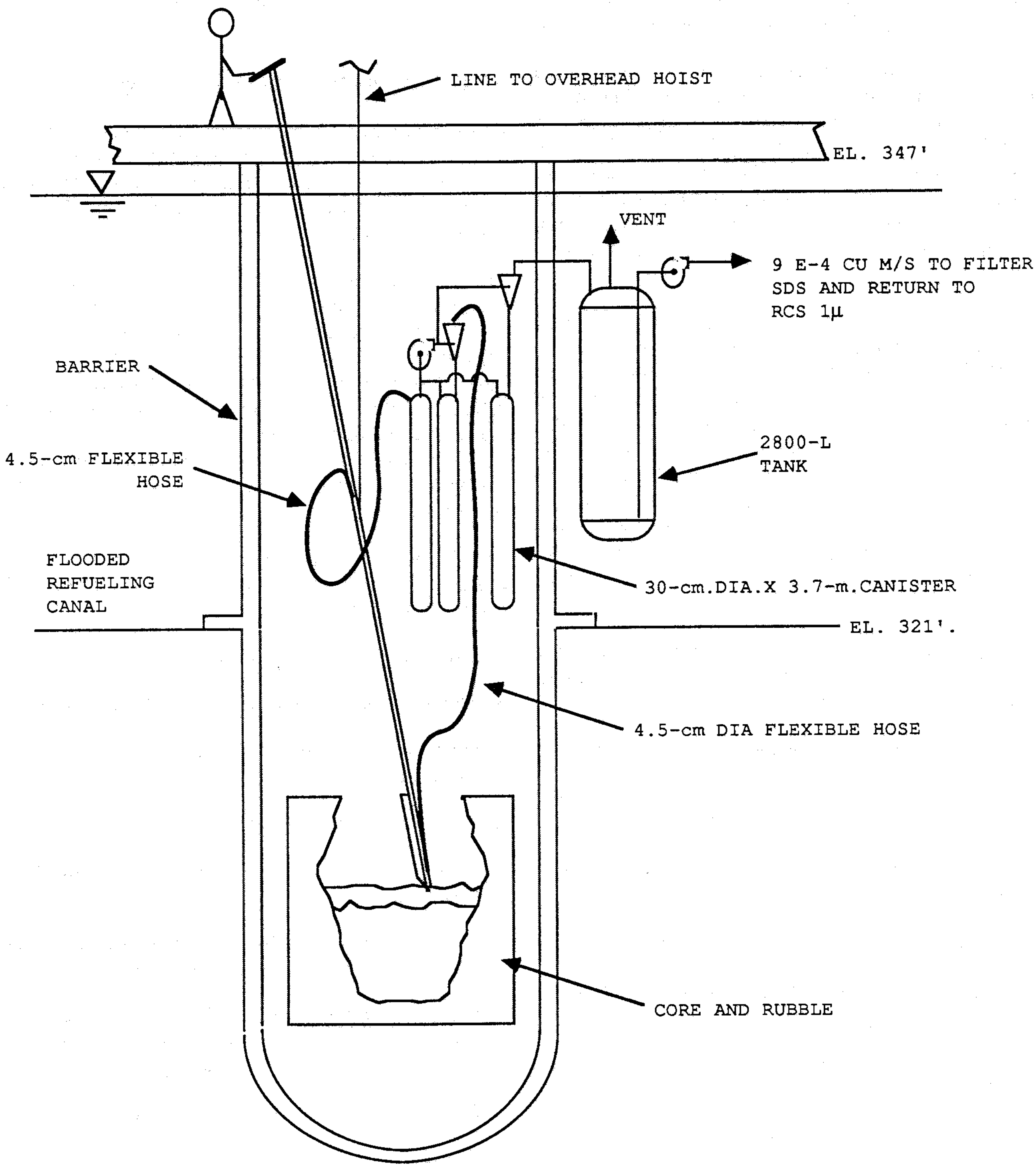


Figure 8-7. Proposed Reactor Core Rubble Removal System

that a lot of rubble in the center region of the upper core had to be removed, the best course of action was to get the head off and to remove the rubble bed. After these tasks were accomplished, the next step would be clearer.

This method would have used a flooded refueling canal for shielding the removal of the reactor vessel head and plenum, while preserving the dry defueling approach as an option (TMI-2 TAAG 1982b, 1983b). The early defueling sequence would be: 1) remove the head and plenum; 2) remove the rubble bed; 3) install a temporary cover over the vessel; and 4) determine the character of the material under the rubble bed before undertaking the next step.

The system consisted of a self-supporting metal tank mounted on the refueling canal floor around the reactor vessel. This tank would have a dedicated suction point for the existing submerged demineralizer system (SDS) and a dedicated vacuum system attached. During this time, plans for a separate defueling water cleanup system (DWCS) were also being developed. The water processing design criteria were so conservative that the SDS was not considered adequate; the new system was developed to handle the expected large volume of water in the flooded refueling canal. See Section 6.5 for the development of the DWCS.

The approach stressed the use of hydraulic vacuum defueling tools and long-handled tools. Some key attributes of the proposal were:

- Integrate defueling and water cleanup strategies.
- Decontaminate only enough to achieve a safe working environment for the operators.
- Work as close to the top of the reactor vessel as practical.
- Use a manually operated vacuum system.
- Design for "blind" defueling; i.e., expect poor water clarity.
- Use existing systems as much possible for water cleanup.
- Develop the next step after rapid completion of the first.

At this point in 1983, plans for an early head removal were eliminated by the delay resulting from safety allegations regarding the polar crane refurbishment (see sections 2.5.3 and 8.4.1.1). Work on an overall defueling strategy did continue, however, as did development work by an offsite contractor (see Section 8.2.7).

8.2.6 Preliminary Defueling Strategy

In the spring of 1983, the project team began to develop a defueling strategy plan that integrated all the existing data and the proposed technical approaches. It was to be a consensus building document that would focus the project's activities.

This effort embraced the bulk of the recommendations of the Debris Defueling Task Force regarding the sequence of the defueling activities. It also accepted the core conditions summarized by that group and by GEND-007. The resulting Preliminary Defueling Strategy (Skillman 1983) was hampered by the fact that the core was actually damaged more severely than either Quick Look revealed or GEND-007 predicted, and that significant quantities of core debris had relocated outside of the core region.

The Preliminary Defueling Strategy set forth a general objective: minimize the time necessary to defuel. It also set limits on this objective:

- Minimize disturbance of the debris bed.
- Remove as many intact fuel assemblies as possible.
- Drop as little material as possible below the lower grid structure.

The sequence of defueling steps was as follows:

- Clear the working area; i.e., remove all shards, debris, and fines.
- Begin removing intact fuel assemblies from the outside in, preferably starting in the least damaged quadrant.
- Place plugs over removed grid support plate holes to prevent new debris from falling into the lower core support assembly/lower head region.
- Remove debris over intact stubs of fuel.
- Remove all intact stubs.
- Clean all exposed surfaces of the lower core support assembly and core former wall.

The central assumptions of this approach were that the core was essentially all located within the original core region, and that the damage was limited to the central third of the core. Based on these assumptions, it was reasonable to attempt to remove all of the intact fuel assemblies first because that would remove the most material quickly and would also provide side access to the damaged fuel assemblies in the center. The least damaged fuel was assumed to be on the periphery of the core. If this was successful, the more difficult assemblies in the center could be grabbed from the side, which would have made removal easier.

New core data, which became available in the fall of 1983, undermined this approach. First, a sonar mapping effort resulted in a topographical image of the core void region. Next, a TV camera was inserted into the reactor vessel to verify the location of a RCS sample suction line. The water clarity in the reactor was much better than it had been during Quick Look. This permitted the first complete video survey of the core void, which was found to be much larger than originally believed.

Very few intact fuel assemblies existed and, in some locations, the core former wall could be seen. The rubble extended across the entire core diameter. These data acquisition efforts rendered moot the defueling plans that assumed a large number of undamaged fuel assemblies.

8.2.7 *Robotic Defueling*

As early as 1981, the DOE had urged the TMI-2 project management to select a single subcontractor to design and supply defueling tools and to pursue the post-head removal defueling operation for the following reasons:

- Cradle-to-grave responsibility would help ensure a thorough and competent job.
- Relevant defueling experience could be applied to reduce the time and resources needed to develop a satisfactory level of expertise.
- Because no one right approach existed, the process of selecting a single subcontractor would focus attention on the earliest possible selection of a defueling approach, which would then have a champion in the subcontractor proposing it.
- One subcontractor would better ensure that tradeoffs between one defueling subsystem and another (e.g., one to control contamination in defueling water) were coordinated. The integration of equipment, procurement, installation, and operations would be better, too.
- An overall approach implemented in a step-wise, integrated manner would allow the DOE to play a more active role sponsoring important activities (Feinroth 1981).

In the spring of 1983, the project management decided to retain Westinghouse Electric Corp. as defueling subcontractor because of its experience in the areas of tool design, robotics, and nuclear fuel handling.

The project team retained control over the tool development program and produced a performance specification for remote/manual tools and fines/debris vacuum systems. This specification identified a defueling plan that was essentially unchanged from the one proposed in 1982. It still utilized a flooded fuel transfer canal and work stations located on El. 347'. The tool placement was done remotely and the tools would need to be designed to defuel the core with remote/manual control.

The specification identified the tools to be developed; however, it also requested that Westinghouse suggest an alternative defueling option. Westinghouse responded with two different approaches: the first was a simple response to the specification; the second was a completely novel approach.

The novel approach was the Westinghouse Candidate System Design (WCSD), which was completely robotic (Westinghouse 1983). Instead of transferring the fuel debris into shipping containers in the vessel or the fuel transfer canal, the WCSD would convert the entire core into vacuumable rubble and transfer it directly into canisters in the fuel handling building.

The WCSD had three main elements: a remotely operated service arm (ROSA), a shredder, and a debris vacuum/transfer system. ROSA would take the place of human operators in moving the tools and debris in the vessel. It was a "human-like", computer-controlled and programmable, electro-servo-powered arm (see Figure 8-8).

This approach had many arguments in its favor:

- It avoided the time-consuming process of placing fuel into shipping canisters within the confines of the reactor vessel and then transferring the canisters into the fuel handling building.

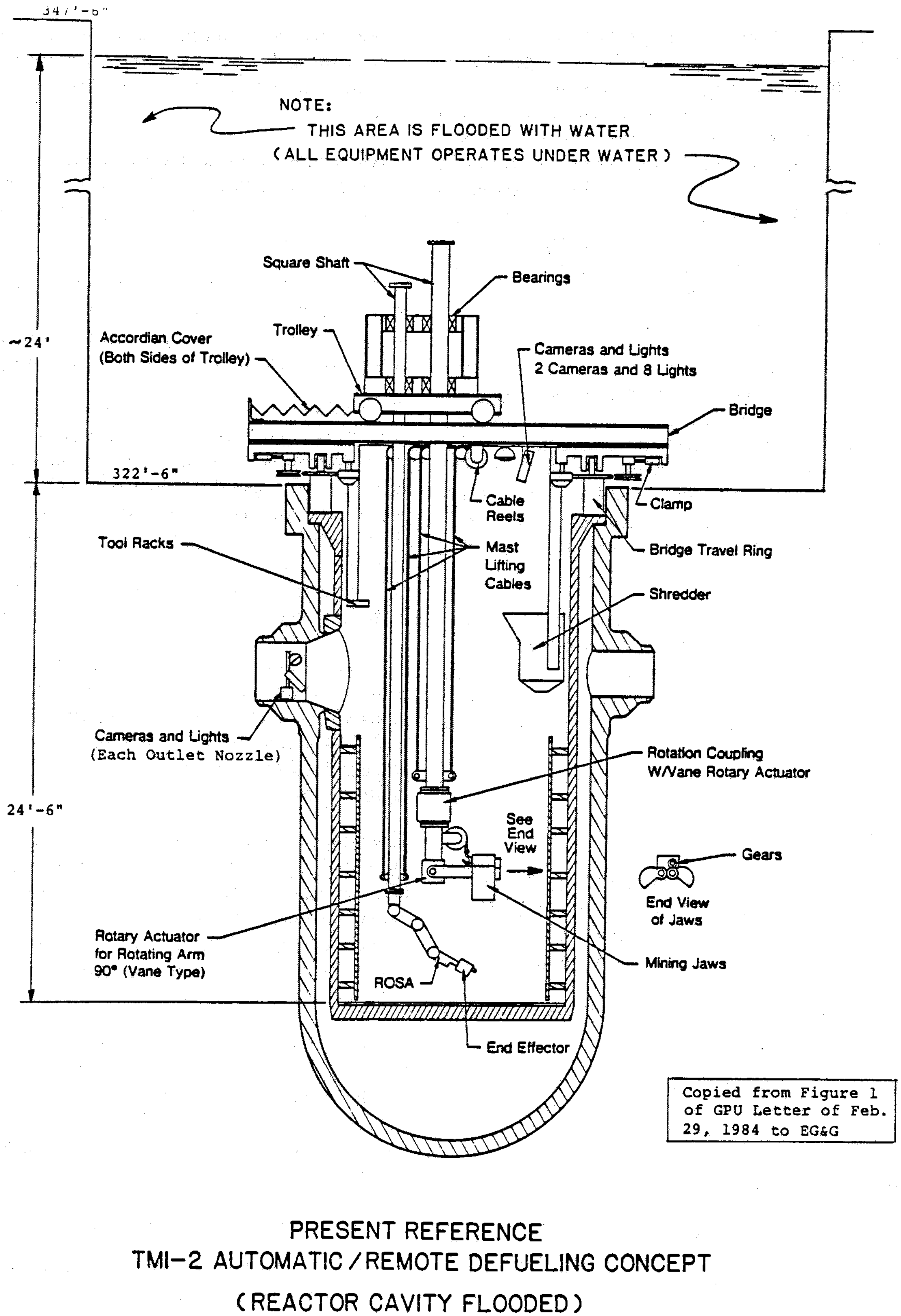


Figure 8-8. Automatic/Remote Defueling Concept

- It avoided a great deal of the in-containment work associated with preparing for or conducting defueling activities.
- It reduced the number of in-containment workhours and could reduce the need to decontaminate as thoroughly as previously required.
- It avoided the most troublesome problems with the remote/manual approach: 1) water clarity was less of a problem because ROSA could be programmed to work blind and because the volume of water to be processed would be vastly reduced; 2) problems associated with tools long enough to be operated from El. 347' were eliminated; and 3) material handling problems associated with moving shipping canisters inside containment did not exist.

One of the most technically challenging elements of the WCSD was the development of the shredder. No one had ever used a shredder to grind up intact fuel assemblies or once-molten core material. A great deal of developmental engineering would be required to develop this concept into a workable design.

The delay in starting defueling was justified by a projected reduction in actual defueling time and by the extremely low radiation exposures associated with a completely robotic approach. The substantial concerns centered on: 1) the release of radionuclides into the reactor coolant system, which would raise general radiation levels and challenge the water cleanup system; and 2) pumping the debris outside of the containment building without first packaging it—a system failure could have very severe radiological consequences.

In early December 1983, a task group was chartered to review all the various defueling system approaches still under consideration (GPUN 1984a). These approaches were:

- Remote/Manual Defueling System—The Remote/Manual Defueling System was essentially the same defueling approach first proposed in late 1981. It had more detail and more specifics, but it was essentially a low volume per unit-time defueling system based loosely on normal defueling practices. Long-handled tools and a selection of remote-powered tools were to be used to move fuel debris into the shipping canisters in the vessel. These canisters would be removed from the vessel and transported underwater through the refueling canal to the fuel transfer mechanisms.
- Remote/Manual Pick-and-Place, and Shredder—This concept was a merging of the WCSD and the remote/manual concepts. Long-handled tools would be used to remove the spiders, end fittings, and other loose debris on top of the rubble bed. This would provide time for the shredder/vacuum system to be developed and tested. The rest of the fuel would be removed by the WCSD approach.
- Robotic, Pick-and-Place with Shredder—This was another merged concept. The ROSA portion of the WCSD would be installed first. It would pick up the loose debris on top of the rubble bed and place it in the shipping canisters. These canisters would be transferred to the fuel handling building in a manner similar to the remote/manual approach. Once this material was removed, the shredder/vacuum system would be installed. The rest of the material would be removed by the WCSD approach.
- Robotic, Mining and Shredder—The Robotic, Mining and Shredder concept was the original WCSD concept. The entire core would be processed through the shredder/vacuum system and transferred directly to the fuel handling building.

The task group recommended that the WCSD concept be adopted in full and that, pending a review of some safety concerns, the preliminary design should begin at once.

At this point, DOE became concerned about the effects of this potential defueling method on its R&D function. If the core was ground up and sluiced into canisters, all of the spatial information regarding fission products and control material would be lost. As a result, DOE began to develop a core sampling program using a core boring machine. This machine later became an integral part of the defueling program as described in Section 8.6.1.2.

8.2.8 Dry Defueling

Substantial questions and concerns surrounded the WCSD. Technical concerns with the shredder/slurry system centered on the development and licensing of an unproven technology and on the uncertainties surrounding the release rate of radionuclides during shredding. In addition, accepting the recommendation meant that the project team would be embarking on a three-year engineering development program before the first piece of fuel could be removed from the reactor.

Rather than attempting to minimize the theoretical overall time necessary to defuel (which was the promise of the WCSD), executive management wanted a method to start defueling as soon as possible. In this way, the project team could deal with the real challenges associated with defueling, not with engineering problems associated with design.

In late March 1984, with funding essentially in place, the polar crane finally requalified, and reactor vessel head lift near, the WCSD concept was rejected. Instead, project management sought a "fast track" defueling concept that could begin moving fuel April 1, 1985. During an intensive planning period, several approaches were studied.

The objective of developing this "fast track" system recognized the R&D nature of the work but sought to recover a slipping schedule and complete the cleanup program by mid-1988 (Clark 1984). The primary elements of the plan were:

- Focus on the earliest practical start of significant fuel removal—April 1, 1985. Help all parties identify and resolve technical and regulatory issues.
- Proceed expeditiously with design and fabrication or procurement of equipment or facilities, while recognizing the need to modify or augment them as work proceeded.
- Work with DOE, EPRI, and others to achieve needed flexibility.
- Accept GPU's decision not to expend resources to preserve the plant or equipment in anticipation of recovery.
- Renew and focus efforts to modify or adopt regulatory requirements to the current TMI-2 situation, which involved lower total risk and significantly different controls.
- Use resources effectively; e.g., minimize expending resources on contingency efforts such as disassembling the upper internals (plenum).

The first plan resulting from this approach began defueling by vacuuming loose debris through the plenum in order to confirm core conditions. Following plenum removal, the refueling canal would be partially flooded for shielding and vacuuming would resume with ROSA positioning the vacuum nozzle. Long-handled tools would be used to support the vacuum system; fuel canisters would be located in the flooded deep end of the refueling canal (Klanian 1984).

This approach was not completely acceptable and so a variation was pursued—dry defueling. It was similar to the one rejected in 1981 (see Section 8.2.2). It could also be implemented quickly with little development work. A Dry Defueling Scheme Task Force was organized with the charter to develop the dry defueling concept far enough along to permit it to be evaluated along with the other concepts. The task force produced a study in late May (Austin 1984). The concept developed was different from any of the others in several ways:

- The fuel transfer canal would only be flooded in the deep end where the fuel transfer mechanisms were located.
- The workers would work on a shielded platform directly above the 1.8-m high internals indexing fixture located on the reactor vessel flange. The platform was to be of a simple design with removable shield plugs.
- The tools would be manually operated.
- The water in the fuel transfer canal and the reactor vessel would be separate, greatly simplifying the water cleanup challenge.

In addition, the dry defueling concept was compatible with the ROSA and the other robotic tools being designed.

A dam was to be installed in the fuel transfer canal deep end to provide the water to shield the plenum and the transfer of fuel canisters, which were to be loaded inside the vessel. The main refueling bridge would be used to transfer fuel canisters from the reactor vessel to the deep end of the canal.

The logic for this method proved convincing:

- The area radiation levels in a dry canal would be, ironically, the lowest in the containment (as low as 10 mR/h).
- The fission products in the reactor coolant would be retained in the minimum volume of water. Lessening the volume of water was crucial to speeding the start of defueling by minimizing equipment development and logistics. (It proved especially important in view of the later problems with water clarity.)

- Fuel handling could begin using simple, long-handled tools and proceeding to more complex methods as necessary.

- The depth of water to the top of the debris bed was shallow enough to allow manual operation of long-handled tools.

- The arrangement afforded the greatest flexibility to adapt to changing needs (Calhoun 1989).

The dry defueling concept was adopted in the early summer of 1984, and all subsequent work was focused on detailed design to develop this concept (Kintner 1984).

Dry defueling was envisioned to comprise two phases, depicted in Figure 8-9; resources were focused on the first:

- Early Defueling—Remove the rubble bed and any solid items contained within or lying on top of this bed (including partially intact fuel assemblies, broken fuel rods, fuel assembly end fittings, control rod parts, and masses of fuel-pellet size material). It was to be accomplished as much as possible by vacuuming, complemented by long-handled tools for a limited degree of pick-and-place (loading fuel into canisters or moving it so as not to impede vacuuming).

- Bulk Defueling—Remove the remaining core material. This would be conducted with long-handled tools, supported as necessary by robotic devices. (Bulk defueling was an accurate description since it did constitute the bulk of core removal and did require a variety of techniques, but the assumption at the time was that it would be relatively straightforward, hopefully consisting primarily of loading damaged fuel assemblies into canisters.)

The design of the defueling water cleanup system (DWCS) remained essentially unchanged, even though the volume of water had been reduced. The DWCS was both to remove radionuclides from the reactor coolant and filter out suspended particles to ensure visibility (see Section 6.2.3).

8.3 Development of Fuel Handling/Shipping Strategy

In parallel with efforts to develop a reactor defueling strategy, the project team also had to develop compatible plans and designs to package and dispose of the core material. The issues were all interrelated, beginning with questions of: What was the final destination of the material? How to ship it? How to package and store it (in whatever condition it existed)?

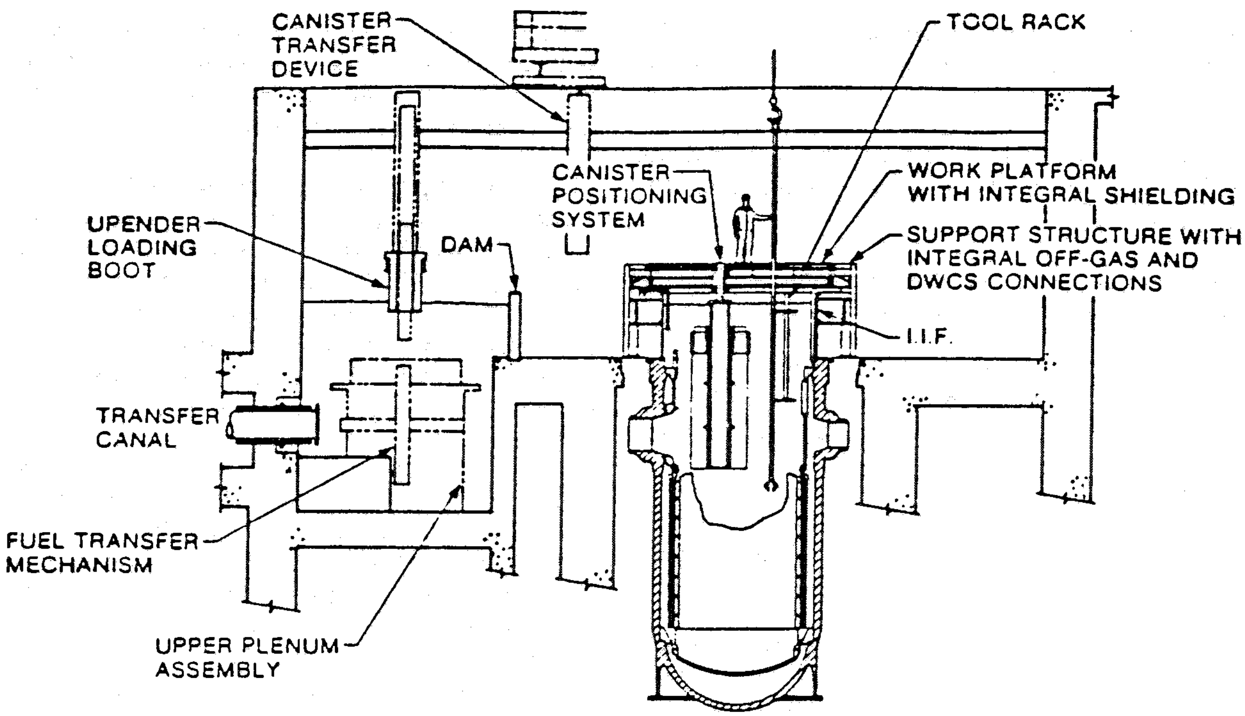
The early scoping plan (Anderson 1980) had recommended temporary onsite storage in either a fuel pool or a cask (caisson). The first reactor disassembly and defueling plan (BNI 1981) had designated fuel pool "A" as the storage area for an indefinite time because no agreement had yet been reached on the final disposition of the core. Both plans had envisioned some type of canister containing the core material.

No specific designs could be made until the question of final destination was resolved. In July 1981, a memorandum of understanding between the NRC and DOE specified the disposal method for TMI-2 radioactive waste, including fuel debris (see Section 6.5). Per that agreement, some of the fuel debris would be taken by the DOE for research while the rest remained in storage on site pending resolution of the national high-level waste disposal issue. In March 1982, the memorandum was revised. To prevent TMI-2 from effectively becoming a long-term waste repository, the DOE agreed to accept all of the TMI-2 core for research and temporary storage.

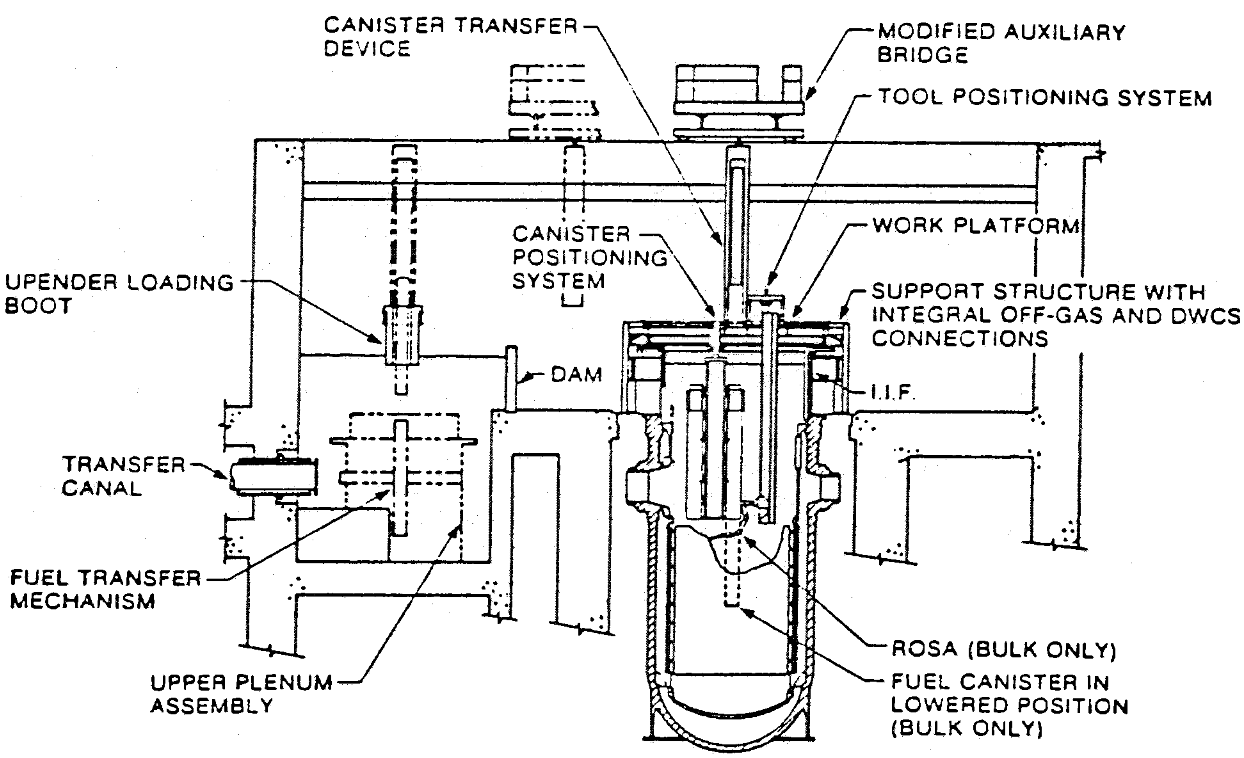
The agreement set the stage for the parallel development of canisters to contain the core debris and a cask to ship it to DOE's facility at the Idaho National Engineering Laboratory (INEL). A contract signed by DOE and GPU in 1984 specified the exact responsibilities for transportation, storage, and disposal (US DOE 1984). The cleanup project provided the equipment and workforce to remove, package, and store the debris on site. DOE furnished the cask, took physical possession and responsibility when loaded, and shipped it to INEL for research and storage until a permanent repository was developed.

Fuel core material was defined as all material contained inside the baffle plates in the original core configuration—the only exception was other material that had become inseparably mixed with authorized core material; e.g., components fused with once-molten fuel, chips, or dross from cutting operations.

INEL was selected as the most suitable site for research and long-term storage of the TMI-2 core because of its unique facilities and historical experience in conducting research for NRC and the nuclear industry on severe accidents and damaged nuclear fuel. It had hot cells to handle the waste and an experienced staff (US GAO 1987). The receipt, research, and storage of TMI-2 core debris at INEL is summarized in *Historical Summary of the*



Early defueling system.



Bulk defueling system.

Figure 8-9. Early and Bulk Defueling Systems

Fuel and Waste Handling and Disposition Activities of the TMI-2 Information and Examination Program (Reno and Schmitt 1988) and detailed in many other publications (see Bibliography for some of the reports and papers).

The core debris was to be stored underwater in the water pit at Test Area North-607, which had originally been planned for use with the nuclear airplane to store, defuel, and dismantle the nuclear propulsion system. A feasibility study was undertaken to evaluate the construction of a TMI-2 fuel recovery plant; it was never built (Evans 1982).

8.3.1 Storage and Shipment

The conceptual alternatives for temporarily storing and then shipping the fuel were laid out in 1982 (Kannard 1982). Three general approaches were considered:

- Transfer the fuel debris to the fuel handling building via the normal method; i.e., the transfer tubes. Storage would either be underwater in fuel pool "A" or dry in the cask loading pit (fuel pool "B").
- Transfer the fuel debris directly outside via the equipment hatch and ship as fast as produced. Variations of this used a cask to provide surge capacity. The potential of providing direct access for the fuel shipping cask into the containment was evaluated, including the effect of licensing a cask that could be loaded and then stored dry outside the plant until shipped. Removing the equipment hatch would permit the cask to be brought directly into containment by rail, turned upright in the deep end of the refueling canal, loaded with canisters, and then removed for storage/shipment. The ability of the floor to accept the heavy load was a concern, as were interferences with other defueling equipment. (Long-term dry storage in casks was eliminated because of uncertainty in licensability and the relatively high cost for a fleet of such casks.)
- Transfer the fuel debris directly outside via the equipment hatch and then store dry in either existing or to-be-constructed storage areas.

Storing canisters loaded with fuel in a flooded fuel pool "A" and then loading them into a cask was determined to be the most efficient and economical method, with the least total occupational exposure. The pool was large enough to ensure a surge capacity in case difficulties arose in the shipping program. This storage alternative also allowed the SDS in the cask loading pit to remain in operation. Another attractive feature was that it deviated least from standard refueling practice and plant design. The only physical impediments were the six large water storage tanks of the tank farm; these were to be removed.

In parallel with the development of defueling plans, the project team continued to evaluate the best means of fuel shipment. As more information about core conditions became available, the design of the canisters and cask was finalized. The challenges were interrelated and pressing:

- A new license (or at least a major modification to an existing certificate of compliance) would be required because casks and contents were licensed together, and no one had licensed a cask for TMI-type fuel debris.
- Several years were required to license a cask.
- Canister design affected and limited the design of the cask.
- Canister design had to be finalized soon to allow canister production in time for the scheduled start of defueling.

The initial choice for shipping was to lease a fleet of eight single-assembly legal weight truck (LWT) casks. (Two rail casks could be used to augment the shipments and eliminate surge storage requirements.) This conclusion was based on the projected financial savings of LWT casks over the higher costs associated with rail freight and cask leasing. The 1980 scoping study had supported this, noting several points in recommending single-assembly LWT casks:

- The cask certification amendment by the NRC should be easier to obtain.
- Fuel being shipped to INEL for research must finally move by truck anyway because of facility limitations.
- The remote handling of the heavy closure heads on rail casks would have been very difficult, even with special handling tools.
- Information developed for the single-assembly cask would be useful in certifying multi-assembly casks if it became apparent that certification of the latter was warranted.

As evaluations continued, the financial and scheduler advantages of shipping the fuel debris by rail were recognized. Instead of modifying and relicensing an existing spent fuel shipping cask, a new rail cask was designed, fabricated, and certified. (A few then-licensed casks were considered, but they were unavailable because of other commitments.) Rail shipment offered the following advantages over truck shipment:

- Significantly lower labor and cost requirements, in large measure because the rail car shipping (with two casks) could take place campaign-style. Also, a rail cask would hold seven canisters to one in a truck. A dedicated staff would not be needed because adequate personnel could be temporarily diverted from other tasks to load a cask. Shared use of the truck bay with Unit 1 would be much easier to schedule.

- Rail transport would require between 20 and 40 shipments, compared with 250 by truck (reducing the chance of an accident) (Morris 1984; US GAO 1987).

Several factors influenced the design of the shipping cask:

- Because of the quantity of plutonium in the core debris, the debris had to be transported in a double-containment package.

- The breached fuel rods could not be considered a form of containment.

- The canisters would not be designed to provide a form of containment because of the need for removable lids and other loading features that made such a qualification difficult (Reno and Schmitt 1988).

Therefore, the cask had to provide both barriers. In addition, a leaktight—though not leakproof—criterion was used.

A fuel transfer cask and transfer cask loading station in the fuel handling building were designed in order to: 1) eliminate the need to bring a large shipping cask into fuel pool "A" for underwater loading; and 2) avoid loading the cask while it rested on the railcar above an exclusion zone. The factors in this decision were:

- The SDS—an important water cleanup system—already occupied fuel pool "B" and would have to be partially removed for underwater loading. The "B" pool would have become as contaminated as fuel pool "A".

- Heavy-load considerations in an area containing vital Unit 1 equipment were a major obstacle. (The overhead crane was not a single-failure type and so would have to be de-rated from 90,000-kg to 45,000- or 55,000-kg. A loaded shipping cask would have weighed approximately 90,000 kg.)

- General contamination levels would have been higher. (Contamination control was a big concern. As it turned out, little decontamination of the cask was required since it was not submerged—in fact, the truck bay was not a contaminated work area during loading operations. Wet loading had looked promising at first because it eliminated the need for a mini-hot cell, transfer equipment, and seismically-qualified uplifting equipment.)

The cask is shown in Figure 8-10; the cask loading station in Photo 8-1. Two casks were ordered, along with skids, railcars, and miscellaneous handling and loading equipment. A series of safety evaluations and drop tests were performed. When these were complete and a functional checkout due, one cask and all the interfacing equipment were assembled at Hanford Engineering Development Laboratory (HEDL), which was convenient and had fewer procedural constraints than TMI-2.

There, all the crews that would work on shipping were trained on the equipment before it was shipped to TMI-2. This resulted in better training, minimized the schedule (18 months to design and build), ensured that all equipment functioned in an integrated fashion, and speeded installation at TMI-2 because of familiarity. Cask loading required an operating crew of 17, plus 11 more for pre- and post-loading work. Loading one cask required approximately 1000 jobhours (Deltete and Hahn 1990).

Because the casks were to be owned and transported by DOE, the legal requirement for NRC certification was debatable. Nevertheless, the DOE decided to license the cask via an NRC Certificate of Compliance since TMI-2 was licensed by the NRC. For an extra margin of safety, the ASME welding code that pertains to the construction of reactors was used, even though the casks would experience little of the same pressure (US GAO 1987). Extensive documentation accompanied each shipment —because of the nature of the core material, some of the specified information about the contents was very difficult to determine; e.g. SNM accountability, curie content, and thermal wattage determination (Deltete and Hahn 1990).

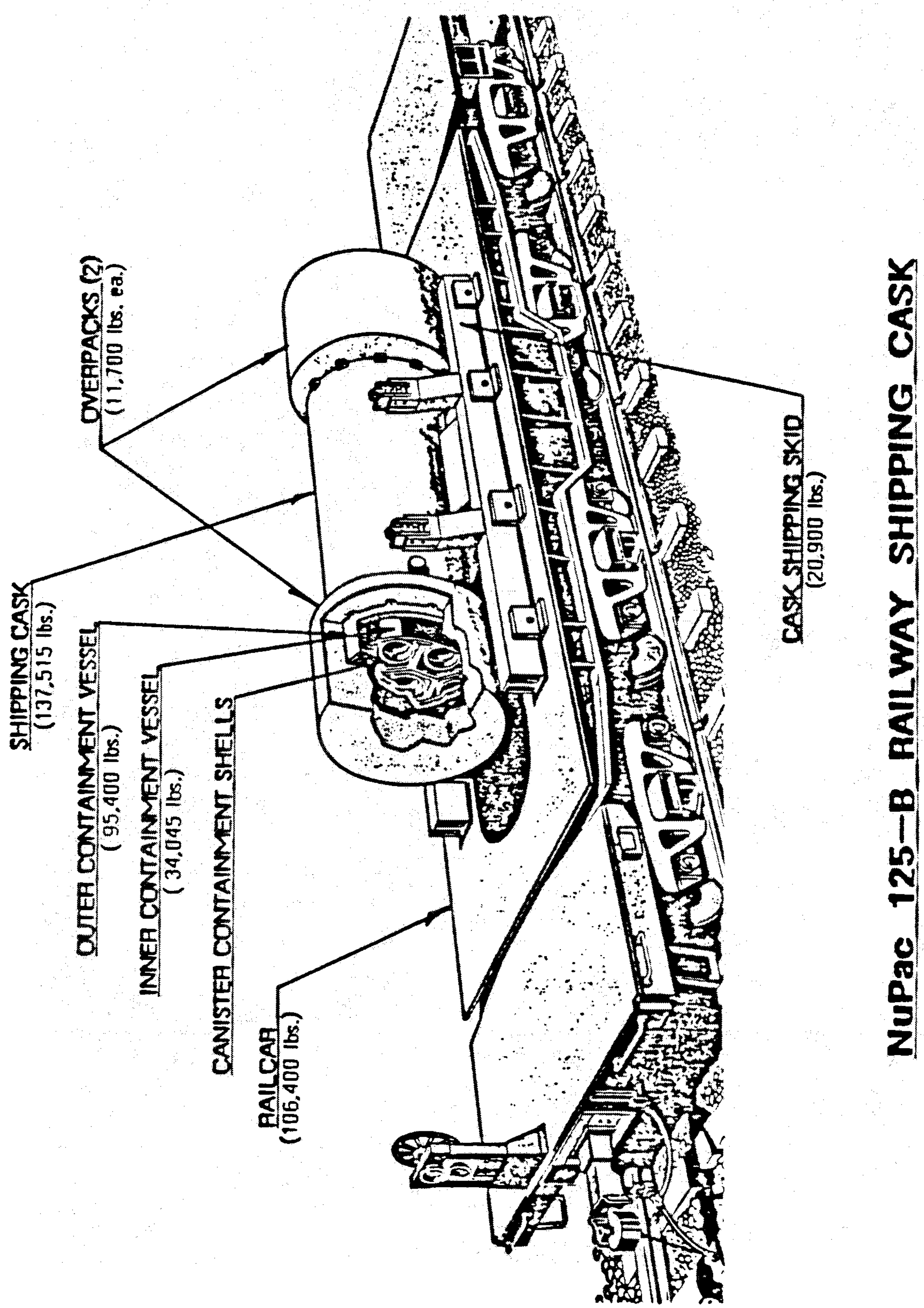


Figure 8-10. Fuel Shipping Cask

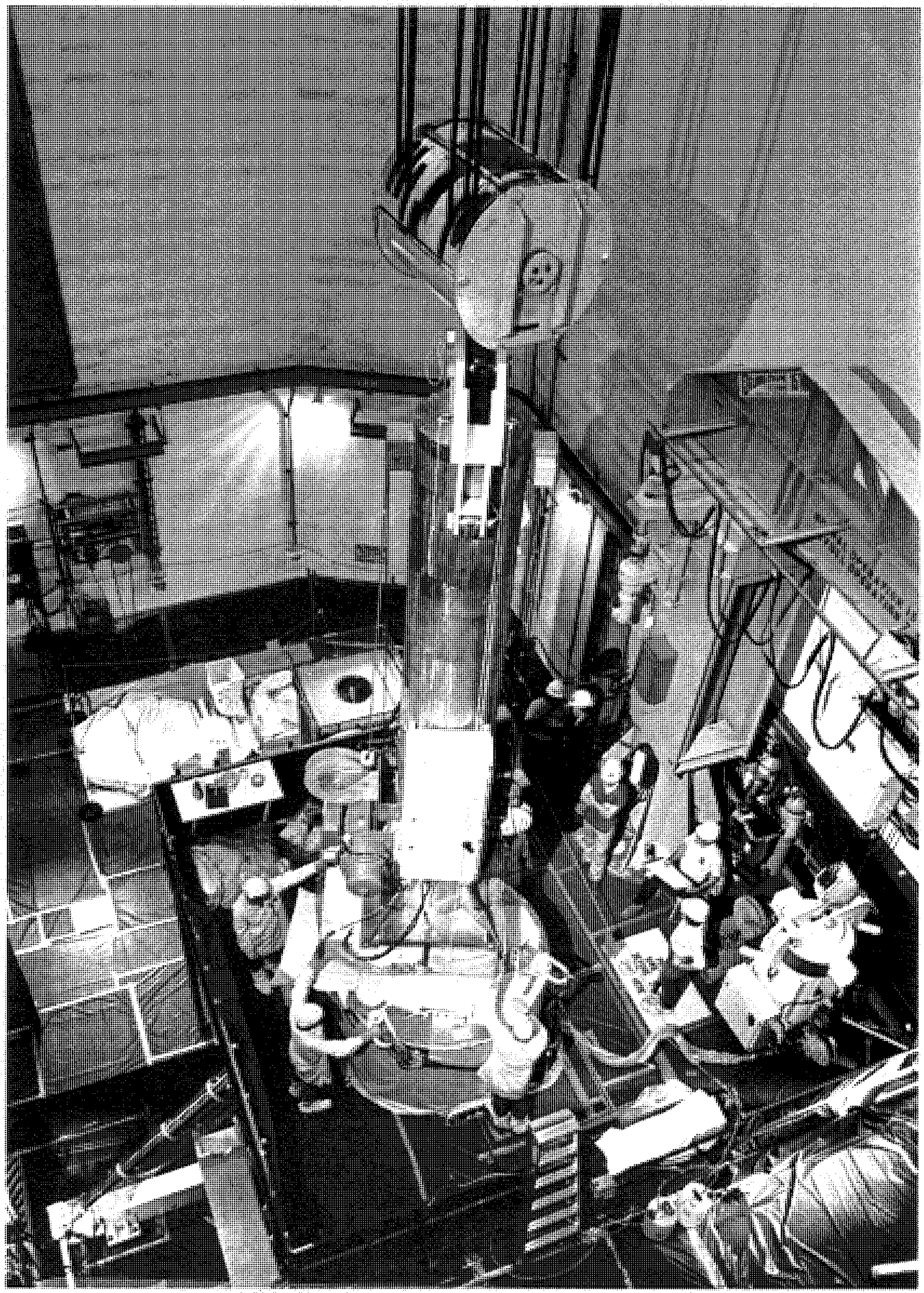


Photo 8-1. Fuel Shipping Cask Loading Station

In 1986, after the shipping campaign began, the project team ordered a third cask, which it leased. Ordering the cask was rationalized on a cost basis to speed the schedule and keep up with the projected rate of defueling. Three casks would provide significant savings, not in terms of freight rate savings for a dedicated train—which remained at the same ton/mile rate—but in the cost of expedited service provided by three-car dedicated trains.

In reality, plans for expediting the fuel shipment schedule with a third cask were never realized. The delays, surprises, and difficult conditions encountered during defueling operations slowed the rate at which loaded canisters were produced. The casks were often on site awaiting the generation of enough canisters for a shipment.

The only cask-related delay in shipment occurred in 1988, when concerns over the O-ring material in the cask resulted in the material being changed to a different grade of elastomer. In spite of meticulous and extensive attention by the DOE to public communication, route safety, and local/state government interaction, the brief delays in shipment that did occur were most often caused by nontechnical factors (e.g., intervener protests along the shipping route or a general dispute between the DOE and railroads over the transport of radioactive materials).

8.3.2 Defueling Canisters

Conceptual work on the defueling canister design began in 1980 (Anderson 1980) and was refined in 1981 (Townes 1981). Three canister alternatives were initially considered:

- Single multi-purpose canister—Selected.
- Multiple single-purpose canisters—The concept was discarded because the core conditions were not defined well enough to design a variety of canisters for specific categories of fuel. It would also have entailed more cost, training time, and difficulty of operation.
- Inner shroud/outer shipping canister system—The concept called for a support shroud to be lowered over the assemblies, which were then to be drawn up into a true canister. It was discarded because the primary motivation would have been to minimize stress on the assemblies after removal—and conditions could not be analyzed to a level at which loading stresses could be predicted.

The primary criterion was to use a simple, single solution unless the data precluded such a design. Since the single multi-purpose canister concept bounded the then-known consensus of core conditions (i.e., GEND-007), it was selected. Either a square or a round variation of the single multi-purpose canister was seen as an acceptable design. An added bonus with this concept was lower manufacturing and inventory costs.

The single multi-purpose canister was originally to have been 450 cm long. The square variation of this canister would have had a 23.3 cm inside dimension, a 2,700-kg capacity, and criticality control by geometry. The round variation would have had a 31-cm inside diameter, a 3,800-kg capacity, and criticality control by neutron poisoning. Neither the square nor round variation was specifically recommended in the early years —licensability had to be weighed against canister loading in order to determine the best choice (i.e., the round variation would be harder to license but easier to load).

In selecting the round canister design, the important factors were criticality safety, structural integrity, and operator ease in loading. From the project team's perspective, the ease of loading predominated since early plans indicated that the defueling critical path was dominated by the many operations involved with loading, closure, and handling of canisters.

Other influencing factors on the canister design were the limiting dimensions and weight restrictions of the fuel transfer system and shipping cask (Rider 1986). Also, the requirement by the DOE that the canisters be stored at INEL vented and under nonborated water with no credit for poisoning further limited design options.

The length of the canister was reduced from 450 cm to 380 cm to ease handling in the reactor vessel. In addition, analysis indicated that licensing the cask for a shorter canister was recommended and would reduce cost. Debate over the proper length of the canister reflected the belief of some that a considerable number of full-length assemblies still existed in the core. This belief was later shown to be unfounded.

A boral insert inside the canister was required to ensure subcriticality under any conceivable accident scenario. With the insert in place, there was just enough room to fit an undistorted end fitting (22.9 cm). This dimension proved to be too small for some types of core debris (e.g., some end fittings had to be placed in storage drums because they would not fit into the canisters without difficult resizing). Some options to modify the canister

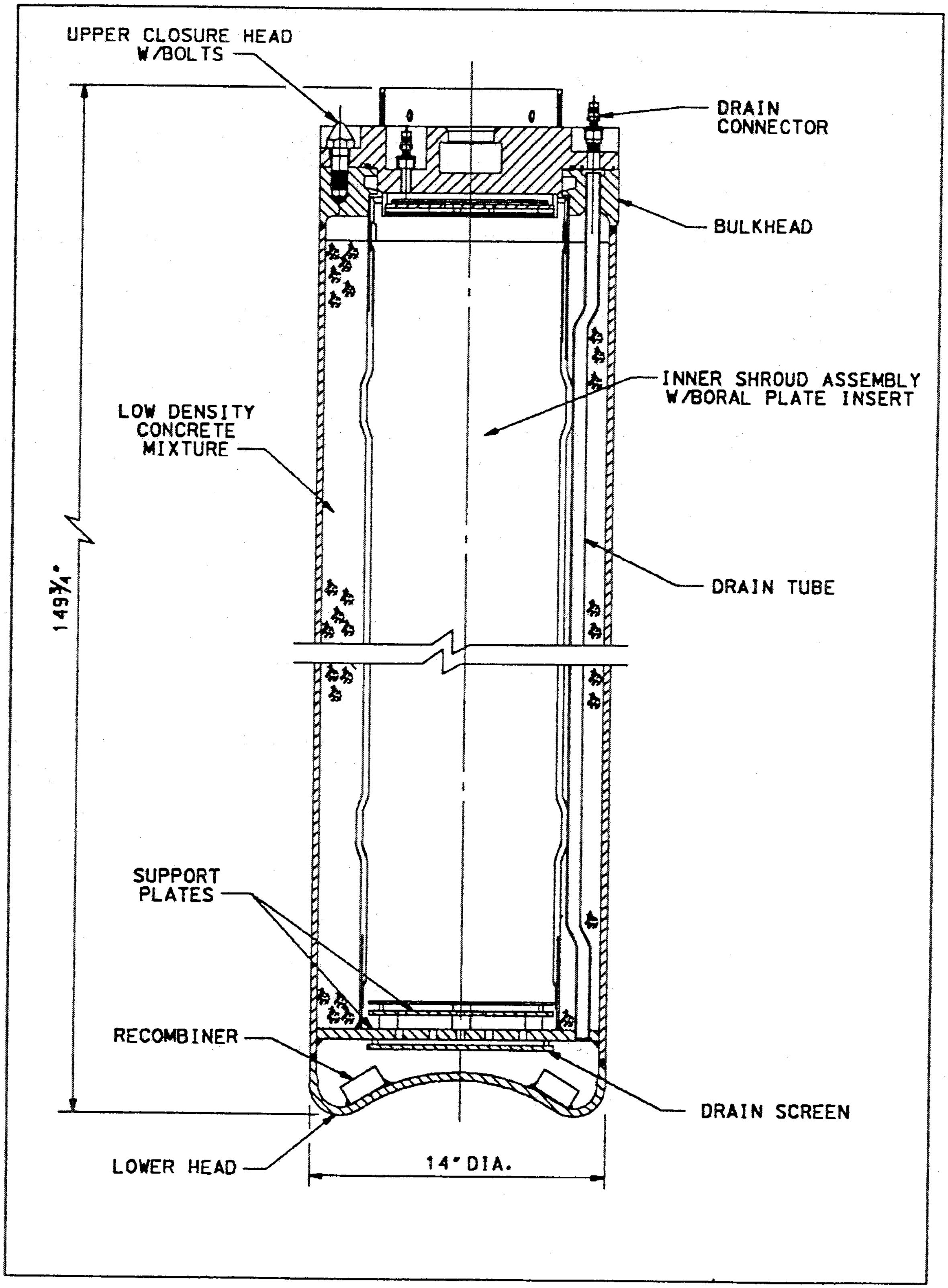


Figure 8-11. Fuel Canister

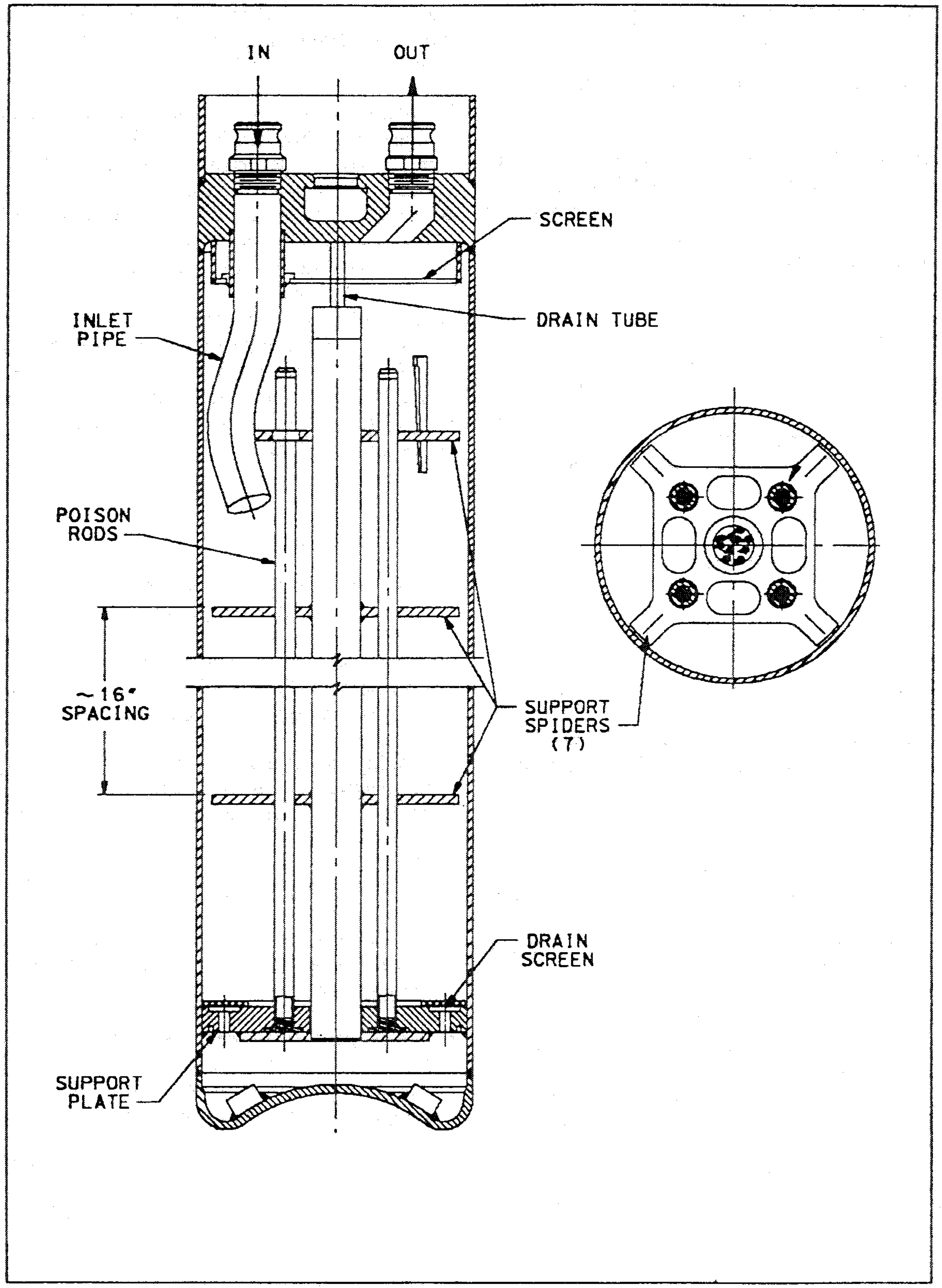


Figure 8-12. Knockout Canister

design based on core conditions and operating experience would have been useful.

Three types of internals configurations produced three different canisters:

- Fuel Canister—This was the basic canister for containing core debris. It had a removable lid and could be loaded with larger pieces of debris or most of a fuel assembly (see Figure 8-11). Since few fuel assemblies were full length, the length limitation was not regarded as a problem.

- Knockout Canister—This canister was to be used in conjunction with a hydraulic vacuum system. Water and smaller pieces of debris were pumped into the canister (see Figure 8-12). As the velocity of the water decreased in the large diameter of the canister, the pieces of debris settled out of the water. Water and fine pieces of debris were then to enter the filter canister.

- Filter Canister—The filter canister captured fine debris on sintered metal filters (which were later precoated). The design and use of this canister in the defueling water cleanup system is described in Section 6.2.3.

Based in part on experience gained during waste shipments of SDS and EPICOR II vessels (see Section 6.5), radiolysis of water in the canisters was expected. Once the canisters reached INEL, they would be continuously vented, but during shipment a problem could exist. Drying the contents of each loaded canister would have been difficult, expensive, time-consuming, and, finally, unnecessary. Consequently, each canister was dewatered before shipment and the debris transported damp. However, special catalytic recombiners were built into each canister to control the accumulation of hydrogen (Reno and Schmitt 1988).

NRC, GPU, and Bechtel Quality Assurance audits of the canister vendor turned up substantial problems regarding the documentation of material used in fabrication. This led to serious delays in fabrication and a narrowly avoided impact on defueling. It also provided a lesson about the dangers of relying on one low-bid vendor for such important work. After a painful period of delays and uncertainty, the contract was withdrawn from the vendor and two new vendors produced satisfactory canisters.

Although the potential lack of an adequate number of canisters or a lack of canister types was a concern, defueling progress was never affected. The effect was an intense scrutiny of all canisters from the first vendor at every step of defueling, and, thus, a significant diversion of resources and attention.

Estimating the number of canisters to be used in defueling evolved into a high art form, and one with significant potential cost. The original estimate (and number of available slots at INEL) was 243—anything over 288 required use of the second half of the pool at INEL. The project's total estimate grew as defueling progressed because the weight loaded into each canister was usually less than anticipated. As important, the relative numbers of fuel, filter, and knockout canisters changed in response to the techniques used and unexpected conditions in the reactor vessel. The final estimate was for between 349 and 360 canisters. (In reality, approximately 340 canisters were used—approximately 270 fuel, 10 knockout, and 60 filter.)

8.4 Defueling System

The guiding principles in designing the defueling system were to keep it as simple as possible, design equipment and operations to ensure the lowest exposure dose to workers, and use existing plant equipment and procedures where possible, developing new tools when necessary. This entailed a large scope of work:

- Existing vital equipment that had been damaged had to be repaired to some extent (e.g., the containment polar crane in order to gain access to the reactor core).

- Fuel transfer equipment had to be modified or new means created to support the fuel removal and shipment strategy (e.g., a canister transfer system and transfer bells in order to move the canisters from the reactor vessel to storage in racks and then load them for shipment).

- New support components were needed (e.g., a rotatable shielded work platform over the vessel, a canister positioning system below the platform, smaller bridges/cranes, canister storage racks).

- New fuel removal tools/equipment had to be developed (as first designed, this meant long-handled tools and a vacuum system for use in dry defueling).

A primary motive for using existing plant equipment was not only to keep operations as familiar as possible, but to minimize radiation exposures by not requiring extensive installation of new components. The general approach was to provide pre-assembled equipment or pre-packaged modifications that could be easily installed. In-containment welding was avoided in order to minimize airborne contamination. Components that were too large to fit through a personnel airlock were designed in subassemblies that were assembled and tested before being disassembled and transported into the containment for efficient reassembly. This approach helped to minimize the time required to prepare for defueling (Rider 1986).

Figure 8-1 shows the plant with the complete fuel removal, transfer, storage, and cask loading system in place. The integrated nature of the system was vital to its success. Once installed, it was used with little change throughout the cleanup—the notable exception was the use of new fuel removal equipment in or above the reactor vessel as defueling progressed and unexpected conditions were encountered. For a more complete description of the initial defueling system, see *Data Report: TMI-2 Defueling Tools and Engineering* (BNAPC 1986).

Preparing the workforce required extensive training in the use of defueling equipment. For this, mockups were vital, especially 1/8-scale mockup of the reactor vessel constructed in the turbine building—the defueling test assembly (DTA). Crews were trained and all major equipment set up and tested in it to ensure that it would fit, work, and not cause interferences.

Directing the defueling operators also entailed special licensing considerations. In addition to the normal staffing requirements for operation of the plant, a senior reactor operator (SRO) was required. GPU then elected to obtain limited licenses for six fuel handling senior reactor operators (FHSROs) to directly control every core alteration activity being performed. Individuals were selected, trained, and licensed by the NRC between October 1984 and the start of defueling in October 1985 (Levin 1985).

A core alteration was defined as the movement or manipulation of any reactor component (including fuel) within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of a core alteration did not preclude completion of movement of a component to a safe, conservative position.

8.4.1 Plant Modifications to Support Defueling

A significant effort went into refurbishing existing equipment such as the containment polar crane and fuel transfer system.

New plant components were installed to support the work. Among these was the new containment air chiller system, which was installed on the air handling system to protect workers from heat stress during summer work hours. Experience had shown that summer temperatures could reach between 299 and 305 K in the containment, thus limiting working time to 1.5 hours per crew. The new chiller system reduced the temperature to 291 K and permitted worker stay times of three to four hours (Burton 1986).

8.4.1.1 Polar Crane Refurbishment

The polar crane was the single most important piece of existing recovery equipment and its refurbishment stood on the critical path to the start of fuel removal. Without a functioning crane, neither the missile shields above the reactor, the reactor vessel head, nor the upper reactor internals (plenum) assembly could be removed—all essential for access to the damaged core. Beyond that, the polar crane would be important for many support activities.

The importance of the crane had been recognized immediately after the accident. Although the details of its condition were not known before the first inspection in 1981, the following was known:

- It had not been subjected to any abusive loading, environmental, or operation conditions during construction or before the accident.
- Its location at a high elevation in the containment—just below the spray headers—had exposed it to some of the most extreme environmental conditions in the building during the accident, both short-term (e.g., the hydrogen burn and a 5-min. containment spray solution of sodium hydroxide and boric acid) and long-term (e.g., 1-1/2 years of continuous wetting from vapor evaporating from the basement).

Corrosion was expected to be the dominant mechanism of damage to mechanical components and the hydrogen burn the dominant mechanism of damage to the electrical components. Exposure to limited-duration temperatures greater than 800 K (1000 °F) were calculated, as well as exposure to tens to hundreds of thousands of rad. Adding to these challenges was the limited documentation on the actual, as-built configuration of the crane.

To direct the refurbishment, a Polar Crane Task Group was established in July 1982. It consisted of members from the cleanup staff and EPRI- and DOE-sponsored consultants from power plant crane manufacturers, who were also involved in an examination and inspection program to support GEND objectives.

The essential question faced by the Task Group related to the extent of refurbishment necessary. Although first recommended, complete refurbishment of all the subsystems would have been expensive and time consuming. Fortunately, the most essential components required for reactor vessel head lift had not sustained extensive damage. These were cleaned and some parts were replaced. If any more refurbishment was to be done, it could wait until decontamination had improved general conditions and airborne contamination was reduced. The following logic guided the refurbishment:

- The 456,000-kg design capacity of the main hoist and bridge system was three times greater than the maximum anticipated load of 150,000-kg for the reactor head and rigging.

- The 23,000-kg auxiliary hoist was of secondary significance and refurbishment would detract from the critical path. A lifting device suspended from the main hook could provide the need capability. (This decision was later changed when the value of the smaller, more utilitarian hoist for defueling support was recognized.)

- The operational features and load paths required could be specifically defined, thereby eliminating the need to provide excessively general operating characteristics; e.g., crane control during head lift would be by strict administrative procedures, the ability to rotate 360° about the polar rail system was not needed, and the crane would be controlled from the pendant—so the operator's cab on the crane itself would not be required (Graber and Lefkowitz 1984).

The refurbishment was nearing completion in the spring of 1983. The load test to qualify the polar crane for head lift was then delayed until early 1984 by allegations related to the applicability of QA procedures and a subsequent NRC investigation, which validated the safety of the refurbishment (see Section 2.5.3).

The load test consisted of removing the four 39,000-kg missile shields over the reactor vessel refueling canal and the 30,000-kg pressurizer missile shield. Although this required the construction of a stand to hold them, the use of in-containment components eliminated the need to bring other weights into a contaminated environment.

8.4.1.2 Fuel Transfer System

The fuel transfer system that normally moved fuel assemblies between the containment refueling canal and spent fuel pool "A" in the fuel handling building was suspect. It consisted of upenders, carriages, and two tubes that connected the buildings. The defueling canisters would be both heavier and larger than assemblies. In tests, the equipment had operated satisfactorily with no load and a dry canal, but poorly with a load and partially filled canal. This had to be corrected before the long-term operational requirements of defueling were imposed.

To upgrade the equipment so that only minimum maintenance would be required:

- Mounting and alignment problems were corrected.

- The carriage chain drive was replaced with a cable drive.

- Equipment (e.g., upenders) was adapted so as to handle the expected asymmetrically-loaded defueling canisters; the upenders and transfer winches were upgraded.

- The carriages were decontaminated, removed, and shipped to the original vendor for rework.

8.4.1.3 Bridges/Cranes

The two fuel handling bridges in the containment and fuel handling buildings were extensively modified —converting them into "canister" handling bridges. Shielded masts were installed on the bridges to allow the out-of-water transfer of loaded defueling canisters. Programmed indexing and interlocks provided safety assurance for handling fuel and a moveable shield collar eliminated radiation streaming at the bottom of the shielded mast during loading or discharging a canister.

In addition, a 4500-kg pendant-controlled service crane was installed to make handling the long-handled tools and other equipment easier. The rails for this bridge crane were installed on the top of the D-rings at El. 367' and provided access to the fuel transfer canal area. Rail extensions provided access to the adjacent El. 347' working area above the reactor vessel. Its installation enabled defueling operations to minimize use of the containment polar crane.

8.4.1.4 Containment Air Control Envelope

A temporary structure was installed outside of the equipment hatch to provide space for staging and packaging equipment and materials related to defueling. This containment air control envelope (CACE) was used instead of the structure originally planned for this location—the containment recovery service building (see Section 8.4.3.1). The CACE permitted equipment and materials to be moved into and out of the containment with a minimum of difficulty through the airlock doors of the equipment hatch.

8.4.2 New Equipment

Equipment specifically designed to support the dry defueling approach was developed.

8.4.2.1 Work Platform

A major component of the defueling system was a rotatable shielded work platform, installed above the reactor vessel. It provided structural support for two jib cranes, the canister positioning system suspended below it, and assorted support equipment.

It was rotated by an electrically powered cable drive assembly; contained a cable management system to tend service hoses and cables; and provided direct access to the reactor vessel via removable panels. A 45.7-cm wide tool slot running across its diameter was used first; a 61-cm wide T-slot running perpendicular to the tool slot was later added (see Figure 8-13).

In addition to the shielding, an important personnel protection feature of the platform was the use of a 4000-scfm filtration unit below the platform to offgas the reactor vessel. This ventilation unit created a negative airflow into the defueling workslot and prevented the airborne radioactivity generated under the platform from affecting personnel working on the platform.

A stationary work platform was considered. It would have been simpler to build and install, and not required any long-lead time components; however, a rotatable platform offered the advantages of more precise alignment of tools and equipment with the core below. Heavy loads could also be picked up in one radial core location and moved to another by rotating the platform (Austin 1984).

8.4.2.2 Tools/Equipment

The design of the initial defueling tools derived from the actions required during the projected "early" and "bulk" phases of defueling; i.e., gripping, lifting, shearing, digging, vacuuming, boring, hammering, chipping, splitting, snaring, coring, torquing, and cutting. Most of these actions were accomplished using a single-function tool:

- Long-handled tools—Heavy-duty tools were attached to the end of a long-handled pole. Hydraulic high-pressure lines, protected within the poles, were used to actuate the end effectors. Light-duty tools (some hydraulically operated) were mounted on aluminum conduit coupled together in sections. Table 8-3 summarizes the general types of long-handled tools available at the start of defueling (see photos 8-2, 8-3, and 8-4 for examples). Support equipment included debris buckets and funnels. The tools were supported by an overhead crane that also provided vertical and lateral motion.

- Vacuum system—The vacuum system consisted of a long-handled nozzle-handling tool, a knockout canister connect assembly module, and stainless steel piping to join the various components suspended beneath and supported by the shielded work platform. (Based on tests, the vacuum system, even with modifications, was not effective in debris removal, primarily as the result of excessive clogging. Various airlift systems were later designed to perform this function.)

- Bulk defueling end effectors—These were to be used on either long-handled tools or a manual tool positioner (which was a post to be supported beneath the shielded work platform and used to deploy the tools, including a remote manipulator). The initial end effectors included: an impact chisel to fracture or bore holes in fused material; an abrasive saw to cut fused material; and a cutting jet or hydrolaser to cut material or components with abrasive grit.

The first set of tools (in a "toolbox") was manufactured off site. Many modifications to the tools were needed because core conditions were not as the designers had thought. After the start of defueling, most long-handled tools were fabricated on site, increasing efficiency and applicability (because of the interaction of the machine shop staff with operators and engineers). The tools and equipment used as defueling progressed are described in Section 8.6, which discusses the actual operations.

The work itself was monitored via a camera and light system located within the vessel. All defueling operations were monitored on TV screens on the work platform and in a command center in the turbine building. The project organizational structures that supported defueling operations are discussed in Section 2.2.4.

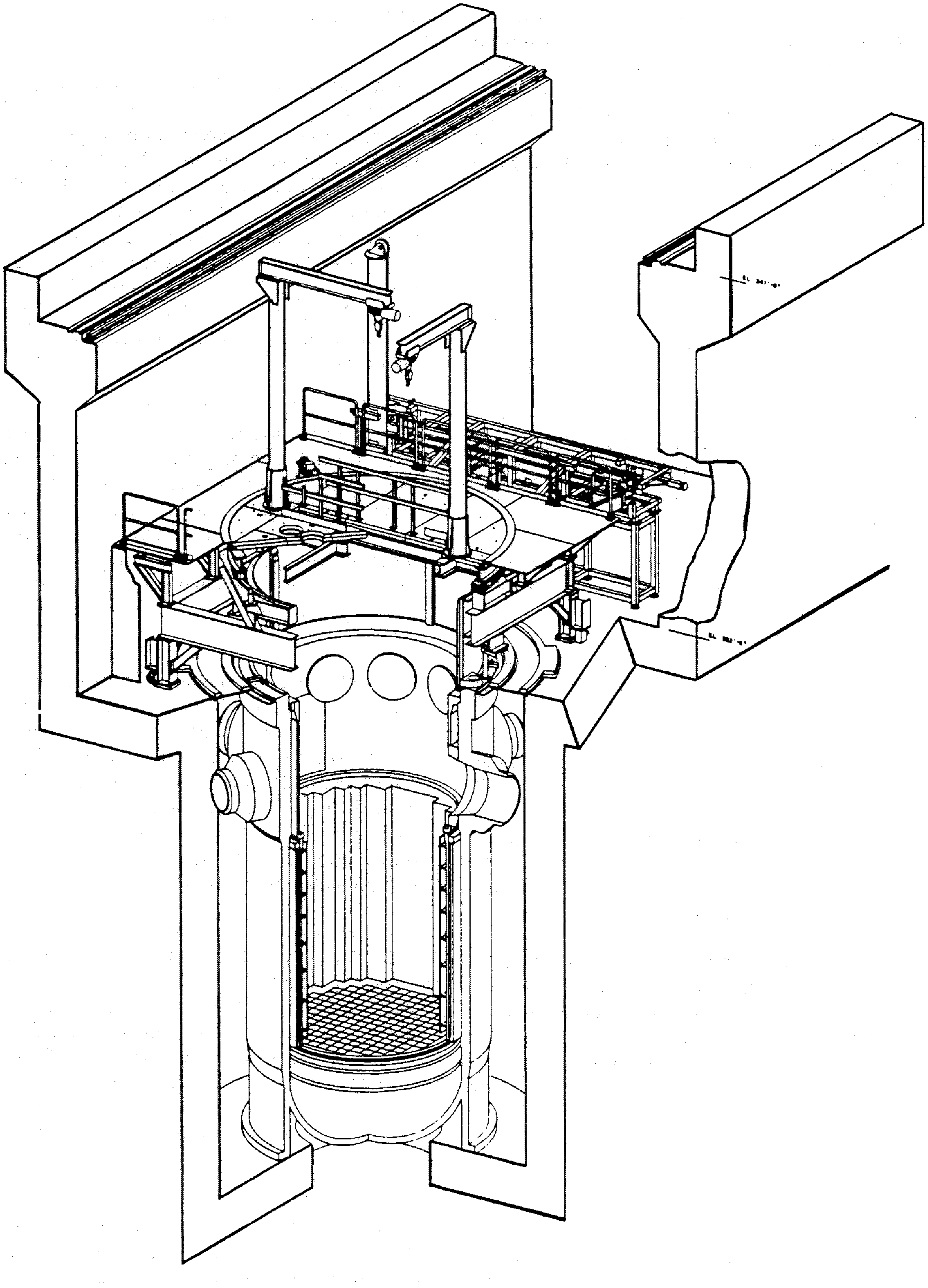


Figure 8-13. Reactor Vessel with work Platform

Table 8-3. Summary of Long-Handled Tools

Light-Duty	Heavy-Duty
Vise grips	End effector handling tool
Bolt cutters	Three-point gripper
Hook tools	Four-point gripper
Socket wrench	Grapple
Debris bucket handling tool	Single-rod shears
Partial fuel assembly tool	Parting wedge
Measuring probe	Heavy-duty tong tool
Light-duty tong tool	Spade bucket tool
End fitting loading tool	Clamshell tool
Banding tool	Heavy-duty shears

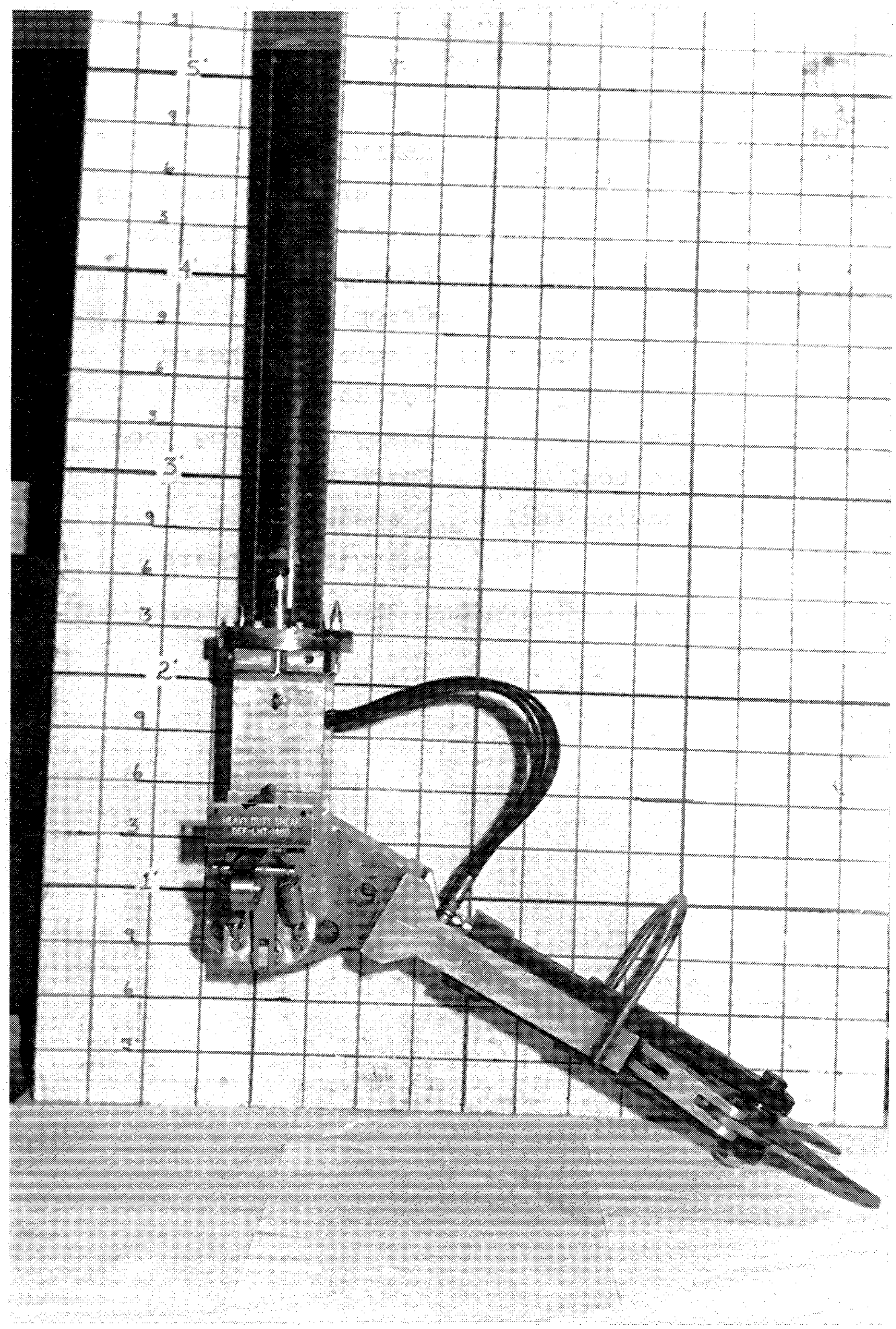


Photo 8-2. Heavy Duty Shears

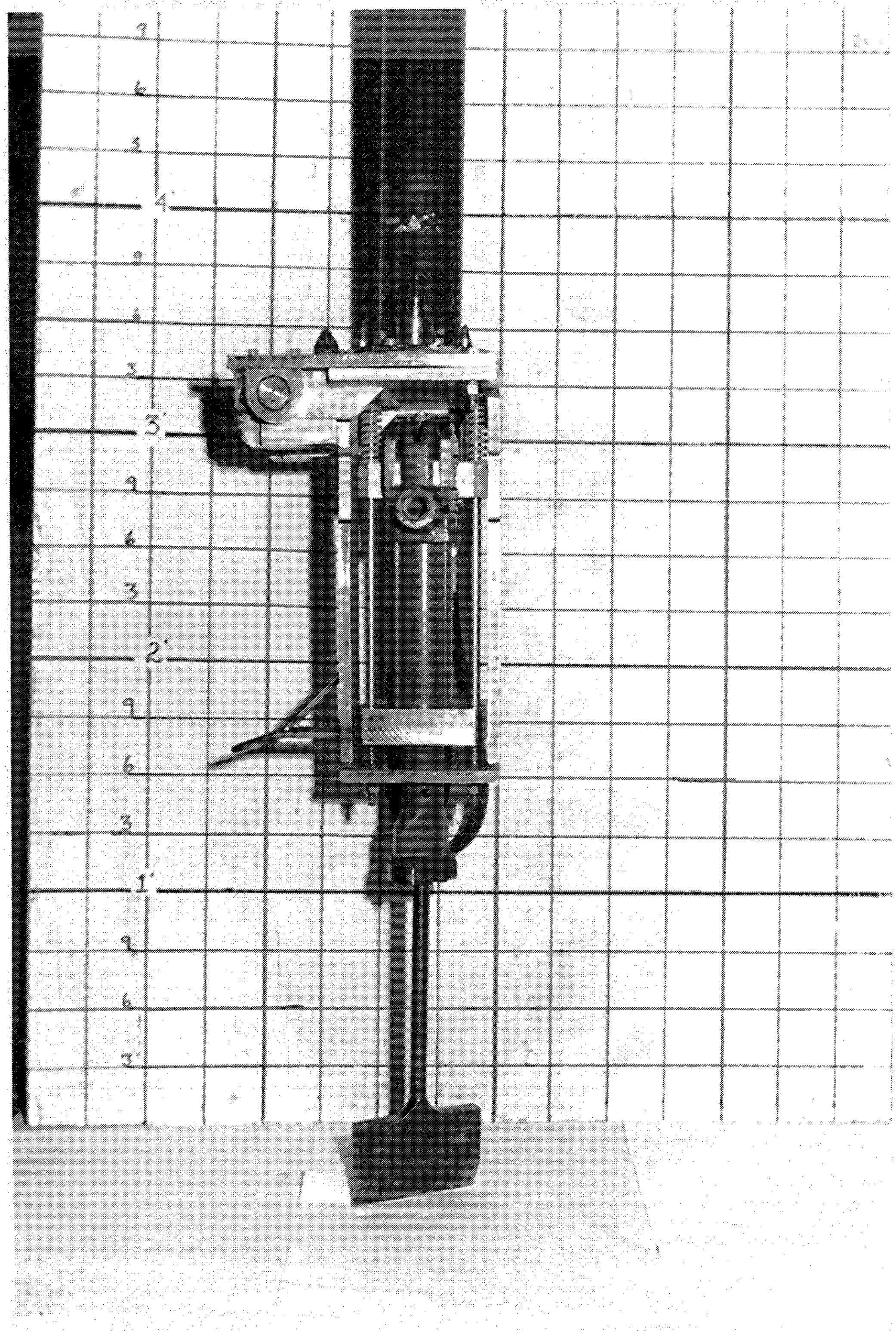


Photo 8-3. Chisel Tool

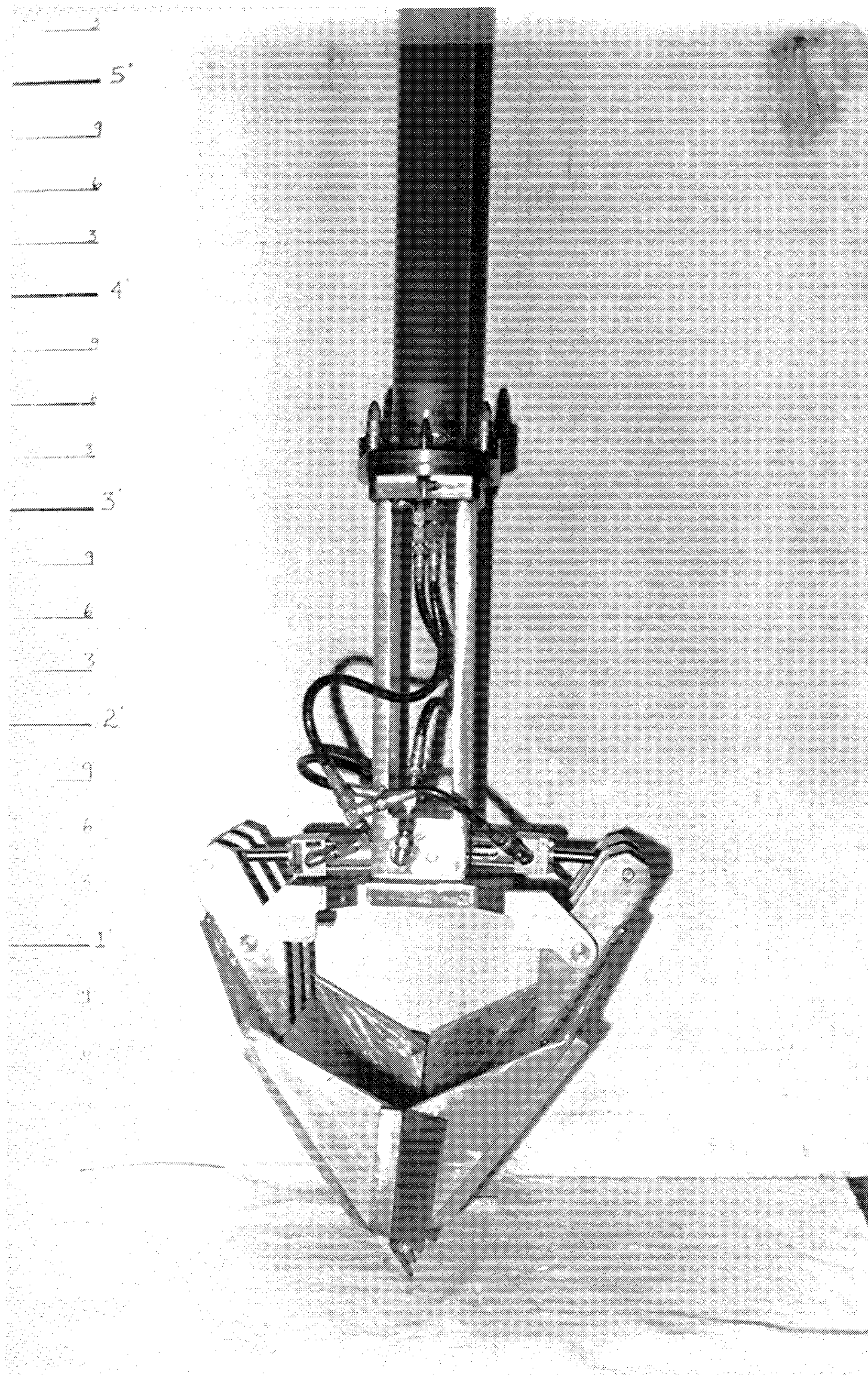


Photo 8-4. Spade Bucket Tool

8.4.2.3 Canister Storage Racks

An adequate surge capacity for storing loaded defueling canisters was important to permit defueling to proceed without concern about external factors affecting the ability to ship or the rate of shipment to INEL. Accordingly, storage space was provided for 252 canisters in spent fuel pool "A" in the fuel handling building and for 11 canisters in the dry end of the containment refueling canal.

The storage racks held canisters at a nominal 45-cm square pitch spacing. Criticality analyses had ensured that this was safe, as was a simple, low-cost rack in a open-lattice modular design. Four modules with an array of nine by seven cells were installed in fuel pool "A" (free-standing and not requiring modification to the pool).

8.4.2.4 Fuel Cask Loading

To prepare and load the defueling canisters in the fuel shipping cask, several new components were needed:

- Dewatering Station—Installed in the fuel handling building to provide the piping, hardware, and services to remove and filter the water from a canister, inert the canister atmosphere, and provide a two-atmosphere shipping pressure.

- Canister Transfer Cask—Transferred single, loaded canisters from storage to the shipping cask. The transfer cask was cylindrical, bottom-loaded, lead-shielded, and fully enclosed. Its 18,000-kg weight was handled by the fuel handling building crane.

8.4.3 Rejected Alternatives

Several facilities were considered but never built or were built in a different form. Their rejection was related to changes in program strategy and greater understanding of containment and reactor vessel conditions.

8.4.3.1 Containment Recovery Service Building

Early plans based on a full-scale decontamination effort preceding defueling had called for the construction of a large support building outside of the containment equipment hatch (BPC 1979). The support building was believed necessary because of the limited workspace available within the containment. The movement of materials and personnel was also inefficient because of the limited capability for using the equipment hatch area on a routine basis.

To meet these needs, a containment recovery service building (CRSB) was designed. The building was to have contained 930 m^2 for decontamination and contaminated tool storage, and an additional 470 m^2 for staging very contaminated equipment removed from containment. It was never built because of budget considerations and a change in project direction to one more focused on fuel removal.

A more temporary variation of this building was considered a few years later; this one was to adjoin the containment air control envelope outside the equipment hatch. This CRSB was still important, but its role was modified to supporting only reactor disassembly and defueling (Tarpinian 1984). It would have done this by:

- Facilitating personnel and material movement

- Providing staging space for equipment entering and leaving the containment (this might include the equipment hatch itself, which was potentially to be removed)

- Providing additional laydown space for contaminated equipment (by this time, the project management had decided not to remove any large equipment, with the possible exception of the reactor coolant pumps)

The decontamination and radwaste reduction functions originally conceived as part of its mission were transferred to the waste handling and packaging facility then being designed (see Section 6.4). Its original sludge and radioactive liquid processing/solidification functions were performed in the auxiliary building.

The second version of the CRSB was to have been a 12- by 34-m Butler-type building with a 9000-kg overhead crane and a truck bay. It would have included a 93-m^2 area for the temporary staging of contaminated equipment and tool storage and general work areas. It was never built because of funding considerations and the use of the CACE instead.

8.4.3.2 Equipment Hatch Removal

Access to the containment was restricted to the 2.7-m dia. personnel airlocks; consequently, the benefits of removing the 7.3-m dia. equipment hatch for improved access were evaluated.

In early 1985, as preparations for defueling gained speed, the project team sought to ease the movement of large defueling components into the containment (e.g., work platform, canister storage racks, service crane). Instead of having to disassemble some of these to move inside, removal of the hatch was proposed. This could have lead to: 1) fewer entries, 2) fewer in-containment activities, 3) improved efficiency, and 4) simplified equipment design—all with ALARA benefits. Also considered was removal of just the personnel airlock within the hatch (GPUN 1985).

The NRC found no reason to object to these plans; however, the project management never requested the final authority to remove the equipment hatch because the resources, time, and worker exposure involved in removing, handling, decontaminating, and storing the hatch would not be offset by improved access.

8.5 Reactor Vessel Preparation

To remove the 177 fuel assemblies from the TMI-2 reactor vessel, several barriers had to be removed: the concrete missile shields above the refueling canal; the vessel head (including upper service structure); and the upper internals (plenum). The plans to do so followed essentially the same course as during a normal refueling outage—the primary differences were the extraordinary amount of preparation and study that preceded the lifts. The reactor head was removed in July 1984, and the plenum in May 1985.

Qualifying the containment polar crane to lift these structures had been a time-consuming undertaking, but with its requalification in early 1984, the pace of the onsite work quickened.

A great deal of study accompanied reactor vessel head lift because of concerns about opening the reactor coolant system to the atmosphere. In retrospect, the operation was straightforward with no technically adverse side effects; but at the time, it consumed a large portion of managerial and engineering resources, along with considerable attention from the news media and public.

8.5.1 Head Removal

In the spring of 1982, a task group was formed to study the options for reactor vessel head removal (GPUN 1982b). The resulting report recommended the basic sequence to be used based on the following criteria:

- Minimize impact on existing and future radiological conditions
- Use existing equipment and techniques when practical
- Ensure adequate decay heat removal and reactor coolant cleanup
- Ensure health and safety practices
- Minimize cost and schedule.

The task group recommended the normal head removal operation be pursued, supplemented by contamination control and radiation protection in view of the little known nature of damage in the vessel and potential radiation problems. As in a normal operation, the refueling canal was to remain dry while the head was transferred to the existing head storage stand on El. 347' of the containment, where shielding would surround it. A boot would be placed under it for contamination control during transfer.

The existing internals indexing fixture (a 1.8-m high collar used to guide components into and out of the vessel) would be modified to fit above the reactor vessel flange, filled with water to provide additional shielding, and capped until the plenum assembly and then the defueling work platform could be installed.

Two aspects of the preparations for head removal were especially important:

- Mockups—Every aspect of the operation was mocked up and trained for—a practice that continued throughout the cleanup before every major evolution in work.
- Readiness Review Committee—As with previous major operations, an intensive committee review of plans, potential problems, and preparedness was conducted by managers, outside consultants, and engineers. This practice also continued throughout the cleanup.

The RCS would not be repressurized after head removal because it would be both too expensive and unnecessary from a safety standpoint. Preparations were made to clean up or flood the refueling canal in the event of unexpected circumstances.

The alternative of a "wet" head lift (i.e., with flooded refueling canal) was evaluated and rejected:

- Advantages—Since defueling plans at the time envisioned a flooded refueling canal, the extensive work needed to flood the canal would be required later anyhow. In addition, a flooded canal (or the installation of a tall shield tank above the reactor vessel) would have provided a guarantee of radiation control.

- Disadvantages—These were persuasive: 1) the potential of severe contamination of the canal or airborne contamination; 2) the resulting limited access to the area; and 3) the fact that the water would have to be drained to prepare for defueling.

Opening the reactor coolant system for the first time required analyses and procedural precautions for issues such as criticality safety, boron dilution of the coolant, releases of krypton, control of hydrogen gas generation, evaluation of the potential for pyrophoric reaction, heavy load handling, and worker dose.

An underhead characterization program, begun in December 1982, revealed that radiation levels would be higher than predicted and control of airborne radioactivity less of a problem than expected. A rapid increase in release of dissolved radioactivity occurred when the system was opened for the underhead characterization program and the reactor coolant became saturated with air. This led to some revisions in equipment and installation sequence but did not affect the basic plan:

- Lead shielding was installed on the reactor vessel head service structure

- Water-filled columns were placed around the head storage stand. (Some of these leaked and the water was replaced with sand—a more effective shielding agent.)

A shielded work station from which to direct operations was constructed on the pressurizer missile shield on top of the "A" D-ring. A misting system was also installed on the refueling canal bridge rails to wet the exposed plenum surface inside the vessel until the internal indexing fixture was seated and filled with water. Finally, a diaper was walked under the reactor head before transfer in order to control contamination. Safety contingency plans for reinstalling the head during all modes of defueling operations were also developed (Jones 1984).

Figure 8-14 and Photo 8-5 show removal of the reactor vessel head and service structure, which took place between July 24–27, 1984. One failure of the polar crane caused temporary difficulties in seating the head on its stand. After the operation, installation of the internals indexing fixture with 1.5 m of water over the plenum effectively reduced exposure rates to rates existing before the head lift (GPUN 1985b).

8.5.2 Upper Internals Removal

Using GEND-007 as a basis, plans for removing the upper internals (plenum) were underway in 1982 (GPUN 1982a). The hypothesized conditions included some debris on the plenum cover plate and internals, some minimal distortion, and some core components fused to the upper core support plate.

The following general criteria and functional requirements were applied:

- The plenum was to be removed in one piece if possible (but without plans for reuse). The techniques for removing the plenum were to be progressively more severe until successful; i.e., first use the polar crane alone; then use jacks to raise the plenum, perhaps supplemented by the crane; then perform partial destructive removal by severing distorted and binding ribs; and, finally, perform full destructive removal with, probably, a plasma arc torch.

- Damage to the reactor vessel was not permitted. Future defueling operations were not to be adversely affected.

- As in a normal refueling operation, the canal was to be flooded. A contamination barrier would prevent extensive contamination of canal water by reactor vessel water. Loose debris from the underside of the plenum was to be knocked off and the plenum bagged to reduce recontamination either during transfer or storage, which was then planned to be in the shallow end of the refueling canal.

With the change in defueling plans to a "dry" scenario, plenum removal plans were changed to storing it in the flooded deep (north) end of the refueling canal, which would be separated from the rest of the refueling canal by a 1.2-m high dam. (Storing the plenum in a shielded tank in the dry shallow end of the refueling canal was also briefly considered.) Reflooding the entire canal remained a possibility if radiological conditions so dictated (GPUN 1984).

The plan to bag the plenum by placing it in an open, gathered sack was discarded. The plan had been the result of concern that the stored plenum would contaminate the refueling canal water. This potential was con-

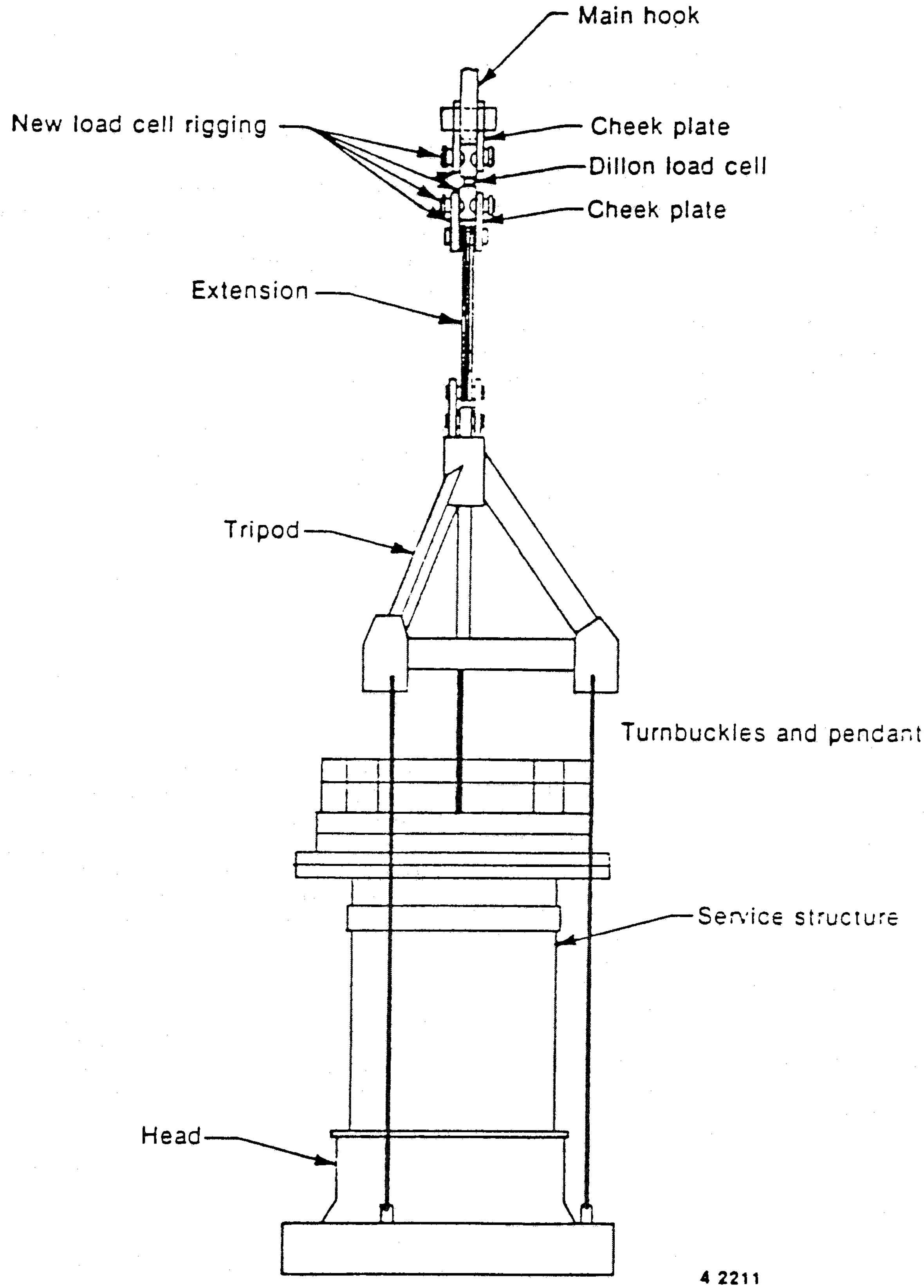


Figure 8-14. Reactor Vessel Head and Service Structure with Rigging

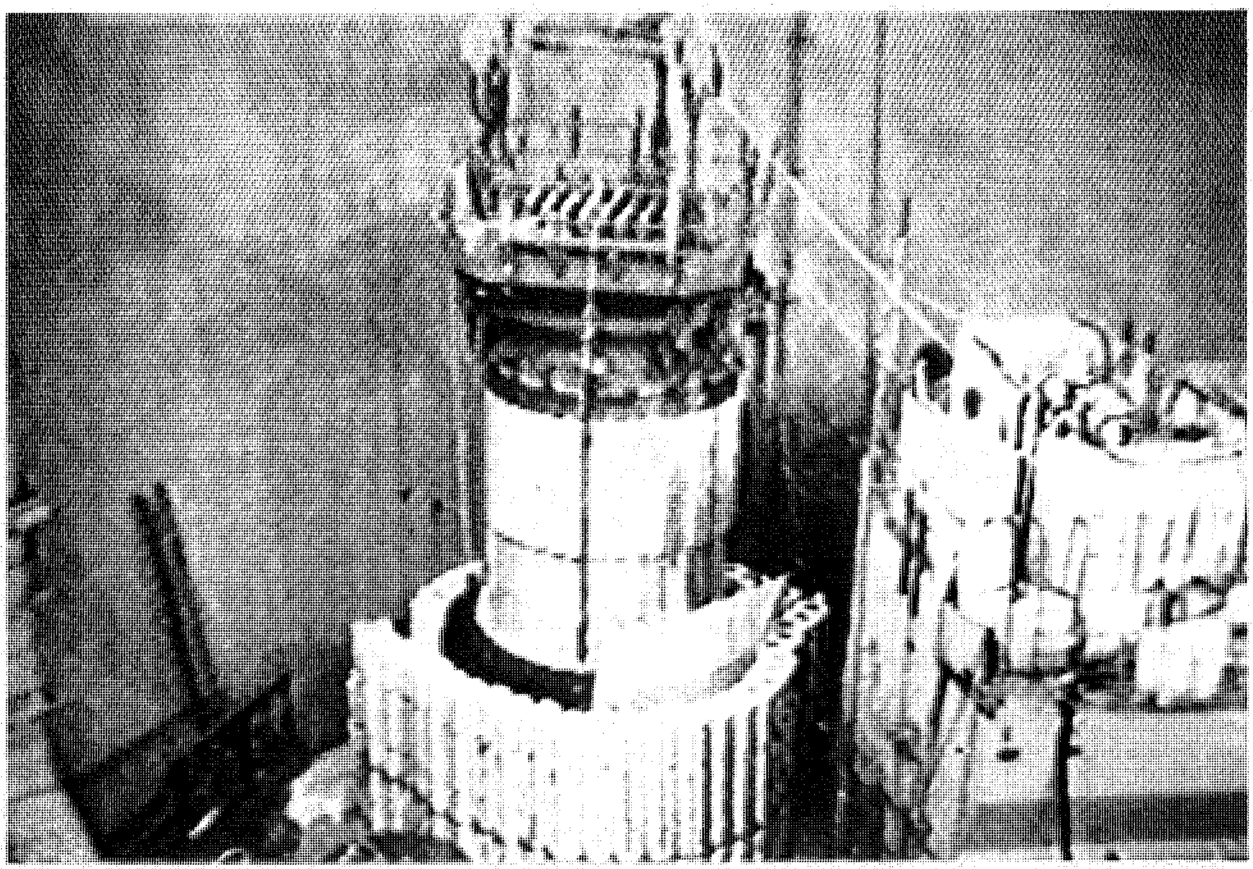


Photo 8-5. Reactor Vessel Head Being Lowered onto Storage Stand

trolled by the ability of the defueling water cleanup system to process the water. Although inexpensive, a soft, drawstring container would have been difficult to install; installation of a hard container would have been expensive, difficult, and entail high worker exposure.

With the head removed, a visual inspection of the plenum revealed that upper end fittings and partial fuel assemblies were adhering to the underside of the structure. In addition, analysis indicated that the plenum was slightly distorted—perhaps in an oval shape—and might not come free during a lift. Consequently, in December 1984, it was jacked up 17.8 cm from its seated position to ensure it would come free. The adhering upper end fittings were then knocked from the underside.

During the final lift in May 1985, extensive precautions were taken to ensure that the plenum did not hang up on any of over 120 potential interference points. It was then successfully lifted from the vessel and stored underwater in the deep end of the refueling canal on a special storage stand (Wilson 1986). Photo 8-6 shows the plenum being removed from the reactor vessel.

8.6 Defueling Operations

The course of defueling operations ran far longer and proved more difficult than predicted. Figure 8-15 shows the general course of defueling in terms of weight of core material removed and types of operation. Photo 8-7 shows a crew of defueling operators on the shielded work platform above the reactor vessel—the hands-on manifestation of a 24-hour-a-day project effort.

Defueling operations usually took the path of least resistance, adapting to unexpected conditions that were frequently very resistant. Productivity was a constant, dominating impetus—how to complete the work as safely and quickly as possible. This sometimes led to choosing equipment or methods that were at hand or familiar in order to begin a new stage of defueling rather than spending the time and resources to develop an untried but potentially more effective technique.

A frustrating learning curve for operators and engineers accompanied each new core region or type of condition that was encountered. An additional frustration to both management and operators was the inability to perform more than one operation at a time from the platform. In spite of constant analyses to find ways of conducting two operations (e.g., vacuuming and plasma arc cutting), space constraints on the platform and in the vessel usually prevented such parallel activities. If space constraints were not the factor, then the side effects of one operation would adversely affect the other (e.g., cause temporary loss of visibility or stir up fines that would foul delicate equipment or obscure visibility).

The known reactor vessel core conditions at various stages of defueling are portrayed in several figures in Section 5.4. Appendix B describes the material composition of each of the major reactor vessel regions containing core debris.

8.6.1 Core Region

This region posed the most immediate and obvious challenge. And it only revealed its true nature during the course of ongoing defueling and data acquisition efforts.

8.6.1.1 Loose Debris Removal

Defueling operations officially began October 30, 1985. The first few weeks of defueling were spent repositioning core debris such as end fittings, burnable poison control rod spider hubs, and partial fuel assemblies—a necessary step to allow the canister positioning system to lower one canister for loading. This work was performed very cautiously while radiological conditions were closely monitored. (By the end of the year, the three- to four-person defueling crews had been increased from two teams per day to three, and to four by the end of January 1986.)

More room to rotate the canister positioning system was then created by cutting and loading the larger pieces of debris. This debris was often too damaged or distorted to fit into fuel canisters and so a chisel driven by a 44-kg slide hammer was devised to separate fused end fittings. Distorted end fittings that could not be fit into canisters were stored in special 200-L drums on El. 347'.

Operations were proceeding pretty much as anticipated when the first of many unexpected turns of events took place: visibility was lost. A combination of microbial growth and fine particulate material clogged the sintered metal filters of the defueling water cleanup system and reduced visibility within the vessel to centimeters. This loss of visibility was to hamper operations from early 1986 until it was restored consistently in February 1987. Section 6.2.3 describes this challenge in detail.

The loss of visibility did not immediately affect the rate of defueling because blind defueling was possible; i.e., a sufficient number of end fittings had been removed from

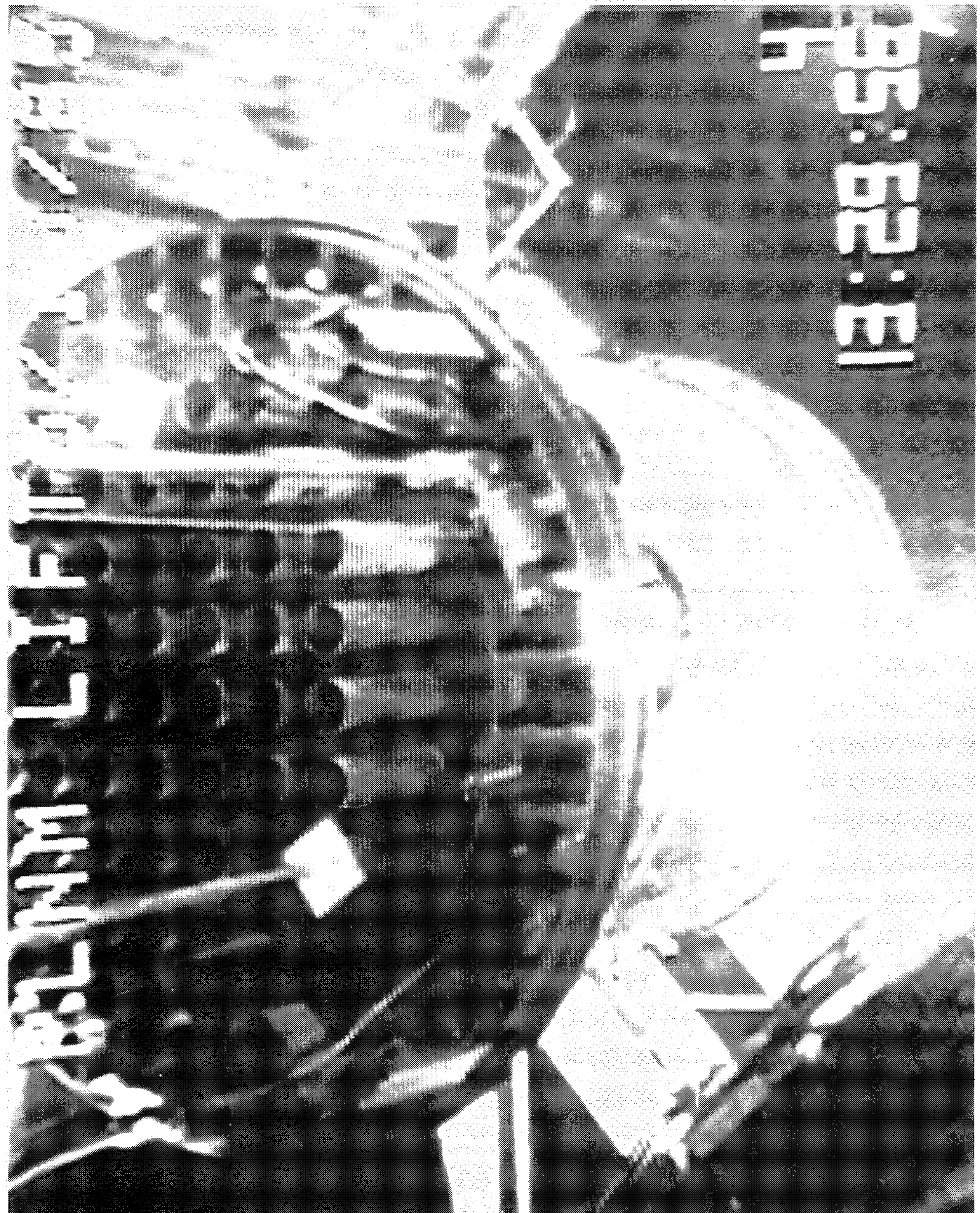


Photo 8-6. Plenum Removal

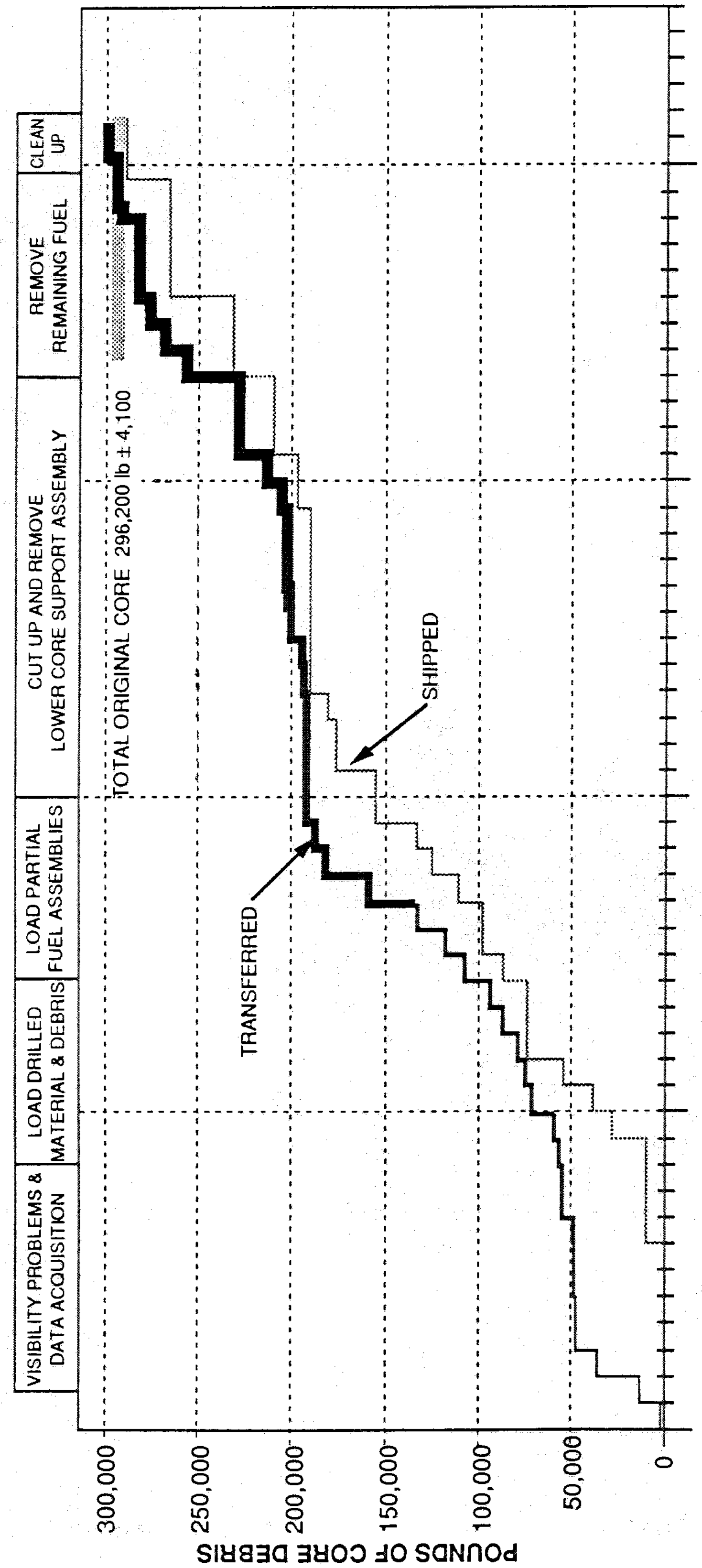


Figure 8-15. TMI-2 Defueling Progress

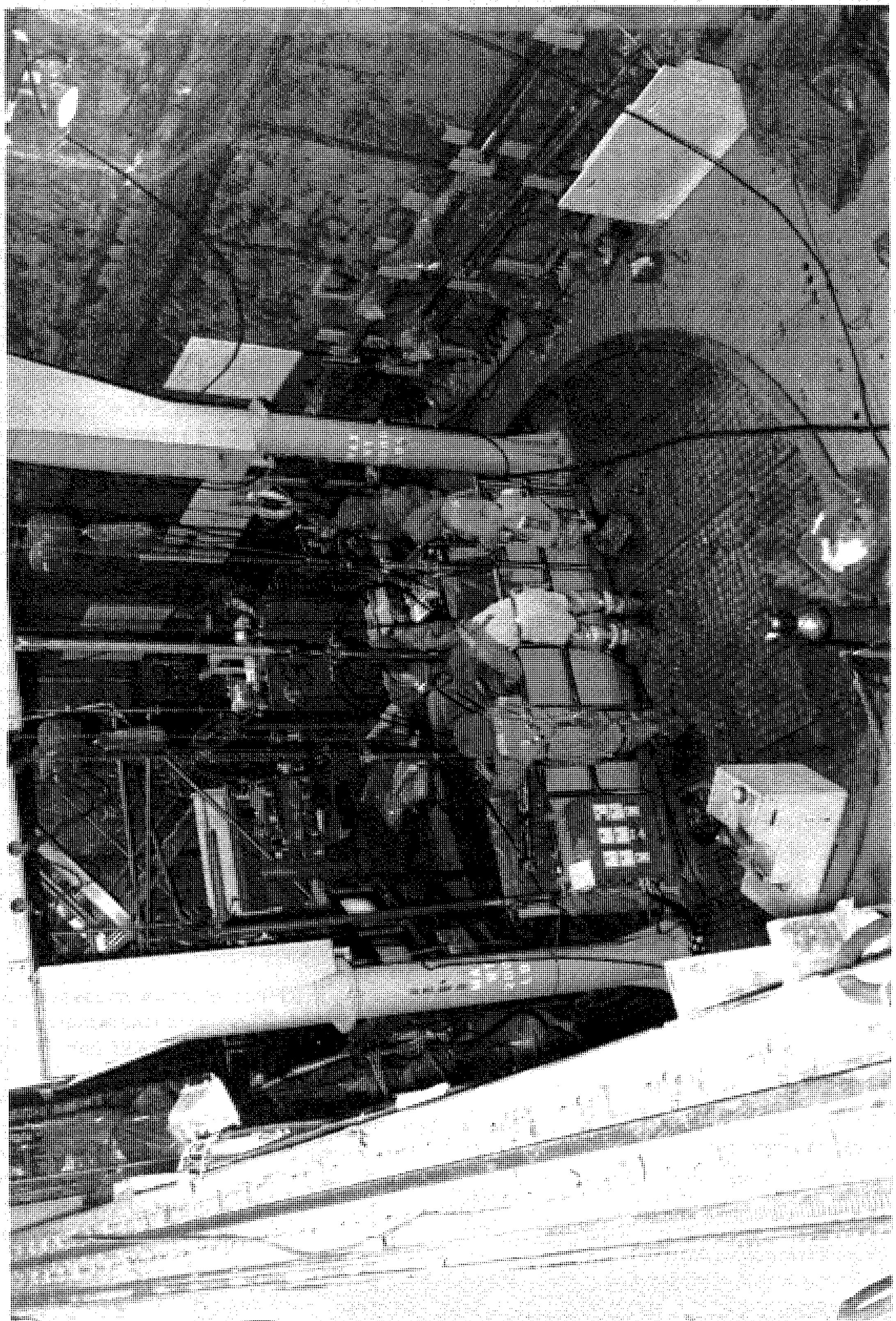


Photo 8-7. Operators on the Defueling Work Platform

the top of the debris bed to permit defueling to continue with a spade bucket tool. For three months, rubble-like loose fuel debris was loaded into canisters at one of the fastest rates ever achieved during defueling.

In April 1986, fuel removal essentially stopped as the tools encountered a hard crust 15 to 30 cm below the original rubble bed surface. Previous probing had established the existence of this crust and started a reevaluation of "bulk" defueling capabilities. Reaching the crust coincided with the beginning of a six-week operation to eliminate the microorganisms (the "bug kill"). The effects of this operation increased visibility to better than 30 cm for a few months before it again declined.

New thinking about tools included new core cutting equipment, hard crust defueling equipment, and incore instrument removal equipment. Also, this crust indicated the need for tools to cut and remove ductile material, which, unlike the brittle material seen to date, required more rigid and precise cutting techniques (BNAPC 1986).

Existing heavy-duty tools and the hydraulic impact chisel were used on the crust. None of the intended tools were strong enough or could deliver sufficient energy to break it up. Fortunately, one did exist.

8.6.1.2 Drilling the Solidified Mass

The solution was the equipment brought on site to take core samples for the data acquisition effort described in Section 5.4.3. The core boring machine (a small commercial drilling rig) was the only equipment on site that could deliver substantial energy to the core debris mass. The potential need to deliver such energy to the core for defueling had been a significant consideration in the development of the machine (see Figure 8-16 and Photo 8-8).

The core data acquisition program had taken place in July 1986, with 10 drill core samples taken from the core region and two from the lower core support assembly. Even though visibility was poor, it was clear enough so that when cameras were lowered into the resulting holes, the results were disheartening. Until that time, a void had been assumed to exist within the solidified material that formed the hard crust—planning had not addressed the possibility of a relatively large monolith. The solidified material existed in a funnel-shaped disc approximately 1.2 m thick at the center of the core and 0.3 to 0.6 meters thick at the periphery. Below that were the stub end remains of fuel assemblies.

The choices for proceeding were:

1. Continue with heavy-duty tools—The hydraulic impact chisel, crust conditioning tool, heavy-duty parting wedge, and 135-kg crust impact tool were all tried over a four-week period—unsuccessfully.

2. Reinstall the core boring machine—It was fitted with solid-faced drill bits and, in October 1986, 409 overlapping holes were drilled to rubbilize the center of the solidified mass (referred to as the "swiss cheesing" operation).

With the mass broken up, defueling resumed in murky water. However, the long-handled tools were still incapable of loading significant amounts of debris. The mystery of this was resolved when some degree of visibility was regained using a temporary reactor vessel filtration system. A video inspection revealed that, although the center mass of solidified material was rubbilized, a remaining peripheral ledge (the "donut") around the drilled area had partially broken up and pieces from it had fallen into the center.

These pieces ("rocks") were extremely hard; some weighed 1200 kg and were over 30 cm in diameter (Rodabaugh 1989). The existing long-handled tools were often destroyed in trying to manipulate the pieces. No obvious solution presented itself because impact forces applied to the rocks were absorbed by the cushioning of the debris bed below.

Several approaches were considered:

- New tools were evaluated; e.g., abrasive water jet; pulsating/cavitating water jet; air chisels; heavy-duty crushing tool; core debris digger; hydraulic clamp lifter; and a modified crust impact tool.

- A shredding machine that was on site as a contingency was reevaluated. (This was not the same one included in the WCSD proposal.) It would have hung from under the work platform; the processed debris would have fallen onto the bed below. It was rejected because of substantial drawbacks: 1) it was never intended to work on rocks; 2) it would have required substantial modifications and maintenance; 3) ALARA and criticality concerns were still major considerations; and 4) approximately three months of testing would be required to determine its feasibility.

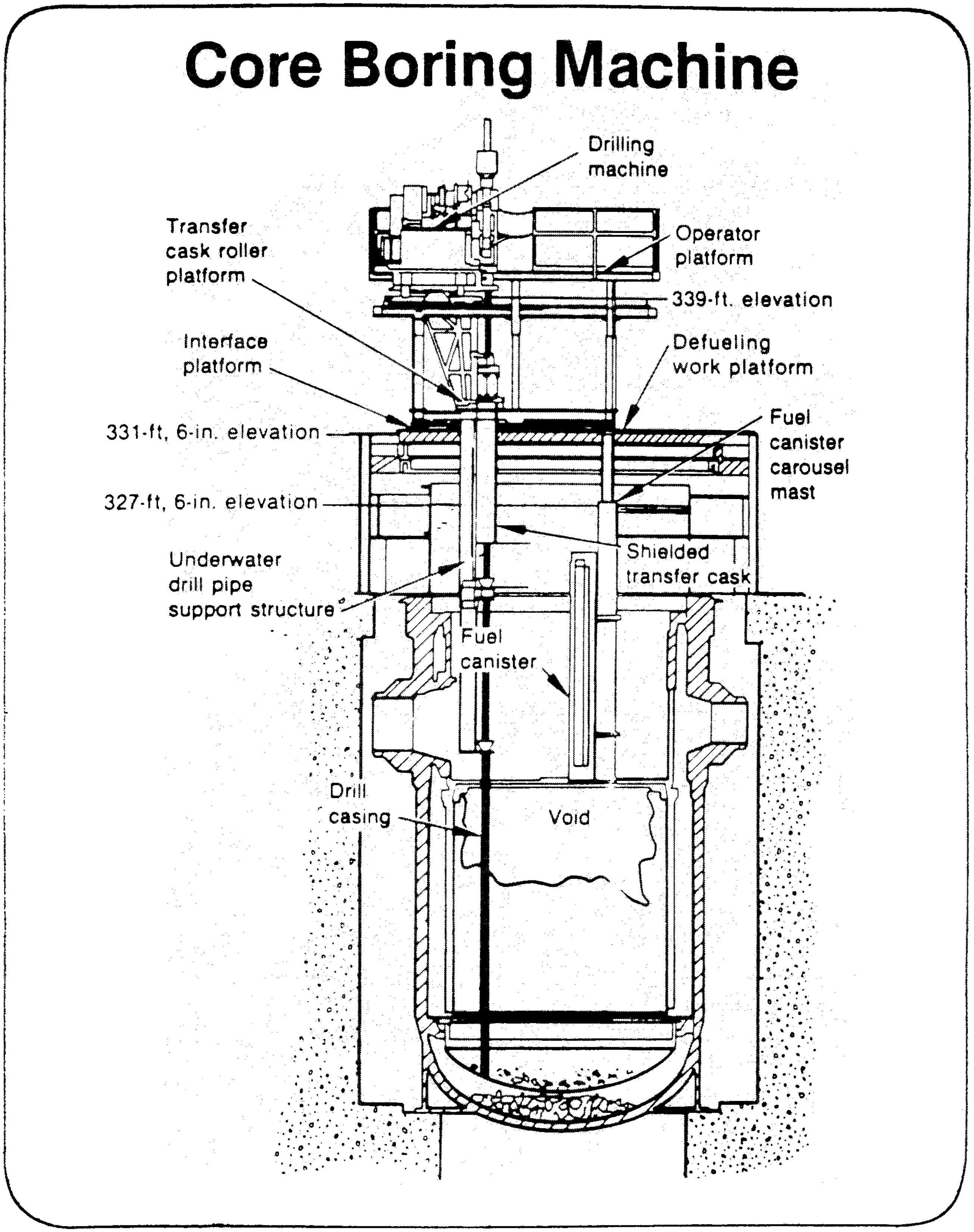


Figure 8-16. Core Boring Machine

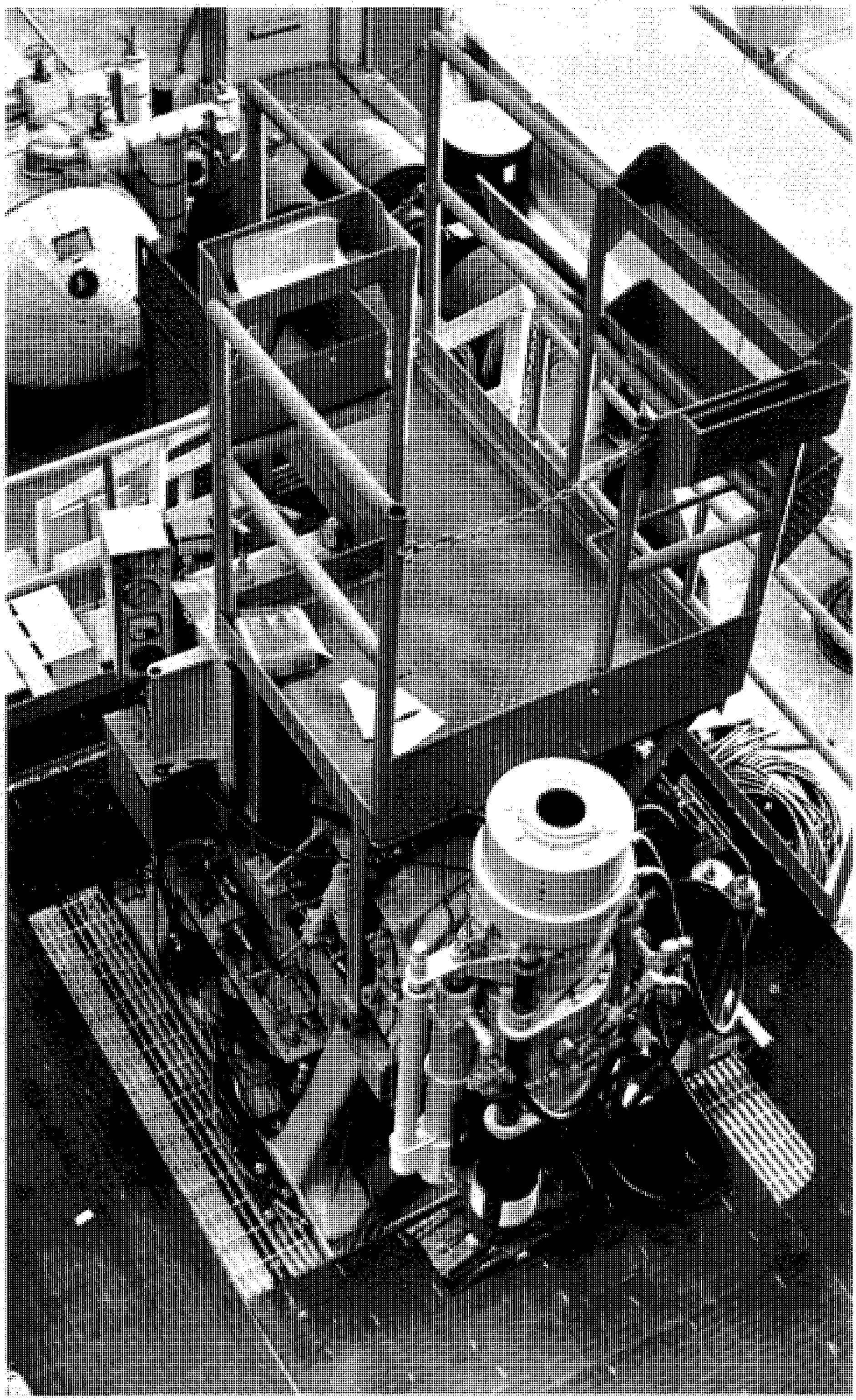


Photo 8-8. Core Boring Machine in Turbine Building

- A method of working around the rocks was considered as a near–term solution to support the schedule. This would have involved the time consuming process of handling the rocks several times and locating a suitable area within the core region to store them (Potts 1987).

In the end, new tooling was fabricated:

- An impact funnel with 2.5-cm thick walls was placed on top of a canister to hold a rock.

- A pneumatic 198-kg jackhammer equipped with a special 3.6-m long chisel bit broke the rock.

- An airlift vacuum, similar to that routinely used for underwater salvage, suctioned up debris as large as 5 cm. Its use helped to increase packing efficiency in canisters already containing oddly shaped pieces of debris (e.g., partial fuel assemblies, end fittings). Several versions of the airlift were eventually used; the basic one had a 10-cm dia. pipe with an air injection nozzle near the bottom. Although very effective, it did cause complete loss of visibility while in use.

After the rocks had been reduced in size, the partial fuel assemblies still standing around the periphery of the core were severed at the debris bed level and sized with shears (if necessary) to fit into fuel canisters.

This period of defueling (late 1986 to mid-1987) saw water clarity regained and considerable progress made.

8.6.1.3 Stub Assembly Removal

As the core drilling data acquisition effort had revealed, "stub assemblies" (partial but mostly intact lengths of fuel assemblies) lay beneath the solidified mass. Of the 177 fuel assemblies, the longest were at the periphery; the shortest at the center. Molten material had flowed among some and fused them together.

The choices for removing these stub assemblies were:

1. Long-handled Tools—Since the original plans had assumed that the fuel assemblies would be removed by long-handled tools, this seemed to be the preferred course—even though the assemblies were considerably shorter than imagined in 1984. Long-handled tools were made to work; eventually, they included the "harpoon" tool (hooked end fittings), single-rod puller, "unicorn" (screwed in from top), side-access lift tool, multiple-fuel element removal clamshell, lower end fitting jack, and the fuel stub end lean-over tool (to tilt an assembly away from others). An ultra-heavy-duty clamshell tool was also considered. This tool could potentially have had other uses, but its development was judged to be unnecessarily burdensome in cost and time.

2. Core Boring Machine—If long-handled tools were unsuccessful, the machine could be used to rubbilize most of the remaining core. However, the effect of drilling apart the rods was unpredictable and possibly undesirable; e.g., a twisted mass of rods that could not fit in a defueling canister might result (Potts 1987).

Removing the stub ends began with good fortune on the western edge of the core, where the partially protruding tops of two stub assemblies could be grappled and the assemblies pulled out. The resulting "beachhead" to the bottom of the core region was not wide enough to permit other tools to be manipulated in it. Consequently, the clam shell tool was used to expand the beachhead by essentially chewing up some adjoining stub assemblies until sufficient room to move was established.

The remaining stub assemblies were lifted out from the bottom. One tool was eventually developed to lift an assembly from the bottom with two fingers after a long spike had been driven into the top. The assembly was then transferred to a loading tool that placed it in a fuel canister.

The pattern of stub assembly removal over eight months of operation was basically catch-as-catch-can; i.e., although several logical removal patterns were analyzed, the defueling operators found that removing the next target of opportunity was more efficient and achieved the goal. By late November 1987, all but two stub fuel assemblies had been removed (R-6 and R-7). These two were so damaged that they were basically amorphous masses whose removal would be difficult and better deferred until the aggressive defueling of the lower components of the reactor vessel.

A side effect of removing the stub assemblies was to have a major negative impact on defueling the regions below. Between 9000 and 18,000 kg of debris in the core region fell from the top of and between the stub assemblies into the lower core support assembly and lower head, where removal was to prove far more difficult than if that debris had been removed while still in the core region. Plugs to fill the holes into the lower core support assembly were tested; however, their installation and maintenance in an area of such great activity was considered difficult and burdensome. Photo 8-9 shows the reactor vessel core region (including the baffle plates) after the stub assemblies had been removed.

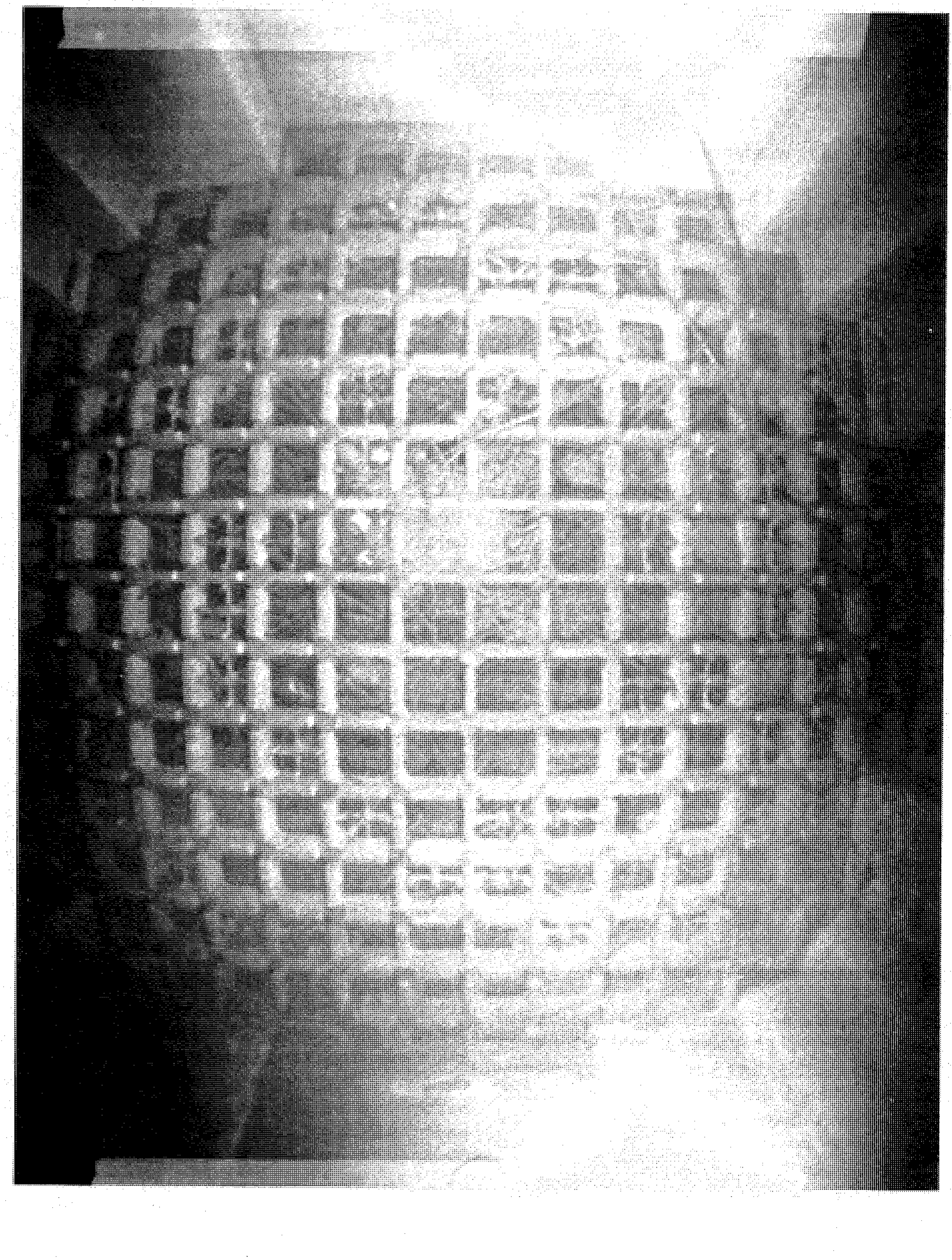


Photo 8-9. Reactor Core Region Looking Down from the Work Platform to the Lower Grid Rib Assembly

8.6.2 Lower Core Support Assembly Region

Defueling the lower core support assembly (LCSA) required new tools and equipment and took well over one year. The time was required, in large part, because of the difficulty in disassembling—under 12 m of water—a five-plate, 0.9-m deep, 3-m dia., stainless steel structure that was not meant to be disassembled from the top. In addition, the large quantity of debris that had relocated there during defueling operations further hindered disassembly. The operation required a new evolution of planning, which eventually rivaled that required to defuel the core region of the vessel.

The LCSA was a massive, complex structure designed to support, position, and provide coolant flow for the reactor core. Figure 8-17 depicts the LCSA, along with the plasma arc equipment eventually chosen to dismantle it. The LCSA contained 48 support posts, 52 incore guide tubes, incore guide spiders, and five lower head inspection holes.

When planning first began, the LCSA also contained a little-understood quantity and type of fuel debris.

The first comprehensive plan for defueling the LCSA (Barrett 1984) had been based on the assumptions of GEND-007 and had assumed that any fuel in the LCSA would be loose debris, with the largest pieces the size of fuel pellets. Three primary alternatives for defueling the entire core support assembly were evaluated:

- In-vessel cleaning/storage—This was the preferred approach, with suction and flushing tools the primary method of defueling. A plasma arc torch was to be used for the limited cutting envisioned. Preference was based on cost-effectiveness and ALARA considerations regarding any ex-vessel transfer operation —the major concern was cobalt-60, a neutron activation product in the stainless steel.

- In-vessel cleaning/ex-vessel storage of large pieces —This alternative had the advantage of making the defueling of the lower head very straightforward. After cleaning, the plan required some in-vessel cutting, removal of major structural components, and storage in the deep end of the refueling canal. (The plan was reexamined in 1986, too, but rejected again because of the complexity of lifting and handling; ALARA concerns; and, by that time, very limited storage space).

- In-vessel cutting/ex-vessel storage of small pieces —Little cleaning but extensive cutting was required, with the relatively small pieces stored in the deep end of the refueling canal. The extent of cutting required was a dissuading factor at the time.

Shortly after the first plan was issued, new data required a great deal of rethinking. Video inspections of the lower reactor vessel head disclosed the existence of a large amount of once-molten solidified core material. Although the physical nature of this material was not known, plans were revised to reflect minimum and maximum design bases (Ales 1985).

The general methods for defueling the LCSA were presumed to be the same as previously considered, only the scope would differ (Porter 1989). The maximum design bases anticipated some areas of extensive structural damage, a ductile matrix of once-molten steel in some of the larger pieces of debris, and up to 9000 kg of core debris in the LCSA.

Eventually, a layer-by-layer defueling approach evolved; it used a tool versus operational matrix for addressing the expected conditions. This approach required that questions constantly be asked:

- Is more defueling necessary?
- Is more inspection necessary?
- Is this the optimum time to do the required work?
- Are the correct tools available? (Porter 1985)

Using this approach, small loose debris would first be removed through available access routes; larger debris broken up with various tools. Adherent debris on accessible structures was also to be removed. The structure would be cut for access where debris could not be removed or broken up. This process would proceed from the top plate to the bottom, with the objective of minimizing cuts (which would be time consuming) and maximizing cleaning. Numerous alternative cutting techniques were investigated:

- Thermic Rod—The preferred method. Excellent for cutting stainless steel components and ductile debris material because of its relative ease of operation and its effectiveness in cutting materials with melting points under 5800 K (10,000°F).

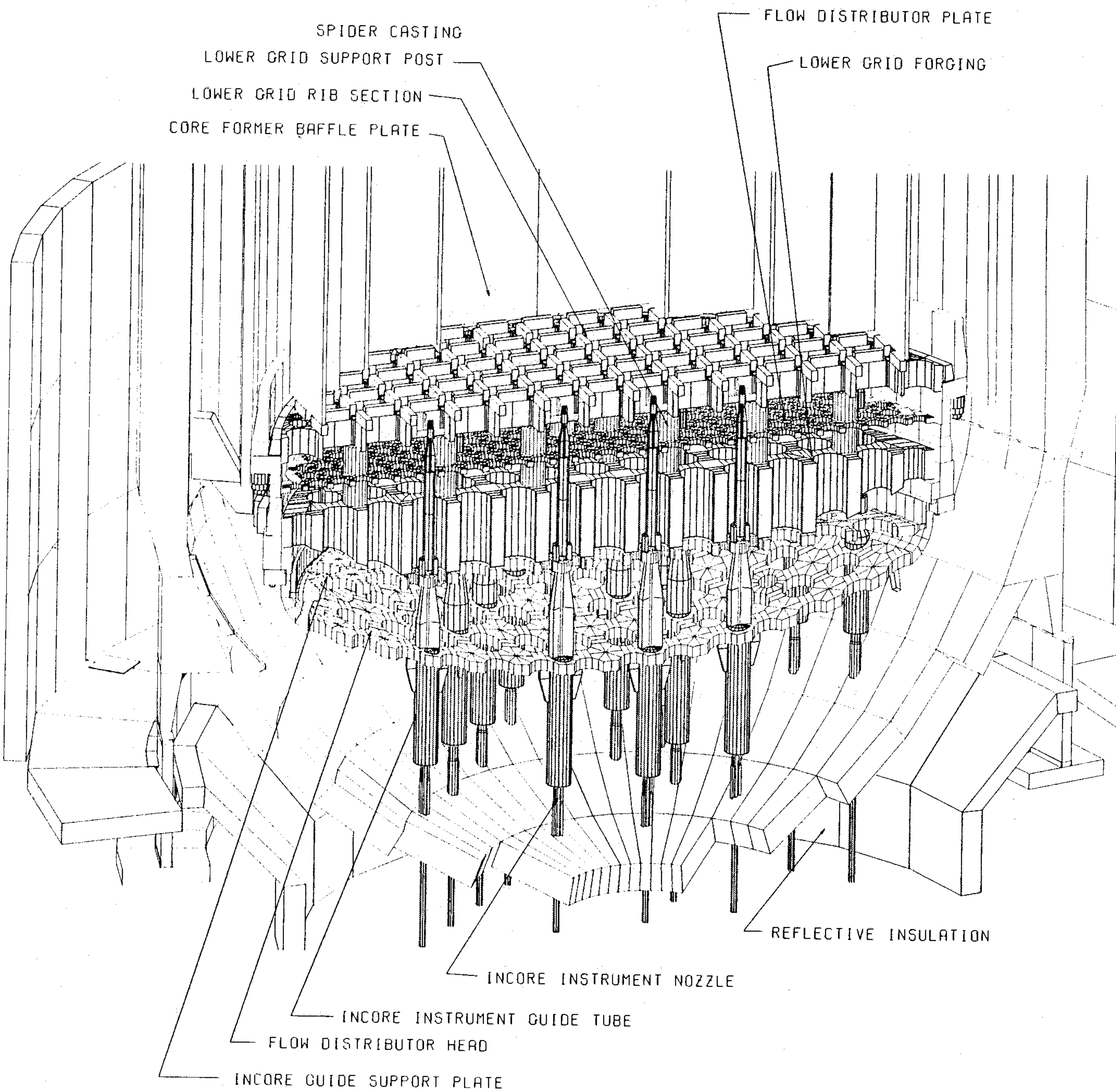


Figure 8-17. Lower Core Support Assembly

- Metal Disintegration Machining (MDM)—An alternative method, using the spark erosion principle for removing metal. It would be very effective providing the structure was relatively free of nonconductive material.

- Water Jet (with or without abrasive)—A secondary, probably effective choice for cutting debris; but it required precise positioning and control. Also, it could add a large amount of abrasive to the volume of debris requiring disposal.

- Plasma Arc—Previously recommended, but the now-anticipated variability in LCSA surface geometry and the poor electrical conductivity of debris-coated structures would likely reduce the reliability to an unacceptable level. (Use of a plasma arc to disassemble a thermal shield at St. Lucie in 1983 had shown that, even on relatively uniform geometry, tool setup and problem solving took much more time than the actual cutting.)

- Arc Saw—Rejected because the system was too large for the LCSA and the high-amperage power supply was not readily available.

- Oxygen Burner—Rejected because of the distance underwater and doubts that enough heat could be generated.

- Mechanical Shearing—Could have limited use but rejected because of the great amount of force required to complete cuts.

- Grinding Abrasive Wheel—Could have limited use but rejected because of the reaction force generated, the size of wheel required, and the slowness of cutting.

- Explosive Cutting—Rejected because of the disruption caused by the explosion, which would produce volumes of gasses and/or pressures. The shock could potentially jeopardize the integrity of the lower reactor vessel head. From a less technical perspective, a skeptical public might be alarmed at the idea of explosions inside the TMI-2 reactor vessel.

- Sawing—This method included a reciprocating power hacksaw, a compact specially designed bandsaw, a circular saw, and variations of these. Rejected because of the tendency of the blades to bind, the difficulty in maintain a sharp edge in borated water, a slow cutting speed, and space constraints.

- Conventional Methods—This included drilling, milling, trepanning, and shaping operations. Rejected because a rigid support structure was required; removing chips from the operational area was difficult yet crucial; borated water had a deleterious effect; and the number of setup, changeout, and relocation operations made it inefficient.

- Laser—Rejected because, without excessive power requirements, the water immediately disperses the cutting beam. Its use would have been experimental, and proven technologies were preferred.

- Ultrasonic Disintegration—Rejected as a method of disintegrating pieces of debris because of its developmental nature.

Later data showed that the structures of the LCSA could be cleaned of adherent debris; this caused a reevaluation of plasma arc technology. To evaluate the candidate cutting technologies, DOE sponsored a series of tests at INEL between December 1985 and March 1986. Underwater tests showed that:

- The thermic rod was very difficult to control, was erratic, and produced severe turbidity problems.

- The abrasive water jet performed well when a debris recovery system was employed on the back side of the cut, but otherwise caused severe turbidity problems.

- The plasma arc torch performed very well and had the least effect on turbidity.

Consequently, the project proceeded to develop the plasma arc torch, with the abrasive water jet as a backup (Porter 1989).

Another step in planning to defuel the LCSA came when the project team incorporated the experience gained in defueling the core region. The difficulty of working under 12 m of highly borated water and the very large quantity of debris relocated to the LCSA by defueling operations in the core region convinced the project team that better access was needed for the entire LCSA/lower head region.

As a result, the layer-by-layer approach was expanded to mean that major portions of each plate would be removed. Specifically, plans called for the first four plates to be cut out and removed; plans for removing the flow distributor plate contained options, depending on lower head defueling plans. This was necessary because of concern about the structural integrity of the lower reactor vessel head, especially the incore instrument nozzle welds to the head.

The size of the opening to be cut in each plate had to be determined:

- Initial plans called for removal of the central 50% to provide the quickest access to the majority of the debris. This would have required approximately 890 cuts and would have resulted in a 1.2 m2, funnel-shaped hole down through the LCSA. It was to this plan that the automated cutting equipment system (ACES) with its plasma arc torch was first designed. In this plan, the pieces would be cut to be handled by the canister handling equipment, although their final disposition was undetermined.

- As the quantity and location of debris in the LCSA became known, a larger diameter access hole was required for defueling. Extensive modifications to ACES resulted from the changed criterion. The larger access hole also doubled the number of cuts required if the plates were to be sized to fit the canister handling equipment—a greatly added stress on the cutting equipment.

Early testing of ACES had raised questions about its performance; i.e., its torch life, cutting ability on incores and support posts, and the reliability of the positioning equipment were all less than desired. Several steps were taken to address the concerns:

- Improve ACES—An intense effort was undertaken to improve the reliability of ACES.

- Decrease the number of cuts—To do this, the size of the pieces cut from the LCSA was increased. Analysis showed that the radiation in air would not be as excessively high as previously estimated. Storage was to be on a rack underwater in core flood tank "A", which was selected because of an acceptable load path to it and its shielded location in a low traffic area. Storage in the containment basement had first been considered but was rejected.

- Develop the core boring machine as an alternative —The use of the core boring machine was evaluated. Mechanical cutting, although rejected in the earlier studies, had many attractive features. Chief among them was the use of a technology that had already proved its worth at TMI-2. In addition, it did not create a water clarity problem, would have few startup problems, and avoided many unknowns. Tests in late 1987 were used to develop the bits and techniques required. The core boring machine proved very adept at cutting out incore guide tubes and support posts —the weakness of ACES—but cutting was only effective when enough space existed for the chips to fall away (Ryan 1988). The reach of the core boring machine (without extensive modification) was also limited and would not permit it to make all the necessary cuts. Also, the large amount of loose debris that had relocated to the LCSA during core region defueling would be a significant obstacle to use.

ACES was originally conceived to work with a robotic service arm—MANFRED—which would deploy tools and handle pieces of the LCSA. This equipment (shown in Photo 8-10) was on site and available. It was never used because of concerns about the complexity of operation and the vulnerability of cables and hoses to accidental severance. MANFRED would also have been extremely difficult to decontaminate whenever it had to be removed from the vessel.

To improve the schedule for disassembly, the final plan combined the best features of the core boring machine (its ability to cut loose vertical cylindrical components) with those of ACES (its ability to make straight vertical or horizontal cuts). The plan was first to have the core boring machine make the crucial vertical cuts by drilling out:

1. 15 incore guide tubes without gussets down through the LCSA. The remaining 37 incores with gussets were not to be drilled out because the limited clearances would make removal difficult.

2. All 48 support posts down through the distributor plate—but not the through the forging because of the volume of chips that would have been generated.

ACES would then be installed, with the core boring machine serving as a contingency tool.

The following concerns had to be resolved before disassembly could begin: criticality control, boron dilution, hydrogen evolution, pyrophoricity of zirconium, nuclear instrumentation interference, release of radioactivity, reactor vessel lower head integrity, basement criticality, and electrical shock. Design changes or analyses addressed all of these issues (Ryan 1989).

As previous industry experience had shown, the overall cutting rate was slow. When the plasma arc was actually

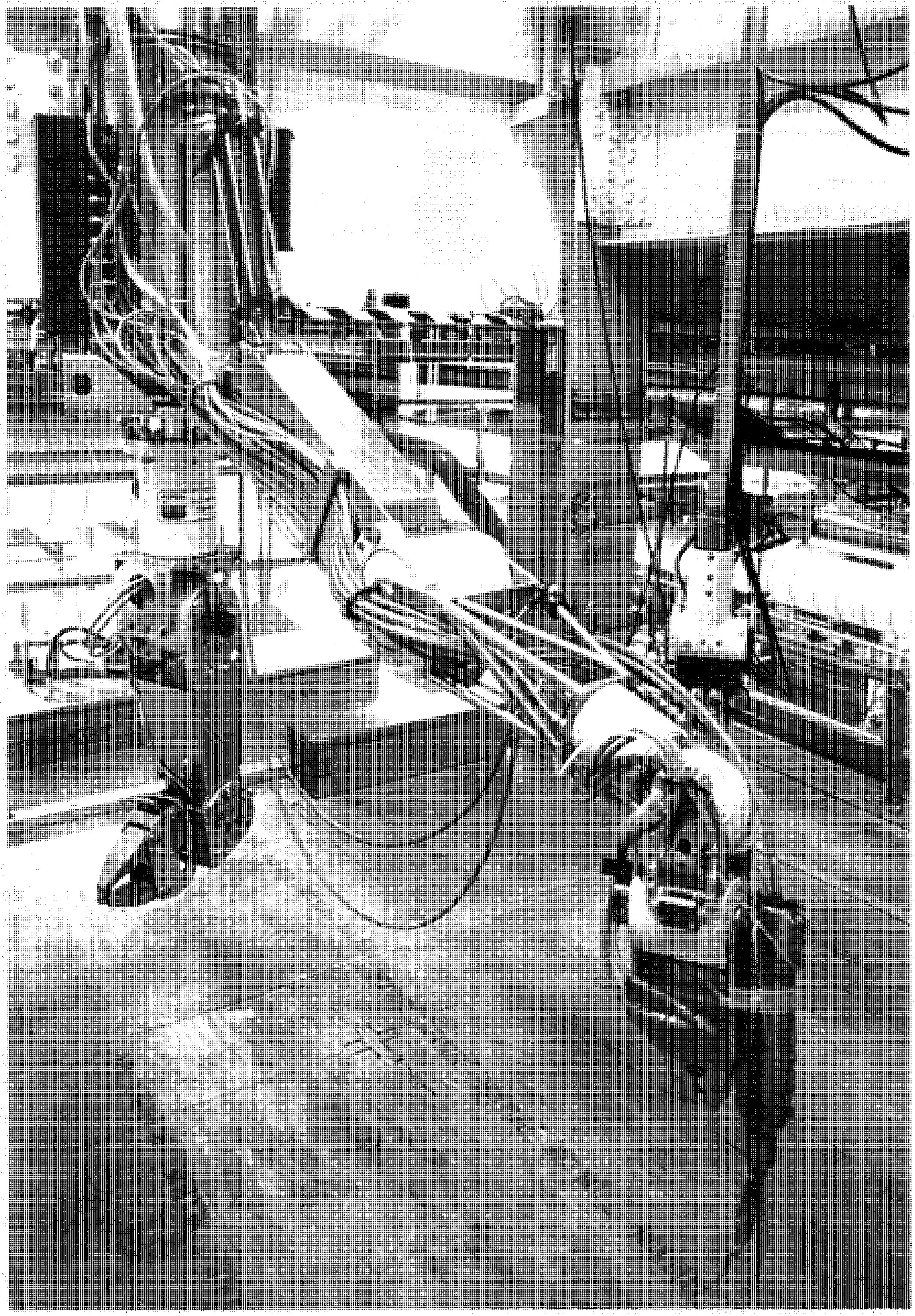


Photo 8-10. MANFRED

cutting, it cut quickly, but setup time and maintenance problems often removed ACES from service. With only one system, considerable time was spent correcting any problem; e.g., to repair the X-Y bridge that supported the plasma arc equipment: two days to remove it, some length of time to repair it, and two days to reinstall it (this improved with experience).

A computer-generated view of ACES, with a cutaway view of the reactor vessel and LCSA is shown in Figure 8-18.

Disassembly and defueling of the LCSA began in January 1988. Defueling with long-handled tools or an airlift was conducted on the large mass of debris above and below each plate as the plates were exposed:

- Lower Grid Rib Plate—Disassembly and removal in 13 pieces: January–April 1988. Not all of the incores without gussets or support posts were drilled out as planned, primarily because the bits became fouled or excessively worn by chips and the loose defueling-related debris; there was no room to discharge the chips created by the core drilling machine. However, additional cuts were made to the lower grid rib plate, cutting it into 13 pieces. The core boring machine was then removed and placed in long-term storage. The pieces were cleaned and removed; the largest piece measured 50 R/h in air at 0.3 m.

- ACES testing—Testing the plasma arc by trimming 72 protruding pieces of the lower grid rib plate: May–June 1988. Life expectancy of a torch: six cuts if it survived the initial cut.

- Lower Grid Distributor Plate—Disassembly and removal of the 2.5-cm thick plate in four pieces: June–July 1988. Cutting was hampered by numerous problems and torch failures, often attributed to debris on the underside of the plate. Redesign of the current controls and feedback loop were required.

- Lower Grid Forging Plate—Disassembly and removal of the 34-cm thick plate in four pieces: August–December 1988. Equipment problems continued, along with building contamination caused during maintenance operations. Difficulty in cutting support posts or incores packed with defueling-related debris. An abrasive saw was used to supplement ACES for vertical cuts. Overall rate of cutting: two to three cuts/day. See Photo 8-11 for a picture of a section of the grid forging being removed from the vessel.

- Incore Guide Tube Support Plate—Disassembly and removal of the 5-cm thick plate in four pieces: December 1988–February 1989.

- Flow Distributor Plate—Disassembly and removal of the 5-cm elliptically shaped plate into 26 pieces: February–April 1989. The large number of pieces was required by the 30 incore guide tubes protruding through, which complicated the cutting pattern. When the cutting was completed, the plasma arc torch was immediately used to make cuts to the core former baffle plates. A large amount of debris from defueling operations rested on this plate and had to be removed by airlift before the cuts could be made.

In the end, the plasma arc torch was made to work through onsite modifications and trial-and-error experience. A plasma arc had never been used to cut so deep in underwater and it had never been used in water containing 5000 ppm of boron, which caused a whole set of new problems. Making it work required patience, persistence, and a developmental attitude.

With the cutting and piece removal complete, the outer, uncut periphery of the LCSA was defueled. High-volume, low-pressure water was used to dislodge loose debris and flush it to the lower head. Newly exposed, resolidified material was broken up with a cavitating water jet.

8.6.3 Lower Head Region

Until February 1985, with the first video inspection, only a small quantity of loose and vacuumable debris was expected in the lower reactor head region. The video inspection revealed an estimated 9000 kg of debris, much of it slag-like. Its elements, friability, and density were not known.

Planning guidance had initially argued for lower head defueling to precede LCSA defueling because removing the bulk of loose debris in the lower head was more important in terms of quantity, ensuring safety, and supporting the shipping schedule (even though new debris might find its way to the bottom during later defueling operations). The sequence was changed to make LCSA defueling a higher priority when large quantities of debris were seen in that structure.

In the spring of 1985, the following approach for defueling the lower head was considered (Skillman 1985b):

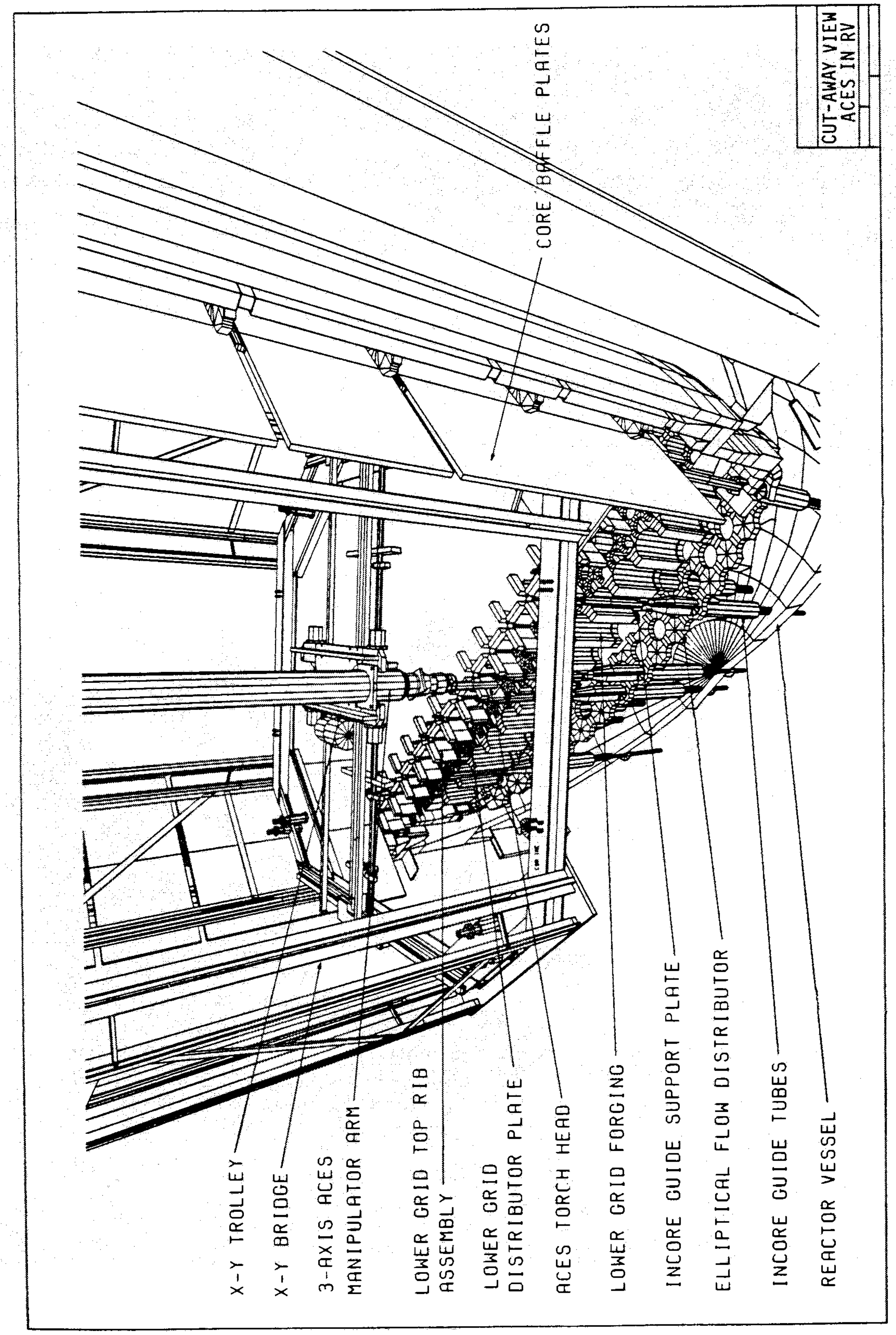


Figure 8-18. Computer View of ACES in the Reactor Vessel

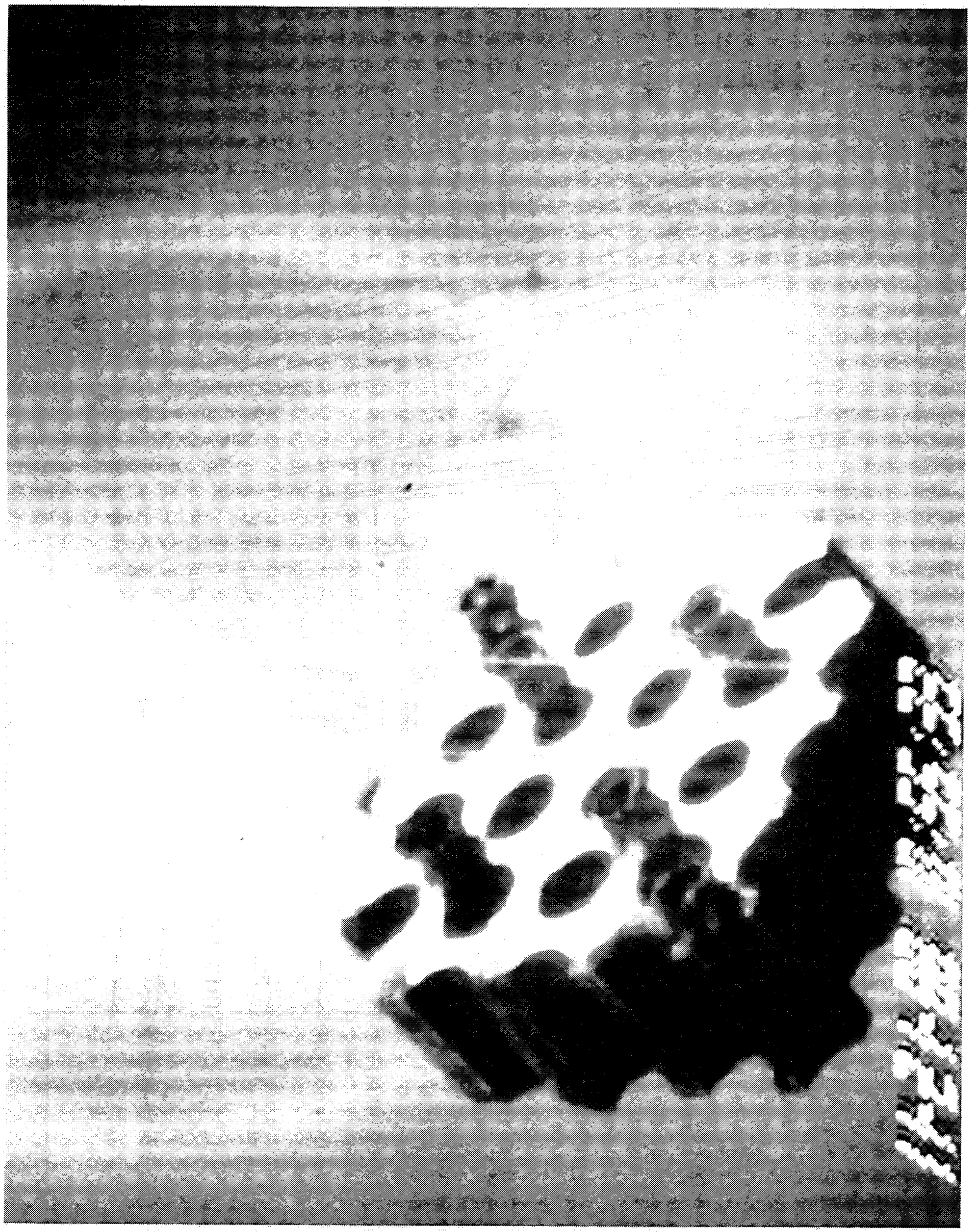


Photo 8-11. Section of the Lower Grid Forging Plate Being Transferred from the Reactor Vessel to Storage

- Clean off the small fuel particles and other debris via vacuuming in order to expose the solidified material. Remove the loose debris as soon as possible to help understand conditions below it and assess lower head integrity.
- Cut and fracture the resolidified material. Prepare to work in and around incore instrument guide tubes and to cut them, as necessary.

From 1985 through the start of lower head defueling in the spring of 1989, an increasingly accurate picture of conditions emerged. However, not only was the knowledge of conditions increasing, so was the quantity of debris. Defueling operations in the core region, especially the stub assembly removal, had caused over 9000 kg of debris to find its way into the LCSA and the lower head. By early 1989, over 25,000 kg were located in the lower head in three general categories as shown in Figure 8-19 (also, see Appendix B).

The approaches to defueling revolved around vacuuming and long-handled tools:

- Vacuum/Airlift—Tests of differently configured active systems were made in late 1987. The most promising (as in the core region) was an airlift system. A passive vacuum system was also considered; it would have had the advantage of not disturbing work in progress within the core region (Kochis 1987).
- Pick-and-Place—Using long-handled tools, the material would to be gripped, scooped, or dug, then placed into a fuel canister. Tools included: 2- and 3-point grippers, debris diggers, spade buckets, and tong tools.
- Material Conditioning—The hard layer of resolidified material had to be broken up for loading by either pick-and-place or vacuuming/airlift. Tools included: cavitating water jet, underwater air chisel, chipping hammer, abrasive saw, displacement water jets, manual impact (slide) hammers, and heavy-duty shears. Three contingency tools were also available: water cannon, abrasive water jet, and the core boring machine.

The sequence was envisioned to be:

- Begin defueling as soon as access was available—This was important to test tools and techniques. Even if LCSA defueling was not complete, beginning lower head defueling gained experience and flexibility.
- Pick-and-place—Necessary to clear material that might clog the vacuum/airlift and limit maneuvering.
- Production vacuuming/airlift—Perhaps interspersed with pick-and-place, removal should begin at the center and work outward.
- Condition large or fused material—Test the conditioning tools and methods as soon as possible to determine if contingency tools would be needed.
- Pick-and-place and vacuuming—Remove the material that has been conditioned.
- Final vacuuming—When all reactor vessel defueling activities have been completed, perform a final vacuuming (Wilson 1987).

Actual defueling operations in the lower head followed the plan fairly closely. Long-handled tools and the airlift rapidly removed thousands of kilograms of loose debris; in May 1989, a record quantity of debris was removed —12,400 kg. The major concern during airlifting was the continual loss of visibility caused by the operation. This problem was address to some degree by modifications to the DWCS in-vessel filtration system, improving recovery time from four hours to two.

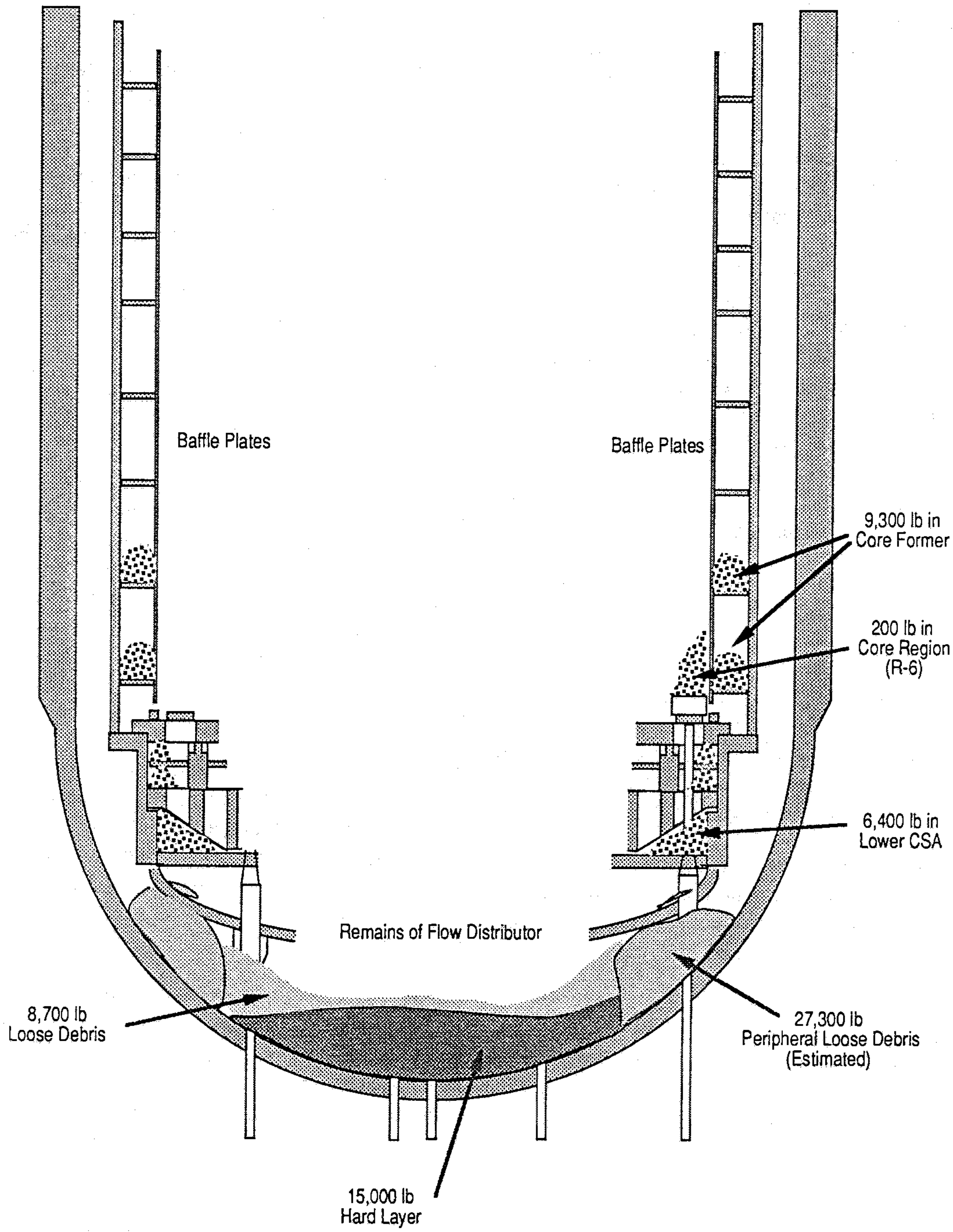
The rapid airlifting of material revealed a monolithic hard mass in the center of the lower head. At first, it appeared very disheartening. Some conjecture held that the consistency of the mass was analogous to "rebar in lava." The possibility of using the core boring machine to break it up was even considered. All efforts were focused on clearing off the hard mass in order to evaluate it.

Removing the loose debris uncovered the irregularly shaped mass of resolidified material, which varied from a few centimeters deep at the edge to 0.6 m at the center. An impact hammer was used on it, starting from the outside and working in. To great relief, the monolith "broke up like a cheap suitcase" (S. Levin, May 1989).

After the pieces of the monolith were loaded into fuel canisters, airlifting continued to clear the lower head.

8.6.4 Core Former Region

As the lower head was being cleared of debris, the project team had to address the last significant location of core debris: behind the core former baffle plates. Initial planning for defueling the LCSA had actually addressed the entire core support assembly, including the core



TMI-2 MATERIAL AT THE BOTTOM OF THE REACTOR VESSEL

Figure 8-19. Lower Head Cross-section with LCSA Cut Out

former region. The entire structure of the core support assembly was to be defueled and stored in the refueling canal (Barrett 1984; Ales 1985).

At the time, little or no fuel debris was believed to lie behind the core former baffle plates. Since large quantities of fuel had been found in the LCSA and lower reactor head region, attention focused on defueling these regions. In the spring of 1987, plans had to be significantly broadened when video inspections revealed a large quantity of debris behind the baffle plates. Figure 5-10 shows the core former region with the baffle plates removed.

Scoping plans laid out two broad categories of methods for gaining access to the region: 1) access holes could be cut in the plates; or 2) entire sections of plate could be removed.

Cutting methods considered to gain access were:

- Plasma Arc Cutting—Although the technology had not yet been proven in the LCSA, it promised to be a fast method. Cutting in the vicinity of fuel, specialized deployment tools, the generation of dross, and water clarity were concerns.
- Electro Contact Cutting—Proof-of-principle tests showed this to be an attractive option, relatively quick and easy to operate. It was pursued as an alternative to the plasma arc torch until the torch proved itself.
- Hydraulic Circular Saw/Wheel—It was a relatively simple technology, but would be slow, wear out many blades, and require a rigid support structure.
- Core Boring Machine—Using this proven technology was tempting, but it would have required major modifications to the design, would not have provided large access holes, and would have been slow.
- Hole Cutter Machine—A custom-designed tool would have drilled a relatively large hole in a relatively short time and been simple to power. The drawbacks were the design and development time and its potential cumbersomeness in operation.
- Thermic Cutting Rod—It was a quick and proven technology. However, the effects of high temperatures on fuel were a concern, as was the probable severe impact on visibility.

If entire baffle plate panel sections had to be removed to gain access to fuel, then other techniques were needed. A large number of lock-welded bolts (756 hex-head and 108 slot-head bolts) held the baffle plates to the core formers; these would have to be broken free or destroyed. The following methods were considered:

- Metal Disintegration Machining (MDM)—Also considered for cutting access holes, the machine was very promising. It would be relatively quick compared to other bolt removal techniques and less expensive. It would, however, require the development of a deployment tool and a large current flow (up to 500 A).
- Electric Discharge Machining (EDM)—The advantages of this nontraditional method of removing metal were that the tool head would not have contacted the work surface (i.e., less changeout) and more industrial experience existed than with the MDM. It would have been more expensive and concerns existed over how to establish a dielectric between work surfaces and the effect of high temperature on fuel debris.
- Mechanical Bolt Removal—Surveys of the industry disclosed that several NSSS suppliers had successfully used mechanical devices to unthread bolts and remove the plates.

The first plan (based upon an estimate of approximately 10,000 kg of debris behind the baffle plates) was to remove as few baffle plates as possible, with a preference for cutting access holes and defueling behind the plates by vacuuming or flushing (McGovern 1987).

In March 1988, plans were changed based on a revised estimate of 4000 kg of debris and a more accurate understanding of distribution and condition. The following was assumed:

- The majority of fuel debris was on the lower former plates; the debris was not uniformly distributed, but no area was free of debris.
- Not all material could be flushed; some was once-molten.
- Once-molten material that flowed behind the baffle plates might have interacted with structural components.
- The majority of the baffle plates were not seriously damaged, except for the large hole melted in the plates in the southeast core region.
- All internal and external horizontal surfaces would eventually have to be flushed.

The final plan, based on time-motion studies, sought to create complete access to the region by removing all baffle plates (Rodabaugh 1989). The plasma arc torch would make a series of cuts; bolts would be removed or, failing that, destroyed; and the baffle plates would be removed in sections. MANFRED, a robotic service arm, could be used to position, hold, or install equipment. This plan was adopted, although simpler deployment tools were used in lieu of MANFRED because of its complex, expensive, and difficult-to-decontaminate nature.

(A variation of this would have been to remove some portion of the baffle plates—e.g., every other one—and attempt to defuel behind the remaining ones. This was rejected because of the uncertainty and difficulty of vacuuming in such a manner.)

How to handle the 4-m long pieces of highly irradiated stainless steel was a serious concern. The peak gamma radiation field (contact in air) generated by the baffle plates was estimated to be approximately 3000 R/h. Several options were examined at different times:

- Cut the plates into small pieces and load them into fuel canisters or other specially designed containers.
- Remove the plates from the vessel and store them either in a remote location (e.g., the containment basement) or in a core flood tank.
- Store them in the reactor vessel so as not to interfere with defueling operations. This was selected, with the plates temporarily hung off other plate sections. It was selected because it required no plant modifications; was most ALARA; required minimum tool development; and did not introduce any new safety concerns (e.g., lift and handling out of the vessel) or difficult failure scenarios. A computer model of the vessel was used to help select how to shuffle them about from hanger to hanger in a manner to least affect defueling and to require the least handling (Kochis 1989).

When to defuel behind the baffle plates was also an issue. A strong case was made for defueling the region at the completion of stub end assembly removal and before LCSA defueling; i.e., defueling from the top of the vessel to the bottom. This would prevent the debris from falling into the cleaned LCSA/lower head region, which was likely if the core former region was defueled last. Equipment and plans were not ready at that point, however, and removing the bulk of the debris (which lay in the other regions) was deemed more important.

As with the defueling of the lower head, actual operations followed plans fairly closely. The core former region was well characterized and accessible, which greatly assisted operations. The real question was how difficult it would be to pull the baffle plate sections from the core former.

In July 1989, production-level removal of bolts from the baffle plates began. The removal of these, in conjunction with the vertical plasma arc cuts made in April 1989, set the stage for lifting each of eight sections out from the core former, defueling it, and suspending it on a side of the vessel. Defueling operations were then conducted on the exposed core former region. Vacuuming, wire brushing, and the cavijet were the most frequently used defueling techniques to remove the debris, which existed in the form of fuel pins, fine particles, and encrusted areas.

8.7 Ex-Vessel Fuel Removal

Removing fuel contained in core debris that had escaped from the reactor vessel was an important but relatively low priority. In the overall sequence of defueling, ex-vessel defueling was to be last unless it could be integrated without adversely affecting in-vessel defueling (DeVine and Negin 1984).

Core debris had been transported from the vessel by both fluid flow and settling phenomena. Potential fuel-bearing locations were first established studying the history of the accident and probable debris transport mechanisms (see Section 1 for figures of the containment and reactor coolant system). The flow through the reactor coolant system had been:

1) Pumped—During the accident and in the first month after the accident by the forced circulation of coolant

2) Burp-oscillation—Movement well into 1980 by the periodic redistribution of coolant volumes—referred to as burping (see Section 3.2).

Approximately 230 kg of fuel were transported within the reactor coolant system by these mechanisms. Additionally, approximately 170 kg were added to this inventory as a result of various defueling operations, for a total of approximately 400 kg.

Arriving at a defueling strategy for this material was, as so often at TMI-2, an evolutionary process closely dependent on data acquisition and analysis. The objective was to ensure that no potential for a recriticality existed.

8.7.1 Technical Alternatives

The escape of some core debris from the reactor vessel was certain—at least once parameters regarding core damage were generated. Speculation as to the actual quantity of debris ranged from a few kilograms to several thousand. As late as 1987, uncharacterized areas such as the J-legs could be hypothesized to contain very large quantities.

The first general defueling approach (not based on significant characterization) was to defuel the reactor coolant system components within the "A" D-ring and then within the "B" (Skillman 1985a). Speculatively, this was to have involved vacuuming the four J-legs, upper tube sheets, once-through steam generator (OTSG) bottoms, and pressurizer bottom. The pressurizer surge, decay heat drop line, and core flood tank injection lines would have been flushed and the reactor coolant pumps disassembled. Nine hundred to 3600 kg of debris were estimated to be in these regions (Skillman 1985b).

Three general techniques were considered: mechanical, hydraulic, and chemical (Moskal 1985). The mechanical techniques assumed fuel would be mechanically mined and loaded into canisters either at the primary location or moved to a central location from surrounding areas. The first three mechanical techniques below were used; several more were considered:

- Remote Manual Pick-and-Place—The primary tool was a pincher or closing scoop type of device. The technique proved to be straightforward and simple, and used familiar, proven tools. The drawbacks were that it was time consuming, and thus expensive in terms of cost and person-rem. Handling small pieces could be awkward.

- Remote Mechanical Pick-and-Place—A simple tool was adapted to a remotely controlled mini-submarine. The technique limited exposure and provided access to difficult areas; however, development and training costs were high and picking up small particles with such a device was sometimes difficult and time consuming.

- Scrap, Push, and Drag—Debris was moved via some type of scraper from peripheral locations to a central one. Inexpensive and generally effective, it still posed the potential of resuspending particles and not doing a thorough enough job.

- Strippable Coatings—Coatings could produce a very clean area and solve some difficult geometries (e.g., OTSG tube sheets), but would generate more waste and raise water chemistry concerns. Information on the technique was in short supply. Equally esoteric and problematic techniques were also rejected; e.g., scavenging debris with eutectic or fusible alloys or using a liquid nitrogen freeze plug.

- Mechanical Disassembly—This was initially the preferred method for defueling a component like a reactor coolant pump and allowing relatively rapid direct access. However, it would have required major decontamination work and special provisions regarding system integrity. Based on final fuel quantity estimates, defueling the pumps proved unnecessary.

Several hydraulic defueling techniques were considered, although only one was used:

- Wet Vacuuming—This was used. It was the most attractive because it used much of the equipment already in use for in-vessel defueling and did not involve much personnel exposure. Some new tooling was developed.

- Flushing with the Reactor Coolant Pumps—Moving the large volume of water would have centralized the debris at the reactor vessel startup filter. However, the condition of the pump was uncertain; ensuring RCS integrity was difficult; the reactor vessel head would have to be reinstalled; and all ex-vessel fuel would not have been captured.

- Flushing by Operating Auxiliary-Motored Reactor Coolant Pumps—Better flow rate control could have been obtained but, besides most of the disadvantages of using the reactor coolant pumps alone, it would be expensive, time consuming, and person-rem-intensive to install new pumps.

- High-Pressure Water Lance—A lance would have been of limited use, specifically in dry areas to flush debris to a central location.

- Flushing Lines by Using a Tank as Source—This was a simple technique using on-hand experience and equipment. The water level in the reactor vessel would have to be lowered and the technique would be limited by line size and flow rate. No fuel-bearing location required this technique for defueling.

Chemical defueling, the third general technique, was rejected. In concept, it meant dissolving fuel and debris in a chemical solution added to the primary system. The coolant would then be processed to recover and remove the fuel. It promised to entail less radiation exposure than the other techniques, but substantial questions existed about its use—particularly regarding its effect on waste management systems and handling of waste forms.

In addition to already published research (e.g., Sjoblom 1986), extensive testing and developmental work would have been required (see Section 7). It did not make sense to attempt to dissolve fuel inside the reactor coolant system and then pump the resulting solution to tanks outside of containment. Including the rinses, there would have been treble the volume and no easy way of disposing of the liquids.

8.7.2 Operations

Ex-vessel defueling operations were finally based on characterization of core debris and the fuel contained in it The programmatic defueling criteria are summoned up in the *TMI-2 Defueling Completion Report* (GPUN 1990):

1) All fuel would be removed that was reasonably accessible within technically practical means.

2) Sufficient fuel would be removed to ensure the absence of the potential for criticality regardless of accessibility and difficulty.

3) Residual fuel would not be removed that was not reasonably accessible by practical means and had been determined to have no significant impact on public health and safety.

Each ex-vessel component required a tailored approach:

- First—Characterize using some mixture of analysis, sampling, remote visual examination, or gamma spectroscopy. The project team proved, by characterization and analysis, that numerous fuel locations throughout the plant did not require defueling in order to meet the criteria described above.

- Second—Plan ex-vessel defueling operations so as to have the minimum impact on in-vessel defueling —especially in terms of resources. For ex-vessel work affecting in-vessel conditions (e.g., water level or clarity, equipment configuration) planners had to seek a "window of opportunity." Primarily, these occurred during interludes between major defueling operations or during equipment changeout.

The amount of fuel relocated to the AFHB was less than 40 kg and the largest single quantities were less than 10 kg. Consequently, there was no dedicated effort to defuel any AFHB component or area. Instead, fuel removal occurred as a byproduct of dose reduction, decontamination, waste processing, sediment transfer, and/or resin removal.

Mockups were prepared and practiced on before actual operations began in the containment. For the quantities of fuel left in all plant locations after defueling, see the *TMI-2 Defueling Completion Report*, submitted to the NRC in 1990. The following five components of the reactor coolant system required defueling to eliminate any potential recriticality during long-term monitored storage:

- Upper Tube Sheets of the Once-Through Steam Generators—The "A" and "B" OTSG upper tube sheets were defueled in the fall of 1987, using pick-and-place long-handled tools and a dry vacuum system (see Photo 8-12).

- Pressurizer Spray Line—Debris in the pressurizer spray line was flushed back into the pressurizer and the RCS cold leg 2A, where some was removed during later defueling operations.

- Pressurizer—In November 1987, defueling was conducted using a submersible pump, knockout canister, and filter canister. An agitation nozzle suspended the fine debris for pumping to the defueling canisters. Later visual inspection indicated pieces of debris up to 5 cm wide, 10 cm long, and 2.5 cm thick remained on the bottom. In the spring of 1988, a remotely operated mini-submarine equipped with an articulating claw and scoop was used to load these pieces into buckets.

- RCS Hot Legs—Defueled in late 1987 and again in 1989 using a combination scraper/vacuuming tool and the in-vessel vacuum system. Residual fuel in the "B" hot leg was later scraped, flushed, and vacuumed into defueling canisters as part of in-vessel defueling.

Photo 8-12. Debris on Tube Sheet of a Once-Through Steam Generator

- Decay Heat Drop Line—In late 1988, the decay heat drop line was defueled using an in-vessel vacuum system. A deployment tool guided the vacuum hose into the line to vacuum loose debris. A hard packed region below the loose debris was broken up with a drain cleaning machine. The material was airlifted to the "B" hot leg for defueling as part of in-vessel work.

In summary, ex-vessel defueling of the reactor coolant system removed greater that 90% of the debris in the pressurizer, decay heat drop line, and hot legs, and 70% of the debris in the OTSGs upper tubesheets.

8.8 Defueling Completion

The following description of the conditions at the end of defueling is extracted from the executive summary of the *TMI-2 Defueling Completion Report* (GPUN 1990):

The *Defueling Completion Report* (DCR) provides the basis for concluding that the TMI-2 facility has been defueled to the extent reasonably achievable and that the possibility of an inadvertent criticality is precluded. As a result of the extensive defueling efforts and the recently completed residual fuel characterization, the following assessments have been made:

- The total quantity of residual fuel is estimated to be less than 1125 kg (approximately 1% of the original core inventory). This fuel is primarily in the form of finely divided, small particle-size sediment material, resolidified material either tightly adherent to components or in areas inaccessible to defueling, and adherent films on surfaces contained within piping, tanks, and other components. The four major plant locations contain the following post-defueling distributions of fuel: AFHB $\leq$ 17 kg; containment (excluding the RCS) $\leq$ 75 kg; reactor coolant system (excluding the reactor vessel) $\leq$ 133 kg; and the reactor vessel $\leq$ 900 kg.

- Evaluation of the ex-vessel residual fuel has demonstrated that insufficient fuel resides in any discrete location to exceed the safe fuel mass limit (SFML) of 140 kg. Further, assuming the residual fuel could accumulate in one ex-vessel area, an unlikely event, the total quantity would not exceed the SFML. In the case of the reactor vessel, a specific analysis was performed to demonstrate that a criticality event could not occur in any configuration of residual fuel.

- With the exception of a final cleanup following the completion of the NRC-sponsored reactor vessel inspection program (VIP), removal of any significant quantity of fuel would require a tedious, labor-intensive effort with an attendant significant occupational exposure. Further, unique defueling techniques such as abrasive cleaning, high-pressure water erosion, chemical cleaning, and component removal and/or disassembly of the primary system would be required. These unique techniques and material requirements would create radioactive waste forms and packages which are not amenable to accepted disposal options and, therefore, could require extended onsite storage or further processing.

Considering the extensive cleanup activity accomplished over the past ten years involving an average workforce in excess of 1000 workers per year, the more than 3.6 million person-hours of cleanup activity, the major effort completed to quantify and characterize the residual fuel, the analyses performed which demonstrate that criticality has been precluded, and the evaluation that continued defueling activities are of no significant benefit to the health and safety of the public, GPU Nuclear concludes that TMI-2 has been defueled to the extent reasonably achievable and that transition to Facility Mode 2, as defined by the TMI-2 Technical Specifications, is appropriate.

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APPENDIXES

Appendix A

TMI-2 TECHNICAL HISTORY CHRONOLOGY

Legend for Topics

1 — Planning & Management
2 — Stabilizing the Plant
3 — Personnel Protection
4 — Data Acquisition and Analysis
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Comments

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Year	Mon	Day	Description of event or milestone	Topic 1	2	3	4	5	6	7	8
1978:											
78	2	8	TMI-2 granted an operating license	√							
78	3	28	TMI-2 achieves initial criticality during low-power testing — 4:37 a.m.	√							
78	9	18	TMI-2 generates power for the first time	√							
78	12	30	TMI-2 added to the rate base at 11 p.m. — 80% power, 735 MW net generation	√							
1979:											
79	3	28	TMI-2 accident begins at 4:01 a.m. — 97% power (ends approx. 4 h later)	√	√						
79	3	30	Heliocopter survey measures 1200 mR/h over plant stack (data misinterpreted)	√	√						
79	3	30	PA Gov. Thornburgh issues a limited evacuation advisory based on measurement	√	√						
79	3	30	Project begins ordering tanks for excess water storage		√			√			
79	3	30	EPICOR I arrives on site		√			√			
79	3	30	HERMAN (robot) arrives from Oak Ridge Y-12 plant		√	√					
79	3	31	Industry Advisory Group formed	√	√						
79	4	1	President and Mrs. Carter arrive to tour TMI-2 with Gov. Thornburgh	√							
79	4	1	Industry Advisory Group first meets	√	√						
79	4	2	Hydrogen recombiner begins removing hydrogen from containment		√			√			

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Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
79	4	3	EPICOR I begins operation		√						
79	4	4	First recovery staff organization developed	√							
79	4	4	Noncondensible gases removed from the reactor coolant system (RCS)		√						
79	4	6	EPICOR II design started		√			√			
79	4	6	Tank farm installation begun		√			√			
79	4	6	Waste gas successfully transferred to containment from aux bldg		√	√					
79	4	6	Reactor coolant pump (RCP) 1A tripped due to vibration; RCP 2A started		√						
79	4	8	WG-1 air filters arrive from Washington		√						
79	4	9	Condensor air extraction filter operational		√						
79	4	9	Gov. Thornburgh lifts evacuation advisory	√	√						
79	4	9	Two 2500-kW diesel generators arrive on site		√						
79	4	9	American Nuclear Insurers preliminary estimate: $140 million of damaged insured property	√							
79	4	11	President Carter appoints 11-member Kemeny Commission	√							
79	4	13	Cooldown of reactor coolant system begins		√						
79	4	19	First cold shutdown acheived		√						
79	4	24	Plant reverts to hot shutdown		√						
79	4	25	WG-1 air filters installed on aux bldg roof and ready to run		√						
79	4	27	Transition to natural circulation begun; RCP-2A tripped; cold shutdown achieved		√						
79	4	27	Find-the-Leak Task Force established		√						
79	5	–	Containment Assessment Task Force established		√	√	√				
79	5	–	Installation of temporary sampling sink begun in fuel handling building		√		√				
79	5	1	Hydrogen recombiner shut down after removing 112 kg of gas in containment		√						
79	5	4	WG-1 air filters begin operation		√						
79	5	5	Work on submerged demineralizer system design begins	√							
79	5	6	Industry Advisory Group disbands	√	√						

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79	5	16	Balance-of-plant diesel generators available		√						
79	5	21	Lancaster files Preliminary Injunction against EPICOR II	√	√			√			
79	5	25	EPICOR II ready for startup testing		√			√			
79	5	25	NRC requires environmental assessment for EPICOR II, prohibits river discharge	√	√			√			
79	5	29	Court orders NRC to perform environmental assessment	√	√			√			
79	5	30	Memorial Day — first standdown by recovery team		√						
79	6	–	Senate approves Hart Subcommittee investigation (1-year project)	√							
79	6	–	Evaporation/solidification facility design authorized					√			
79	6	–	Technical Advisory Group formed by DOE	√				√			
79	6	20	Waste gas leak traced to compressors		√						
79	6	20	Mini-condensate system started up		√						
79	7	–	NRC suspends TMI-1's operating license (reactor shut down 2/79 for outage)	√							
79	7	–	Decision to build mini-decay heat removal system		√						
79	7	6	NRC requests Project begin groundwater monitoring system		√						
79	8	7	First low-level waste shipment to Richland, WA		√			√			
79	8	1	TMI Generation Group formed	√							
79	8	24	First sample of water from containment basement obtained by ORNL				√	√			
79	8	28	Five workers receive extremity exposures over federal limits operating valves	√		√					
79	9	–	Temporary sample sink operational	√	√						
79	10	–	Routine entries into auxiliary building without respirators		√	√		√			
79	10	–	Richland, WA disposal site temporarily closed	√	√			√			
79	10	–	Reactor coolant system begins to burp as it cools		√						
79	10	3	NRC finishes environmental assessment for EPICOR II	√				√			
79	10	16	NRC issues Order and Memoradum to permit operation of EPICOR II	√				√			
79	10	18	NRC requires resin solidification facility for EPICOR II resins	√				√			
79	10	22	EPICOR II begins operation		√			√			

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Topic

Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
79	10	30	Kemeny Report issued	√							
79	10	31	Barnnwell, SC site closed to TMI-2 waste	√				√			
79	11	–	Facility Decontamination Workshop sponsored by DOE in Hershey, PA	√					√		
79	11	–	Richland, WA and Beatty, NV burial sites open for TMI-2 radwaste	√				√			
79	11	5	Interim staging facility for radwaste in use		√			√			
79	11	10	Initial CCTV and radiation measurements in containment (via penetration R-626)		√	√	√				
79	11	13	Project requests authorization to purge containment	√	√	√	√				
79	11	21	NRC issues statement of policy to prepare PEIS	√							
79	12	–	Hot chemistry laboratory in service				√				
79	12	6	Project informs NRC of intent to enter containment airlock on Jan. 31, 1980	√	√	√	√				
79	12	12	Project commits to groundwater monitoring program		√						
79	12	12	Summary Plan for Decon and Defueling sent to Hart Subcommittee	√							
79	12	18	NRC withholds approval to purge containment pending environmental assessment	√	√	√	√				
1980:											
80	1	–	DOE Technical Integration Office established at TMI-2	√							
80	1	10	First EPICOR II vessel transferred to "A" module of waste acres	√				√			
80	1	24	Rogovin report issued	√							
80	1	29	NRC notice of requirement to solidify wastes or use high integrity container	√				√			
80	2	–	Management Plan for TMI-2 Radiological Controls Program issued.	√		√					
80	2	7	GPU omits 1st Quarter dividend on common stock for first time in GPU history	√							
80	2	11	NRC issues 30 Proposed Recovery Technical Specifications	√							
80	2	11	Leak in auxiliary building releases small amount of Kr-85		√	√					
80	2	27	Lancaster Agreement prohibits river discharge of accident-generated water (AGW)	√	√			√			
80	3	–	Construction of TMI-2 administration building begins	√							

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80	3	4	NRC approves opening of containment airlock	√	√	√	√				
80	3	13	Containment airlock opened for surveillance		√	√	√				
80	3	13	Construction of two processed water storage tanks (PWSTs) begins					√			
80	3	20	Makeup pumps shutdown; standby pressure control (SPC) system started up		√						
80	3	24	GPU stock all time low since accident — 3 3/8	√							
80	3	24	Met Ed (GPU) and unions sign cooperative work agreement for cleanup	√							
80	3	25	Project requests permission to enter containment without purge		√	√	√				
80	3	26	GEND agreement signed by GPU, EPRI, NRC, and DOE	√							
80	4	–	Eight monitoring wells installed for Groundwater Monitoring Program	√	√						
80	4	15	Project requests permission to enter containment on April 24, 1980	√	√	√	√				
80	4	23	Project notified that SCBAs not approved by NIOSH — postpones entry		√	√	√				
80	5	–	Seven more observation wells installed for Groundwater Monitoring Program	√	√						
80	5	–	NRC issues final environmental assessment of containment purge	√	√	√	√				
80	5	–	Start of installation of SDS					√			
80	5	15	Union of Concerned Scientists issues report on purge	√							
80	5	16	NRC approves Project request to enter containment on May 20, 1980		√	√	√				
80	5	20	Containment entry attempted 8 pm; inner airlock door fails; mission scrubbed		√	√	√				
80	6	–	Analysis of soil reveals cobalt and cesium not related to containment		√						
80	6	–	COT A tank available for water storage					√			
80	6	12	NRC approves purge — GPU decides to postpone entry until after purge	√	√	√	√				
80	6	23	Sholly appeal to halt purge filed in US Court of Appeals, Wash. DC	√							
80	6	26	Sholly appeal denied	√							
80	6	28	Containment purging begins		√	√	√				
80	7	–	GPU signs contracts with two Bechtel companies	√							
80	7	–	Draft PEIS issued	√							
80	7	11	Containment purging completed; ~46,000 curies of Kr-85 vented		√	√	√				

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80	7	15	Project requests permission to enter containment	√	√	√	√				
80	7	22	NRC approves containment entry	√	√	√	√				
80	7	23	1st containment entry — two technicians, 22 min.		√	√	√				
80	7	30	2nd Quarter figures show first loss in GPU history	√							
80	8	8	GPU estimates 5-year, $500 million cleanup	√							
80	8	15	2nd containment entry — surveys			√	√				
80	9	11	GPU Nuclear Corporation incorporated	√							
80	9	12	GPU informs NRC of scaledown in cleanup operations	√							
80	9	15	GPU Nuclear Group approved to run TMI-2; P.R. Clark Executive Vice President	√							
80	9	18	PA PUC Order prohibiting use of customer revenues for cleanup	√							
80	10	–	DOE Technical Integration Office at TMI-2 fully operational	√							
80	10	16	3rd containment entry — surveys			√	√				
80	10	29	Mini-decay heat removal system operational		√						
80	11	5	EPICOR I ceases processing Unit 2 water		√			√			
80	11	7	Revised cleanup estimate adds 2 years; to cost $1 billion including inflation	√							
80	11	10	Sholly decision by US Court of Appeals	√							
80	11	13	4th containment entry — photos, decon experiment			√	√				
80	11	19	NRC Advisory Panel for the Decontamination of TMI-2 chartered	√							
80	12	2	EPICOR II completes processing 2.1 milliom L of contaminated aux bldg water					√			
80	12	9	End of loss-to-ambient test		√						
80	12	1	First transfer of an EPICOR II vessel to Module "B" in waste acres					√			
80	12	11	5th containment entry — decon tests, first visual inspection of polar crane			√	√				
1981:											
81	1	5	RCS decay heat removed via loss-to-ambient mode		√						
81	2	–	First sequenced defueling plan issued	√						√	

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81	2	3	6th containment entry (plus on 2/5) — surveys, decon tests, sample			√	√				
81	3	–	Government endorses expanded DOE role at TMI-2 for FY 1982	√							
81	3	9	Final PEIS issued	√							
81	3	11	Project requests exemption from resin solidification requirement	√				√			
81	3	16	TMI-2 Safety Advisory Board established; Dr. J. Fletcher named chairman	√							
81	3	17	7th containment entry (plus 3/19 & 20) — water sample & RV head survey			√	√				
81	3	23	NRC approves deletion of resin solidification requirement	√				√			
81	4	8	8th containment entry — photo open stairwell for pump installation			√	√	√			
81	4	23	First low-level EPICOR II vessel shipped to Richland, WA					√			
81	4	29	NRC Commissioners reserve decision on AGW; other decisions to NRC site staff	√				√			
81	4	30	9th containment entry — SDS sump sucker installed			√	√	√			
81	5	–	Base estimate of core conditions published (GEND-007)	√			√			√	
81	5	–	SDS installation essentially complete						√		
81	5	14	10th containment entry — survey/samples; first attempt to operate polar crane			√	√				
81	5	19	First EPICOR II high-activity prefilter vessel shipped to BCL					√			
81	5	28	11th containment entry — safety equip on polar crane; first gamma spectroscopy use			√	√				
81	6	3	DOE announces intent to take SDS wastes	√				√			
81	6	18	NRC approves operation of SDS	√				√			
81	6	25	12th containment entry — surveys and samples			√	√				
81	6	28	Last of original 22 low-level EPICOR II vessels shipped to Richland, WA					√			
81	7	–	DOE proposes Gross Decontamination Experiment	√			√		√		
81	7	8	PWSTs available					√			
81	7	9	Gov. Thornburgh proposes shared funding plan for cleanup	√							
81	7	12	SDS operation begun to test system					√			

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81	7	13	13th containment entry — first in situ inspection of polar crane			√	√				
81	7	15	Initial Memorandum of Understanding (re: radwaste disposal) signed by DOE/NRC	√				√			
81	7	23	14th containment entry — first woman; TV maintenance; surveys			√	√				
81	7	25	First transfer of water to PWSTs					√			
81	8	–	Programmatic decision for "early core removal" over decontamination priority	√					√	√	
81	8	27	15th containment entry (and 9/3) — TV maintenance; surveys; inspect air coolers			√	√				
81	9	11	EPICOR II begins operating in second mode to support SDS					√			
81	9	11	Unusual Event declared when 2000 L of coolant leak to containment basement	√		√					
81	9	23	SDS begins processing containment basement water					√			
81	9	24	16th containment entry — surveys for decon & defueling			√	√				
81	10	9	Government (OMB) agrees to federal participation in cleanup funding	√							
81	10	29	First containment entry (17) under accelerated program — decon preps	√		√	√		√		
81	11	–	EPRI Site Office established	√							
81	11	–	TMI-2 Technical Assistance and Advisory Group (TAAG) established	√							
81	12	21	New $2-million siren system turned over to local counties	√							
1982:											
82	1	–	P.R. Clark becomes Executive Vice President, GPU Nuclear	√							
82	1	1	GPU Nuclear Corp. becomes licensee for TMI-1, -2, and Oyster Creek	√							
82	1	8	Unusual Event declared because of blow down of clogged air line	√							
82	2	19	Unusual Event declared because of faulty instrument readings	√							
82	3	4	Containment Gross Decontamination Experiment begins (completed 3/19)				√		√		
82	3	5	Completed processing 2.5 milliom L of containment basement water through SDS					√			
82	3	12	Completed processing 2.5 mil. L of containment basement water through EPICOR II					√			

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Topic

Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
82	3	15	Revised Memorandum of Understanding signed by DOE/NRC — DOE accepts core	√				√		√	
82	3	22	Unusual Event declared because of leak in makeup system	√		√					
82	3	25	Last entry related to Gross Decon Experiment				√				
82	4	–	Reactor vessel visual exam technique demonstrated on Unit 1 reactor				√				
82	4	–	First EPICOR II prefilter shipped to INEL					√			
82	4	22	Installed well head pumps through a tube to remove additional basement water					√			
82	5	5	Visual look at El. 282' in containment from open stairwell.				√	√			
82	5	21	Processing of reactor coolant system water via SDS begins					√			
82	5	21	First SDS vessel sent to PNL for vitrification					√			
82	6	23	Three-day axial power shaping rod tests conducted				√			√	
82	6	23	First sample of floor silt from containment basement via the open stairwell				√				
82	6	23	NRC issues guidelines for core accountability	√						√	
82	7	14	Partial draindown of the reactor coolant system initiated for Quick Look				√				
82	7	19	Polar Crane Task Group established	√						√	
82	7	21	Quick Look at TMI-2 core (also on 8/6 & 8/12)				√			√	
82	8	1	B. Kanga of Bechtel becomes Director, TMI-2	√							
82	8	3	SISI (robot) surveys makeup & purification (MUP) demineralizer cubicles				√				
82	8	12	First mechanical and electrical inspection of the polar crane				√				
82	9	1	Integrated project organization formed from GPU Nuclear and contractors	√							
82	9	1	10 CFR Parts 19 and 20 entered into the Federal Register	√							
82	9	3	Completed first processing of all accident water					√			
82	9	4	Fishing rod used to lower TLDs into purification demineralizer cubicles				√				
82	9	14	Neutron dosimeters used in "A" MUP cubicle (completed 10/13)				√				
82	10	–	Collimated Janus probe for continuous gamma spectrometry in "A" MUP cubicle				√				
82	10	–	Vacuum skid installed for dewatering SDS vessels						√		

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82	10	–	Dose Reduction Task Force established	√		√	√		√		
82	10	1	100th containment entry			√	√		√		
82	10	14	Containment dome washed with water						√		
82	11	–	Troll ball flushing of containment basement complete (begun in May 1982)						√		
82	11	–	Upper corridors of aux bldg accessible in street clothes			√			√		
82	11	18	All control rod drive mechanishm (CRDM) leadscrews uncoupled				√			√	
82	12	–	Auxiliary building El. 281' corridors released for access without respirators						√		
82	12	–	Most polar crane equipment tested							√	
82	12	–	Readings beneath the vessel head indicate about 500 R/h				√				
82	12	–	All axial power shaping control rods verified uncoupled				√			√	
82	12	–	FRED (robot) purchased				√				
82	12	–	Dose Reduction Program established	√		√			√		
82	12	3	Thermocouple inserted from above verifies in-core thermocouples				√			√	
82	12	8	Four axial power shaping rods uncoupled				√			√	
82	12	10	Air suits used instead of ice vests for polar crane team — results good			√					
82	12	16	Interim solid waste staging facility (carport) available for use					√			
82	12	17	First refill of the reactor coolant system				√				
82	12	31	First highly loaded SDS vessel shipped to Rockwell-Hanford, WA					√			
1983:											
83	1	–	Once-through steam generator (OTSG) "B" filled, recirculated, and sampled				√				
83	1	–	EEI creates voluntary program providing $150 million for cleanup	√							
83	1	–	Operational tests conducted of the polar crane							√	
83	1	–	Neutron shield tanks removed from around the reactor vessel							√	
83	1	4	Public Law No. 97-415 signed; Sholly decison rendered moot	√							
83	1	4	Punctured neutron shield tank to prevent water accumulation from decon ops						√		

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Comments

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83	1	11	Containment El. 282' sludge sampled from El. 305'				√				
83	1	11	Flushed the reactor service structure (above head)						√		
83	1	26	Project revises cleanup completion date to mid-1988 for $975 million	√							
83	2	–	Pyrophoricity raised as an issue by TAAG				√			√	
83	2	6	Containment equipment hatch used for personnel and material access							√	
83	2	11	Reactor vessel flange and seal plate inspected — results positive				√				
83	2	13	Inspection of the polar crane wire rope and main hook — results positive							√	
83	2	18	Shield wall installed in containment for elevator and enclosed stairwell			√				√	
83	2	20	Sampled, vented, and purged gas from both purification demineralizers					√			
83	2	22	US Supreme Court vacated Appeals Court decision on Sholly	√							
83	3	–	Part A of Dose Reduction Program in containment complete						√		
83	3	–	New dosimetry system in place using Panasonic 4 element			√					
83	3	–	Management and safety allegations made related to polar crane refurbishment	√						√	
83	3	6	Contents of both purification demineralizers sampled				√		√		
83	3	6	Spent fuel pool "A" entered to measure radiation and contamination levels			√	√				
83	3	15	First core accountability plan submitted to NRC	√						√	
83	3	25	200th containment entry			√	√		√	√	
83	4	–	First major meeting on fueling shipping cask (DOE/GPU)	√						√	
83	6	9	Decision made not to reflood the containment basement for dose reduction			√		√	√		
83	7	–	Request for proposal for legal-weight truck shipping cask issued	√						√	
83	7	–	Project commits to major revamping of procedure system	√							
83	7	–	Five CRDM leadscrews were sectioned and removed for shipment to INEL				√			√	
83	7	12	Last EPICOR II prefilter vessel shipped; total of 50 containers					√			
83	8	–	Drained the reactor coolant system to 0.6 m above upper plenum							√	
83	8	–	Visual inspection (camera) of plenum top surface — relatively clean				√			√	

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83	8	–	Radiation dose rates under the head in air showed no unexpected conditions				√			√	
83	8	–	Cutting and packing 12 neutron shield blocks for shipment completed					√			
83	8	–	Sonar topography mapping of reactor core void region performed				√			√	
83	8	20	25-day reactor coolant system draindown completed for data acquistion				√				
83	9	–	Cutting and packing mirror insulation from vessel head for shipment completed						√		
83	9	9	First sample of TMI-2 core obtained				√			√	
83	10	–	Robotic/shredder defueling system proposed by Westinghouse	√						√	
83	10	–	Second video exam of core reveals more clearly the true extent of damage				√			√	
83	10	13	300th containment entry			√	√		√	√	
83	11	–	P.R. Clark becomes President and CEO of GPU Nuclear	√							
83	11	–	E.E. Kintner becomes Executive Vice President of GPU Nuclear	√							
83	11	–	TMI-2 Core Shipping Technical Working Team established	√						√	
83	12	–	Respirator cleaning facility starts operation; contractor phased out			√		√			
83	12	–	Reactor coolant drain tank samples obtained				√				
83	12	13	Japanese FEPC announces 5-year, $18 million plan to participate in cleanup	√							
1984:											
84	1	–	Revised administrative procedure system reflects integrated organization	√							
84	1	–	Operation of the expanded decon facility in the aux building begins					√			
84	2	–	Remote reconnaissance vehicle (RRV-1) (Rover) delivered to site			√	√				
84	2	–	Criticality Task Force established to review options for defueling	√			√			√	
84	2	6	J.F. O'Leary becomes Chairman of the Board of GPU Nuclear	√							
84	2	29	Polar crane passes load test for head lift							√	
84	3	–	Project management rejects proposed defueling strategies	√						√	
84	3	16	Reactor vessel head studs partially detensioned							√	
84	3	20	TMI-2 core transport, storage, and disposal contract signed by DOE/GPU	√						√	

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84	3	24	Certificate of Compliance obtained for EPICOR II HIC	√				√			
84	4	–	Video-enhanced overlapping photos of core void obtained				√			√	
84	4	–	Tech spec approved for 4,350 ppm boron during defueling				√			√	
84	4	3	First EPICOR II HIC buried at Richland, WA					√			
84	4	13	Canal to vessel flange seal plate installed							√	
84	4	17	NRC rejects first core accountability plan	√						√	
84	4	18	DOE and Japanese FEPC sign agreement for participation	√							
84	5	–	Jib crane on El. 347' of containment load tested							√	
84	5	–	Decision to defer MUP resin removal because of limited fuel quantity	√				√			
84	5	15	Polar crane ready for use							√	
84	5	17	Tank farm removal from "A" fuel pool begins					√	√	√	
84	5	29	Decision for dry defueling with long-handled tools	√						√	
84	6	–	EEI BOD approves $25 million/yr for 6 years	√							
84	6	–	TMI-2 Program Strategy document issued	√							
84	6	–	Decision to conduct additional containment entries on back shifts	√							
84	6	14	Steam vacuum decontamination system first tested in aux building						√		
84	6	23	Containment chillers placed in service			√					
84	6	24	9-day draining of the RCS to support head lift completed							√	
84	6	28	Reactor vessel head studs final detensioning							√	
84	6	28	First entry (#396) without respirator			√					
84	7	3	400th containment entry			√	√		√	√	
84	7	5	Project announces accelerated schedule; defueling to be complete in 1987	√						√	
84	7	6	Reactor vessel head studs removed							√	
84	7	22	Control rod drive leadscrews parked in housings							√	
84	7	24	Reactor vessel head removed in 2-day operation							√	
84	7	27	Internals indexing fixture installed on vessel flange							√	
84	8	–	Defueling test assembly (mockup) ready for training							√	

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84	8	–	RCS water processing begins via the internals indexing fixture					√		√	
84	8	–	Purchase order placed for two rail fuel shipping casks	√						√	
84	8	–	W.D. Travers becomes NRC Project Director	√							
84	8	15	Gov. Thornburgh urges NRC not to allow restart of Unit 1	√							
84	8	20	F.R. Standerfer of GPU Nuclear becomes Director, TMI-2	√							
84	8	27	19-day campaign to solidify 10 vessels completed							√	
84	9	7	Polar crane taken out of service because of unauthorized brake modification	√						√	
84	9	30	Initiated MUP demineralizer elution					√	√		
84	10	–	Major scabbling campaign begins in containment (completed 6/85)						√		
84	10	–	Computer analysis indicates considerable fuel debris in lower head				√			√	
84	10	18	Polar crane authorized for limited use							√	
84	10	18	Six tank farm tanks removed from "A" spent fuel pool						√	√	
84	10	25	Reactor vessel upper plenum inspected in preparation for removal				√			√	
84	11	1	NRC begins polar crane brake inspection	√						√	
84	11	9	RRV-1 first characterizes containment basement			√	√				
84	12	–	Strategy plan for decontamination of containment basement issued	√					√		
84	12	–	Long-term program to locate ex-vessel fuel begins				√				
84	12	–	Training of fuel handling SROs begins	√						√	
84	12	–	Completed decontamination and refurbishment of spent fuel pool "A"						√	√	
84	12	1	500th containment entry			√	√		√	√	
84	12	4	EEI finalizes contribution program — $150 million over 6 years	√							
84	12	6	Plenum jacked 6.4 cm							√	
84	12	11	Plenum jacked free of reactor vessel (18.5 cm) for inspection				√			√	
1985:											
85	1	–	Decision to use HICs in defueling water cleanup system (DWCS)	√				√		√	
85	1	–	Gamma spectroscopy indicates <2.5 kg fuel in upper tube sheet of OTSG "A"				√				

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85	1	–	Final design review for complete defueling system	√		√				√	
85	1	5	SPC system nitrogen bottles (truck-sized) removed from "A" pool area					√			
85	1	10	Polar crane returned to service							√	
85	1	14	Three workers receive skin exposures over admin limits in SIVR			√					
85	2	–	Last of 46 EPICOR II prefilters buried in HICs at Richland, WA					√			
85	2	18	Agreement reached by EPRI/EEI/GPU stressing EPRI technology transfer role	√							
85	2	20	Phase I of video inspection reveals resolidifed mass in RV lower head				√			√	
85	2	22	EG&G scientists report some core melt at TMI-2	√			√			√	
85	3	–	DOE approves construction of a TMI-2 abnormal waste storage facility at INEL	√						√	
85	3	–	Steam vacuum decontamination begins in the aux building						√		
85	3	–	Dose reduction to support defueling complete (air coolers done by 9/85)			√			√	√	
85	3	21	Wire probing of reactor vessel instrument penetration tubes; one gamma-scanned				√			√	
85	4	–	Cork seal between containment and AFHB replaced by asphalt & polyurethene seal					√	√		
85	4	–	Defueling platform and vacuum system delivered							√	
85	4	–	Upper plenum flushed in the reactor vessel with high pressure water							√	
85	4	12	Polar crane auxiliary hoist ready for use							√	
85	4	12	MUP demineralizer resin elution completed					√	√		
85	4	18	Project requests exemption from certain requirements for SNM shipping	√						√	
85	4	26	Last original accident–related SDS vessel shipped					√			
85	5	–	Self-propelled "mole" used to decontaminate diesel generator building drains						√		
85	5	–	Refurbishment/testing of fuel transfer system completed							√	
85	5	2	600th containment entry			√	√		√	√	
85	5	14	Fuel transfer canal filled to El. 327'							√	
85	5	15	Plenum removed and stored in the fuel transfer canal							√	

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85	5	29	NRC Commissioners vote 4-1 in favor of Unit 1 restart	√							
85	6	–	Robot LOUIE used for survey of the MUP cubicles				√				
85	6	–	Containment air control envelope is ready for use							√	
85	6	3	Last of 5 (of the 6) tank farm tanks shipped to EG&G Idaho (between 5/31–6/3)					√	√		
85	6	7	U.S. 3rd Circuit Court of Appeals issued stay order against TMI-1 restart	√							
85	6	8	TMI-1 in hot standby status	√							
85	6	25	Containment service crane load test							√	
85	6	28	Completed 1-1/2 month boration of PWST 1 for "A" pool fill							√	
85	7	–	Audit findings against canister vendor delay fabrication/shipment	√						√	
85	7	–	Core boring machine arrives on site				√			√	
85	7	–	Phase II video inspection of lower reactor vessel head completed				√			√	
85	7	–	Lower head debris samples obtained				√			√	
85	7	–	Project concludes that ~800 curies of Cs-137 remain in MUP demineralizers					√	√		
85	7	1	Defueling platform and components installed							√	
85	7	9	Abnormal Waste Contract signed by GPU and DOE	√				√			
85	7	10	Tie-in of DWCS to SDS piping completed					√		√	
85	7	11	Fuel transfer canal filtration system placed in service					√		√	
85	7	12	Nitrogen blanket removed from steam generators in preparation for opening				√				
85	7	15	Containment service crane available for use							√	
85	7	23	Hydraulic spray nozzle used to disturbed debris in reactor vessel lower head				√			√	
85	7	24	NRC staff concludes no significant offsite health effects from accident	√							
85	8	–	Canister handling bridges and storage racks received on site							√	
85	8	1	Revised core accountability plan submitted to NRC	√			√			√	
85	8	27	U.S. Third Circuit Court of Appeals upholds NRC order allowing TMI-1 restart	√							
85	9	–	First defueling canisters received on site							√	

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85	9	–	RRV-1 obtains sediment samples from containment basement				√				
85	9	3	TMI-2 Strategic Planning Task Force established	√							√
85	9	5	NRC completes polar crane brake inspection	√							
85	9	5	PA Health Dept reports no evidence of increased cancer among TMI area residents	√							
85	9	19	U.S. Third Circuit Court of Appeals authorizes Unit 1 restart, effective 9/2/85	√							
85	9	24	Supreme Court Justice Brennan continued stay order against TMI-1 restart	√							
85	9	24	Unit 1 commences receiving new fuel	√							
85	10	–	NRC licenses five TMI-2 limted-to-fuel-handling SROs	√						√	
85	10	1	700th containment entry			√	√			√	
85	10	2	U.S. Supreme Court upholds (8 to 1) NRC order allowing Unit 1 restart	√							
85	10	3	TMI-1 initial restart criticality — 1:30 p.m.	√							
85	10	16	Three-day ANS Executive Conference on TMI-2 held in Hershey, PA	√							
85	10	17	NRC approves revised TMI-2 core accountability plan	√						√	
85	10	21	"A" spent fuel pool filled with approximately 775,000 L of water from PWST					√		√	
85	10	30	Defueling begun with core alterations/preparing the debris bed							√	
85	10	30	"A" spent fuel pool filtration started up					√			
85	11	–	Construction of the waste handling and packaging facility begins					√			
85	11	12	Loading of first fuel canister begins							√	
85	11	13	Reactor vessel filtration portion of DWCS started up					√		√	
85	11	26	First concrete core samples obtained from containment basement by RRV-1				√				
85	11	28	Hydrogen peroxide added to "A" spent fuel pool to control microorganism growth					√		√	
85	12	23	Auxiliary building corridors released for clean access (except basement)						√		
1986:											
86	1	–	Integrated test of fuel transfer cask/shipping cask conducted at HEDL							√	

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86	1	–	NRC Commissioners request AGW dispoal recommendation from GPU	√				√			√
86	1	–	First criteria for cleanup completion and "interim monitored storage" developed	√							√
86	1	–	Video shows pressurizer to contain a maximum of 12 liters of removable fines				√			√	
86	1	6	TMI-1 at 100% power	√							
86	1	6	First fuel loaded into fuel canister							√	
86	1	6	7-day, unsuccessful demonstration of vacuum system in reactor vessel completed					√		√	
86	1	12	First fuel canisters transferred from containment to "A" spent fuel pool							√	
86	1	14	Biogrowth first seen in reactor vessel during canister positions system exam					√		√	
86	1	27	NRC issues Final Recovery Technical Specifications and formally amends license	√							
86	2	–	DiBioContamination Task Force established					√		√	
86	2	8	RV pool filter put in operation; shutdown after 4 h with high radiation levels					√		√	
86	2	24	Revision to the TMI-2 organization creates new Defueling Department	√						√	
86	3	–	Shredder received on site							√	
86	3	–	Fuel shipping cask No.1 arrives at TMI-2							√	
86	3	–	Computerized radiation mapping system in operation w/3-D containment model			√	√				
86	3	28	Visual inspections of upper tube sheets of the two steam generators completed				√			√	
86	4	–	Analysis shows debris on upper tube sheet of "A" OTSG has little fuel				√			√	
86	4	–	Gamma scans indicate little fuel in the letdown coolers				√			√	
86	4	–	Gamma scans indicate no fuel in two areas of the containment basement				√			√	
86	4	1	Fuel handling building crane load test completed							√	
86	4	11	NRC issues Certificate of Compliance to DOE for 125-B rail cask	√						√	
86	4	21	Fuel shipping cask No. 2 arrives at TMI-2							√	
86	4	25	Reactor vessel filtration/bug kill operation begins (completed 5/18)					√		√	

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86	4	30	900th containment entry			√	√			√	
86	5	–	TMI-2 receives good marks from NRC SALP	√							
86	5	22	Defueling operations resume after "bug kill"					√		√	
86	6	–	Dr. R.Q. Marston becomes head of SAB; Dr. Fletcher resumes as NASA Administrator	√							
86	6	20	Defueling suspended for Core Stratification Sampling Program preparations				√			√	
86	7	–	Strategy for Recovery Program Completion & Post-Recovery Configuration issued	√							√
86	7	2	Workhorse (Remote Working Vehicle) arrives on site						√		
86	7	3	Core Stratification Sampling Program conducted in RV (completed 7/27/86)				√			√	
86	7	20	1st fuel cask shipment (7 cans, ~1100 kg)							√	
86	7	31	Project proposes evaporation of AGW to NRC	√				√			√
86	8	9	1000th containment entry			√	√			√	
86	8	11	Attempts begin to break up resolidified mass in core with long-handled tools							√	
68	8	31	2nd fuel cask shipment (14 cans, ~4400 kg)							√	
86	9	–	TMI-2 Water Clarity Group established					√		√	
86	9	9	Gross flush of containment basement by RRV begins (completed 11/86)						√		
86	10	–	MUP system flushed, substantially reducing dose rate						√		
86	10	–	Program master schedule revised to show defueling complete by December 1987	√						√	
86	10	20	Swiss cheesing operation on resolidified mass (completed 11/15/86)							√	
86	11	–	Carbon-based hydraulic fluid in tools replaced with water-based fluid					√		√	
86	11	21	Resumed defueling with long-handled tools							√	
86	12	2	Post-defueling monitored storage (PDMS) plan submitted to NRC for information	√							√
86	12	12	Water scarification of containment basement walls begins						√		
86	12	14	3rd fuel cask shipment (14 cans, ~8400 kg)							√	

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1987:											
87	1	–	Waste handling and packaging facility begins operation					√			
87	1	8	Successful startup of reactor vessel DWCS using coagulant addition/filter aid					√		√	
87	1	11	4th fuel cask shipment (7 cans, ~4100 kg)							√	
87	1	13	TMI-2 TAAG disbands	√							
87	1	26	Airlift equipment first used in reactor vessel							√	
87	2	–	Decon facility on El. 347 of containment in operation						√		
87	2	1	5th fuel cask shipment (7 cans, ~3700 kg)							√	
87	2	4	Sediment removal begins from aux bldg sump (completed 5/87)					√			
87	2	9	RCS turbidity at its lowest point (0.75 NTU)					√		√	
87	2	10	Major data acquisition in lower head and core former regions (completed 2/23/87)				√			√	
87	2	15	6th fuel cask shipment (7 cans, ~3900 kg)							√	
87	2	24	Loading upper sections of peripheral fuel assemblies begins							√	
87	3	–	Fuel cask train/automobile accident in St. Louis	√						√	
87	3	3	RRV-3 arrives on site						√		
87	3	18	First fuel assembly (A-6) lifted from lower grid							√	
87	3	22	7th fuel cask shipment (14 cans, ~8500 kg)							√	
87	3	23	GEND group discusses additional TMI-2 research efforts of value	√			√				
87	3	25	NRC Advisory Panel votes 5-4 (1 abstention) against AGW evaporation	√				√			√
87	3	31	Robotic removal of sediment begins in containment basement						√		
87	4	2	GPU declares first dividend since Feb. 1980 — 15 cents/share	√							
87	4	8	EPICOR II completes second mode supporting SDS					√			
87	4	15	EG&G scientists report that 35% of core melted during accident				√			√	
87	4	15	Reactor vessel lower head sampling program proposed to NRC Commissioners	√			√				
87	4	28	Tech Spec Change Request 53 submitted to NRC — 3 facility modes proposed	√							√

Legend for Topics

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Comments

Prepared by the EPRI TMI-2 Technical History Project

Year	Mon	Day	Description of event or milestone	Topic 1	2	3	4	5	6	7	8
87	5	–	One third of core debris removed							√	
87	5	–	Temporary reactor vessel filtration system removed and stored					√		√	
87	5	7	J.F. O'Leary becomes Chairman and CEO of GPU	√							
87	5	11	Production-level removal of stub fuel assemblies begins							√	
87	5	20	First Cuno filter (abnormal waste) shipped to INEL					√			
87	6	–	Automated cutting equipment system (ACES) plasma arc torch arrives on site							√	
87	6	1	Removal of "A" spent fuel demin resins begins (completed 6/5/87)						√		
87	6	2	Deborating and condesate evap. demin resin removal begins (completed 6/12/88)						√		
87	6	21	8th fuel cask shipment (14 cans, ~5900 kg)							√	
87	6	30	PEIS supplement on AGW issued	√				√			√
87	7	–	Sediment removal operations in containment basement complete (73 cu m removed)						√		
87	7	24	Cleanup demin resin removal begins (completed 7/31/87)						√		
87	7	26	9th fuel cask shipment (14 cans, ~5300 kg)							√	
87	7	26	Robotic gross flushing operations continued in containment basement						√		
87	8	–	First shipment of dry active waste (DAW) to Barnwell, SC since moratorium	√				√			
87	8	–	Fuel shipments temporarily suspended by dispute between RR and US Government	√						√	
87	9	–	Gross robotic flushing of containment basement completed						√		
87	9	–	Over half of fuel debris removed from reactor vessel							√	
87	9	–	"A" OTSG upper tube sheet defueled							√	
87	9	13	10th fuel cask shipment (14 cans, ~5700 kg)							√	
87	10	–	First defueling of "B" OTSG upper tube sheet							√	
87	10	5	MUP resin sluicing operation begins						√		
87	10	21	Soviet delegation tours TMI-2; enters containment	√							
87	10	25	11th fuel cask shipment (14 cans, ~6400 kg)							√	
87	10	26	Second Cuno filter shipped					√			

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Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
87	10	27	Third fuel shipping cask received on site							√	
87	11	–	First planning meeting with NRC research section on lower head sampling	√			√				
87	11	–	Schedule revised to show defueling complete 4th qtr 1988; PDMS 2nd qtr 1989							√	
87	11	–	Pressurizer defueling attempted with vacuum						√		
87	11	3	Scarification of containment basement walls begins (complete 2/10/88)						√		
87	11	15	12th fuel cask shipment (7 cans, ~3500 kg)							√	
87	12	–	Core region defueling complete; 176 of 177 stub assemblies removed							√	
87	12	–	J. F. O'Leary dies; W.G. Kuhns re-elected as Chairman/CEO of GPU	√							
87	12	–	"A" and "B" RCS hot legs and decay heat drop line partially defueled							√	
87	12	–	BOP diesel generators sold to Brazil	√							
87	12	15	Core former region data acquisition program begins (completed 12/17/87)				√			√	
87	12	20	13th fuel cask shipment (first triple-cask; 21 cans, ~10,000 kg)							√	
87	12	31	1500th containment entry			√	√			√	
1988:											
88	1	16	Dismantling of lower core support assembly (LCSA) begins							√	
88	1	18	ASLB rules that project must defend AGW evaporation disposal plan in court	√				√			√
88	2	–	Third (and last) Cuno filter shipped to INEL					√			
88	2	–	Fabrication of evaporator authorized	√				√			√
88	2	–	NRC reorganizes site office; reduces onsite staff	√							
88	2	7	14th fuel cask shipment (21 cans, ~9300 kg)							√	
88	2	22	Missouri Senator raises fuel shipping concerns; shipments temporarily halted	√						√	
88	3	–	At Paris meeting, OECD supports NRC lower head sampling program	√			√				
88	3	–	Concrete shield pad poured over SIVR floor to lower dose rates						√		
88	3	28	Mini-Rover submarine first used to defuel pressurizer							√	

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Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
88	4	–	Two SDS vessels shipped to Battelle Columbus; transloaded to HIC for burial						√		
88	4	10	15th fuel cask shipment (21 cans, ~1900 kg)							√	
88	4	11	Core boring machine removed after cutting LCSA grid section into 13 pieces							√	
88	4	23	Grid rib section pieces transferred to core flood tank "A" in 6-day operation							√	
88	5	10	Trimming grid rib section with plasma arc torch begins							√	
88	5	18	Core former bolt removal tool tested in reactor vessel (removes 5 bolts)							√	
88	5	22	16th fuel cask shipment (21 cans, ~4700 kg)							√	
88	5	23	Defueling operator falls partway into reactor vessel; no overexposure	√		√					
88	5	27	NRC approves Tech Spec Change Request 53 (for 3 Modes)	√							√
88	6	9	Trimming grid rib section with plasma arc torch completed							√	
88	6	14	Containment basement block wall fill-and-drain begins (completed 8/88)				√		√		
88	6	14	Pressurizer defueling completed using Mini-Rover							√	
88	6	16	GPU submits license and Tech Spec Change Request for Mode 4	√							√
88	6	18	Cutting lower grid distributor plate begins							√	
88	7	2	Lower grid distributor plate cut into 4 pieces (88 cuts)							√	
88	7	8	Lower grid distributor plate pieces moved to core flood tank in 4-day operation							√	
88	7	21	SDS removed from operation (temporarily restarted to support fill-and-drain)					√			
88	8	–	Thaxton plug problem with canisters discovered, temporarily delays shipment	√						√	
88	8	–	Last of 19 SDS vessels shipped to Battelle for transloading into HICs					√			
88	8	–	Post-defueling monitored storage safety analysis report submitted to NRC	√							√
88	8	–	ASLB issues Memorandum and Order on AGW — hearings to be held	√				√			√
88	8	8	Plasma arc torch begins cutting lower grid forging							√	
88	8	11	First shipment of LSA waste to Oak Ridge, TN for supercompaction					√			

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88	9	–	NRC approves modified security plan combining Unit 1 and 2 security	√						√	√
88	9	–	RRV-1 begins transfer of scarification debris from containment basement						√		
88	9	–	Efforts to sluice resins from MUP vessels suspended after limited success						√		
88	9	–	Final fuel measurements of lower head of "A" and "B" OTSG and J-legs made				√			√	
88	9	–	Drilling of candy cane begun to defuel decay heat drop line							√	
88	9	7	NRC Advisory Panel votes 8-2 against endorsing PDMS w/o more information	√							√
88	9	31	First two post-defueling survey reports (plenum and letdown coolers) submitted to NRC	√							√
88	10	–	Decontamination beyond defueling support deferred until defueling complete	√					√	√	√
88	10	2	M.B. Roche of GPU Nuclear becomes Director, TMI-2	√							
88	10	24	NRC Advisory Panel recommends against accepting PDMS to NRC Commissioners	√							√
88	10	28	W.G. Kuhns announces cleanup to cost $973 million	√							
88	10	31	ASLB hearings on evaporation begin	√				√			√
88	10	31	ANS/ENS meeting with TMI-2 Topical: Materials Behavior and Plant Recovery Technology	√							
88	11	–	GPU named Electric Utility of the Year by EL&P magazine	√							
88	11	–	Two thirds of core debris removed							√	
88	12	–	Fuel characterization of OTSGs continues				√			√	
88	12	18	17th fuel cask shipment (21 cans, ~2700 kg)								√
88	12	22	Plasma arc torch begins cutting incore guide tube support plate							√	
1989:											
89	1	–	Abnormal waste shipments on hold pending DOE final disposition plan	√				√			
89	1	7	Incore guide tube support plate removed in four pieces							√	
89	2	2	ASLB final initial decision recommending approval of AGW evaporation	√				√			√
89	2	15	Topographical survey of debris bed in lower reactor vessel head region				√			√	
89	2	19	18th fuel cask shipment (21 cans, ~5900 kg)							√	

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89	2	21	Lower reactor vessel head debris bed probed				√			√	
89	2	28	Plasma arc torch cutting of flow distributor plate begins							√	
89	3	–	Reorganization creates new Engineering Department	√							
89	3	–	Draindown of "A" and "B" OTSGs (secondary sides) complete								√
89	3	31	Flow distributor plate cut into 26 pieces							√	
89	4	4	ASLB rejects joint interveners request for stay of issuance on evaporator	√				√			√
89	4	12	Baffle plates cut into 8 sections (32 cuts) by plasma arc torch							√	
89	4	13	NRC orders ASLB final initial decision on AGW disposal immediately effective	√				√			√
89	4	21	Flow distributor pieces removed to storage; pick-and-place resumes							√	
89	5	–	New record monthly rate of debris removal (12,400 kg)							√	
89	5	–	Resolidified material in lower head breaks apart "like a cheap suitcase"							√	
89	5	–	Farewell ceremonies for last of departing Japanese engineers (43 over 5 years)	√							
89	5	1	Airlifting defueling operations begin in the lower head							√	
89	5	9	New in-vessel filtration system installed					√		√	
89	5	23	S.H. Hoch becomes Chairman and CEO of GPU; Chairman of GPU Nuclear	√							
89	5	24	2000th containment entry			√	√			√	
89	6	–	Extensive testing of baffle plate and LCSA defueling tools							√	
89	6	18	19th fuel cask shipment (21 cans, ~9500 kg)							√	
89	6	27	LOUIE I robot returned to DOE			√	√				
89	7	–	Extensive damage to some incore nozzles seen				√			√	
89	7	–	In-vessel filtration system modified to act as a vacuum					√		√	
89	7	–	95% of core debris removed from vessel							√	
89	7	3	Cracks observed in lower head cladding; load handling restrictions in force				√			√	
89	7	14	Production-rate baffle plate bolt removal begins (864 originally in plates)							√	
89	8	8	First phase of baffle plate bolt removal complete (831 of 864 removed)							√	

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Year	Mon	Day	Description of event or milestone	Topic 1	2	3	4	5	6	7	8
89	8	10	Reactor vessel hot legs defueled (completed 8/14)							√	
89	8	13	20th fuel cask shipment (21 cans, ~14,300 kg)							√	
89	8	14	Processed water evaporator for AGW delivered to site					√			√
89	8	17	TMI-2 GORB disbands, subcommittee of TMI-1 GORB created for TMI-2	√							
89	8	26	First video of cracks on lower reactor vessel head made with color camera				√			√	
89	8	28	Kerf cleaning between baffle plate sections begins w/abrasive saw							√	
89	9	25	Two workers handle core debris cleaning decon facility, exceed federal limits	√		√					
89	9	26	First baffle plate section (NW) removed; core former region defueing begins							√	
89	10	–	Measurement of residual fuel on end fittings in storage drums completed				√			√	
89	10	10	Installation of processed water evaporator completed	√				√			√
89	10	18	GPU Nuclear corporate reorganization recognizing phasedown of TMI-2	√							
89	10	27	Eighth and final baffle plate section (NE) removed							√	
89	10	29	Defueling of core former region behind baffle plates complete							√	
89	11	1	Evaporator testing begins with surrogate solution; technical problems exist	√				√			√
89	11	7	Lower core support assembly flush and cavijet defueling complete							√	
89	11	28	Ten-day defueling hiatus begins following unplanned exposure to worker's hand	√		√				√	
89	12	9	TMI-2 Safety Advisory Board disbands	√							
89	12	16	Lower head airlift/vacuum complete; bulk defueling complete (11:54 a.m.)	√						√	
89	12	17	21st fuel cask shipment (21 cans, ~10,800 kg)							√	
89	12	24	Core flood lines, hot & cold legs defueled; ex-vessel defueling complete	√			√			√	√
1990:											
90	1	30	Final reactor vessel cleanup and inspection completed — 8 a.m.	√			√			√	√

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Year	Mon	Day	Description of event or milestone	1	2	3	4	5	6	7	8
90	1	30	TMI-2 Vessel Inspection Program (VIP) begins — lower head sampling				√				
90	3	–	Post-VIP cleanup to ensure <1% of original fuel inventory remains in plant	√			√			√	√
90	4	15	22nd and final fuel cask shipment (20 cans, ~7,800 kg)	√						√	√
90	4	26	TMI-2 enters Mode 2 licensing condition ("defueled")	√						√	√
90	4	27	TMI-2 enters Mode 3 licensing condition ("fuel shipped off site")	√						√	√

Appendix B

POSTACCIDENT REACTOR VESSEL CONDITIONS

The following description of the postaccident conditions in the TMI-2 reactor vessel is extracted from the *Defueling Completion Report* (GPUN 1990):

The original core inventory included approximately 94,000 kg of UO_2 and 35,000 kg of cladding, structural, and control materials. Accounting for the oxidation of core materials during the accident and for portions of the upper plenum structure that melted, the total amount of postaccident core debris was estimated to be approximately 133,000 kg.

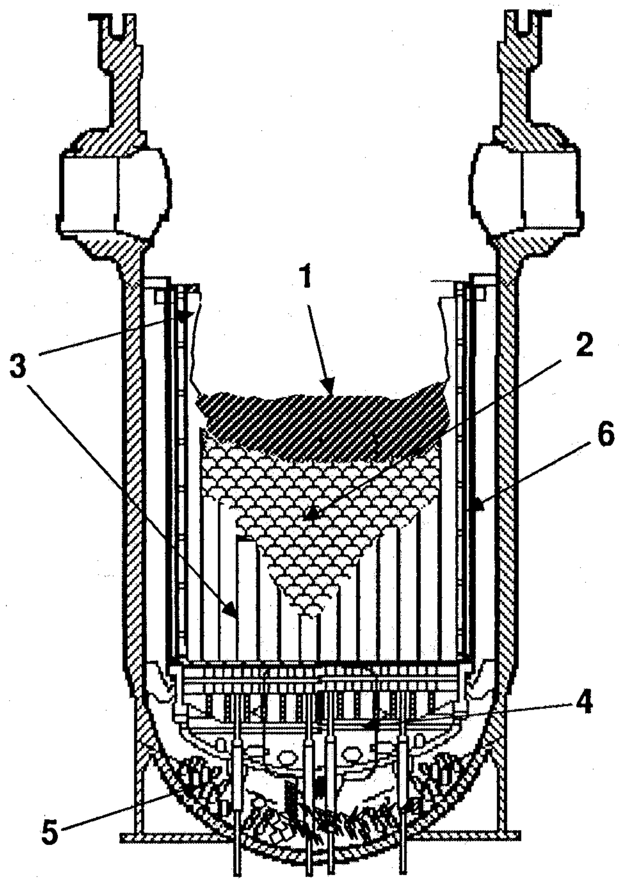
During the accident, peak temperatures ranged from approximately 3100 K at the center of the core (molten UO_2), to 1244 K immediately above the core, and 723 K at hot leg nozzle elevations. Approximately 50% of the original core became molten. Lower portions of three baffle plates on the east side of the core melted and some of the molten core material flowed into the core bypass region. Approximately 30,000 kg of molten materials flowed from the core to the core bypass region and through the lower internals. Approximately 19,000 kg came to rest on the reactor vessel lower head (see Figure B-1).

Upper Plenum Assembly—The upper internals (plenum) had two damaged zones. Localized variations of damage were evident in each zone. For example, in the limited area above one fuel assembly, ablation of the stainless steel structure was observed; however, grid structures adjacent to the ablated zone appeared undamaged. In some regions, the once-molten grid material had a foamy texture, which occurs when stainless steel oxidizes near its melting point. A once-molten mass close to this grid material appeared to be unoxidized, suggesting that some of the hot gases exiting the core were oxygen deficient. The damage to the upper plenum assembly indicated that the composition and temperature of the gases exiting the core varied significantly within the flow stream. Only a small quantity of fuel debris was measured within the upper plenum.

Core Region—A core void or cavity existed at the top of the original core region. Below that, a bed of loose debris rested on a resolidified mass of material that was supported by standing fuel rod stubs. The stubs were surrounded by intact portions of fuel assemblies. A previously molten, resolidified mass was encapsulated by the distinct crust of material in which other fragments and shards of cladding could be identified.

The core void was approximately 1.5 meters deep, with an overall volume of 9.3 cubic meters. Of the original 177 fuel assemblies, 42 partially intact assemblies were standing at the periphery of the core void. Only two of these fuel assemblies contained more than 90% of their full-length cross-sections with the majority of fuel rods intact. The other assemblies suffered varying degrees of damage ranging from ruptured fuel rods to partially dissolved fuel pellets surrounded by once-molten material.

The loose debris bed at the base of the core cavity ranged in depth from 0.6 to 1 meter and consisted of whole and fractured fuel pellets, control rod spiders, end fittings, and resolidified debris totaling approximately 26,000 kg. Beneath the loose debris bed was a large resolidified mass approximately 3 meters in diameter. This mass varied in depth from 1.5 meters at its center to 0.25 meters at its periphery, and contained approximately 33,000 kg of core debris. The center of this solid metallic and ceramic mass consisted of a mixture of structural, control, and fuel material that reached temperatures of at least 2800 K and possibly as high a 3100 K during the accident. The upper crust of this mass, which consisted of the same material and also reached 2800 K, contained intact fuel pellets near the periphery. The lower crust consisted of once-molten stainless steel, zircaloy cladding, and control rod materials resolidified in flow channels surrounding intact and partially dissolved fuel pellets. The thickness of this lower crust, based on initial video examinations, was estimated to be approximately 0.01 meters on the average. The resolidified mass was



ZONE	DESCRIPTION	ESTIMATED QUANTITY (Kg)
1	Upper Debris Bed	26,000
2	Resolidified Mass	33,000
3	Intact Assemblies (Partially or Fully Intact)	45,000
4	Lower CSA	6,000
5	Lower Head	19,000
6	Upper CSA	4,000
	TOTAL	133,000

Figure B-1. Postaccident Estimated Core Material Distribution

shaped like a funnel extending down toward the fuel assembly lower end fittings near the center of the core.

The standing, undamaged fuel assembly stubs extended upward from the lower grid plate to the bottom surface of the resolidified region of the once-molten materials. These stubs varied in length from approximately 0.2 to 1.5 meters. The longer partial fuel assemblies were located at the periphery of the resolidified mass. On the east side of the core, one fuel assembly was almost completely replaced with once-molten core material; this indicated a possible relocation path into the lower core support assembly and core bypass region for the once-molten material. The standing fuel assembly stubs and peripheral assemblies constituted approximately 45,000 kg of core debris.

Upper Core Support Assembly—This region consisted of vertical baffle plates that formed the peripheral boundary of the core; horizontal core former plates to which the baffle plates were bolted; the core barrel; and the thermal shield. There were a number of flow holes in the baffle and core former plates through which coolant flowed during normal operations. On the east side of the core, a large hole approximately 0.6 meters wide and 1.5 meters high, and extending across three baffle plates and three core former plates was discovered. Adjacent baffle plates on the east and southeast were warped, possibly as a result of the high temperatures and the flow of molten material in the bypass region.

It was concluded that molten core material from the core region flowed through the large hole in the baffle plates into the upper core support assembly and then circumferentially throughout the assembly. It also flowed downward through the flow holes in the core former plates into the lower core support assembly (LCSA) at nearly all locations around the core. The majority of the molten material appeared to have flowed into the LCSA on the southeast side through the hole in the baffle plate and through the southeast core former plate flow holes.

The circumference of the core region (i.e., the area behind the baffle plates) contained loose debris throughout. The depth of debris varied from approximately 1.5 meters on the east side to a few millimeters on the southwest side. There appeared to be a resolidified crust on the upper horizontal surfaces of the three bottom core former plates; this crust varied in thickness from approximately 0.5 to 4.0 cm. It is estimated that approximately 4,000 kg of core debris were retained in the upper core support assembly region. In the small annulus between the core barrel and the thermal shield, fine particulates were observed but no major damage to these components was seen.

Lower Core Support Assembly—The LCSA region consisted of five stainless steel structures. The structures varied in thickness from 0.025 to 0.33 meters, with flow holes 0.080- and 0.15-meter dia. flowholes. Some molten core material flowed through these structures and came to rest on the lower head. There were approximately 6,000 kg of resolidified material dispersed at various locations on the circumference of these structures. In several places, resolidified material completely filled the flow holes; columns of once-molten material were observed between the plates. The largest accumulation of resolidified material appeared to have flowed into the LCSA from the east side of the core. Although most of the material was seen on the east to southeast side, many columns of resolidified material were also seen in the LCSA around the periphery of the core beneath the core bypass region.

Lower Reactor Head Region—The debris in the lower head region accumulated to a depth of 0.75 to 1 meter and to a diameter of 4 meters. The spatial distribution of the material was neither uniform nor symmetrical. The surface debris had particle sizes that varied from large rocks (up to 0.20 m) to granular particles (less than 0.001 m). The larger rocks, especially in the northeast and southwest regions, were located near the periphery. The debris pile was lower at the vessel center than at the periphery, with granular or gravel-like material observed in the central region of the vessel. A large resolidified mass was identified between the loose debris bed and the lower head of the reactor vessel. This mass was approximately 0.5 meters thick in the center and 1.7 meters in diameter. A large cliff-like structure formed in the northern region from once-molten material. The cliff face was approximately 0.38 meters high and 1.25 meters wide. Approximately 12,000 kg of loose core debris and 7,000 kg of agglomerated core debris relocated into the lower head.

Appendix C

EXISTING DECAY HEAT REMOVAL SYSTEM

One of the main concerns with using the installed decay heat removal system was that the system was not leak-tight. Under normal conditions, the leakage through the seals and valves posed no radiation threat. However, with the concentration of radionuclides in the reactor coolant being so high, any leak, no matter how small, would render the system impossible to repair and might force the evacuation of the auxiliary building. For this reason, the project team decided to perform a pre-operational test of the existing decay heat removal system to detect and eliminate leaks.

This required entries into the decay heat removal areas. The first problem facing the operators was the fact that the decay heat pumps and heat exchangers were in a pit beneath the auxiliary building basement that had been flooded with intermediate-activity water (1-100 µCi/ml). This flooding resulted in surface contamination and dose rates on the order of 200 mR/h in the decay heat vaults. Above the vaults, the dose rates from various discrete sources were on the order of 1 R/h.

All entries were via the diesel generator building located to the west of the fuel handling building with doorways into El. 305' and 282' of the fuel handling building. In mid-April 1979, a lean-to tent was provided for the workers to change into anti-contamination clothing, and the decontamination teams began making entries after the Easter weekend.

On May 4, 1979, a closed-circuit television system was installed to monitor the decay heat removal system for leaks, which were noted when the system was pressurized. Remedial actions were taken to repair the leaks to make the system leak-tight; however, there were doubts about how effective the repairs would be.

If the decay heat removal system had been used, even in June 1979, the dose rates would have been prohibitively high above the vaults. Consequently, a task order was issued in early April to design a shielded cover for the decay heat removal vaults. This shield was designed, but it effectively cut off the air flow to the vaults. The decay heat removal pump motors were air-cooled and the shield caused them to overheat. Another task order was issued to provide a forced air ventilation system to cool the vaults. This system was designed, but it would have caused serious airborne contamination problems if a pump seal failed.

Also, starting up the decay heat removal system would have forced the evacuation of the south end of the fuel handling building where the decay heat removal pipes penetrated the containment. This would have forced the termination of all work on the alternate decay heat removal system. Since these were unacceptable side effects, the plans to utilize the decay heat removal system were abandoned in late May 1979. Although the system remained on line and ready to start, it was not used.

Appendix D

ALTERNATE DECAY HEAT REMOVAL SYSTEM

Another decay heat removal system was proposed to avoid contaminating the installed decay heat removal system. This new alternate decay heat removal (ADHR) system was to be a completely redundant, full-sized system. It was to be tied into the existing decay heat removal system piping just outside of the containment building. The new lines were to be routed out of the fuel handling building into a new structure designed to house the heat exchangers, pumps, and control systems.

In addition to decay heat removal, the ADHR was to have many other functions, including providing:

- Cooled auxiliary spray to the pressurizer for complete depressurization of the reactor coolant system after shutdown of the reactor coolant pump. A second potential function of the auxiliary flow was to provide cold water injection to the top of the core.
- Filtration/purification/chemical addition for reactor coolant
- A means of draining the containment sump by transferring water to treatment or disposal facilities
- A means of degassing the reactor coolant system
- A means of monitoring the reactor coolant system boron concentration.
- Surge capacity for solid reactor coolant system volume changes
- Gas and liquid sampling capability.

The ADHR was designed with redundant components so that decay heat removal would not be lost assuming a single active failure or during equipment maintenance. It was to operate assuming a loss of offsite power.

Work on the ADHR began in early April 1979, with a survey of available equipment around the country. The lines that tied into the existing piping were installed and a hole was cut through the fuel handling building basement wall. A pit was dug next to the west side of the fuel handling building and the new isolation valves were installed. The rest of the system was to be fabricated and brought to the site as skid-mounted assemblies.

As the design proceeded, it became clear that the original concept was too ambitious to be implemented. The specific activity of the reactor coolant was over 840 μCi/ml in May 1979. The hazard to personnel and to the public from even a small leak from the ADHR system was a serious concern. The installed purification system was no longer operable and no plans were being developed to put it back in service. Thus, the radioactive material in the coolant would be removed only by decay and leakage. The specific activity of the ADHR water had to be assumed to be the same as the activity of the reactor coolant system.

Since dumping steam to the condenser was working better than expected, much of the original urgency abated. The NRC insisted on many safeguards against leaks and equipment failures in the new facility. The planned ADHR building was to be a two-story structure 30 m long by 14 m wide. The heat exchangers were to be located in the basement, and the controls were to be located on the ground floor of the new building. It was to have 1.5-m-thick walls for shielding, seismically able to withstand the impact of an airplane crash. Also, it was to be located over the safety-related cooling piping from the river water pumphouse. This made it extremely difficult to construct. The pipes would have had to be exposed by excavation and the building built around them. As the original concept grew more complex, the schedule slipped.

By mid-summer, it was clear that the ADHR concept was too ambitious, posed serious potential environmental risks, and was over-sized. The decay heat generation rate in the reactor coolant system had decreased to well below the system design capacity and other measures were proving effective in removing the decay heat. Since the cost could not be justified, the system—scheduled to be put in service in December 1979—was put on hold in July. Although the equipment skids were fabricated, they were never shipped to the site. The ADHR system was replaced by the mini-decay heat removal system; some of the equipment was returned to the original owners in the early summer of 1980.

Appendix E

STEAM SYSTEM MODIFICATIONS

On April 6, 1979, an effort began to develop methods for using the secondary side cooling systems to cool the plant indefinitely. This criteria development program resulted in four approaches. Designs for all of these were begun in an attempt to develop enough design data to make a clear choice:

- Short-term cooling using the "A" steam generator and the condensate pump—cancelled because the performance characteristics of the condensate pumps were unacceptable for long-term performance.
- Long-term low-pressure cooling using the "B" steam generator—cancelled because of concerns about the potential of leakage in the "B" steam generator.
- Long-term "A" cooling using 600-psi design—cancelled because it was not possible to make the necessary tie-ins while the "A" steam generator was in use.
- Long-term "B" cooling using 600-psi design—partially installed.

The system design criteria for the long-term "B" steam generator system were issued April 14, 1979. The system used the "B" steam generator and was to be operated as a water/water heat exchanger. A high-pressure closed-cooling water system removed the heat from the steam generator and transferred it to the nuclear service closed cooling water system. This water was cooled by the mechanical draft cooling tower located to the west of the plant.

The new closed cooling water system was designed for 600 psig so that all leakage would be into the reactor coolant system in order to keep the radioactive contamination inside the containment. The main steam piping would have to be filled with water so new pipe hangers had to be added to accommodate the extra weight. The system was to utilize one of the existing condensate demineralizer beds, but this was abandoned as impractical. A small, portable cleanup demineralizer was designed to remove radioactivity and to maintain water chemistry in the steam generator.

The system was to be installed in phases. The first phase would install the system equipment and enough local controls and instruments to permit manual operation. The second phase would install automatic controls and remote instrumentation to permit the system to be operated from the control room. The system was installed in the basement of the turbine building by the middle of May and scheduled for July startup (see Figure E-1 and Photo E-1). Construction of the second phase was put on hold in May 1979.

There were concerns about the system causing a boron dilution event if the "B" steam generator had been damaged by the accident. Samples of condensate taken from the "B" steam generator were contaminated, but not from leakage. This concern caused doubts about the appropriateness of using the system. Although these would later prove to be without foundation, the system was effectively abandoned in place awaiting the completion of the mini-decay heat removal system.

In May 1980, an unfavorable evaluation was made of the mini-decay heat removal system. A subsequent report recommended that the long-term "B" cooling system be used instead of the mini-decay heat removal system. This rekindled interest in long-term "B" and several tasks were undertaken to get the system operational. However, the decay heat of the core had diminished to the point where soon no active cooling systems would be necessary to cool the reactor.

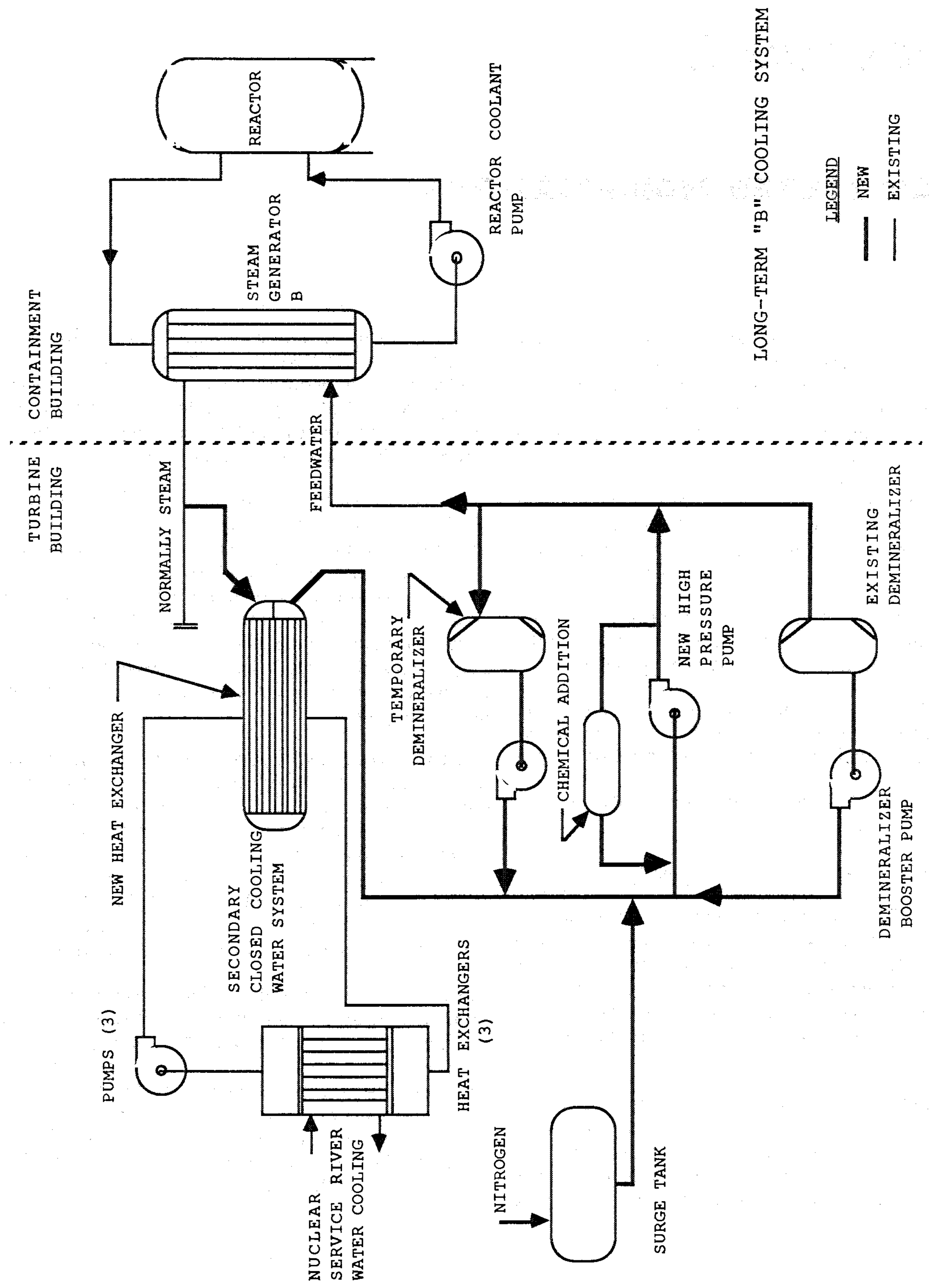


Figure E-1. Long-Term "B" Cooling System

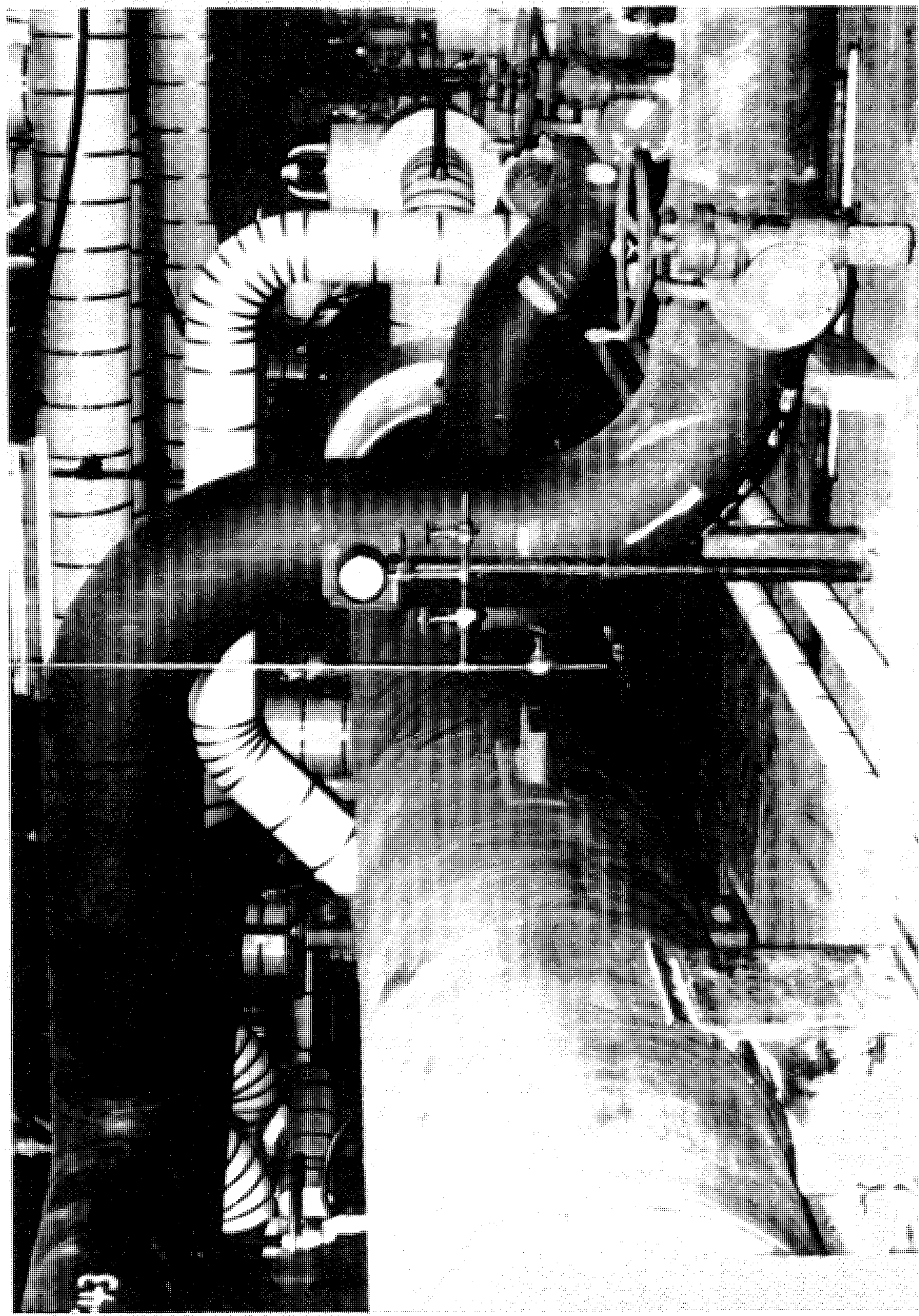


Photo E-1. Long-Term "B" Cooling System Heat Exchanger

Appendix F

MINI-DECAY HEAT REMOVAL SYSTEM

In late June 1979, a mini-decay heat removal (MDHR) system was proposed. At that time, the ADHR system was encountering scheduling delays, and the decay heat from the core had decreased to approximately 800 kW. In the time frame expected for completion of the ADHR, the decay heat would be less than 200 kW. Figure F-1 shows the general design of the MDHR and Photo F-1 shows the MDHR heat exchanger. Table F-1 compares the sizes and capacities of the ADHR and the MDHR systems.

The proposal was to provide a small, skid-mounted decay heat removal system to be installed in the plant. The system would have the capacity to remove a maximum of 1 mW of decay heat. Its advantages over the ADHR were:

- Small piping and components would allow offsite construction and ease the shipment of the skids.
- The MDHR system was not safety-related.
- The system skids could be located in the existing buildings.
- The electrical power consumption would be less and the components would be easier to install.
- The cooling water requirements would be less and could be supplied by the existing nuclear service water.
- The system could be available for use in approximately 90 days.

The MDHR was never operated, although it was maintained in the technical specifications as a backup decay heat removal system and as a method of dealing with a potential loss of coolant accident in the reactor vessel.

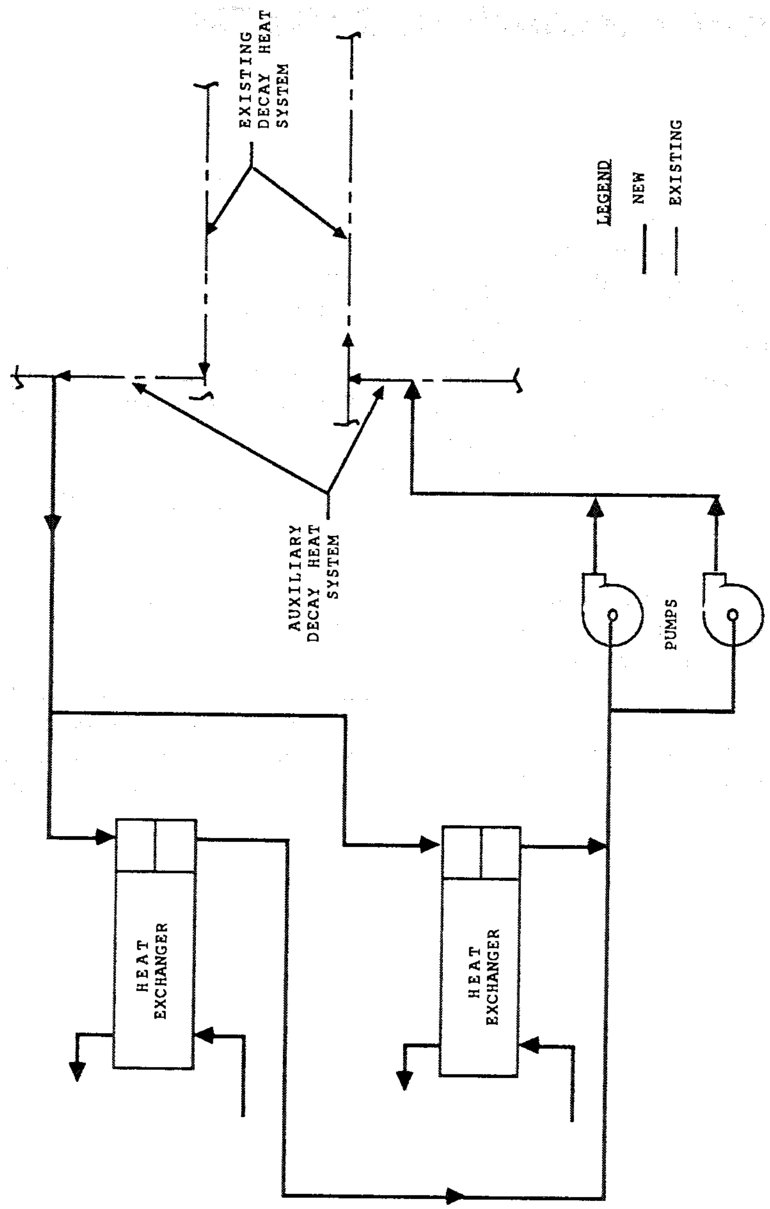


Figure F-1. Mini-Decay Heat Removal System

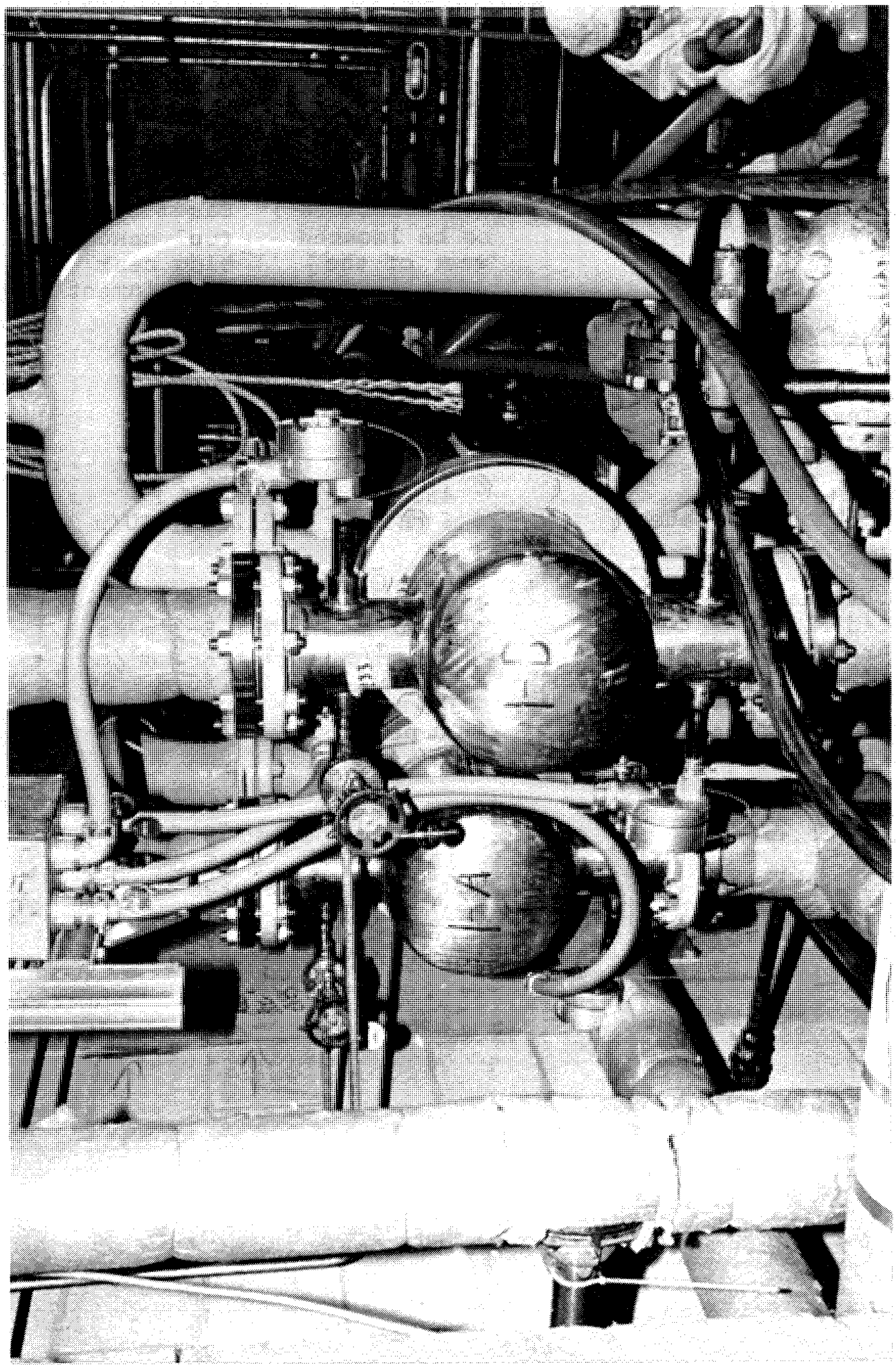


Photo F-1. MDHR Heat Exchanger

Table F-1. Comparison of ADHR and MDHR

Feature	ADHR	MDHR
Location:	New structure to be located west of the AFHB	Basement of fuel handling bldg near containment
Closed Cooling Water:	ADHR subsystem	NSCCW
Heat Exchanger:		
No. Installed	2	2
Shell Design Press.	175 psig	175 psig
Tube Design Press	600 psig	235 psig
Design Temp	450 K	366 K
Heat Transfer Rate	28.6 mBTU/h	2.25 m BTU/h
Pump:		
No. Installed	2	2
Rated Flow	0.2 m^3/s	7.6 E-3 m^3/s
Operating Flow	>0.13 m^3/s	7.6 E-3 m^3/s
Discharge Head	107 m	60 m
Motor Horsepower	300 kW	11 kW
Design Pressure	600 psig	240 psig
Design Temperature	450 K	366 K
Pipe:		
Inlet Size	20 cm	5 cm
Outlet	15 cm	5 cm
Weld Type	Butt welds	Socket welds
Code Class	ANSI B31.7	ANSI B31.1
Material	304 SS	304 SS
Seismic Class	OBE	OBE

Appendix G

MINI-CONDENSATE SYSTEM

The condensate booster pumps at TMI-2 had a capacity of 0.6 m^3/s. One condensate pump was providing water to the steam generator from the condenser. As the decay heat of the core decreased, the condensate pump had to be throttled to prevent over-filling the steam generator. The pump could have begun to suffer excessive wear as a result of cavitation and inadequate flow. To avoid the consequences of a potentially ruinous failure, new low-flow condensate pumps were installed in parallel with the existing pumps.

This alternate condensate system was begun in the late summer of 1979. It had three primary functions:

- Provide alternative condensate pumping capability to support the long-term steaming of steam generator "A" in the event of the failure of the existing condensate pumps
- Provide a demineralizer on the discharge of the alternate condensate pumps for polishing the condensate to remove suspended and dissolved impurities
- Provide a source of condensate to feed a temporary auxiliary boiler system.

The alternate condensate pump system was located in the turbine building sump pit at El. 278' and consisted of two redundant trains, each containing a small centrifugal pump and a full-flow condensate polishing demineralizer. Each pump took suction from a common header connected to the cold condenser hotwell via an existing drain line. The system discharged to the "A" steam generator control station.

If the installed condensate pumps failed, these new pumps could continue to supply cooling water to the steam generators for as long as necessary.

Appendix H

STANDBY PRESSURE CONTROL SYSTEM

Initially, two alternatives to inventory, pressure, and chemistry control of the reactor coolant system were pursued:

- An active pressure/volume control system—similar to the existing makeup system
- A passive reactor coolant solid system pressure/ volume control system—similar to the pressurizer.

Design of the active system was assigned on April 5, 1979. The passive system was combined with the active system on April 10. The original schedule called for the passive portion of the system to be completed by April 15 and for the active portion to be completed by April 30.

It soon became clear that no accessible pipes would permit both injection to and letdown from the reactor coolant system. The only lines that could be used for high-pressure letdown from the reactor were part of the existing makeup system that were severely contaminated, partially blocked, and required for the continued safe cooldown of the plant. The only lines available for high-pressure injection were the existing high-pressure safety injection lines and they had check valves in them, which made them unsuitable for letdown.

To expedite construction, the task was divided into two separate systems: 1) the injection portion of the original system consisting of a remote pressurizer, and injection and chemistry control subsystems; and 2) a letdown and purification system. This permitted the work on the makeup portion to proceed while allowing the engineers more time to work on the letdown portion of the system. The letdown portion was eventually cancelled due to the difficulties involved in finding a suitable alternative to the partially blocked letdown line.

The engineering and design effort proceeded in two phases. Phase I was to develop an operational system that used local instruments and manual controls. Phase II was to automate the control function of the system and provide a control panel in the control room. This was done when it was realized that the longest lead time activity was buying and installing the remote controls and instruments in the control room. Dividing the system construction into two phases permitted the system to be available to the plant at the earliest possible time.

The heart of the SPCS was three 3400-L surge tanks arranged in series. The first two tanks were maintained full; the third tank was partially filled with a nitrogen overpressurization. This was done to minimize the potential of getting noncondensible gases into the reactor coolant system. The nitrogen cover gas was provided from two banks of six 1.8-m^3 capacity cylinders. These tanks fed a common manifold and the pressure was controlled by one of four pressure regulators (two 100% automatic and two 100% manual regulators).

The surge tank level was maintained by a variable speed charging pump, which had a capacity from 1.2E-04 to 6.3E-04 m^3/s, or by one of two redundant 2.5E-03 m^3/s charging pumps. The charging pumps took suction from a 19,000-L capacity storage tank that was filled with borated water from a borated water batching tank.

The bulk of the SPCS was located in the new fuel storage pool in the fuel handling building (El. 331'). The charging water storage tank, the borated water batching tank, the borated water transfer pump, and the nitrogen supply cylinders were located on the north end of the operating elevation of the fuel handling building (El. 347'). This area was selected because it was close to the tie-in point, was readily accessible for construction crews, and was inside an existing building.

The tie-in point for the injection line was in the pipe chase in the basement of the fuel handling building (El. 282'). The general area dose rates in the pipe chase were 20 R/h. This was a difficult location but could not be avoided. The problem was further complicated when the designers were informed that makeup pump "1C" might be used in the future. Since the tie-in point could not be isolated from the discharge pressure from pump "1C", the SPCS tie-in piping needed to be protected. The tie-in was made using prefabricated spool pieces and a minimum of welds in the pipe chase. Portable shield walls and system flushing were used to control the dose rates during the tie-in.

Appendix I

CONTAINMENT KRYPTON-85 VENTING

Preparatory Steps: The TMI-2 containment purge required use of two systems, one for slow venting and the other for more rapid venting. For slow venting, the project team modified the existing hydrogen control system (HCS); for rapid venting, the "B" train of the containment air purge and purification system was modified. Both systems sent the gas to the station vent, through which all releases were monitored. Use of one train versus the other was based on the in-containment krypton concentration and on existing meteorology. For either route, the flow rate upper limit was not to exceed an offsite integrated dose of 15 mrem beta skin or 5 mrem total body, or an offsite dose rate more than 3 mR/h beta or 1 mR/h whole body.

Prior to venting, HCS modifications included the following. The original HCS exhaust fan, which has a capacity of 7E-02 m^3/s, was replaced with a larger fan with a design flow equal to that of the filter train; i.e., 4.7E-01 m^3/s. The purge flow rate, originally controlled by a throttle valve, was modified to include an additional valve that provided fine control of the flow rate over the entire 4.7E-01 m^3/s flow range. Interlocks were added to the modified HCS to provide rapid isolation in case of equipment failure or high radiation concentrations at the fan discharge. Instruments were added to indicate, record, or alarm filter train differential, pressure, exhaust flow rate, and effluent radiation levels.

To compensate for a potential recirculation problem between the modified exhaust line and a containment makeup air line, the building's air cooling fans were operated continuously during the two-week venting project.

One of the most important aspects of the purge project was the project's ability to accurately determine the radioisotopic inventory of the building atmosphere and to measure precisely the effluent radioactivity concentrations. There were two modes of measuring this: the sampling system that existed before the accident; and the glove box that had been installed in a containment penetration (R-626) after the accident to collect atmospheric samples and test re-entry equipment before deployment.

Samples could be collected from either location for use in gas, particulate, radioiodine, or tritium analyses. The sampling system drew samples from two points in the containment, located approximately 3.7 m east and west of the north-south centerline of the containment dome (El. 469'). However, following the accident, the project team was unable to determine whether the drain valves located on the sample lines inside the containment were open or closed, thus they were unsure whether samples were being drawn only from El. 469' or were also being drawn from the area near the drain valves, at approximately El. 317'. The uncertainty was alleviated by installing a new sample line that tied the existing sample panel to the containment pressure sensing line through its isolation valves at El. 354'. Subsequent samples were drawn from both the original sample lines and the new line.

The stack monitor, which was tested and calibrated prior to the purge, was the official instrument used to record the radiation releases during venting. The readout was on the turbine deck, just outside the TMI-2 control room. Readout capabilities included: instantaneous, 10-min averages, hourly averages, and daily averages for beta particulates, iodine, and gaseous activity in effluents. However, because of a series of false alarms from the stack monitor, the project team first attempted to reprogram the monitor and, when this proved insufficient, installed two other particulate monitoring systems.

The first was a bypass particulate grab sample system, whereby air was pulled from the stack sample line through a particulate filter that was removed for immediate analysis and replaced at 15-min intervals during venting. The second system, which required NRC approval before its use, was more complex and provided instantaneous readings of particulates being released every 1000 seconds. This second system included a sodium iodide (NaI) detector, which transmitted to a single-channel analyzer that could distinguish krypton-85 gamma activity from that of other isotopes by reading cesium-137 energy peaks. Readout from this system was adjacent to the stack monitor readout at the turbine deck. The alarm signal was set as the particulate release rate (i.e., the difference between current and previous readings) of 150 counts, which corresponded to a stack concentration of 5.8E-10 μCi/cc, or one-tenth the instantaneous particulate release rate technical specification limit. Once the real-time particulate monitoring system was approved, the bypass particulate sampling system was used as a backup.

The gamma/Geiger Mueller (G-M) monitor, which was the normal containment purge unit area radiation monitor, was temporarily mounted near the modified HCS filter train and exhaust fan to detect leaks during the purge. For this purge task, the alarm was set at 10 mR/h. The monitor had a local readout and alarm signal, and also indicated, recorded, and alarmed in the control room.

The plant vent stack cap was removed and the supplemental ventilation system atop the auxiliary building was shut off. Prior to removing the stack cap and terminating the supplemental ventilation system service, workers inspected all ductwork between the inlet of the exhaust fans and outlet of the filters for potential leak paths, and sealed and tested all leaks; re-balanced the AFHB ventilation systems to provide correct flow rates; re-tested the AFHB exhaust filters to comply with the technical specifications; and retested the service building HEPA filter.

A computer routine was developed to calculate the allowable venting flow rate in real time. It performed atmospheric dispersion and radiological exposure calculations and computed accumulated beta skin and whole body doses for each of the 16 sectors around the plant. The routine used meteorological data, supplied at 15-min intervals, and actual monitored release rates. GPU also augmented its Radiological Environmental Monitoring Program (REMP) to directly monitor offsite radiation levels. This had never been done at TMI before the accident.

Personnel Requirements. Personnel requirements for the actual venting effort involved a core staff of 36, divided into 12-h shifts. Others were associated with pre-venting planning and analyses and reporting of results after venting. One control room operator was dedicated to the venting and conducted all venting operations; one assistant operator was dedicated to monitoring readouts on the turbine deck. They used two-way radios to communicate. Shift foremen supervised all venting operations.

Shift engineers were responsible for monitoring and ensuring safe conduct, maintaining a complete record of the krypton discharge, and informing the environment assessment command center of venting flow rate changes and in-containment krypton concentrations, etc. NRC staff were also on round-the-clock oversight duty.

Each person associated with the containment purge was required to take training classes and to read and understand the operating procedure, functional test procedures, and the flow print. Practical walk-throughs were performed using the operating procedure as guidance. Oral exams were used to ensure that staff understood the venting system.

Hundreds of thousands of dollars were spent on radiological environmental monitoring equipment, particularly the radiation environmental laboratory.

Monitoring and Analyses. Once venting commenced, station vent particulate filters and charcoal cartridges were changed daily. Particulate filters were submitted for gross beta, Ge(Li), gross alpha, and strontium-89/90 analysis. Charcoal cartridges were submitted for Ge(Li) analysis only. In addition, daily gas samples of the effluent taken.

The environmental monitoring that began during the purge was the most diligent program of its kind ever undertaken by the industry. Daily news releases were issued by the EPA, GPU, and the Pennsylvania Department of Environmental Resources. EPA and NRC held daily news conferences to report the findings of the Citizens Radiation Monitoring Program, a local group that was formed before the venting activities but did lent additional credibility to the program.

Fifty residents of the surrounding area volunteered to wear TLDs during the 14-day venting program. Based on the TLD readings, the total gamma doses to which these individuals were exposed during the period were within normal background range.

Three TLD systems were used to monitor environmental releases off site during the purge. These TLDs were used only during the actual venting period. Locations for the TLDs were based on population and meteorological parameters.

Onsite monitoring commenced as well to ensure worker safety. The shutdown criterion was set at a dose rate equivalent to 10 mRad/h (beta) to an individual outside the protected area. Onsite surveys were conducted round-the-clock at least once per hour from June 28 to July 1, 1980. These data were used to characterize onsite radiological parameters, particularly in terms of the purge rate and the TMI site meteorology. As of July 1, the onsite monitoring program was relaxed, and downwind survey measurements were taken every four hours.

Appendix J

TRANSCRIPT FROM FIRST CONTAINMENT RE-ENTRY

This transcript is from the first postaccident entry made into the TMI-2 containment on July 23, 1980. The two technicians making the entry were M. Benson and W. Behrle. The 22-minute entry began with Benson opening the containment airlock. Difficulties with the communications equipment account for some to the blank times or misunderstandings.

Benson: Benson to Base—I'm ready to turn the handwheel, over.

Command: Roger, out.

Benson: Benson...the airlock is equalizing, over...now...over.

Benson: Benson to Base—Still equalizing, over.

Command: Roger, out.

Behrle: Behrle to Benson—I'm ready Michael, over.

Command: Base to Behrle—Are you commencing to open the door now? Over.

Behrle: Behrle to Base—The inner airlock door is opening, over.

Command: Roger, you're on the clock, over.

Behrle: Behrle to Base—I read 400 mrem about 6 ft [1.8 m] inside the building, head height, over.

Command: Roger, out.

Behrle: Behrle to Base—I have entered to building, over.

Command: Roger, out.

Behrle: Behrle to Base—A swipe area reads 250 mrem gamma.

Command: Base...This is base—We did not copy, over.

Behrle: "A" swipe area reads 250 mrem; "B" swipe area, swipe area reads 2 rem.

Command: Roger, we copy.

Benson: Benson to Base—I have taken the "A" and "B" swipes; the readings are one rad, over.

Behrle: Behrle to Base—The red area reads between 400 and 600 mrem, that is shoulder height to floor, on contact with floor they all read about the same, over.

Command: Roger, we copy.

Benson: Benson to Base—Do you copy? over.

Command: We copied the last transmission, over.

Benson: I have taken the A, B, and C swipes; I am taking D, over.

Command: Base did not catch the last end, repeat please, over.

Behrle: Behrle to Base—The floor drain on contact next to the ramp reads about 5 rem on contact, over.

Benson: Benson to Base—Do you read me? Over.

command: Roger, we copied.

Benson: I have taken all four swipes, over.

Command: Roger, we copy; you took all four swipes, over.

Benson: The surface readings for all the swipes are one to two rads, over.

Command: Roger, one to two rads, over.

Benson: The general area still reads zero rads (words?), over.

Command: Base to Benson—We need a digital on you and Behrle, Benson first, over.

Behrle: Behrle to Base—My digital reads 23 mrem, over.

Command: Base to Benson—We did not get a transmission out of you. Could you give us your dosimeter, over.

Benson: Benson to Base—19 mrem, over.

Benson: (Word?) That's 19 mrem, over.

Command: Roger, 18 mrem, over.

Behrle: Behrle to Base—The high pressure injection line above the D -ring reads about 3 rem on contact.

Command: Roger, we copy the 5 rem, we don't know the area, over.

Behrle: Behrle to Base—The D-ring reads about 300 mrem on contact at 5 or 6 different locations, over.

Command: Roger, that's the yellow area. By the way, you are 5 minutes 35 seconds into entry, over.

Behrle: Behrle to Base—It's the red to yellow area, over.

Command: Did not copy, over.

Command: Base to Benson, Base to Benson—Do you have any beta readings? Over.

Benson: Benson to Base—At the edge of the red area on contact with the floor I am reading 4 rad, over.

Command: Roger, we copy. Four rads—edge of the red.

Benson: In the general vicinity, in the area it's 1 rad, over.

Behrle: Behrle to Base—The general area radiation levels in orange are about 500 mrem, over.

Command: Roger, we copy. You are now 7 minutes 22 seconds into entry, please come back with your dosimeter readings, over.

Behrle: Behrle to Base—My dosimeter reading is 43 mrem, over.

Command: Roger.

Behrle: Behrle to Base—Benson's air pressure reading is 1800 pounds, over.

Benson: Benson to Base—I read 40 mrem, over.

Command: Roger, we copy.

Command: Base to Behrle—Was that the elevator door? Over.

Behrle: Behrle to Base—The floor drain in yellow—reads 2 rem, over.

Behrle: Behrle to Base—The stairwell door has been blown open by the explosion, over.

Command: Base did not copy, over.

Behrle: The stairwell door by the elevator has been blown (word?) open by the explosion. It reads 400 mrem, over.

Command: Roger, we copy, over.

Command: Base to Benson—Are you getting pictures? Over.

Behrle: The radiation reading in the stairwell is 8 rem, 8 rem, over.

Command: Roger, we copy, 8 rem in the stairwell. Are you getting pictures? Over.

Benson: Benson to Base—I have taken several pictures. The reading in the stairwell is 4 rads, over.

Command: Roger, we copy. Four rads in the stairwell, over.

Command: Base to Benson and Behrle—You are now 10 minutes 22 seconds into entry. We need your dosimeter readings, over.

Behrle: Behrle to Base—My dosimeter reads 73, 73 mrem, over.

Command: Roger, 43 mrems, over.

Benson: Benson to Base (interruption by Behrle: 73 mrem, over)

Command: We had dual transmission. Was that 83 mrem, Behrle? over.

Behrle: Affirmative, over.

Benson: Benson to Base—I have 60 mrem, over.

Command: Roger, we copy—60 mrem, over.

Darryl: Osdon to Base—I have completed wiping down the inner reactor door and seal area.

Command: Base—We could not understand the last transmission and we don't know who it came from, over.

Darryl: Osdon to Base—I have completed wiping down the inner reactor door and seal area.

Command: Roger, we copy.

Behrle: Behrle to Base—The highest reading over the hatch, over the hatch is 10 rem, over.

Command: Roger, 10 rem, over.

Benson: Benson to Base—The beta reading over the hatch is 6 to 7 rad, over.

Command: Roger, we copy—67 rads.

Behrle: Behrle to Base—At the edge of the hatch it is 4 rem, over.

Command: Base could not copy, try again, over.

Benson: Benson to Base—The beta reading over the hatch is about 7 rad, over, over.

Command: Roger, we copy.

Behrle: Behrle to Base—The flood pipe reads 3 rem on contact, over.

Command: Roger, we copy. You are now 13 minutes 10 seconds into entry. We would like a dosimeter reading, over.

Behrle: Behrle to Base—108 mrem, over.

Command: Roger, we copy.

Command: Base to Benson—We need a digital dosimeter on you, over.

Benson: Benson to Base—116 mrem, over.

Command: Roger, we copy, 115, over.

Behrle: It is 1400 mrem at the aircooler.

Command: We did not copy, please repeat.

Behrle: Air cooler.

Command: We copy, 115 mrads, I mean mrem beta.

Behrle: Behrle to Base—General background radiation readings in blue is 700 mrem, over.

Command: Roger, we copy, over.

Behrle: Behrle to Base—Ramp? on contact is reading 1200? mrem.

Command: Roger, we copy.

Command: Base to Benson and Behrle—You are now 16 minutes 3 seconds into entry, please give us your dosimeter readings.

Behrle: Behrle to Base—The D-ring on contact reads 400 mrem, over.

Command: Roger.

Behrle: Behrle to Base—My digital dosimeter reads 135 mrem, over.

Command: Roger, we copy.

Command: Base to Benson—We need your dosimeter reading, over.

Behrle: Behrle to Base—The floor drain in blue reads 8 rem, over.

Command: Roger—Benson, we need your dosimeter, over.

Benson: One hundred forty three, over.

Command: Roger, Benson, we copy, 143 mrem.

Command: Base to Benson and Behrle—Start procedure to exit the building, over.

Behrle: Behrle to Base—Affirmative, over.

Command: Roger, we copy your acknowledgment, over.

Benson: Benson to Behrle—Don't forget the light, over.

Command: Base did not copy, come back, over.

Benson: Bill, don't go around behind the core flood tank.

Command: Base to Benson and Behrle—You are now 18 minutes 45 seconds into entry, over.

Command: Base to Benson and Behrle—Please notify us when you get to the airlock, over.

Command: Base to Benson and Behrle—Please give us your location, over.

Command: Base to Darryl, Base to Darryl—Where is the entry team? Over.

Darryl: Benson and Behrle on (word?) for the airlock commencing to shut the outer door.

Command: Roger, we copy, you're getting ready to shut the outer door. Is that affirmative? Over.

Benson: Benson to Base—My dosimeter reading is 172 mrem, the other....

Command: Base to Behrle—What is your dosimeter reading? Over.

Command: Base to Behrle, Base to Behrle—What is your dosimeter reading? Over.

Behrle: One hundred seventy six.

Command: Roger, we copy—176.

Benson: Benson to Base—My digital dosimeter on my left arm is 175 and the right arm dosimeter reads 216, over.

Command: Base to Behrle—Keep it in your head, we can't copy all the information, over.

END