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TMI-2 TECHNICAL INFORMATION AND EXAMINATION PROGRAM 1983 ANNUAL REPORT

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ABSTRACT

The Department of Energy's Technical Information and Examination Program at Three Mile Island Unit 2 continued the research and development work begun on the Island in 1980. The work concentrated in six major areas: waste immobilization, reactor evaluation, data acquisition, information and industry coordination, core activities, and EPICOR II and waste research and disposition.

The program assists in resolving specific problems at TMI-2 while developing techniques and broadening understanding of accident consequences to improve the overall safety and reliability of nuclear power. The Technical Information and Examination Program aims to communicate applicable information to the nuclear power industry to ensure that the industry can avail itself to the maximum amount of information possible.

ACKNOWLEDGMENTS

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TMI-2 TECHNICAL INFORMATION AND EXAMINATION PROGRAM 1983 ANNUAL REPORT

PROGRAM HISTORY AND PURPOSE

1983 marked the fourth year of operation of the Department of Energy's (DOE) Technical Information and Examination Program (TI&EP). This program was created to learn from the data acquired and techniques developed during the recovery of the Three Mile Island Unit 2 (TMI-2) nuclear power plant. In March 1980, the GEND group established the TI&EP to obtain and distribute information which would contribute to the industry's knowledge of accident sequences, and to coordinate research on ways to respond to the technical challenges associated with this effort.

The plant at TMI continues to be a focus of significant national and international attention. Nearly every system and component in the Reactor Containment Building and support facilities was affected in some way by the accident and thus recovery activities involve every major building and numerous systems and components.

During four years of operation, the TI&EP has played an important role in the progress of the recovery at TMI-2 pursuing its goals of obtaining, developing, and distributing vital information to its target audience. General Public Utilities Nuclear Corporation (GPU Nuclear) took its first major step to recovery when, in July 1980, it succeeded in safely venting 44,000 Ci of ^{85}Kr from the Unit 2 Reactor Building environment. This achievement paved the way for the frequent manned entries into the building needed to accomplish the cleanup tasks ahead. By August 1980, GPU Nuclear had decontaminated water which had collected in the Auxiliary and Fuel Handling Buildings as a result of the accident. Through interagency agreements between the NRC and DOE, the TI&EP acquired high specific activity waste canisters as resource material to complement ongoing research and disposition programs in radioactive waste management.

Since the TI&EP was created, it has obtained and analyzed data on the accident and its aftermath and distributed that data to the nuclear power industry. The TI&EP's primary data gathering and distributing arm has been the Data Acquisition Program (DAP).

All major activities of the TI&EP are documented in formal technical reports distributed through the established DOE report publication mechanism. Program findings are communicated directly to the industry by computer conferencing and information networks, engineering society meetings, and nuclear industry conferences.

The DAP activities have contributed to the progress of the cleanup while obtaining information of benefit to research and development (R&D) programs aimed at improving reactor safety and reliability. Some of these programs have included extensive research into the possible causes and characteristics of the hydrogen burn incident, examination of the survivability of vital safety and control instrumentation and electrical equipment, and in-depth studies of the fission product migration paths during and after the accident.

In October 1981, the TI&EP initiated R&D programs based on the technical challenges related to the recovery and cleanup. Those two programs are referred to as: the Waste Immobilization Program and the Reactor Evaluation Program. The Waste Immobilization Program was designed to focus on disposition of abnormal wastes generated at TMI-2 as a result of the accident and recovery activities. The Reactor Evaluation Program was chartered with investigation of the various reactor disassembly techniques including head removal, plenum removal, and fuel removal as well as characterization/data acquisition necessary to support the disassembly.

While Auxiliary Building water processing was underway, GPU Nuclear and its contractors with the assistance of several DOE contractor experts, developed an effective technique for processing the more highly contaminated water in the Reactor Building basement and Reactor Coolant System (RCS). Per the NRC-DOE Memorandum of Understanding, DOE agreed to use the wastes generated by this cleanup system in its research programs, and thus provided a disposition mode for the wastes, allowing the NRC to permit GPU Nuclear to begin processing. Cleanup of the basement water, as well as processing contaminated water from other locations in the plant, marked another significant milestone toward total cleanup which was accomplished with TI&EP assistance. The waste containers generated by this processing are now part of a rigorous DOE program to examine effective ways of immobilizing and disposing of high specific activity nuclear plant wastes.

The main focus of the TMI-2 cleanup now is the characterization, examination, and ultimate removal of the reactor core and fuel. GPU Nuclear and it's contractors have been actively involved in evaluating ways to disassemble a reactor under such conditions. Much of the R&D work performed by GPU Nuclear, it's contractors, and TI&EP scientists and engineers, has centered around the actual condition of the core and dispersion of fission products resulting in new data on reactor behavior in the event of an accident such as the one at TMI-2.

The year 1983 saw major accomplishments toward the goal of successful fuel removal. The shape of the damaged core and internals was successfully mapped using the Core Topography System. The first actual samples of damaged core material were obtained when engineers from DOE's Reactor Evaluation Program entered the Unit 2 Reactor Building in September and October. During the summer, the TI&EP and GPU Nuclear reached a major milestone in the TMI-2 recovery program when the last of 50 EPICOR II canisters and the last Submerged Demineralizer System (SDS) liner were shipped from the island.

But perhaps more important, the DOE programs in 1983 while assisting in moving the entire cleanup effort closer to a successful conclusion, demonstrated that every new challenge this unique situation presents can be met. Today's nuclear technology can get the job done. In addition, DOE's Programs made significant progress toward a more knowledgeable, more reliable nuclear power industry.

SIGNIFICANT ACCOMPLISHMENTS IN 1983

With the accomplishments of 1983, the TI&EP's R&D program at TMI continues to move steadily forward. Each step of the way, important questions about the nature and impact of the accident have been answered.

Waste Immobilization Program

One area in which the TI&EP has demonstrated significant progress is its Waste Immobilization Program. In late August, the last container used to decontaminate accident water was shipped from TMI, marking a milestone in the program. A total of 63 highly radioactive containers have left the Island since May 1981, taking with them 95% of the radioactivity released from the RCS during the accident. These containers have gone to DOE-sponsored facilities for research in safe disposal methods for the wastes.

Fifty of the radioactive containers that have left TMI, called EPICOR II prefilters, used mostly organic materials to decontaminate more than 1900 m^3 of accident water from the Unit 2 Auxiliary and Fuel Handling Building. Sent to the Idaho National Engineering Laboratory (INEL), these prefilters were examined for short- and long-term effects of ionizing radiation on various types of ion exchange media and on the canisters containing those media.

Three other containers, called SDS liners, were used in a study of vitrification, whereby the inorganic media that removed the radioactive products were mixed with chemicals, heated, and then cooled to form glass logs. Tests on these glass logs have proven them capable of successfully immobilizing the radioactive contaminants.

The remaining ten SDS vessels have been sent to Richland, WA, for a monitored burial demonstration program. The vessels were loaded into concrete overpacks or high integrity containers. One of them will be monitored for pressure, temperature, moisture, and fission product particulates—to determine the long term performance after burial.

Also in 1983, engineers continued to characterize the demineralizer vessels in the TMI-2 Makeup and Purification System. They sampled contents from both the A and B vessels, as well as from the void above the resin beds. Liquid and solid samples were analyzed at the Oak Ridge National Laboratory (ORNL), after which a plan was developed to elute the high activity radionuclides and then package the less radioactive vessels' contents.

Reactor Evaluation Program

Programs toward head and plenum removal also met with success in 1983. Engineers took radiation surveys in the area below the reactor vessel head and used video cameras to inspect the underside of the head and top of the plenum for debris which would make head removal difficult. They also took samples from the head and plenum areas to analyze the makeup of the particles found there. Sample material fines from the upper plenum cover were subjected to spark, flame ignition, and impact tests and were found to have no pyrophoric reaction. The resulting information confirmed plans to remove the reactor vessel head without flooding the area with water for safety reasons.

Major advances were also made in designing the tools to inspect and remove the plenum assembly. Designs for the plenum inspection tools were approved, and fabrication was started. Engineers also started design reviews for equipment to lift the plenum. Other tooling that reached the preliminary design phase included plenum transfer and storage hardware, equipment to knock off partial fuel assemblies hanging from the plenum, and cleaning equipment.

In other Reactor Evaluation Program supported activities, efforts were initiated to restore spent fuel pool A in preparation to hold the fuel during shipment preparations. Efforts included evaluating health and safety aspects of removing the tank farm from the fuel pool, surveying all potential sources of contamination to personnel, and decontaminating the tank farm and preparing temporary storage areas for decontaminated components.

In activities toward fuel and core debris removal, a water cleanup system concept was developed for which proof-of-principle tests were conducted, and preliminary designs were proposed for fuel removal tools and fuel canisters.

Data Acquisition Program

As part of the Data Acquisition Program, the instrumentation and electrical components were evaluated to determine the effect of actual accident and postaccident environmental conditions on specific safety instruments, electrical systems, and selected plant hardware. Engineers are using their findings specifically to learn how these conditions affect the ability of a system or component to function during and after an accident. Motor-operated valves, air cooling fans and their associated inlet damper limit switches, pressurizer heaters, solenoid valve assemblies, and pressure switches were among the targets of this study.

In an ongoing task, Instrumentation and Electrical (I&E) engineers have been gathering information on how the cables and connections in Unit 2 have performed under accident and postaccident conditions. Many of the components tested to date have shown abnormalities caused mainly by water damage. Test results will help engineers define the extent to which cable and connection problems affect the rest of the reactor plant systems and will help to improve cable and connection design, manufacturing, and installation procedures.

Major achievements have also occurred in the refurbishment of the polar crane. After nearly a year of inspections, analyses, repairs, replacements, and no-load operational testing, the polar crane is structurally, mechanically, and electrically ready to be load tested.

In another major area of Data Acquisition, Radiation and Environment (R&E) Program engineers worked toward defining the source term in Unit 2. Source term is the concentration and distribution of radionuclide activity. Engineers measured the concentrations of certain fission products deposited on building surfaces. The amount of nuclides was a very small percentage of the total core inventory.

An important R&E project completed in 1983, was the study of the hydrogen burn event. When the TMI-2 reactor core overheated, the zirconium encasing the fuel reacted with steam, liberating large quantities of hydrogen to the building, where it later ignited. This event, called the hydrogen burn, was found to cause little damage to the Reactor Building and no damage to the safety systems. The damage was fully consistent with TI&EP expectations. Also as a result of their studies, engineers are confident that no explosion occurred.

Information and Industry Coordination

Serving as a link between the TI&EP and the nuclear power industry, the Information and Industry Coordination (I&IC) group in 1983 used several technology transfer mechanisms to inform well over 5000 nuclear industry professionals of progress being made at TMI and DOE laboratories. Technological developments from DOE programs have altered industry perspectives on many issues confronting the nuclear community today. During the year, the I&IC transmitted information on improper installation practices, equipment degradation caused by moisture intrusion, waste management techniques, and worker heat stress in radioactive environments. I&IC involvement with industry regulatory agencies and standards setting groups has brought TMI-2 data to bear on equipment qualification standards and accident damage prediction codes.

INEL TMI-2 Programs

The INEL was the site of considerable progress in TMI-2 R&D programs, especially in the areas of core data acquisition and EPICOR II liner disposition.

Three control rod drive lead screws removed from TMI-2 were examined at the INEL for data on the plateout of fission products and on peak temperatures experienced during the accident. Also in 1983, TI&EP engineers took a major step toward analyzing the makeup of the damaged core. Using equipment designed by EG&G Idaho, the first actual samples of the rubble bed were taken from inside the reactor.

One of the most revealing tasks of the year resulted in the first accurate map of the inside of the damaged core area. A sonic sensing head used ultrasonic signals to obtain hundreds of thousands of data points, which model makers reconstructed into a three-dimensional model. The core topography system model will allow engineers to more carefully plan the removal of the damaged core internals.

Efforts at the INEL also have been instrumental in furthering the EPICOR II and Waste Research and Disposition Program. Three of the EPICOR II prefilters have been used in experiments to immobilize their contents; the remainder will be buried in special high integrity containers (HICs) at commercial disposal facilities. A prototype container underwent a number of integrity tests, including being loaded with sand and dropped on to soil similar to that found at the commercial disposal facility near Richland, WA. Last September, one of the EPICOR II liners was retrieved from temporary storage and sealed in one of the HICs and is now awaiting authorization for permanent commercial disposal.

WASTE IMMOBILIZATION PROGRAM

The Waste Immobilization Program includes three major activities of interest. These three are the Zeolite Disposition program for shipping and examining ways to dispose of SDS zeolite ion exchange beds, the Abnormal Waste Technology program for characterizing special wastes at TMI and developing methods for removing them, and the Resin Disposition program for characterizing and examining ways to dispose of resins from the EPICOR II water processing system. DOE will use special TMI-2 wastes in R&D work with benefits applicable to the nuclear industry.

Zeolite Disposition

The SDS was developed to process accident-generated water predominantly contaminated with $^{134,\ 137}Cs$ and ^{90}Sr. Commercial nuclear waste processing companies developed the SDS for GPU Nuclear with technical assistance from TI&EP-coordinated DOE laboratory personnel at ORNL and Savannah River Project. The SDS uses inorganic zeolites to remove radionuclides from the contaminated water and concentrates them in a form which is suitable for safe shipment to DOE Laboratories for R&D disposition programs.

By the end of 1983, 14 vessels from the SDS were expended and shipped off the Island. The characterization by curie content of these vessels is shown in Table 1. These vessels were continually vented during storage at TMI because hydrogen and oxygen gases were generated from radiolysis of residual water in the vessels. The vessels had to be prepared with catalyst for gas recombination, since the generation rate could produce unacceptable concentrations during shipment.

The 1982 tests by Rockwell Hanford Operations (RHO) determined that platinum-palladium catalyst pellets coated with alumina oxide would safely recombine these gases back into water. A vacuum outgassing system, developed by Westinghouse Hanford Company (WHC), was installed at TMI to remove residual water in the vessels to ensure that the catalysts would function during a shipping accident involving vessel inversion. A tool developed by RHO was used to remotely connect the submerged liners to the vacuum system and then

Table 1. Expended SDS vessel curie loading as of September 1983

Vessel Number	$^{134,137}Cs$ (Ci)	^{90}Sr (Ci)	Total Cs + Sr (Ci)	Total w/daughters (Ci)
D10011	44,317	2,061	46,378	88,158
D10012	57,176	2,003	59,179	112,635
D10013	49,281	1,974	51,255	9[illegible],151
D10014	28,534	2,585	31,119	59,761
D10015	5,767	1,012	6,779	12,896
D10016	57,156	1,869	59,025	112,622
D10017	30,312	1,021	31,333	59,542
D10018	2,451	309	2,760	5,273
D20022	2,599	1,006	3,605	7,198
D20026	3,947	1,074	5,021	9,692
D20027	7,084	8,124	15,208	29,732
D20028	43,333	1,660	44,993	86,334
D20029	1,792	4,626	6,418	12,705
D20031	546	3,512	4,058	8,082

to add catalyst pellets to the Johnson screen cup below the vessel's vent port (see Figure 1). The 1983 SDS vessel shipments from TMI were all prepared using the vacuum outgassing and catalyst addition method developed by the DOE laboratories.

To prepare each vessel for shipment, GPU Nuclear removed the vessels from their underwater storage locations to an underwater station for the remote dewatering processes required before shipping. The first step involved forcing water out of the vessel using nitrogen gas. This procedure removed any free standing water in the bottom of the vessel. The next step was to connect the vessel to the vacuum outgassing system (see Figure 2) using the RHO tool (see Figure 3). This system removed additional water, called "sponge water," which was not chemically bound to the zeolite but which could, over time, drain down and accumulate at the bottom of the vessel. The vacuum outgassing process removed water by vacuum distillation and was performed to ensure that no free standing water would be in the vessel during shipment. The platinum-palladium catalyst used in SDS vessels will not function if submerged in water, and concerns that free standing water would cover the catalyst if a vessel were inverted during a transportation accident were eliminated by vacuum-outgassing free standing water in the vessels. After completing water removal, the catalyst pellets were added through the special RHO designed tool connected to the vessel which has a funnel for catalyst loading.

After the catalyst was added, the tool was removed from the vessel and the vacuum system was connected to the vessel's vent hose. A vacuum was

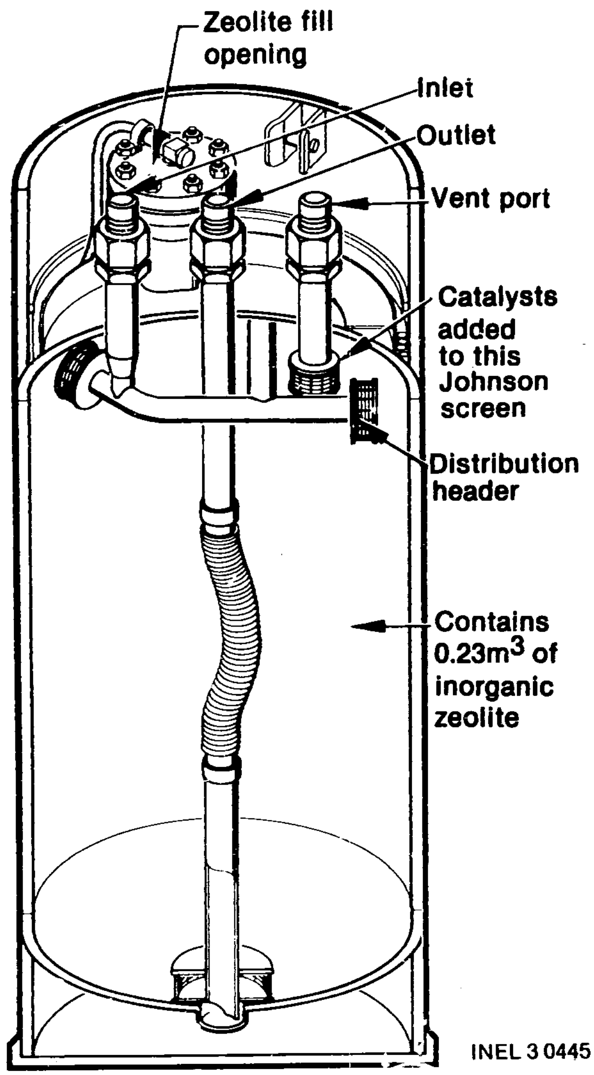

Figure 1. Cutaway view of an SDS liner showing Johnson Screen to which catalysts are added.

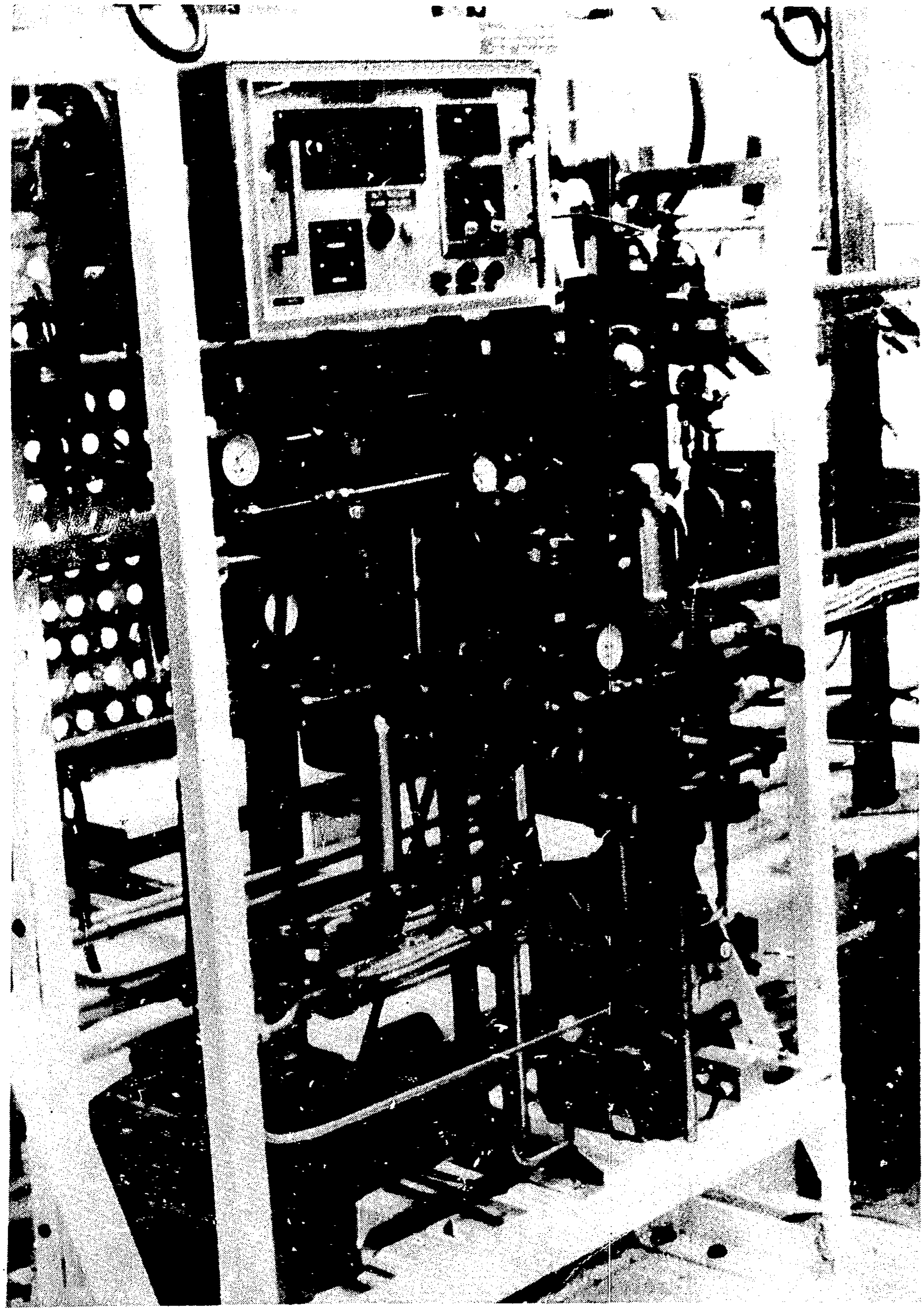

Figure 2. Vacuum system designed by Westinghouse Hanford Company.

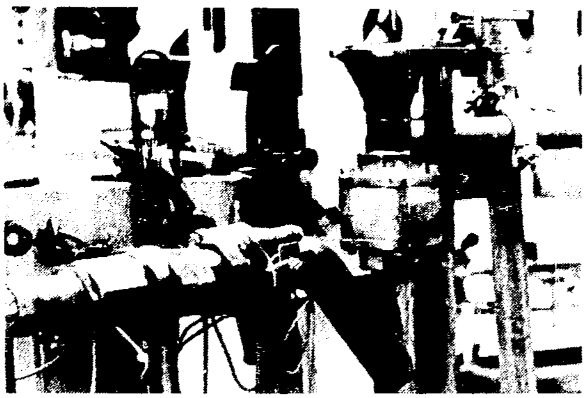

Figure 3. TMI catalyst addition/offgassing tool.

established and the vessel was isolated to monitor pressure. With the catalyst in the vessel, the hydrogen and oxygen was then recombined back into water. If the vessel's pressure did not increase (as it would have without the catalyst present) the catalyst was determined to be working properly. As a final check before shipment, samples were obtained and analyzed for hydrogen and oxygen concentrations to ensure that combustible gas mixtures were not present during shipment.

The last steps in preparing the vessels for shipment were the addition of a rupture disc and filter assembly at the end of the vessel's vent hose and installation of a lifting bail. The assembly was an added precaution for burial in the RHO Monitored Burial Demonstration Program. Nonstoichiometric gas generation could result in an increase in pressure in a vessel. The rupture disc would relieve this pressure by allowing the gases to pass through the filter assembly before being exhausted to the vessel's overpack. After the rupture disc and filter assembly was installed, the vessel was placed in the shipping cask, the vessel's vent hose was carefully coiled on top of the vessel, and the lifting bail for remote vessel removal was installed.

A major milestone of shipping the first 14 original SDS vessels was reached ahead of schedule in August when the last SDS liner used to decontaminate accident-generated water left the Island. The cumulative curies shipped in SDS vessels from TMI is shown in Figure 4. These highly radioactive vessels were used in DOE sponsored programs for research on safe disposal methods for the wastes. These programs are discussed in the following sections.

Zeolite Vitrification Demonstration Program. In December 1982 and January 1983, GPU Nuclear and DOE shipped the two highest curie content SDS vessels, D10012 and D10016, from TMI. These vessels went to Battelle Pacific Northwest Laboratory (PNL) at Richland, Washington for testing and disposition research.

These vessels, each loaded with 113,000 Ci of radioactive fission products, underwent studies at PNL to characterize the radioactive inorganic ion exchange media zeolites in preparation for vitrification. In the vitrification process, the zeolites and glass-forming chemicals are fed into a canister in a furnace, where the mixture is heated to approximately 1050°C, causing vitrification. After the mixture cools, the canister serves as the container for the final waste product, a glass column which is a stable form for the SDS zeolite waste. The TMI zeolites were vitrified and results indicate that the radioactivity is being successfully contained in the glass log. This procedure establishes vitrification's feasibility as a disposal option for TMI's highly loaded radioactive wastes.

Vitrification Demonstration. Following careful analysis of zeolite-to-glass-former ratios for each vessel's zeolites, PNL proceeded with mixing in glass formers with the zeolite for vitrification. The operations summary is shown in Table 2.

Vitrification operations on the first high-activity vessel (D10012) were started with 315 kg of feed to be processed. The feed rate with the first vessel (D10012), initially $\sim$10 kg/h, was slowed to $\sim$5 kg/h when the can was two-thirds full to prevent overfilling, and was reduced to 1 to 2 kg/h to fill the can. A total of 191 kg of feed was vitrified. A new canister was installed and the remaining 124 kg of feed left in the mixer were vitrified. After the second high-activity vessel (D10016) was dried and blended with glass-forming chemicals, this new canister was installed, and 180.5 kg of feed were processed.

The feed consisted of the zeolite and glass formers which were fed to the in-can melter system shown in Figure 5. During the vitrification process, the level of glass in the canister was verified by the total amount of feed fed ($\sim$30-kg feed/1-m glass) and by the use of the gamma level detector. A gamma detector probe was inserted in a hole in the cell wall and aimed at the canister in the furnace. Plugs were located at the four levels in the canister.

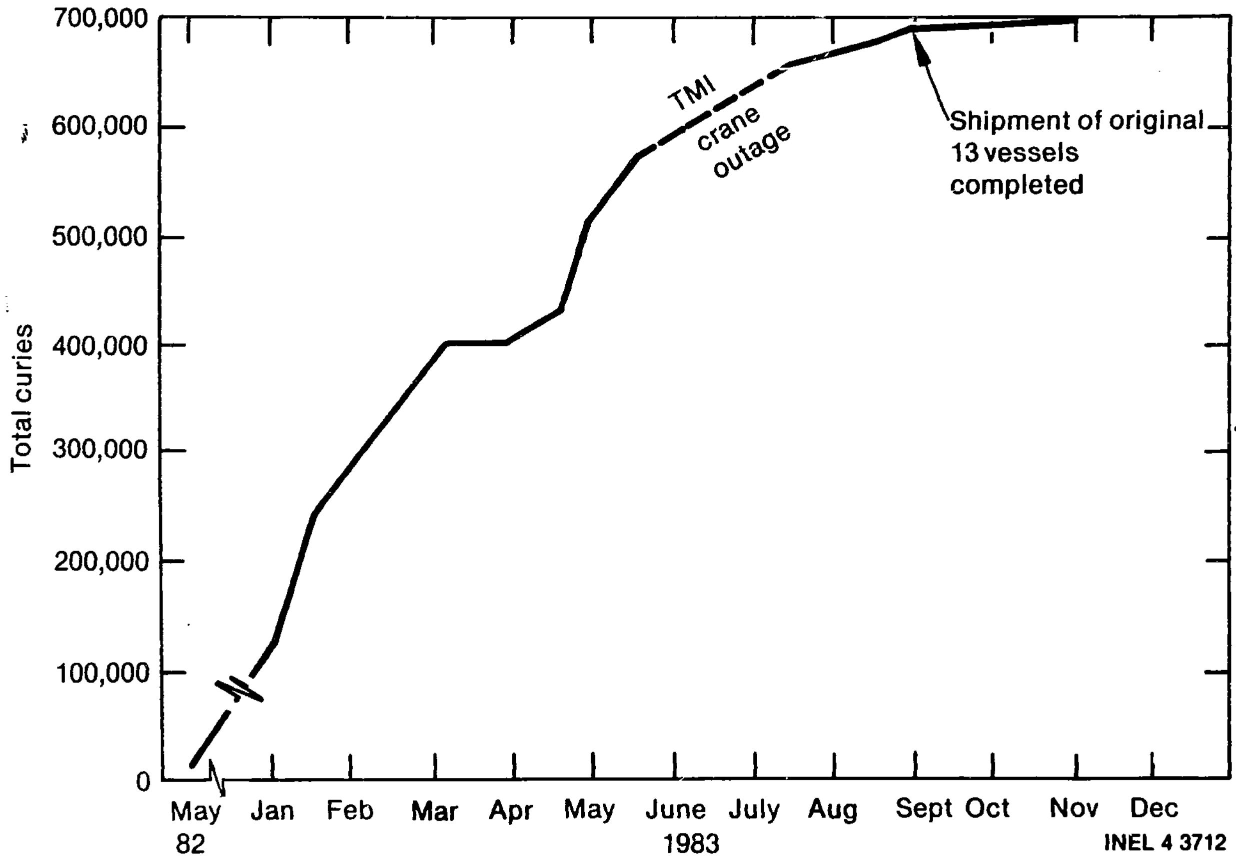

Figure 4. Cumulative curies in SDS vessels shipped from TMI.

The probe monitored the radiation background. When the glass level in the canister reached the height of the probe, the radiation reading increased several orders of magnitude due to the concentrated activity of the feed. The probe was then moved to the next higher plug location.

Zeolite vitrification was followed by several tests designed to characterize and evaluate the results of the experiment. Glass core samples were taken 0.3 m from the bottom, 0.3 m from the top, and from the middle of the glass in the canister.

Leach behavior and microstructure of radioactive vitrified zeolite material were studied. The leach tests were conducted in deionized, deaerated water according to the MCC-1P Static Leach Test Method. Specimens were tested to compare the leach behavior of the radioactive vitrified zeolite contained in three separate canisters with that of the nonradioactive zeolite tested previously, and with a Materials Characterization Center reference glass. The following conclusions were reached:

- The normalized mass loss of cesium and of the glass-former elements from the radioactive vitrified zeolite after the 28-day leach test is significantly lower (factor of 2 or more) than that from the nonradioactive reference glass MCC 76-68.

- The average leach rate for the radioactive vitrified zeolite is slightly lower than the rate for the nonradioactive vitrified zeolite, for all elements reported and all leach test periods longer than seven days.

- The leach behavior of radioactive vitrified zeolite from all three canisters is essentially the same.

- There is no significant systematic difference in leach behavior relative to location of radioactive vitrified zeolite (top, middle, or bottom) in the canister tested.

Table 2. Operations summary

Description	Feed (kg)	Maximum Feed Rate (kg/h)	Canister Fill Height (m)	Glass (kg)	Canister
First High-Level Vessel (D10012)	191	12.3	2.0	167	Z-2
Remainder First High-Level Vessel and part of Second High-Level Vessel	194	13.6	2.0	164	Z-4
Second High-Level Vessel (D10016)	180.5	10	1.9	161	Z-5

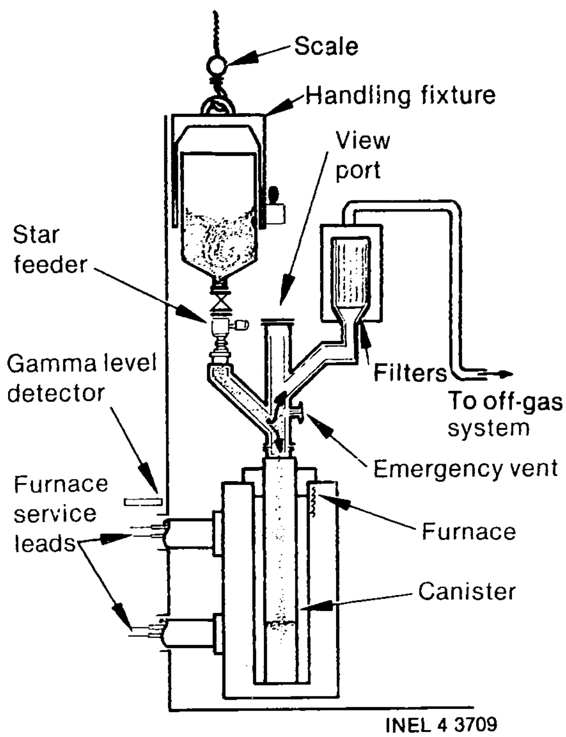

Figure 5. Zeolite vitrification system.

- Microstructure of the radioactive material was consistent with that observed in the nonradioactive material, and no second phase components were detected.

Monitored Burial Demonstration Program. SDS vessels shipped to Richland, Washington were also received by RHO for use in a monitored burial demonstration program. The SDS vessels were individually loaded into concrete overpacks, shown in Figure 6, prior to burial at a DOE waste site Long-term monitoring of one of these vessels will be performed to determine the performance after burial. Parameters of interest are pressure, temperature, moisture, and fission product particulates.

System pressure will be monitored by installing two pressure transducers on the vessel's vent hose. The transducers are being installed and checked at PNL. The electrical leads will be routed through the overpack and topsoil to an above ground monitoring station. System temperatures will be monitored by an extensive array of thermocouples to make measurements on the SDS vessel's shell, the overpack, and between adjacent overpacks. The leads extend to the above ground monitoring station.

System moisture and fission product monitoring will be accomplished by installing two stainless steel tubes running from the inside bottom of the concrete overpack to a few feet above the surface of the topsoil. These tubes will be valved and capped at their upper ends and used periodically for removing gas samples to determine water vapor content and for particulate fission product monitoring.

Abnormal Waste Technology

In 1982, DOE's TI&EP initiated an effort to provide technical support to GPU Nuclear in

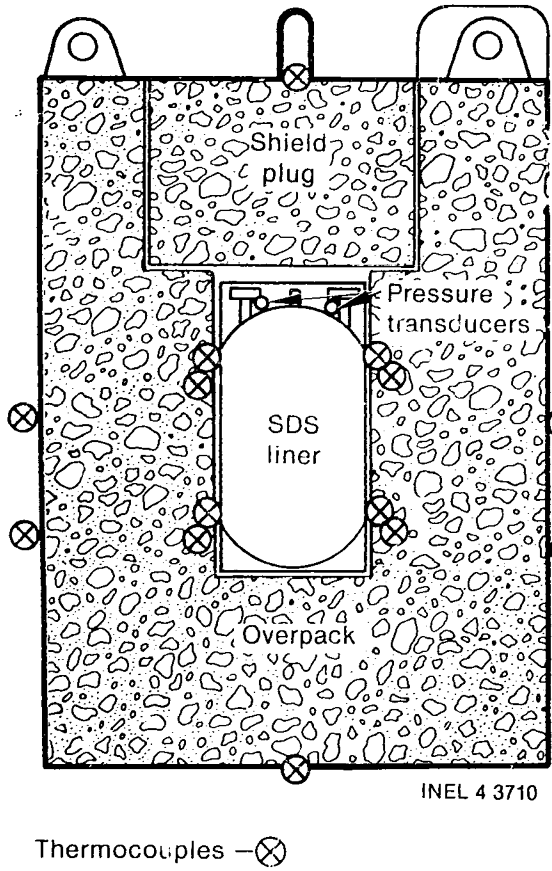

Figure 6. SDS liner overpack.

removing, packaging, and shipping abnormal waste from TMI. The term abnormal waste designates that waste which is not routinely generated at nuclear power plants and is not suitable for commercial disposal. At TMI-2, this abnormal waste includes wastes in filters, tanks, equipment, and sumps suspected of containing greater than 10 CFR 61 radionuclide concentration limits for commercial shallow land burial.

DOE acceptance of abnormal waste for R&D or reimbursable disposition will require characterization activities, including analyses for curie content, transuranic (TRU) contamination level, dose rate, quantity, and physical form. The criteria for acceptability as low-level waste are presently those requirements in effect at the Washington State burial site, i.e. not greater than 100 nCi/g of Pu waste. Current Washington State requirements will be used to classify abnormal waste until a regional burial site located in the northeastern United States is opened later in the decade or the new NRC regulations in 10 CFR 61 are implemented by Washington State.

Purification System Abnormal Waste. The TMI-2 Makeup and Purification System demineralizers were in use for at least 18 h, 35 min. during the TMI-2 accident. This system processes Primary Coolant System water and returns it to various locations within the Primary Coolant System and storage tanks. GPU Nuclear has estimated that about 175 m^3 of highly contaminated reactor coolant passed through the organic resin beds. The fission products and fuel debris resulted in high radiation levels in the demineralizer cubicles and prompted concern over the degree of subcriticality in the vessels. The 1982 robot inspections of the cubicles and fuel assessments using three independent techniques eliminated concern over criticality and permitted further characterization in 1983.

The first samples to be obtained by GPU Nuclear were from both the A and the B vessels and consisted of gas samples from the void space above the resin beds. The results of the gas sample analyses performed by GPU Nuclear and a DOE laboratory confirmed that there were high hydrogen and low oxygen gas concentrations and a substantial quantity of nondiatomic gases. These results had been predicted by resin irradiation studies performed at PNL in 1982 to determine the effects of temperatures and irradiation on organic resins. From a comparison of the actual gas samples to the scoping study results, it was concluded that conditions in the demineralizer after the accident were such that the resins in both vessels were wet when irradiated.

After the vessels were gas sampled, the vessels were purged with nitrogen gas to remove hydrogen gas. The next step was to purge the resin fill line with nitrogen gas in preparation for obtaining resin and liquid samples from the vessels. The resin fill lines are 7.62-cm diameter pipes through which new resin is sluiced from the floor above into the vessels. These fill lines were cut just outside the demineralizer cubicles and sampling equipment was placed into the vessel through the fill line and the fill line's diaphragm valve.

First attempts to obtain resin and liquid samples were made with a vacuum sampler. A teflon tube was used to obtain a 100-mL sample of liquid from the B vessel. A sample could not be obtained from the A vessel since it was dry. The difficulty in trying to obtain an A vessel sample led to the use of a radiation tolerant, fiberoptic scope to see what was in the vessel. The scope, inside a polyethylene guide tube, was pushed into the vessel through the resin fill line and passed easily through the resin fill line diaphragm valve. The fiberoptic scope and guide tube paths are detailed in Figure 7. Observations during the fiberscope inspection concluded that the A vessel contains a boron crystal coating on top of the bed of resin. The center of the bed has a large void that appears to be above the resin sluicing outlet line. The resin in the bed is agglomerated and amber colored below the crystalline crust.

Using mechanical probes and a vacuum sampling system, a 10-g solid sample of the A vessel resin was obtained. This sample had radiation readings of 3 rad/h beta and 150 mR/h gamma. The mechanical probe inserted into the B vessel found the resin bed approximately 0.30-m below the top of the water and 0.46-m thick. Estimates of the resin and water levels in the B vessel are shown in Figure 8. Samples from various depths in the resin bed resulted in a 75-mL slurry with approximately 50 mL of solids. Radiation readings taken without shielding at the top of the sample shipping container were 40 rad/h beta and 800 mR/h gamma.

The liquid and resin samples obtained from the B vessel in March and April and a solid sample obtained from the A vessel in April were analyzed at the ORNL. Selected preliminary results are listed in Table 3. All samples confirm that the fuel content of the demineralizer resins is very low, but the presence of transuranics in the resins means they may have to be handled as TRU wastes. The samples also indicate cesium activity levels that far exceed known values for any other accident-generated waste in the plant. The ^{137}Cs activity ranged from 220 μCi/g in the A vessel solid sample to 16.9 x 10^3 μCi/g in the B vessel solid sample obtained in April. Cesium will have to be removed from the resins before existing plant systems can be used to transfer the resins out of the A and B vessels.

In normal plant operations, demineralizer resins are removed by sluicing them through existing sluicing piping to spent-resin storage tanks. Tests

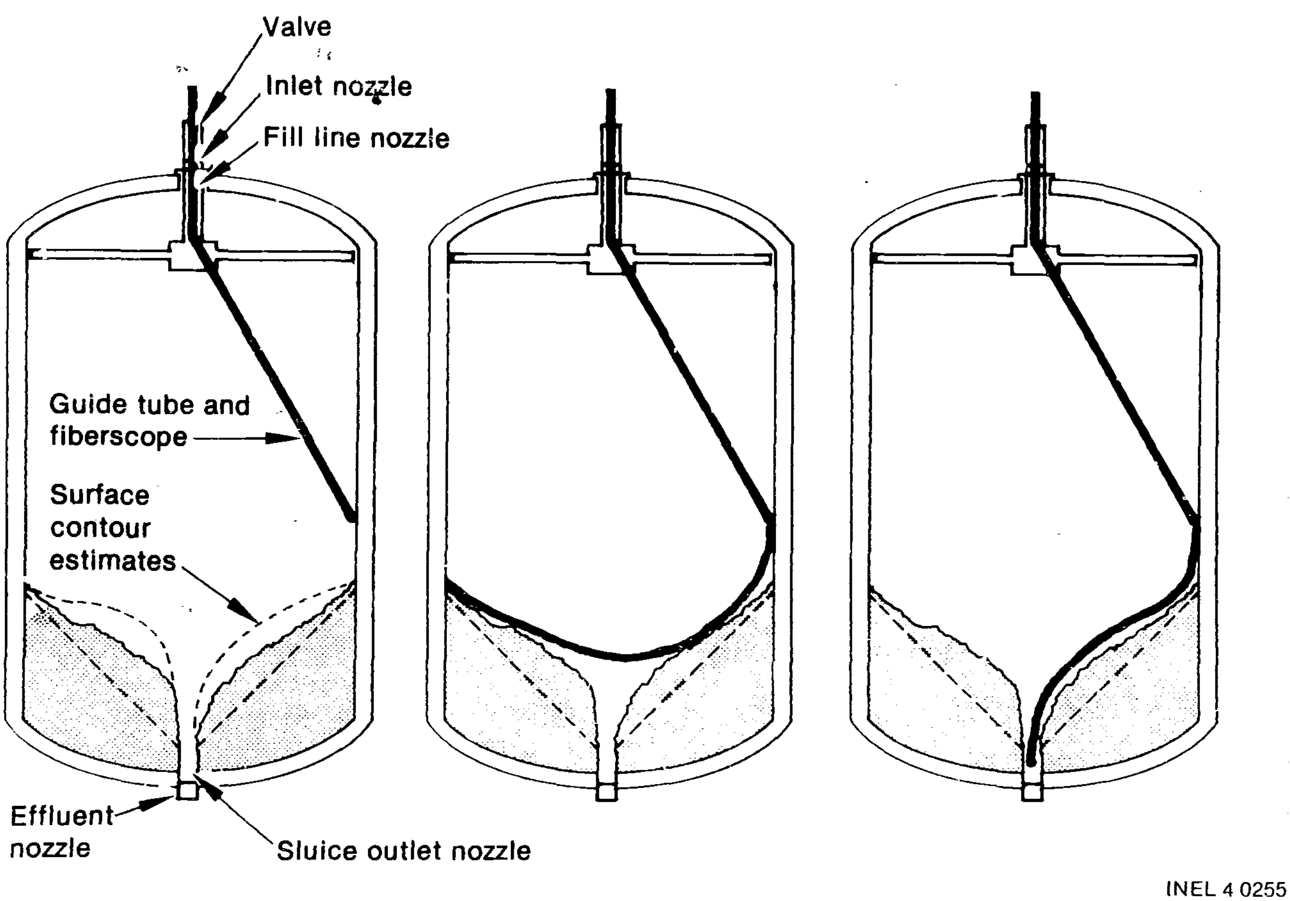

Figure 7. Pathways of fiber optic boroscope during examination of demineralizer A vessel internals.

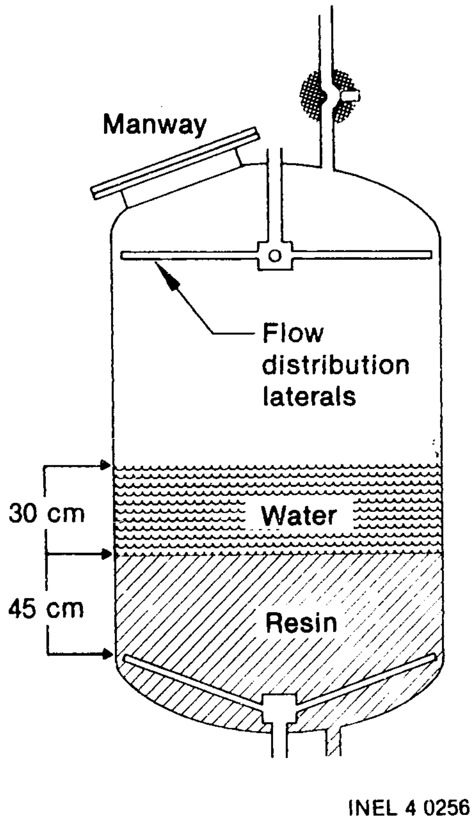

Figure 8. Estimates of resin and water levels in B vessel.

performed on irradiated resin confirm that the B vessel resins at TMI are sluiceable. However, the high activity would make normal sluicing a high-radiation exposure task for plant workers. The DOE Waste Immobilization Program participants, EG&G Idaho, ORNL, and WHC together with GPU Nuclear have developed a two-phase plan to first remove the cesium from the resins, and then sluice the resins from the vessels for packaging.

During Phase 1, ^{137}Cs will be removed from the resins and processed through the plant's SDS. To accomplish cesium removal, engineers will add water to the vessels to rinse the contents. The resins will be rinsed with borated water and "fluffed" with nitrogen gas, and then the water will be decanted. Essentially the same operation will take place during elution; however, chemicals such as sodium borate will be added to the flush water to remove additional radioactive cesium from the resin. During both the rinse and elution steps, the flow rate of water through the vessels will be restricted to below 0.32 L/s, a rate slow enough to ensure that very little of the resin will be carried out with the rinse water. To guard against particle carryover to other parts of the plant, a filter will be installed in the flow path to remove resin and fuel particles out with the water.

Table 3. Sample analysis of resins in the A and B demineralizer vessels

	March 1983 Sample		April 1983 Sample		
Element	B Solution (ppm)	B Solid (ppm)	B Solution (ppm)	B Solid (ppm)	A Solid (ppm)
Cs	30	—[a]	30	100	100
Sr	1	—[a]	<1	1	4
U	0.064	1,620	0.109	283	1,250
Pu	0.72 E-3	3.550	0.64 E-3	0.787	3.520
Isotope	(μCi/g)	(μCi/g)	(μCi/g)	(μCi/g)	(μCi/g)
^{134}Cs	181	778	101	1,130	15
^{137}Cs	2,640	11,200	1,480	16,900	220
^{90}Sr	14	490	9,460	880	200

a. No analysis conducted.

Engineers estimate that about 7.6 m^3 of water will have to flow through each demineralizer vessel before the cesium activity is significantly reduced. The water will be added to the vessels in 0.76-m^3 batches, and each vessel will be rinsed three times and then eluted seven times. Because the cesium concentrations are so high, the discharge stream from each vessel will have to be diluted with additional process water immediately after the rinse water leaves the demineralizer cubicle. The entire procedure can be repeated if it appears that still more cesium could be removed from the resins. Cesium removal will reduce the dose rates both in the demineralizer cubicles and along the sluice path to the spent-resin storage tanks. Removal of the cesium will also minimize the handling problems associated with the packaging of the sluiced resins for shipment.

Once the resins have undergone rinsing and elution, Phase 2 of the removal plan will begin. In this phase, the actual sluicing, packaging, and disposition of the demineralizer resins will occur.

The ORNL information contained in the preliminary results have allowed waste management planners to develop this two-phase resin removal process. Additional samples will have to be obtained from the dry resin bed in the A vessel before engineers can confirm that the two-phase process developed on the basis of the wet resins in the B vessel will work equally well for the dry A vessel resins. Barring any complications, the demineralizer content cleanup process should begin in mid-1984. The removal of the resins from the plant will be a significant milestone in the TMI-2 cleanup.

Resin Disposition

When DOE first prepared to ship ion exchange media canisters from the EPICOR II water processing system off TMI in 1981, there were 50 canisters. By the end of July 1983, all of the original 50 EPICOR II canisters had been shipped.

The canisters are prefilters from the EPICOR II water processing system at TMI-2, which decontaminated 1900 m^3 of accident water from the TMI-2 Auxiliary and Fuel Handling Buildings. The curie loadings of the canisters after processing accident water range from 160 to 2200 Ci. EPICOR II is a three stage ion exchange cleanup system. The first stage uses liners called prefilters, while the second and third stages are called demineralizers.

The first canister left the Island in May 1981 for characterization studies at Battelle Columbus Laboratories, where researchers concluded that the canister had suffered minimal damage as a result of exposure to the radioactive ion exchange media it contained. The liner then went to the INEL for further characterization (see EPICOR AND WASTE RESEARCH AND DISPOSITION PROGRAM Section). After that first shipment, regular shipments to the INEL began in October 1982 and continued at a rate of three to six a month until completion in July 1983. At the INEL, researchers are studying the short- and long-term effects of ionizing radiation on various types of ion exchange media and on the canisters containing those media. They are also pursuing disposition in a high integrity container as an alternative to solidification.

REACTOR EVALUATION PROGRAM

The Reactor Evaluation Program at TMI-2 focuses on development of technology for reactor defueling and disassembly so that the core can be removed offsite for examination. Data on the reactor core and internals are essential for evaluating accident prediction codes and reactor design, operation, and recovery standards as well as for direct TMI recovery planning. The techniques and equipment developed under this program provides basic technology for defueling a damaged reactor core. Figure 9 illustrates the steps in preparing the reactor core for defueling.

The TI&EP's Reactor Evaluation Program is divided into four major tasks: reactor vessel head removal, plenum removal, fuel and core debris removal, and core transportation.

Reactor Vessel Head Removal

Reactor vessel head removal was delayed because the polar crane was unavailable. Consequently, an alternate plan was implemented, substituting a small-capacity lifting apparatus in lieu of the polar crane for certain activities. The change in lifting capacity, work method, and scope in some cases required the revision of previously prepared safety, engineering, working, and operating documents, as well as the generation of new documents. As a result, completing the underhead characterization task was delayed by about six months. Head removal activities will resume early next year, with reactor vessel head removal scheduled for the end of FY-84.

Underhead characterization consisted of preparation and training, and acquiring data on videotape. Radiation measurements were taken with miniature ionization chambers and thermoluminescent dosimeters. Also, samples of fine materials were taken from the upper plenum cover. Data were obtained with the RCS water level lowered to just above and then just below the top of the plenum; the control rod drive mechanism (CRDM) was removed from the H-8 CRDM location to provide greater access for inspection equipment. Findings from underhead characterization are contained in a GPU Nuclear data report, *Underhead Data Acquisition Program*, TPO/TMI-110, December 1983, which concludes the planned dry method of head removal, versus flooding the canal, is satisfactory, provided that appropriate shielding measures are taken to minimize radiation exposure during head removal and storage and IIF installation. Expected radiation levels approximately 1.2 m from the inner diameter of the reactor vessel and approximately 1.5 m above the reactor vessel flange plane may be 8 to 20 R/h gamma with the reactor vessel head removed. These expected levels are somewhat higher than the level of 8.6 R/h used for planning purposes and thus have been factored into head removal plans.

One concern was that upon reactor vessel head removal, a layer of pyrophoric material, which might be located on the top of the plenum cover, would ignite. To address this concern, a sample was obtained. The sample material fines from the upper plenum cover had no pyrophoric reaction when subjected to spark, flame ignition, and impact tests. The upper cover of the plenum did not appear to be distorted, and only a layer of light debris was found adhering to the surfaces of the plenum and underside of the head, eliminating the need to flush these surfaces.

GPU Nuclear studied three methods of evaluating the effect of parked lead screws on work area dose rates and the cost of implementation. The methods were to: (a) lift and store the reactor vessel head with lead screws in the parked position (within the head and service structure), (b) lower the lead screws to the floor with the reactor vessel head in the storage stand, and (c) remove the lead screws prior to reactor vessel head lift. The study concluded that removal of the lead screws would result in increased man-rem dose rates and increased costs. Lifting the reactor vessel head with lead screws parked (fully raised and locked into position) then lowering the lead screws to the floor in the storage stand could be beneficial if more shielding was added at the storage stand at added cost. A decision regarding lead screw removal should be made early in 1984.

Four lead screws at different CRDM locations were parked to obtain radiation level data at selected points on the outside of the service structure and to evaluate the radiation dose rate effects on the parked lead screws. The radiation survey results showed an increase from about 80 to 350 mR/h in the canal seal plate area. While two of the lead screws were lowered with some

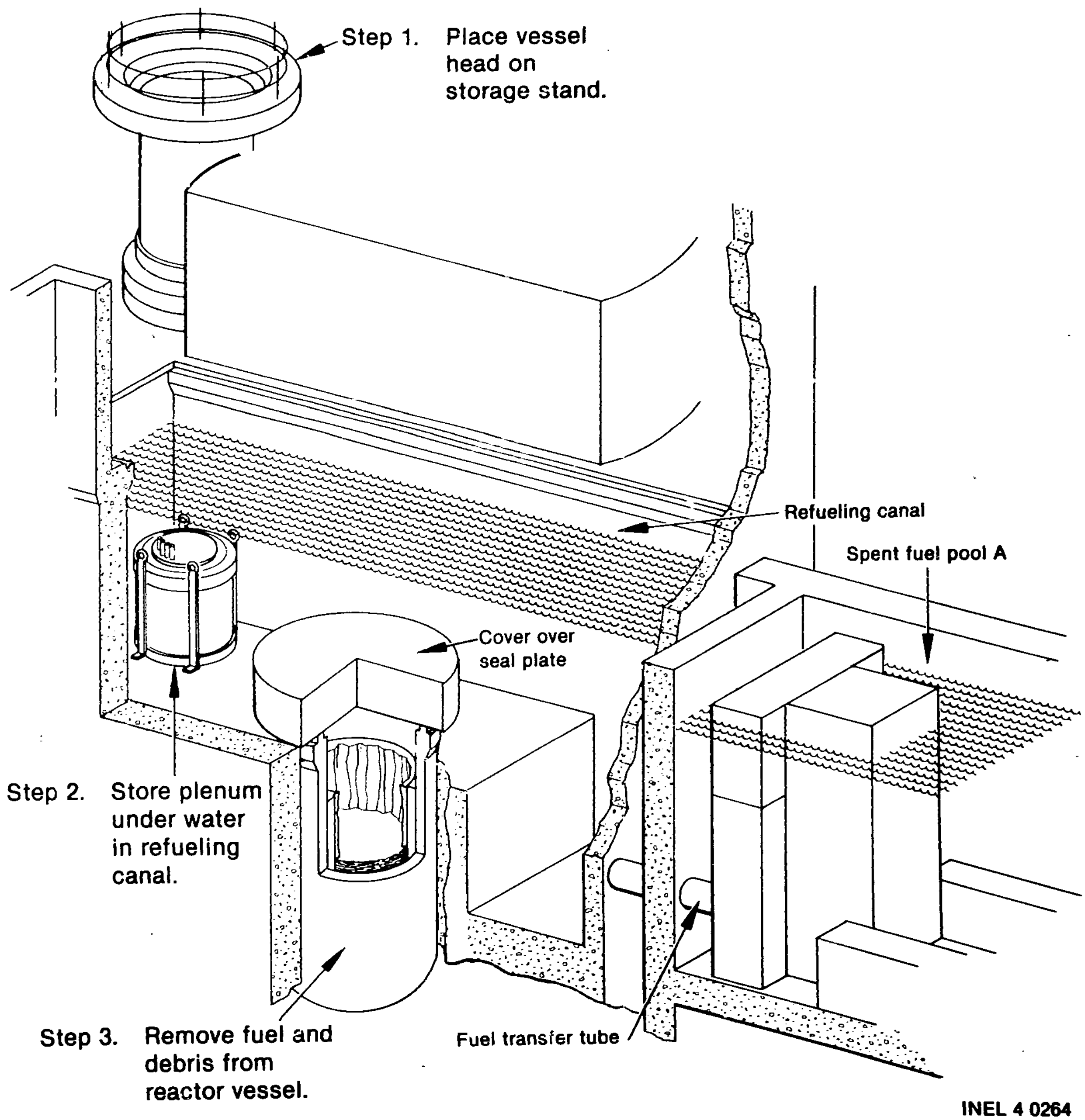

Figure 9. Major steps in preparing the TMI-2 reactor vessel for defueling.

resistance, a third could not be fully lowered because it met an obstruction. Engineers concluded that the lead screw bayonet coupling was contacting the control rod guide support plates. The lead screw was fully withdrawn and now remains in the parked position. The fourth lead screw was lowered without difficulty.

The neutron shield tanks and insulation surrounding the reactor vessel head were removed, and the reactor vessel head service structure and refueling canal were decontaminated. Also, service structure hoists and power distribution panels were installed, and the steam generator secondary sides were recirculated and drained in preparation for primary side draining. In addition to these activities, design modifications were completed, materials procured, and installation work initiated for the canal seal plate and defueling water cleanup system; the head lift lugs and lift tripod were inspected; the reactor vessel head hold-down nuts and studs were cleaned and lubricated; the stud tensioner was modified and successfully load tested,; and the source range detectors were calibration checked.

Also in 1983, engineers conducted a head drop analysis, verifying adequate structural capacity of the reactor components below in support of reactor vessel head lift.

Plenum Removal

Plenum removal work in 1983 concentrated on five major areas: plenum removal tools, plenum removal operations, criticality analysis and the Safety Evaluation Report (SER), fuel transfer mechanism modifications, and refurbishment of spent fuel pool A. DOE-funded work to refurbish spent fuel pool A was completed at the end of the year. Remaining work on spent fuel pool A will be funded by GPU Nuclear. Work in the other four areas was still in progress at year's end. The following sections present the results of this work for each of the five areas.

Plenum Removal Tools. During the past year, the option of destructive removal of the plenum was dropped from the program for tooling design purposes. The plenum will be removed intact. All current tooling is based on the assumption that plenum removal will require a jacking force of no more than four times the plenum weight. Therefore, the design capacity of the jacking system is 240 tons. The plan is to first jack the plenum up from its current seated position and then lift it out of the reactor vessel using the polar crane. The plenum will be stored in a bag underwater in the shallow end of the refueling canal (see Figure 9). To date, four tool groups have been identified as necessary to perform this job; they are: inspection tools, initial lift tools, transfer and storage hardware, and cleaning equipment. By the end of 1983, designs for the first two tool groups were completed. Also, appropriate test assemblies were identified for testing key aspects of the job.

Inspection tools will be used to perform key inspections of the area between the reactor vessel flange and the plenum assembly flange and the gap between the loss-of-coolant accident (LOCA) restraint bosses for each hot leg. They will aid in the search for debris on important horizontal surfaces such as the jack seating surfaces and the plenum support flange at the keyways. The tools will also be used in inspections of the vent valves to check for distortion, the bottom annulus between the plenum assembly and core support shield to check for debris, the upper grid ring flange joint, the control rod guide assembly to the upper grid joint, and the core cavity boundary. Modifications to these inspection locations can be made if necessary.

In conjunction with these inspections, which primarily define current plenum status, plenum removal progress will be monitored. The core void video inspection (see Figures 10 and 11) and topography work performed in 1983 revealed that a large number of partial fuel assemblies are hanging from the bottom of the plenum. A new camera system will be used to monitor and verify the removal of these partial fuel assemblies, and several cameras placed at key locations around the plenum will be used to monitor the progress of the initial lift.

The inspection tools required to perform these operations were designed, and some of the components were already procured before the year's end. Babcock & Wilcox (B&W) selected the appropriate cameras and other inspection tools, such as auxiliary lights, that would meet inspection location access restrictions. All cameras are radiation hardened and will be available to monitor defueling and other operations after the plenum is removed. Video cassette recorders will provide videotape archiving of camera views.

Initial lift tools for the plenum consist of four, 60-ton hydraulic jacks with associated controls and placement equipment. This lift equipment was designed in 1983 and will be fabricated in 1984. While the jacks will be custom designed, because of their planned location, the hydraulic cylinders will be standard shelf components. The jacks will lift the plenum slowly and in small increments and will be hand-pumped by four operators at a central pump station. The load on each jack will be continuously monitored for sticking and to prevent angular jamming of the plenum. Also, the top of the plenum will also be continuously monitored to prevent angular jamming. Extensometers were designed into each jack to determine rise. In the event of a hydraulic failure, the plenum will be supported by a manual follower, which automatically raises to back up the hydraulic jack. After the plenum is raised about 6 cm, its underside will be examined to determine core separation. If necessary, fuel assemblies still adhering to the plenum will be removed, after which the plenum will be jacked to about 23 cm. Following a final

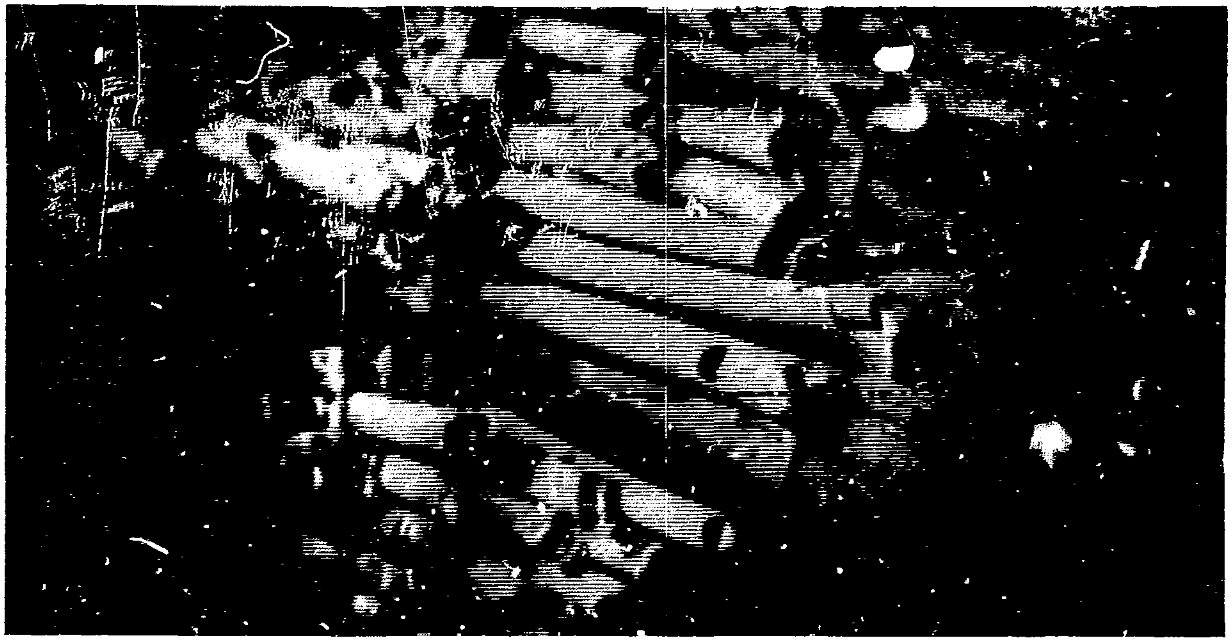

Figure 10. In this still photograph from a videotape, partial fuel assemblies are shown hanging from the bottom of the plenum, which is seen on the left of the screen.

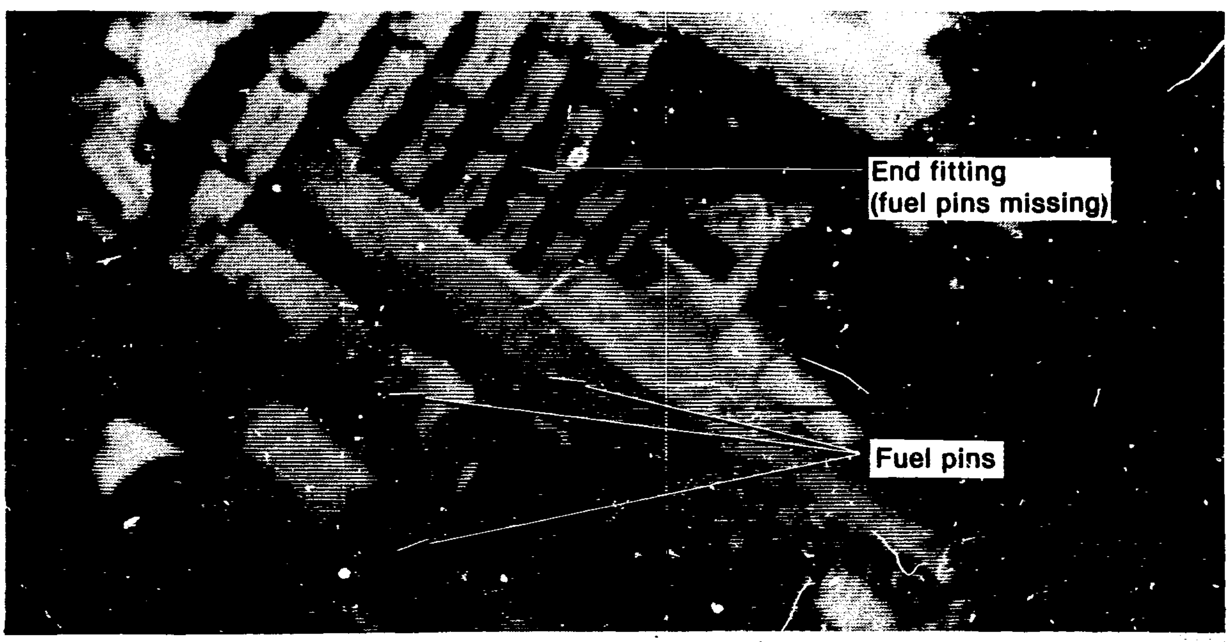

Figure 11. This still photograph from a videotape shows uneven damage across a fuel assembly.

clearance check, the hookups to the jacks will be severed, the storage enclosure will be positioned over the plenum, the polar crane will be rigged to the plenum, and the refueling canal will be flooded.

Transfer and storage hardware, which underwent preliminary design review in 1983, will be used to move the plenum from the reactor vessel to the shallow end of the refueling canal and to secure its storage until further disposition. The equipment consists of rigging to attach the plenum to the polar crane, a transfer contamination barrier, and a plenum storage ring.

The studs attaching the plenum lifting legs to the plenum may have degraded and could not be certified as capable of lifting the plenum without a difficult and costly in situ certification program. Therefore, the preliminary design focuses on a three-point lift with the new, equally spaced lifting blocks locked under the plenum cover assembly support flange.

The design calls for one fixed length pendant and lifting block assembly and two positioner pendant and lifting block assemblies, all of which will be attached to the existing handling fixture (tripod). During the lift, the plenum could tilt and jam, but the positioner pendant and lifting block assemblies are designed to compensate for tilting if it occurs, thereby preventing jamming. The design includes all new installation and monitoring equipment required to rig and lift the plenum from the reactor vessel.

The transfer contamination barrier (TCB) is a container in which the plenum will be placed to control radionuclide release into the refueling canal. This semi-rigid plastic bag will be assembled in the Reactor Building by being placed over a truss-like frame. The frame will be erected in the Reactor Building from a collapsed condition small enough to pass through the personnel hatch. Access holes for the handling equipment pendants and monitoring equipment will be provided in the top of the container. The bottom of the TCB and top of the IIF will be closed by using sheets of reinforced plastic. The plenum storage ring consists of a segmented, stainless steel ring with integral bearing pads to support the plenum assembly above the canal floor.

In addition to the major equipment items described above, support or accessory equipment items such as positioning poles and hooks will be used for the plenum final lift and transfer to canal storage. In many instances, the final lift operations will call for equipment used in previous operations.

A tool was designed to knock off the partial fuel assemblies hanging from the plenum. The tool is currently undergoing proof-of-principle testing to determine if it will meet the design requirements. The preliminary design was based on the capability to shear a complete fusion weld on the two close tolerance points between the plenum and the top portion of the fuel assembly, on each of the four sides of the fuel assembly. Calculations showed that such a design was feasible. However, B&W was asked to test and verify its capability. Subsequently, GPU Nuclear videotaped the top of the core void showing that, in some cases, a fuel assembly may be missing on one side of the fuel bundle while several inches of fuel pins are hanging down on the other side (see Figure 11). The impact of this information on the design of a tool with a comparatively small contact area has not been analyzed. The videotapes also suggest the possibility of a better bond than originally postulated between the fuel assembly and the bottom of the plenum.

The hydraulic debris removal equipment considered early in 1983 was put on hold after video examinations revealed that the top of the plenum was covered with very little debris. If debris is found in other areas, such as the keyways, specialized hydraulic vacuum or wash equipment may be required.

The design of several tools in these four groups must be verified. Consequently, B&W identified the appropriate equipment to perform the verification tests. For example, the four jacks will lift a load of 240 tons. Also, a prototype jack will be overload tested in accordance with American Society for Testing and Materials (ASTM) specifications.

Plenum Removal Operations. By year's end, EG&G Idaho agreed in principle to support operations involving the initial inspection and initial lift of the plenum. At this time, the contract negotiations are nearly complete, and work should commence early in 1984.

Criticality Analysis and Safety Evaluation Report. Engineers expect some displacement of core materials when the plenum is removed, especially when the partial fuel assemblies are knocked off of the bottom of the plenum, thus the need existed for a thorough, well documented criticality analysis for the SER.

Studies were initiated to determine if the current safety analysis takes into account the conditions anticipated during plenum removal, and if the feasibility of using soluble or insoluble poisons other than the current concentration of natural, soluble boron in the RCS. While these two studies probably will not provide all the information required in order to obtain an approved SER, they should provide the baseline information necessary to define what additional analysis or plant condition changes are required.

Fuel Transfer Mechanism Modifications. The fuel transfer mechanisms currently installed at TMI-2 are similar to mechanisms installed at other B&W plants. B&W and TAAG have recommended use only after replacing the existing underwater air motor driven chain drive with a single winch cable drive mounted above water, as this installed design has exhibited lack of reliability.

Several new design features were recommended which will further increase the reliability of the mechanism beyond models currently in use at other plants. Certain unique criteria must also be met in the modified design since the transfer mechanism will be handling canisters of fuel material instead of fuel assemblies. The canisters not only have a different physical dimension, they can also have variable total weight and center of gravity.

The engineering approach to the unique features is to design a holding mechanism to attach canisters instead of fuel assemblies, limit total canister weight to the transfer mechanism's current design capability, and modify the design to permit reasonable variability in center of gravity for a maximum weight canister. The design work on this approach was initiated in late December 1982 and is continuing. During the past year, the decision was made to purchase a new Fuel Transfer System from a cancelled PG&E plant instead of reworking the existing drive and upender units. The final design interface between the existing TMI-2 system and the new equipment requires verification of the installed equipment. Refurbishment of spent fuel pool A must be well along prior to this verification; consequently, DOE began support of refurbishment in 1983.

Refurbishment of Spent Fuel Pool A. The Fuel Pool Waste Storage System was designed to store up to 418 m^3 of radioactive waste water from the Reactor Building sump and the miscellaneous waste holdup tank without contaminating the fuel storage pool. The system consists of two levels of tanks, associated valves, manifolds, pipes, pumps, and shielding. The lower level is composed of two 95-m^3 tanks, and the upper level consists of four 57-m^3 tanks. Each level of tanks has a standpipe and pumping system. The standpipes serve as the paths through which liquid wastes can be transferred from one set of tanks to another, or to the EPICOR II system or SDS. More than 4,077 m^3 of accident generated water passed through the Fuel Pool Waste Storage System since the accident.

In refurbishing spent fuel pool A, the Fuel Pool Waste Storage System was to be decontaminated and removed and the fuel pool repaired and recertified for use in defueling operations. DOE supported the work performed in 1983. GPU Nuclear will continue the refurbishment support in 1984. Following are the accomplishments achieved in 1983 and the tasks that still need to be completed:

1. Analysts evaluated the health and safety considerations necessary for removing the tank farm from spent fuel pool A. The Technical Evaluation Report defined safe load paths for crane transport of materials, safety provisions for removal of shielding and tanks, administrative controls, and actions to be taken to protect the SDS during lifting operations. The report also addressed disposal of decontamination wastes and considered the possibility of releasing nuclear materials to the atmosphere in the Fuel Handling Building as well as outside the building after components are removed.

2. Each activity that required exposure of personnel to the pool environment or which was believed to be a potential source of contamination was surveyed for radiation characterization.

3. Temporary storage locations were prepared for the various contaminated components.

4. An alternate sump pump flow route was established from the Reactor Building sump to the SDS. The new piping route was installed and ready for operation before the existing route was removed from service.

5. The two lower tanks contained a quantity of liquid waste that could not be removed with the existing steam eductor system, so a revised pumping arrangement was engineered to replace the eductors.

6. The two silica gel driers and four charcoal filters that make up the vent filter system were sampled to characterize each for disposal and then were removed.

7. The tank farm was flushed with water which resulted in low tank radiation levels. The techniques and procedures for chemical flushing were approved so that the tanks could be further decontaminated.

8. Eight of the sixteen concrete shield slabs were removed from the pool. The undersides of the slabs were decontaminated to an acceptable level for safe storage. Also, all of the concrete blocks and lead bricks were removed.

9. The tanks, standpipes, and piping will be disconnected and disassembled for handling and storage. After being decontaminated, the tanks will likely be stored for reuse or shipped off site as low level waste. The piping and standpipes will be cut into segments for shipment or storage.

10. The pool liner will be inspected for damage and repaired as necessary and then will be tested to confirm its integrity before the pool is released for use.

Fuel and Core Debris Removal

The TI&EP continued defueling the water cleanup system and initiated three new activities dealing with fuel removal tooling, fuel canisters, and canister storage racks.

Defueling Water Cleanup System. A System Description Document was prepared to detail a system that could maintain water clarity and control radionuclides in the fuel transfer canal during defueling. Concurrently, engineers prepared a conceptual design, whereby the reactor vessel water and canal water would be processed separately through ion exchange systems and filters. Two key documents were issued during the course of these efforts: *Planning Study on Long-Term Water Management* and *Technical Plan for Defueling Water Cleanup System*, the latter containing the system's detailed conceptual design. Proof-of-principle tests on filters of the type proposed for the water cleanup system were conducted to support this design, and the functional requirements and criteria were defined. The Quality Assurance classification, licensing, and regulatory requirements were also identified.

The water cleanup system concept consists of a filtration and soluble processing subsystem serving the reactor vessel and contamination barrier and a filtration and soluble processing subsystem serving the fuel transfer canal and spent fuel pool A. Lastly, engineering schedules and piping and instrumentation diagrams were developed and equipment locations identified, completing the DOE-funded support for the defueling water cleanup system.

Fuel Removal Tooling. A reference approach to defueling was established based on a 1983 report titled *Preliminary Design Study for Plenum and Fuel Removal*, compiled by the TMI-2 Reactor Disassembly and Defueling Task Group. A Technical Specification for the fuel removal system and the fines and debris vacuum system was issued in March. The reference design, as set forth in the specification, deals with actual fuel removal equipment of two types: mechanical equipment and vacuum and separation equipment. For both systems, positioning masts, manipulators, and/or long-handled tools will be required. Mechanical tooling may include grapples or other lifting gear, strongbacks for supporting damaged assemblies, scoops or buckets for large debris, chisels and other separation devices, and canister-loading equipment.

The fines and debris removal vacuum system will be designed to remove fines as small as 10 μm and debris of approximately the size of fuel pellets, separate the debris from the water, and load the debris into canisters. The defueling water cleanup system, the conceptual design for which was completed in 1983, represents an important interface with respect to the defueling system. Vacuum system effluent, containing fines smaller than the system is designed to handle, and soluble fission products will be handled by the water cleanup system.

Westinghouse Electric Corporation was selected as the contractor to design and fabricate fuel removal tooling. Westinghouse then formally submitted two proposals. The first proposal, *Cost Proposal for Fuel Removal and Fines/Debris Vacuum System*, represented a manual, remote defueling scheme in accordance with the Technical Specification's reference system. The second proposal, *Westinghouse Candidate System Design*, suggests a more automated approach in comparison to the reference design system. This automated system, which also complies with the controlling specification's requirements, uses robotic arms to position a vacuum hose and load components too large for vacuuming into a shredder. A vacuum hose is connected to the shredder outlet. All debris is then pumped in a slurry out of the Reactor Building for separation and canister loading. Figure 12 shows an early design of this proposed fuel removal tooling. This defueling method requires little manual work in the Reactor Building. Other aspects of this defueling concept, such as the separation, manipulator end effector, and video equipment, are similar to those in the reference approach.

At the end of 1983, the fuel removal tooling task was well into its preliminary engineering, concept selection phase, at the end of which a defueling system design will be committed. In the meantime, the robotic shredder system design will undergo proof-of-principle testing. This preliminary engineering effort will be completed in early 1985.

Fuel Canisters. The design requirements document and a technical specification for fuel and filter canisters were developed during the year and issued in the first quarter of 1983. The fuel canisters are designed as shipping and interim storage containers for the reactor vessel fuel and debris. The filter canisters are designed for use in the fines and debris vacuum system and defueling water cleanup system. A canister interface control document was also issued at that time. B&W was selected to design the canisters and provide two prototypes. The reference fuel canister design under consideration by GPU Nuclear consists of a circular outer shell with an inner square shroud supporting the neutron-absorbing material. This "full size" design assumes that the core contains full length, full cross-section fuel assemblies that will have to be shipped. Based on the core topography results, TI&EP engineers have suggested that an evaluation of a smaller cross-section canister be performed prior to engaging in final design.

A simpler canister design with a smaller geometry would have an inherent safety feature and prevention of criticality would not be an issue. These "inherently safe" canisters might also offer an additional benefit: available DOE shipping casks could be used to transport these smaller canisters.

Canister Storage Racks. GPU Nuclear developed a technical specification to purchase fuel canister storage racks to go in the refueling canal and spent fuel pool A, where the canisters will be stored until they are shipped off site. Vendor bids for rack design and fabrication were taken and evaluated.

A study is currently underway to determine if one single rack design will meet the needs at TMI as well as the INEL, where the fuel assembly material and fuel fines and debris will eventually be stored until their disposal.

Core Transportation

DOE and GPU Nuclear negotiated a draft contract for DOE's acceptance of the TMI-2 fuel core. Under the terms of this draft agreement, GPU Nuclear will prepare the fuel canisters for shipment, and DOE will procure the casks and schedule their shipment. The TI&EP drafted a plan organizing the shipping program keeping in mind that close coordination is required between GPU Nuclear at TMI and EG&G Idaho at TMI and the INEL so that fuel may be cost effectively transported for further R&D efforts. Implementation of the plan during the next several years will ensure safe shipments from TMI to the INEL shortly after defueling starts.

Among specific activities initiated in 1983, the TI&EP reviewed bids submitted for suitable shipping casks and conducted safety-related studies for control of hazards associated with the fuel shipments. Also, GPU Nuclear conducted logistical studies of the cask shipping rate, and INEL engineers did storage studies and formed a technical working team with members of the various organizations involved in preparations for the shipments. The team will coordinate the design or modification of various fuel handling systems both at TMI and the INEL.

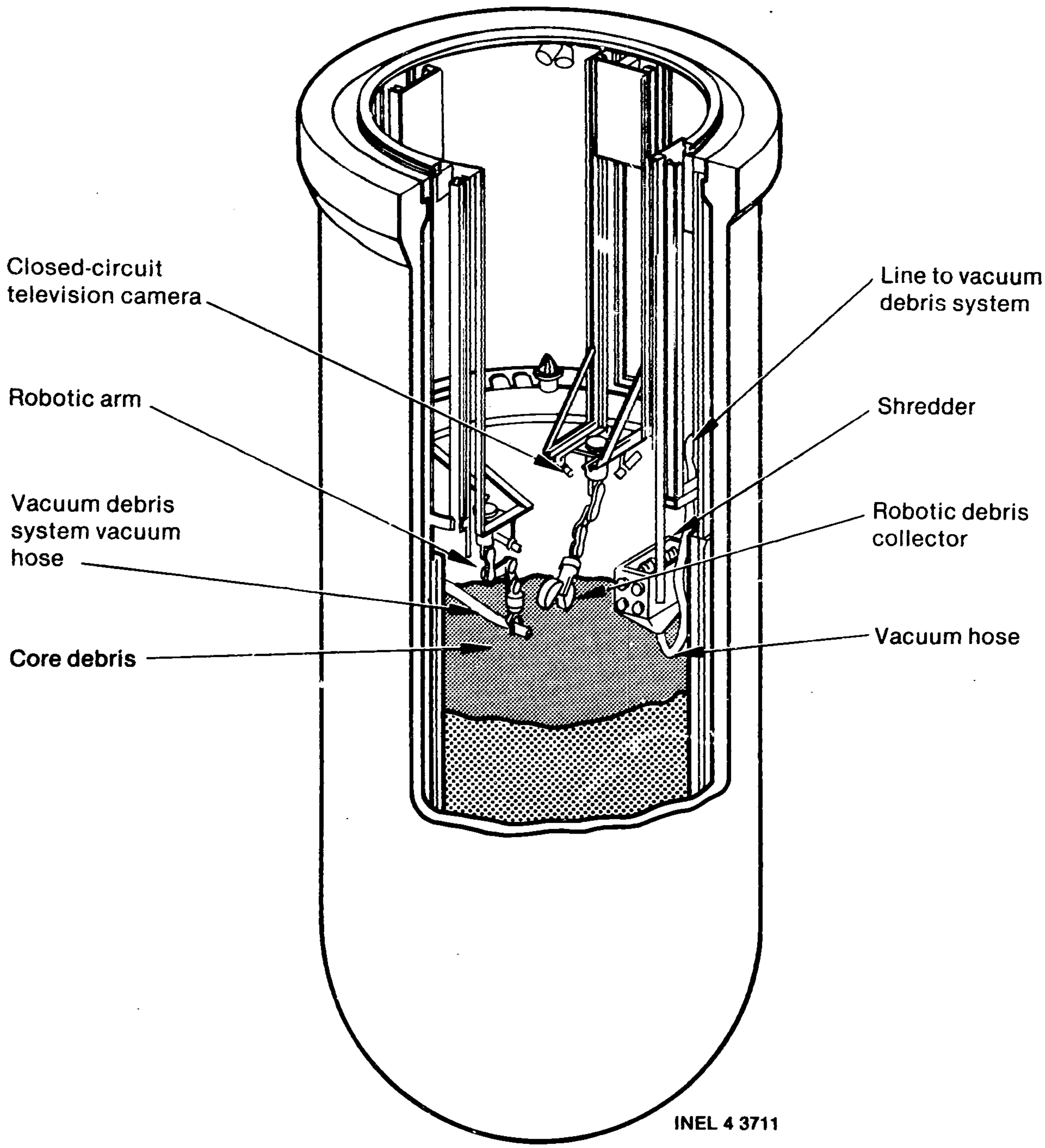

Figure 12. The Westinghouse Candidate Design System uses robotic arms to defuel the reactor vessel.

DATA ACQUISITION PROGRAM

Instrumentation and Electrical Components

The TI&EP Instrumentation and Electrical Program continued evaluating the effect of actual accident and postaccident environmental conditions of specific safety instruments, electrical systems, and selected plant hardware to determine how these conditions effect the ability of a system or component to perform their intended functions during and after an accident. I&E personnel have performed in situ tests, coordinated component and sample removal from the Reactor Building, and directed various offsite laboratories that have the expertise, facilities, special capabilities, or a combination of these attributes, to conduct detail examinations and evaluations. I&E findings have provided and will continue to provide a valuable insight into the adequacy of the instruments and equipment, the adequacy of current instruments standards and qualification procedures, the importance of plant construction control, and maintenance and operating procedures. In addition, the information and results from these efforts are being transferred to the nuclear community by means of individual investigative and topical reports, presentations at technical and professional meetings, and through established computer and telecommunications links with the nuclear power industry.

Electrical Components and Discrete Devices. Many of the electrical components and discrete devices in the Reactor Building are safety-related equipment or are representative of safety-related equipment used in other nuclear power plants. Since some of these components and devices have failed during or after the accident, I&E personnel developed a program to test and examine selected equipment to determine failure mechanisms and make recommendations to change designs and installation and maintenance procedures that could prevent reoccurance of similar types of failures in the future. The electrical components and discrete device task includes in situ testing, sample removal from the Reactor Building, and detailed offsite examinations.

To assess the survivability of motor-operated valves under LOCA conditions, I&E personnel performed in situ tests on a sampling of 22 motor-operated valves of which 18 are class 1E safety qualified. Seventeen of these valves are located in the Reactor Building basement and the remainder are located at various elevations above ground level.

In examining the 17 valves in the Reactor Building basement, which was flooded for a year following the accident, it was determined that 5 of the valves had been submerged. All 5 were class 1E qualified valves and they exhibited severe degradation as a result of the submersion. Analysis of the test data indicates that these valves experienced total failure of their motors, cables, and limit switches.

Testing of the other valves in the basement showed a slightly wet or damp condition, but no insulation damage. These valves may exhibit slight abnormalities when operated, but should function properly once they have dried. Test results of the five valves located above ground level indicate that only slight abnormalities exist and that they suffered little or no degradation. This slight degradation should not impair normal operation of these valves.

In situ testing was conducted on the five Reactor Building air cooling fans and their associated inlet damper limit switches. The running current of all the motors was slightly above the preaccident value, but still within the expected normal plant operating range. During in situ testing of the fan that tripped on thermal overload in August 1979, I&E personnel determined that this trip was due to faulty thermal overload relays. New thermal overload heater blocks have been installed and the fan will be returned to normal operation following testing. In addition, the inlet damper limit switches were tested and appear to be intact and electrically operational. However, the dampers themselves are not functioning properly due to mechanical problems.

In situ testing was also conducted on the pressurizer PORV and heaters. The PORV malfunctioned mechanically and stuck open during the accident. The in situ tests determined that both of the valve solenoid coils are functioning normally; however a bypass limit switch remained open when the valve was de-energized. This abnormality is probably due to an oxide buildup on its contacts and prevents operation of the valve. Two other limit switches, used to indicate whether or not the coils are energized, are in the proper position and the associated circuitry is functioning normally.

In the 13 groups of pressurizer heaters consisting of 117 heater elements, only one element was found to have an open circuit. In addition, some cables exhibited an unacceptable insulation to ground resistance with the indication that 15 of the cables may be wet. A detailed analysis of the test data is in progress.

Two types of electrical components and discrete devices from the TMI-2 Reactor Building were examined at the INEL. These components were two valve solenoid assemblies and two pressure switches. The detailed examination of the two air handling valve solenoid assemblies revealed that the lead wire insulation of both solenoids was degraded by radiation exposure; the insulation was discolored and has lost some of its tensile strength. Examiners determined that Reactor Building pressure suppression spray fluid had intruded into the housing of one solenoid, causing extensive rusting of the solenoid shell as shown in Figure 13. The fluid inside the shell also intruded into a hermetically

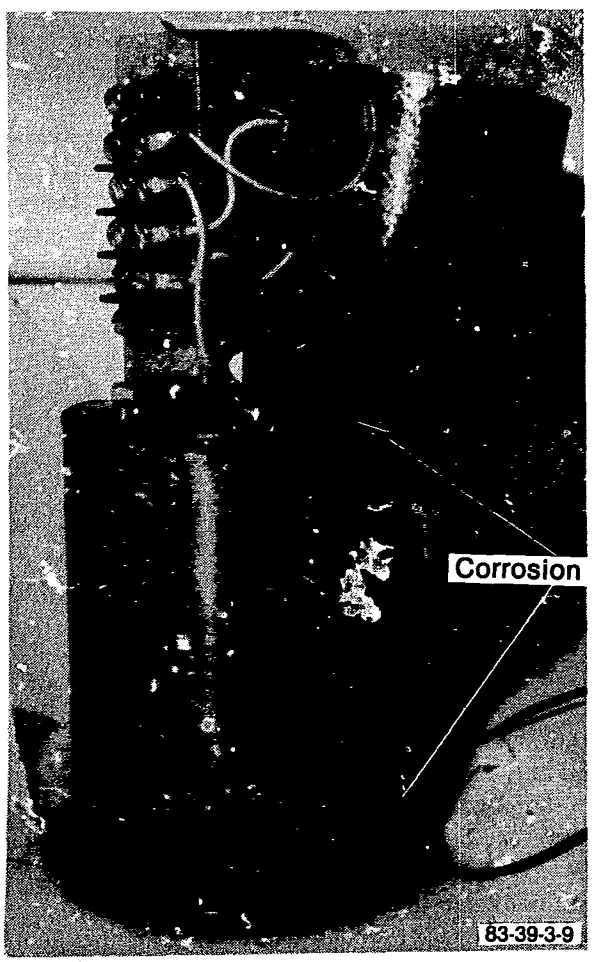

Figure 13.- Solenoid shell rust caused by water intrusion.

sealed reed switch, causing severe corrosion and crumbling of the contacts. Water intrusion into the solenoid housing was a result of poor installation geometry. The terminal box, connected to the solenoid by a flexible conduit, collected water when the Reactor Building pressure suppression system spray was actuated. The terminal box is located above the solenoid housing and allows the water to channel down the conduit into the solenoid housing. The absence of a drain in the solenoid housing trapped the water that led to corrosion of the coil shell. All conditions in the other solenoid appear to be in the as-manufactured state.

Two pressure switches, used in the nitrogen manifold and manufactured by Static-O-Ring, were also subjected to detailed examination. Both devices were found to function normally. There was some superficial rusting on the inside and outside surfaces, but it did not affect electrical or mechanical characteristics.

Resistance Temperature Detectors. The data recorded by the Reactor Building resistance temperature detectors (RTDs) during and after the accident provide temperature profiles that are key to understanding the accident and subsequent plant conditions. Testing during 1983 centered on the RTDs in the RCS. Loop Current Step Response tests, designed to determine the response time and conditions of the RTDs, were performed on 13 RCS RTDs by ORNL. Identical response time testing was conducted at the Crystal River Unit 3 facility RCS RTDs. This facility, a sister to TMI-2, uses the same type of RTD assemblies and was in cold no-flow condition, much like the condition of TMI-2 during initial tests. The purpose of the Crystal River test was to provide a basis for data comparison. Results from the two test series have been compared with each other, as well as with laboratory bench tests on similar RTDs. Preliminary conclusions indicate that:

- The temperatures measured by the TMI-2 RTDs are reasonable. That is, the TMI-2 RTDs have not undergone a major shift in calibration.

- The resistance of the lead wires extending from the RTDs to the temperature transmitters was found to be normal.

- The resistance-to-ground has suffered degradation. A likely cause is moisture ingress in the RTDs or cables.

- The heat transfer resistances and the response time of the TMI-2 RTDs appear to have experienced degradation. This will be verified by data from the final testing at Crystal River.

The testing and analysis will be completed when a final series of tests is performed at the Crystal River facility in a hot functional operating mode. Results from these final tests will allow a correlation to be established between the cold, no-flow test data and the operation condition data and will allow for more meaningful conclusions to be drawn relative to the TMI-2 RTD conditions.

Plans are underway for removal of a dual element RTD and thermowell from the RCS A-loop hot leg. The sensors are two which exhibited extremely low resistance-to-ground during the in situ testing. Removal and examination of the sensors will make it possible to determine if the low readings were caused by a structural failure of the sensors or were due to an anomaly in the cabling, Reactor Building penetration, or both.

I&E studies performed on RTD/thermowell combinations have shown that using Never-Seez as a thermal conductor in the thermowell can cause gradual degradation in the response time of the assembly. Documentation of the TMI-2 RTD assembly installations is not specific with respect to using Never-Seez. Removing and flushing this thermowell would resolve the unknown of using Never-Seez and will assist in the evaluation of the survivability of the TMI-2 RTDs.

Time response tests will be conducted on a complete thermowell/RTD assembly to obtain laboratory confirmation of the in situ testing performed and to completely document the overall assembly postaccident condition and system performance. In addition, the Radiation and Environment Program will also examine the thermowell because it represents an internal metal surface that was in contact with reactor coolant and could be analyzed for deposition of fission products and possibly reactor control rod materials.

Pressure Transmitters. The 58 pressure-sensitive transmitters in the Reactor Building are safety-related instruments. Twenty transmitters failed during or after the accident. I&E personnel undertook the task of examining the transmitters to determine the cause of failure and to possibly suggest design changes that could prevent similar failures in the future.

In 1982, two transmitters had been removed from the Reactor Building and examined at the INEL. While neither unit was LOCA-qualified, it appears that the seals in one pressure transmitter were adequate to withstand the pressures, temperatures, and steam encountered during the accident, while those in the level transmitter were not. Three additional transmitters were removed from the Reactor Building and examined at the INEL for comparison and corroboration of the previous conclusions. Although the transmitters were not designed and fabricated to withstand a LOCA, investigators determined that they are sufficiently similar to the LOCA-qualified transmitters (which are in an inaccessible location in the Reactor Building) to provide information on failure mechanisms.

Examination of the three transmitters at the INEL supported the conclusions drawn from the earlier 1982 testing program. The Foxboro pressure transmitter, designed to monitor pressure in the core flood tanks, appeared to have survived the accident and the subsequent Reactor Building environment. Pressure calibration of the transmitter indicated that it was operational and in calibration. The internals of the transmitter shown in Figure 14 were well sealed from any moisture and appear to be in their as-manufactured condition.

The Bailey level transmitters, designed to monitor water levels in the core flood tanks, were not opera-

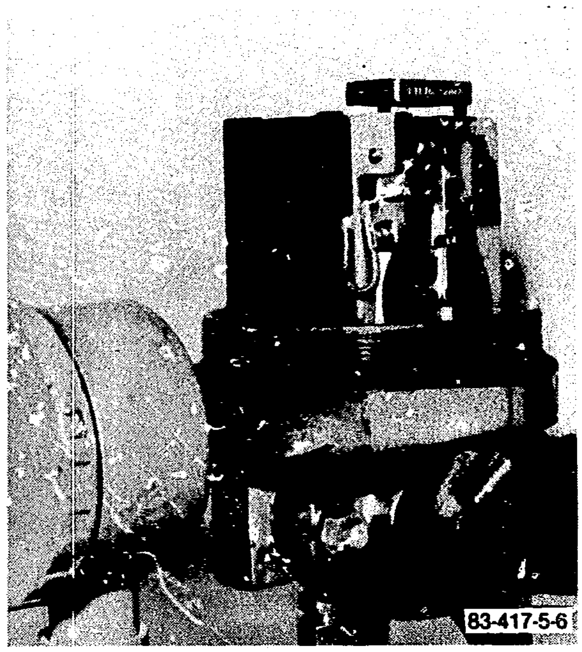

Figure 14. Internals condition of the Foxboro transmitter.

tional when examined at the INEL. Detailed examination of these transmitters revealed extensive damage to the internal components due to moisture intrusion. Some of the electronic components, as shown in Figure 15, were badly corroded with some of the leads actually corroded away. The pressure ports and gaskets of the corroded transmitters were examined for sources of potential leaks, but were found to be in good condition. Inadequate sealing of the electrical conductors as they entered the transmitters appear to be the most likely sources of moisture in the transmitters.

The mounting configuration of the transmitters on the core flood tanks is such that one Foxboro pressure transmitter and one Bailey level transmitter are located adjacent to one another in the same instrument rack. The conduits from each of the transmitters extend vertically for approximately one meter where they enter a common junction box. Removal and examination of the remaining Bailey core flood tank level transmitter, the adjacent Foxboro pressure transmitter, their common junction box, and the interconnecting conduits will provide an opportunity to determine if water entering the junction box could travel by way of the conduit into the electronics enclosure of the level transmitter. All the examination results will be considered together in an evaluation to determine why some of these pressure transmitters failed.

Incore Instrumentation. Since incore instruments are an integral part of the TMI-2 core, the damage patterns that they sustained may strongly correlate to core damage. I&E personnel performed tests on all the incore instrumentation in an effort to determine the extent of damage in the lower portions and

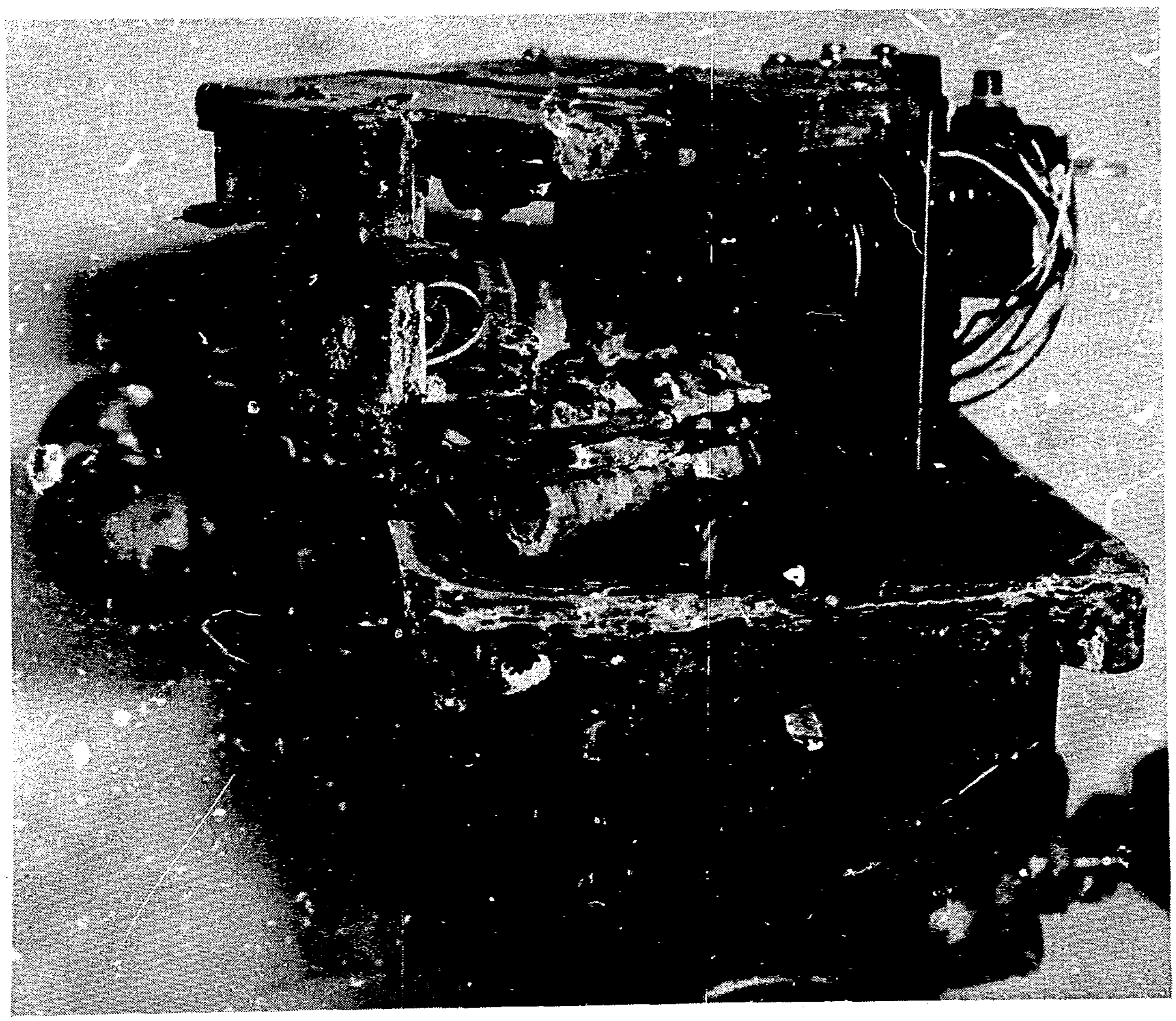

Figure 15. Severely corroded internals of Bailey transmitter.

visually inaccessible regions of the core. The incore instrumentation consists of detector assemblies located in instrument tubes or strings threaded up through the bottom of the reactor vessel into the center of 52 individual fuel assemblies throughout the core. Each detector assembly contains seven self-powered neutron detectors (SPNDs), one background detector, and one thermocouple. Each instrument string is 39-m long, and terminates at a seal table on the Reactor Building at the 347-ft elevation, above the vessel head, but below the highest point of the hot leg piping.

During 1983, I&E personnel conducted in situ tests on seven incore instrumentation assemblies. The tests were performed to supplement earlier tests in two respects. First, the tests provided more exact data on the out-of-reactor lead-in cables to remove uncertainties from the original tests, thereby allowing for a more precise quantification of incore instrumentation and providing a better assessment of core damage. Second, a new frequency response test, developed at the INEL, provided length measurements of the incore SPNDs to supplement the length measurements of the incore thermocouples. These data will also be useful in assessing core damage.

Although the data analysis is not complete, some conclusions and correlations can be made. The 364 SPNDs and 52 background detectors were evaluated together as a group of 416. Only 22 of these detectors appeared to be operational and these were mainly located in the lower areas of the core. While 331 detectors still exhibited the open circuit condition they need to operate, test data indicate that moisture had entered the insulation rendering their output unreliable. Fifty-six detectors had short circuits and were not operational. The remaining seven detectors appeared to have an open circuit, but the insulation resistance was not high enough for the detectors to be considered operational.

Present data indicate that all 52 thermocouples in the strings failed. Twenty-six had open junctions with moisture in the insulation and two had open junctions with dry insulation. The remaining 24 thermocouples have formed new junctions as a result of accident or postaccident conditions. The locations of the new thermocouple junctions vary as much as 4.3 m. The longest new junction is only 1.7-m shorter than the original while the shortest new junction is 6.2 m shorter than the original. Correlating the thermocouple data with the SPND data implies that the lower regions of the core could have also sustained damage during the accident.

Cables and Connections. Late in 1981, cable systems became a major focus of the I&E Program because (a) information on the survivability of the electrical cables was required for use in recovery planning, since all plant controls and monitoring information flow through cables, (b) analysis of these systems provides valuable information for monitoring postaccident conditions, developing cleanup plan parameters, and ensuring plant safety, and (c) the nuclear industry in general has a parallel need to asses conditions in order to predict near end of life performance for aged facilities.

Through its Cables/Connections Task, I&E Program personnel seek to determine the effects of the TMI-2 accident and postaccident conditions on the cables and connections within the Reactor Building. Cables and connections are defined as all components in a given electrical channel, or circuit, from the Reactor Building penetration assembly up to but excluding the instrument or component at the end of the channel. The components encompassed by this definition include penetration assemblies, penetration boxes, terminal boxes, terminal blocks, splices, bulk cable, and connections. The information and understanding generated as a result of the Cable/Connections Task group effort will reflect specific responses of this hardware to the TMI-2 LOCA and post-LOCA environment. While these task results will be specific to the TMI-2 facility, they will help to formulate basic guidelines regarding the response of cable and connection components to a LOCA event. Such guidelines will serve to help government regulators and the nuclear industry improve the reliability of these components, as well as help them re-evaluate qualification testing procedures and regulatory requirements in light of this actual LOCA event data.

A representative sampling of the electrical channels are being subjected to appropriate in situ electrical scan tests. The intent of these tests is to (a) identify anomalous channels for more comprehensive in situ characterization tests, (b) identify impaired channels for removal and laboratory examination, and (c) estimate what fraction of the channels may have been impaired. To date, 116 channels with 20 different cable types totaling 6,225 meters in length, have been tested in the mass scan. These consist of unique types of power, signal, and control cables passing through 11 Reactor Building penetrations.

Although data analysis is continuing, the one most predominant cable anomaly encountered during the tests is a shift in the cable's characteristic impedance. This anomaly may be caused by moisture ingress through the cable's insulation. Additional in situ tests are being developed for those cables exhibiting an anomaly during the mass scan test.

Once an anomaly has been exhibited in a specific length of cable, it may be difficult to locate the damaged portion of the cable for removal from the Reactor Building for detailed examination. This difficulty stems from the fact that the cable trays in the Reactor Building contain many cables, most of which are only identified at the end. In order to locate the cable to be removed within a particular cable tray, I&E personnel developed a cable tracing device. The cable tracer allows the technicians to positively identify the cable for cutting and removal. The device is currently undergoing proof-of-principle tests.

Seven sections of cable were removed from the Reactor Building and shipped to offsite laboratories for examination. The cables were not identified for removal by the mass scan testing, but were made available through other recovery efforts and their representative nature was judged to be sufficient to perform some offsite analyses. The sample sections include a power cable for a control rod drive motor, a signal cable for a control rod position indicator, an extension cable for a control rod drive thermocouple, a power cable for a reactor head fan, a control cable for the small jib crane, and two instrument cables for radiation monitors. The cables are undergoing both electrical and mechanical examinations.

Polar Crane Refurbishment. Following nearly a year of inspections, analyses, repairs, replacements, and no-load operational testing, the polar crane at TMI-2 is structurally, mechanically, and electrically ready to undergo load testing. The 500-ton Whiting polar crane installed in the TMI-2 Reactor Building was rendered inoperable as a result of the March 1979 accident. Figure 16 provides a schematic of the crane's main components.

Because of the polar crane's strategic importance in removing the reactor vessel head, GPU Nuclear through the Polar Crane Task Group decided inspections, refurbishments, and tests should be aimed at restoring functional capabilities of strictly the bridge, trolley, and main hoist mechanisms using pendant control. The industry experts mak-

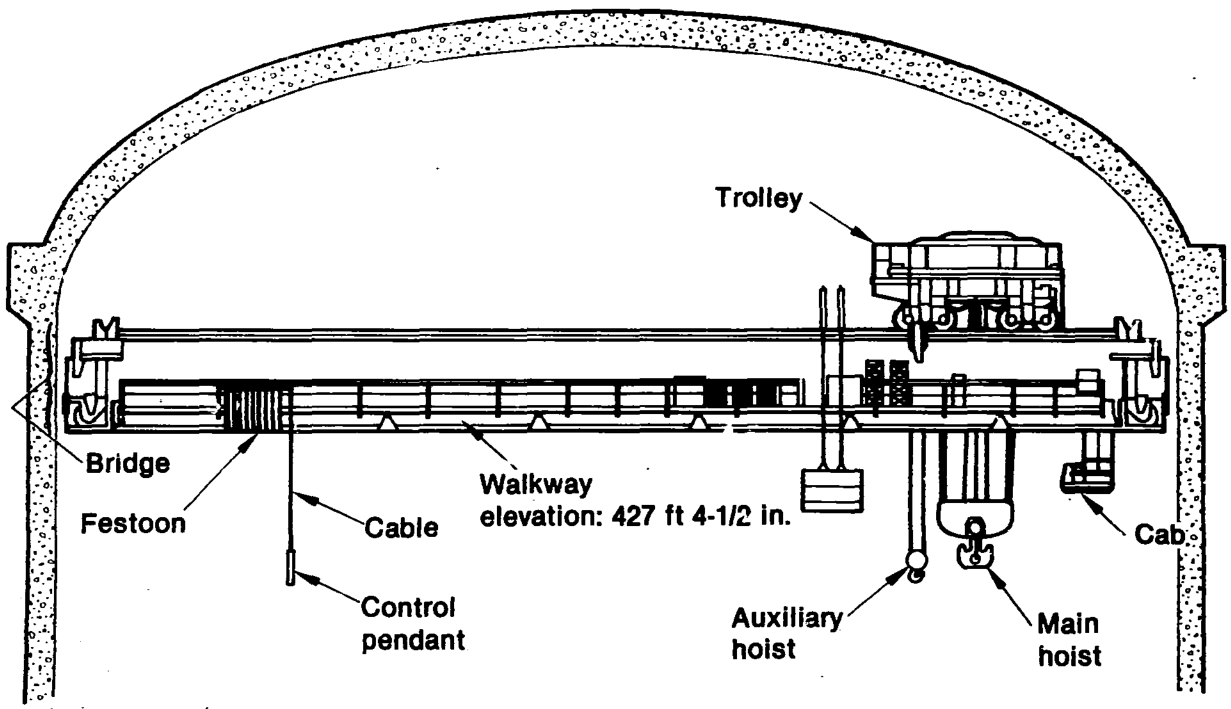

Figure 16. Major components of the TMI-2 polar crane.

ing up the task group agreed that concentrating recovery efforts to these operating functions would help to control costs, save time, and minimize man-rem exposure, while achieving the main objective of the polar crane recovery project, that is, re-establishing those crane motions necessary to move the missile shields and reactor vessel head.

Much of the damage to electrical systems and components appeared to be the result of the hydrogen burn. Unlike the reactor control components, which thus far have not exhibited any damage from the hydrogen burn, the polar crane was installed without the normal reactor grade equipment or precautions. The power and control conductor-collector system from the crane bridge to the trolley showed extensive damage. In some areas, the supporting insulators fractured, and in other cases, the insulating sheath material had softened enough that the conductor rail clip lost its grip. This entire system was replaced with a new flexible cable loop system rather than repair the existing system. The cables, approximately 30-m long, provide power and control interconnections between the bridge and trolley. A similar cable bypasses the crane conductor system to supply three-phase power to the bridge. This was a major cost savings and a reduction in personnel exposure to radiation in the restoration of the crane.

Among the electrical equipment that was replaced in-kind were 15 electrical relays and contacts in various control cabinets, which were corroded or malfunctioning. Five trolley and bridge accelerating resistor banks were also replaced because of corrosion or moisture related problems. Figure 17 shows a typical resistor bank installation and Figure 18 shows the damage to resistor bank insulation and windings.

The crane control pendant, which was superficially charred during the hydrogen burn but found in subsequent offsite examinations to have retained its functional capabilities, was replaced. The crane festoon, which was totally destroyed by the hydrogen burn, was also replaced. The control pendant, suspended from a cable, hangs down to the 347-ft elevation in the Reactor Building and provides a means to remotely operate the crane in lieu of operation from the cab. The cable hangs from a trolley system at the walkway handrail. The pendant and cable may be trolleyed back and forth along the walkway by means of the festoon.

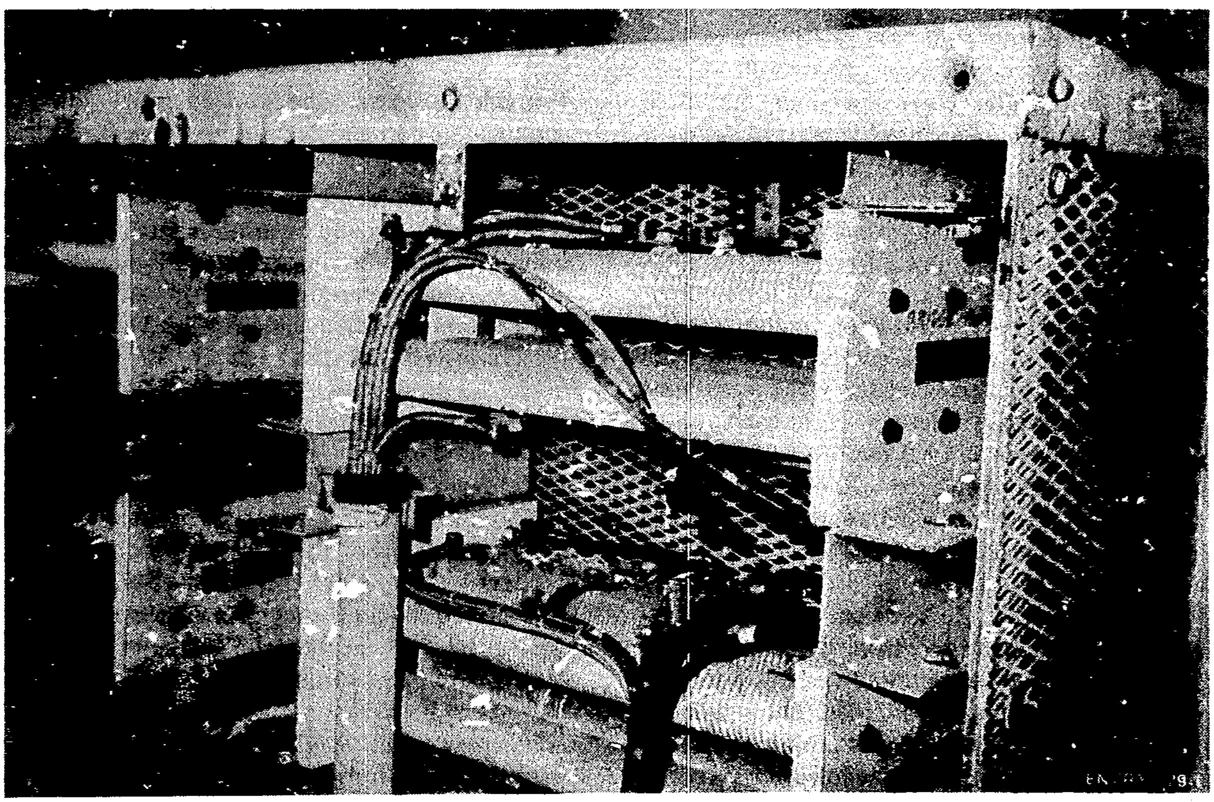

Figure 17. Typical polar crane resistor bank installation.

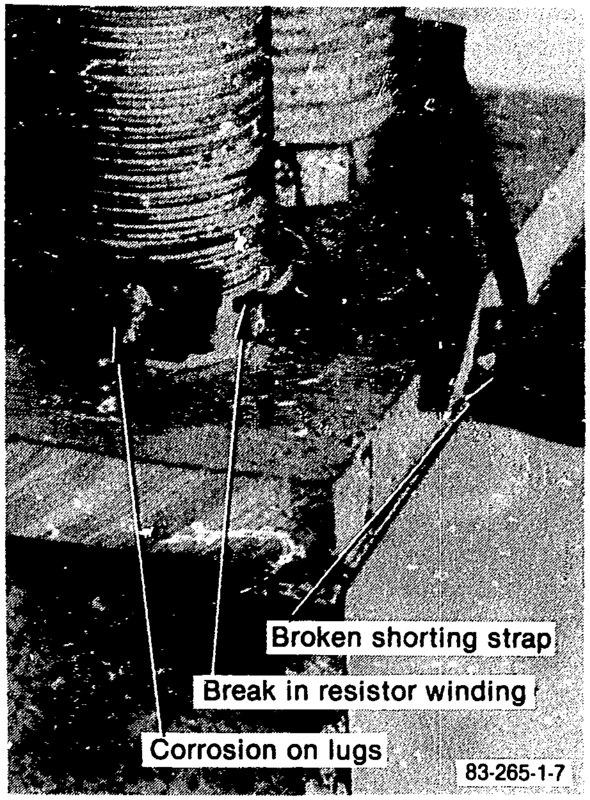

Figure 18. Insulation corrosion and winding breakage in the polar crane.

None of the crane's motors or clutches needed to be replaced, but were not declared electrically operable until after corrosion films had been removed from the slip rings. The metal conduit, which houses and routes electrical wiring around the polar crane, was not damaged by the accident, and none of the internal wiring showed signs of distress.

Mechanical damage was minor, as compared to the damage to the electrical components, with the only significant replacement being the main hoist magnetic drum brakes. Because of the extent of the corrosion and their critical importance to safety, these brakes were replaced without determining whether they were operable. The brake wheels, however, only needed cleaning.

Once recovery of the electrical and mechanical components of the bridge, trolley, and main hoist was complete, they were tested without a load to verify their operating capabilities. The remaining effort in the polar crane recovery project is full load testing of the polar crane. Scheduled for spring of 1984, the test calls for the hoist to lift about 210 tons so it may be certified to 170 tons—the approximate weight of the reactor vessel head. A frame holding five missile shields, with an approximate total weight of 192 tons, as well as cables and associated lifting equipment will provide the weight. The test load will be lowered, stopped and held by the hoist brakes, and finally lowered to the floor. Then it will be transported 3 m out and back by the trolley and at least 3 m from side to side by the bridge. The load will be lifted again after rotating the bridge 180 degrees, completing recertification of the crane for reactor vessel head removal.

Radiation and Environment

The Radiation and Environment Program consists of three major areas of interest; Primary Systems, Reactor Building and Support Systems, and Accident Evaluation. Program objectives, defined by those areas of interest, include measurement of transport and deposition of fission products from the reactor core to various associated systems and the Reactor Building environment, and assessment of the accident effects on the plant. The data resulting from the Radiation and Environment program can significantly enhance current understanding of nuclear plant accident environments and the phenomena that contribute to those environments.

Primary Systems. Engineers collected data concerning transport and deposition of core fuel debris, fission products, and activated corrosion products in the RCS and its associated components. Fission product transport and deposition data collection is important to the radionuclide mass balance and source term programs and will also provide information on radionuclide distribution to support the removal of the vessel head.

The fission products made their way from the reactor core to the basement through the RCS, the PORV, and finally through the reactor coolant drain tank (RCDT) rupture disk. These areas are the major sources for core debris and fission products outside the reactor vessel. Material transported to and through these primary systems during the accident were the source of all fission products released to the plant environment.

Reactor Coolant Drain Tank. The RCDT was the major pathway for release of accident water to the Reactor Building basement. A component of the Reactor Coolant Leakage Recovery System, the

RCDT collects, cools, and stores leakage discharged from the PORV, reactor coolant pump seals, and reactor vessel closure head gasket. Total water release volume based on the borated water storage tank inventory during the accident, along with the sump pump volume, RCDT volume, and the sump pump transfer volume, indicates that 1009 m^3 of borated water was originally introduced into the basement during the first three days following the accident. For over two years following the accident, in-leakage yielded an additional 715 m^3 of water to the basement or 71.4% of the total basement water volume of 1724 m^3. Analysis of samples from the Reactor Building basement floor have shown significant levels of fission products and measurable quantities of fuel, fuel cladding, and core debris associated with liquid and solid samples.

To access and inspect the RCDT, holes were drilled through the concrete floor on the 305-ft elevation and the upper surface of the elbow of the tank's vent line. This allowed engineers to perform visual surveys and to take liquid and particulate samples from inside the tank.

A video inspection of the area of the tank below the vent line was conducted in December using a fixed focus camera by technicians on the 305-ft elevation floor. A dark particulate sediment was nonuniformly distributed on the bottom of the tank. Following this inspection, a 140-mL sample of liquid was extracted about 0.6-cm from the bottom of the tank.

A second video inspection of the inside of the tank was made using a camera equipped with remotely controlled iris and focus. The quality of the image was much improved over the previous inspection. The sizes of the particles were larger than had been predicted for particles released through the pressurizer. The sediment was not uniformly distributed and appeared to be less than 0.32-cm thick. A sample of sediment and water was obtained from the area of the bottom surface of the tank that is directly below the vertical section of the vent line. The sample was described to be milky in appearance which implies that particulate matter was collected.

Preliminary gamma spectroscopy analysis results for the liquid sample were obtained during December. Radionuclide concentrations of detected radionuclides are presented in Table 4.

Table 4. RCDT preliminary sample analysis results

Radionuclide	Concentration (μCi/mL)
^{134}Cs	5.0 ± 0.19 E-2
^{137}Cs	9.9 ± 0.08 E-1
^{190}Sr	3.8 ± 1.3
Gross Beta	5.0 ± 0.14
^{3}H	3.5 ± 0.07 E-2

These results are similar to the concentrations measured in the RCS prior to its drain-down in August. This implies that there was significant RCS water leakage through the RCDT to the Reactor Building basement. The sediment will be analyzed early next year for gamma-emitting radionuclides, ^{90}Sr, ^{129}I, tellurium, and fuel. Its mass density and particle size distribution will also be measured. Fractions of the filtered solids will be sent to HEDL for the scanning electron microscope analysis. These data will be used to estimate the maximum flow rate through the PORV and the average RCS leak rate since the time of the accident. In addition, the results should provide information on the long-term leach rate data for fission products transported to the RCDT in fuel particles.

Mobile Gamma Ray Spectrometer. A Technical Evaluation Group (TEG) meeting held to develop a plan and priorities for future TMI-2 sampling, recommended that gamma spectral assays of RCS piping and components be performed. It was anticipated that these in situ gamma spectral measurements will help quantify the distribution of fuel and fission products throughout the RCS and will provide information on selection of physical samples.

To achieve this measurement objective, the EG&G mobile gamma ray spectrometer that uses a high-resolution, hyper-pure germanium detector will be used. Initial calibration of the spectrometer was performed by The Pennsylvania State University (PSU) during July. The calibration provided absolute counting efficiency curves for energies from 80 keV to 3 MeV for point and plane sources at several different fixed distances from the detector. The counting efficiency curves that are the end product of these calibration measurements will be used to convert spectral data to radionuclide concentrations. The quantification of fuel will be achieved using the established proportionalities between radionuclides such as $^{144}Ce/Pr$ and $^{154,155}Eu$ and fuel.

Following the completion of the calibration work at PSU, the gamma ray spectrometer was loaned to the Power Burst Facility at the INEL. It was used to quantitatively assay the test train discharge steam line after the completion of the Severe Fuel Damage 1-1 experiment. The concentrations of radioiodines and telluriums plated out on the internal surface of the steam line were measured at about a dozen locations. These data will improve current understanding of fission product transport and deposition mechanisms.

On-line operating procedures and a training manual for the system were prepared and plans were approved by GPU Nuclear for the collection of spectra from selected tanks and pipes located within the Auxiliary and Fuel Handling Building and from standardized sources that will be prepared onsite. These data will supplement other calibration data previously obtained and will assist GPU Nuclear in determining appropriate cleanup measures for these components.

The logistics of gamma scanning the pressurizer, the bottom of which is the location having the highest probability of containing a significant amount of fuel, and also of gamma scanning the once through steam generators were examined and a procedure for performing the scans was developed. Since the polar crane will have to be used for these measurements, it is likely that the gamma scans will not be performed until mid 1984. These scans will allow an estimate to be made of the total fuel and fission product inventories of these RCS components.

Reactor Building and Support Systems. The Reactor Building and Support Systems are areas to which fission products were released from the RCS, primarily through the PORV, then through the RCDT rupture disc to the Reactor Building basement. The support systems were the principal barrier to fission product release to the outside environment following release of accident water to the basement. Analyzing samples to determine the location, quantity, and radioactive nature of fission products released to each system provides information about the radionuclide release deposition and mitigation capabilities within the building environment which will provide additional information for the ongoing radionuclide mass balance program.

Reactor Building Air Coolers. The Air Cooling System circulates air through the Reactor Building. Operating continuously, the system circulated the Reactor Building atmosphere through fans into the air cooling plenums and past air cooling coils that use circulating river water as a cooling medium. Four of the system's five motor driven air fans were operating when the accident occurred. Because of heat exchange and circulation processes, the fans' internal surfaces and cooling coils contain radionuclide deposits and airborne particulate matter that plated out on surfaces during the accident. Radiation surveys and gamma spectral measurements at the top of the air coolers near the fans indicate radiation levels a factor of 10 higher than the general area radiation surveys which would indicate a larger fraction of radionuclide deposition.

During March and April 1983, the five access panels on the Reactor Building air coolers were removed from the Reactor Building. Because the air coolers were in operation during and after the accident, it was believed that any deposits on them could provide some explanation of the atmosphere and environment during the time of the accident. Following inspection of the air cooler interior surfaces above the cooling coils, smears were obtained from each access panel and scrape samples were obtained from only two panels. In addition, smear and scrape samples were obtained from air cooler A intake louver. The smear sample indicated activity levels of 460-mR/h gamma and 5-R/h beta. Preliminary analytical results of these samples, which are being analyzed at the PNL onsite mobile laboratory, indicate the presence of mixed fission products, including $^{144}Ce/Pr$, a fuel precursor. Also ^{137}Cs levels were found to be as high as the 5.5 μCi/smear sample taken from air cooler A's intake louver. Smear and scrape sample analysis at PNL will be completed during early 1984. Fission product surface deposition analysis has been initiated and will be completed during 1984.

Reactor Building Basement Samples. Samples obtained May 14, 1981 from the approximately 2470 m^3 of water discharged to the Reactor Building basement revealed that the liquid was a uniform solution; no stratification of dissolved fission products occurred from top to bottom in the

2.44 m of water. Samples from the basement floor included a solids fraction that contained greater than 90% of the insoluble fission products and core materials.

In January, three additional samples taken from different areas of the basement floor yielded three different amounts of solid material as shown in Figure 19. Samples 1 and 2 are both brown and similar in consistency, as compared to Sample 3 which is grey and contains three large particle agglomerations. These samples indicate that solids are apparently very heterogeneous in distribution on the basement floor. The analysis results show that the principal radionuclide in liquid and solids are ^{137}Cs and ^{90}Sr respectively. Also present are fuel, control rod material, cladding, and other materials found in fuel elements. Two Reactor Building sump pump discharge samples were collected during August and will be analyzed in early 1984. Total radiation readings for these samples were 4 R/h gamma and 70 R/h beta.

This sampling effort will contribute to the calculation of the total solids in solution in the sump water. Combined with the measured concentrations of radionuclides in the solids fraction of these samples, the data will allow calculation of the total radionuclide inventory of the recirculated sump water. Sequential measurements of this type made during decontamination and processing would provide the desired total mass of the sediment in the basement.

Surface Deposition and Environment. The Reactor Building and internal system surfaces were exposed to released fission products and core materials to varying degrees during the course of the accident. A "hot house" effect occurred in the Reactor Building when large quantities of hot reactor coolant with entrained fission products were released with steam, to the much cooler building atmosphere.

Entrained fission products were released and distributed by a number of mechanisms depending on the nature of the elements. Volatile fission products, such as xenon, were released and remained in the building atmosphere except for small fractions which dissolved in the liquid coolant. Soluble fission products such as cesium were probably carried in water droplets and later ran into the basement

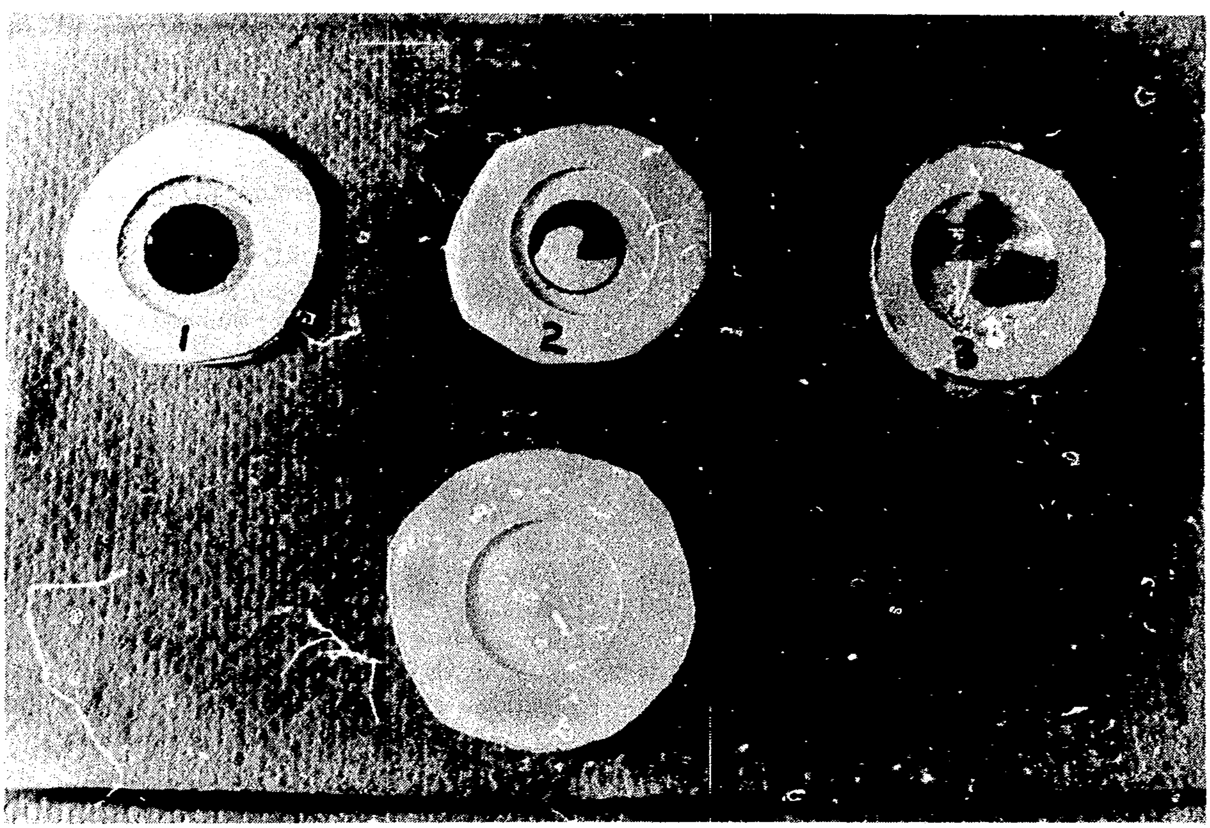

Figure 19. Reactor Building basement filtered solid samples obtained January 11, 1983. The filters are numbered according to sample location, the blank filter is for comparison.

as droplets condensed on cooler upper building surfaces. Fission products such as iodine have both volatile and soluble chemical species that were transported and deposited by both droplet and solution mechanisms. Insoluble fission products such as cerium, or core products such as uranium and silver, were carried by mechanical action as particulates or aerosols in water droplets. Radiation surveys show that vertical dose rate levels are about a factor of 10 lower than horizontal surfaces which retained fission products after the water on the surfaces evaporated.

The emphasis on surface deposition sampling has been to characterize steel and painted and unpainted concrete, to determine how deposition occurs on each of these major surfaces. Surface deposition studies conducted during 1983 characterized activity deposition on each of the major TMI-2 Reactor Building surfaces.

Characterizing the deposition on Reactor Building surfaces is occurring in the basement and all areas from the ground level up. In the levels of the building from the ground, or 305-ft elevation, up to the polar crane area, there was evaporation, condensation, and standing water on surfaces, which later evaporated. As a result, radionuclides in this region were heterogeneously distributed. Additional surface samples were collected from this region during September, using a concrete boring tool that in each case extracted a core of concrete 5 cm in diameter and about 5-cm long. A concrete core sample taken on the 305-ft elevation floor in front of the air coolers is shown in Figure 20. Analyses of these core samples is ongoing and is being directed to determine the extent to which radionuclides migrated into the concrete.

The analyses of the milled samples collected during 1981 and 1982 were completed and published in October 1983 as GEND 037, *Surface Activity and Radiation Field Measurements of the TMI-2 Reactor Building Gross Decontamination Experiment.*

The sampler was designed to allow both collection of loose particulates from a surface and milled subsurface samples containing both paint and concrete or steel. The beta and gamma exposure rates that were measured are displayed in Figures 21 and 22, which are, respectively, the floor plans of the 305- and 347-ft elevations. Sampling locations indicated by circles correspond to horizontal surfaces and those indicated by triangles correspond to vertical surfaces.

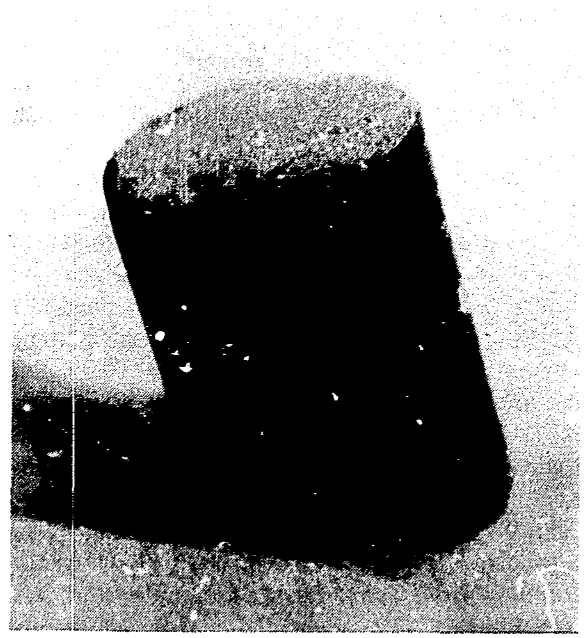

Figure 20. Concrete core sample taken from the 305-ft elevation.

Prior to decontamination, floors on both Reactor Building elevations were considerably more contaminated than walls. The mean December 1981 concentrations of ^{137}Cs and ^{90}Sr on the 305-ft elevation were, respectively, 3.6 ± 0.9 and 0.17 ± 0.04 $\mu Ci/cm^2$ and their concentrations on the 347-ft elevation floor were 2.5 ± 0.7 and 0.3 ± 0.2 $\mu Ci/cm^2$, respectively.

Sufficient surface activity data was collected before decontamination to make possible calculations of the fractions of the total core inventories of certain fission products that were deposited and which remained on Reactor Building surfaces. The total activities of ^{137}Cs, ^{90}Sr, ^{129}I, and ^{125}Sb were calculated using predecontamination surface concentrations and the surface areas of horizontal and vertical concrete and steel surfaces within the Reactor Building. These activities were divided by their respective core inventories that were calculated using the ORIGEN-2 computer code. The fractions of the core inventories of these nuclides deposited on Reactor Building surfaces prior to decontamination are all very small. The calculated values of the fractions of the core inventories for these four nuclides are 3.5 ± 0.4 E-4, 2.4 ± 0.8 E-5, 5.7 ± 0.5 E-4, and 5 ± 2 E-5, respectively.

Hydrolasing or high pressure water spraying the Reactor Building floors and walls, reduced the mean ^{137}Cs surface activity on the 305- and 347-ft eleva-

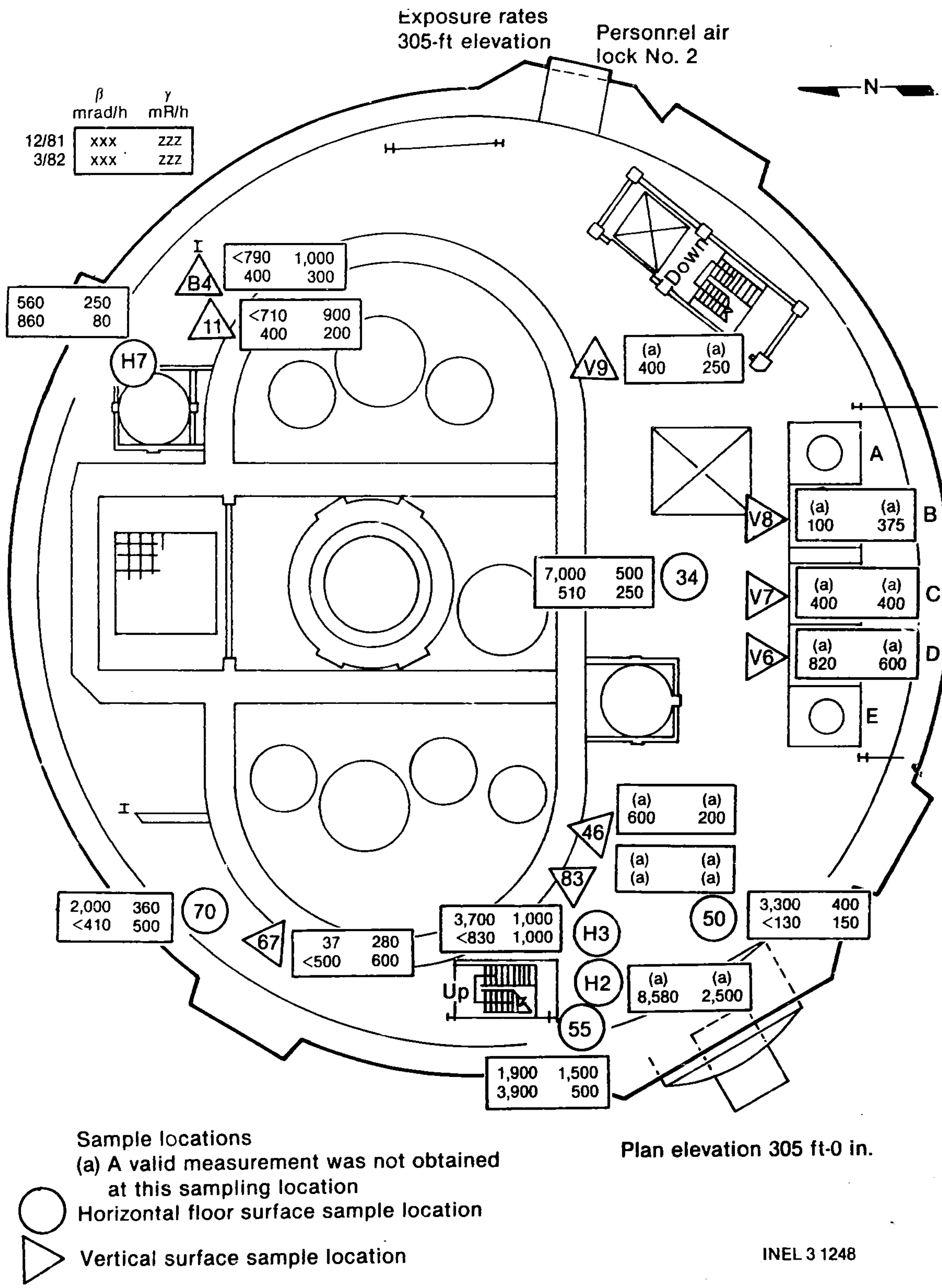

Figure 21. December 1981 and March 1982 beta and gamma exposure rates at the 305-ft elevation (shown on the 305-ft elevation floor plan).

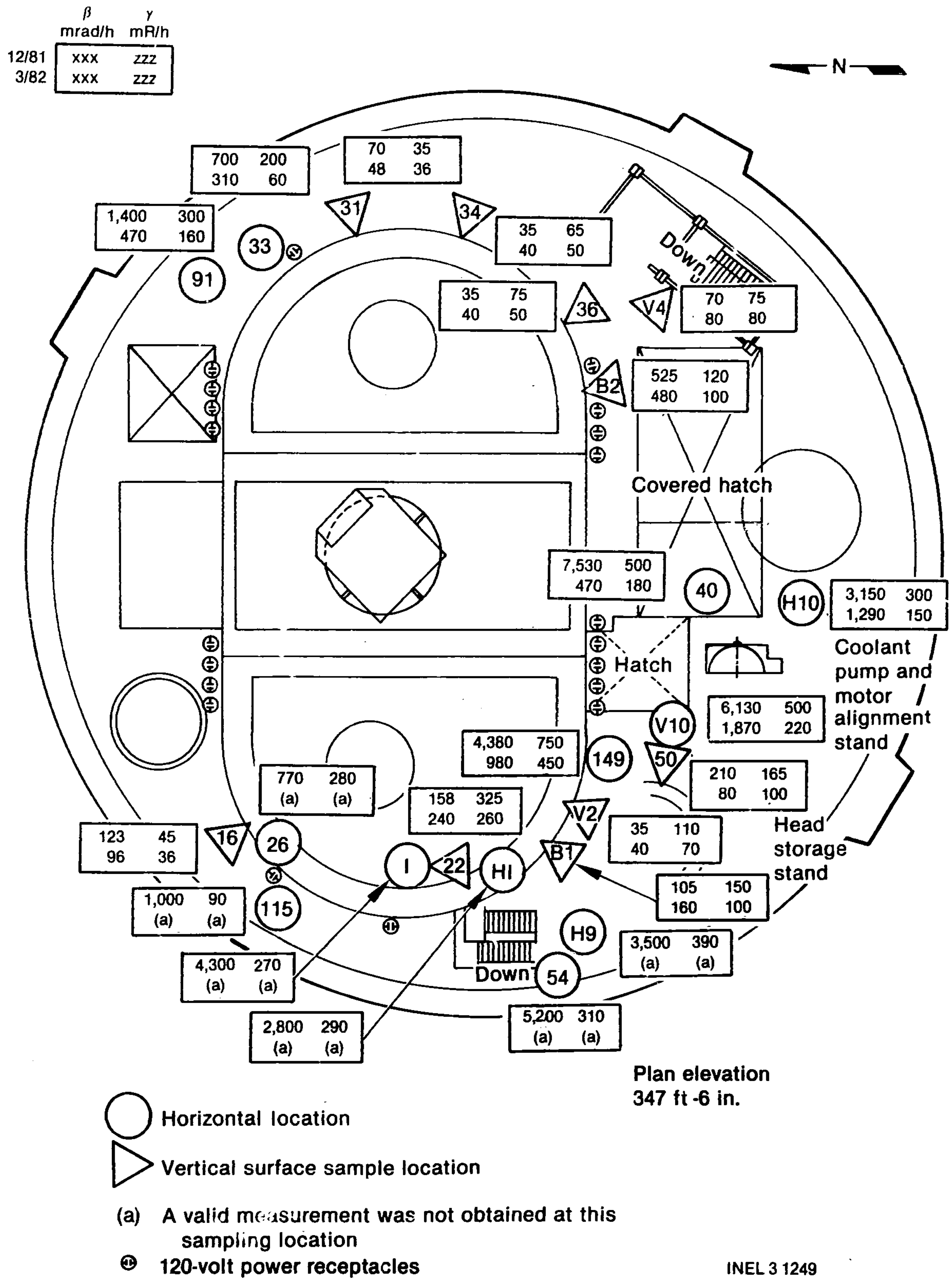

Figure 22. December 1981 and March 1982 beta and gamma exposure rates at elevations 347-ft, 6-in., 367-ft, 4-in., and 369-ft 6-in. (shown on the 347-ft 6-in. elevation floor plan).

tion floors by a factor of 20 and 13 respectively. The mean surface concentration of ^{90}Sr following decontamination was reduced by a factor of 30.

The decontamination effectiveness achieved for ^{60}Co, ^{90}Sr, and ^{137}Cs activities fixed to Reactor Building floors was about the same but it was less than half as effective on fixed ^{125}Sb and ^{235}U. The data also show that particulate ^{125}Sb activity was less effectively removed than the particulate ^{90}Sr and ^{137}Cs activities. Since ^{137}Cs is soluble in warm water and ^{125}Sb and ^{235}U are not, it is likely that standing water played as important a role in removing ^{137}Cs and ^{90}Sr activity from floor surfaces as did the abrasive action of the water emitted from the hydrolaser.

One of the primary motives for performing beta and gamma exposure rate measurements at sampling locations was to determine how well the beta and gamma radiation fields correlated with surface activity. If it could be shown that the fields immediately above a surface were linearly related to surface activities on that surface then it would be possible to infer surface activities using the results of a quick and simple measurement. The surface activity measurement results revealed that ^{137}Cs was the predominant radionuclide on Reactor Building surfaces both prior to and following decontamination and that the concentrations of other radionuclides were, on the average, proportional to ^{137}Cs concentration. Therefore, ^{137}Cs was chosen as the nuclide having the best potential for being simply related to both the gamma and beta radiation fields.

Least-squares fits of the data were performed to determine the correlation between surface activity concentrations and local exposure rate. Of the data sets fit with a power function, the December 1981 beta exposure rates set exhibited the best correlation. These data are shown in Figure 23. Given the relatively long half-lives of the predominant nuclides, the power function and the calculated coefficients were used to estimate ^{137}Cs (and consequently other nuclides) surface concentrations on Reactor Building surfaces at locations that were not sampled.

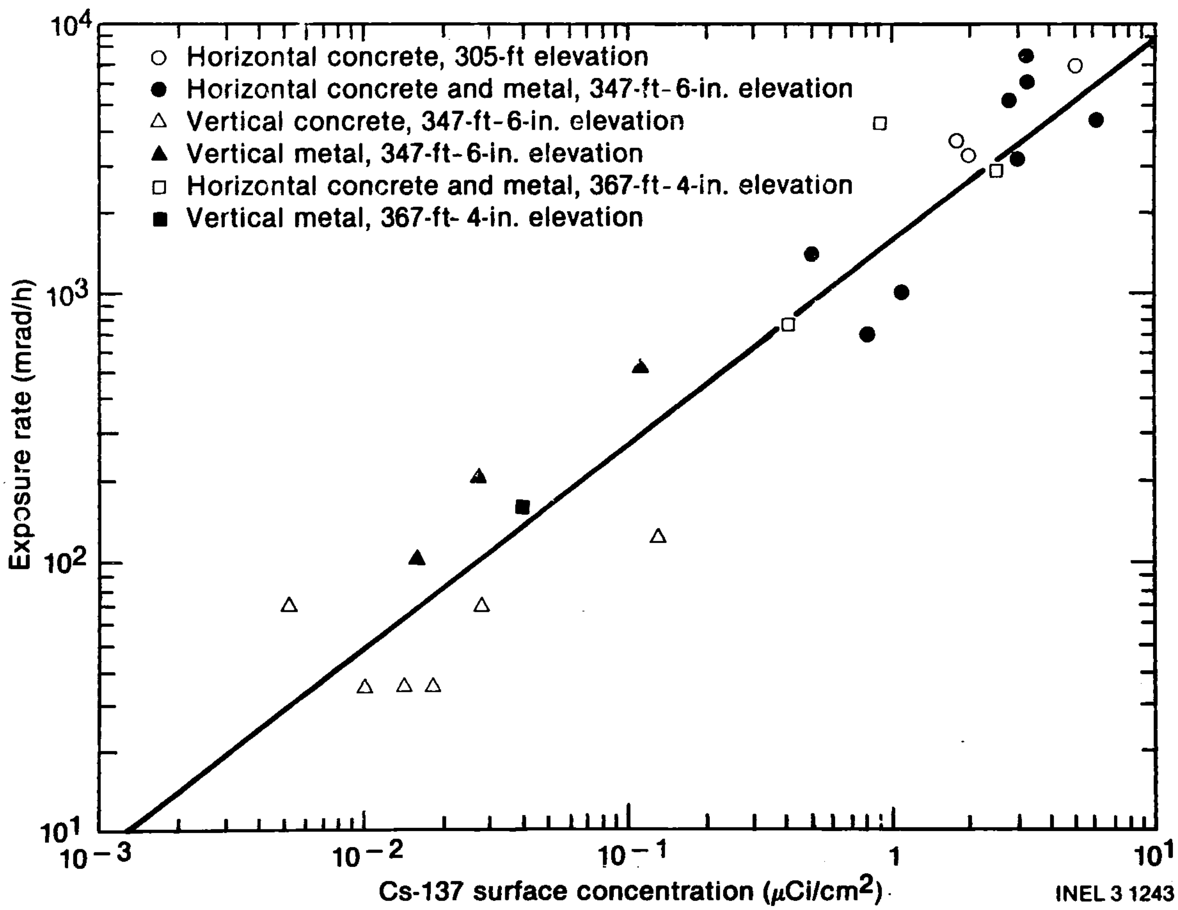

Figure 23. December 1981 ^{137}Cs surface concentration versus beta exposure rate.

Reactor Building Visual Surveys. Because the Reactor Building basement is still highly contaminated, characterizing that region of the building was restricted to the use of remote sampling equipment and closed circuit television cameras. During each survey, the camera was lowered through a penetration in the floor and video surveys were made of the region of the basement in the vicinity of the penetration. Since the camera could be operated from above, visual information was obtained with a minimum man-rem exposure.

Reactor Building basement visual surveys were performed during September. The objective of this task is to provide needed information concerning the conditions of building structures and equipment with special emphasis placed on determining the distribution of basement water sediment and the condition of cable trays, piping, and conduit runs located just below the 305-ft elevation.

Video surveys indicated that relatively large areas of the basement floor are clear of solids, but there is evidence of rust and corrosion. Inspection of the basement ceiling revealed the presence of debris, discolored surfaces, and charred cables. During another survey, the RCDT leakage transfer pumps, the baffle wall that encloses them, and the floor and walls in the general area were examined. The pumps appeared rusted but were intact, walls displayed a familiar bathtub ring, and sediment on the floor appeared to be uniformly distributed.

The video surveys showed no obvious indications of serious physical damage resulting from the accident. Some corrosion of carbon steel and painted surfaces was observed. The bathtub rings left after the basement water was processed are distinct and are present on all vertical surfaces that were exposed to the water. In 1982, sediment was observed on the basement floor to be spread evenly in a thin layer over the floor. By April 1983 large areas of the basement floor were apparently bare with no visible solids deposit. The migration of solids could possibly be attributed to SDS basement water processing and additional water added from hydrolasing off the floors and walls on the 305- and 347-ft elevations.

During February, EG&G obtained air samples in the TMI-2 Reactor Building to determine type and size of aerosols for estimates of recontamination. The samplers used were settling samplers and cascade impactors. The settling samplers were placed in-containment for a period of 4 hours. The cascade impactor sampling system was modified and recalibrated using aerosols of known sizes, and also placed in-containment.

The particle size distribution was found to be bimodal. Particles greater than 20 μm aerodynamic diameter accounted for 30% of the activity. Particles less than 5 μm aerodynamic diameter were associated with 60% of the activity. Aerodynamic diameter means that the particles settled or impacted as would a unit density spherical particle of the given diameter. Particles having aerodynamic diameters of less than 5 μm have actual diameters of 2 to 3 μm or less and were spherical and white. Raman analyses performed on these particles displayed characteristics of boric acid. In conclusion, the large highly settleable particles mostly originate in the air cooling fans and the smaller particles are mainly the result of the cleanup operation.

Accident Evaluation Damage Assessment.

Hydrogen Burn. The hydrogen burn investigation at TMI-2 was concluded with work performed by Factory Mutual Research Corporation to review unanswered questions concerning the burn phenomena. Factory Mutual investigators concluded that there was no reason to doubt the estimate of 370 kg of hydrogen in the Reactor Building at the time of the burn. Mixing calculations indicated that hydrogen was well-mixed throughout the Reactor Building with the average hydrogen concentration being in the range of 7 to 8% hydrogen by volume. There were somewhat leaner concentrations in the unventilated enclosed stairwell, and richer concentrations in the vicinity of the release site (RCDT rupture disc) and the open stairway. Ignition is generally agreed to have occurred in the Reactor Building basement, but the precise location and ignition source cannot be pinpointed. Flame propagation occurred upward and outward from the burn origin and eventually encompassed the entire volume of the building.

Also, during the last year, a task was instituted at Sandia National Laboratory to perform a "post-test prediction" of the TMI-2 hydrogen burn event. This task will use the HECTR computer code to provide a comparison of predicted and measured Reactor Building conditions during and after the burn. Industry Degraded Core Rule Making (IDCOR) is also using results from the TMI-2 hydrogen burn investigation to evaluate and benchmark the MAAP computer code.

Accident Evaluation Mass Balance. Principle efforts during the last year have been to determine what specific data is needed from TMI-2 to better define severe accident source terms, to lay out an organized/prioritized program to obtain the needed data, and to obtain data during "windows of opportunity." Radioiodine behavior at TMI continues to be a significant question. Tellurium behavior during and following the accident is at present undefined, but it is important to obtain data on its release and transport characteristics. Cesium was found plated out in significant quantities on TMI-2 leadscrew surfaces. An analysis effort with severe accident behavior computer codes was conducted to quantify predicted core damage state, hydrogen release, and fission product release, transport, and deposition, and to maintain a close tie between the code development community and TMI-2 data.

Radioiodine and radiocesium releases at TMI-2 total about 20 to 25% and 50% respectively of the core inventory. Current predictions imply that the same fractions of radioiodine and radiocesium would be released from the core in a severe accident. The factor of 2 difference measured at TMI-2 could be due to retention of iodine in the fuel/structural material/control material debris, retention of iodine on RCS surfaces, i.e., upper plenum, or errors in the iodine concentration data.

Efforts are underway to locate the not yet accounted for radioiodine at TMI-2. The search is also being made in the Reactor Building basement and RCDT. Also, attempts are being made to quantify errors in the original iodine data.

During the past year, it was determined that a considerable amount of radiocesium was retained on surfaces of leadscrews located in the reactor vessel upper plenum. If other similar surfaces in the upper plenum region retained a similar amount, up to 5% of the core inventory may have been retained in these locations. Information developed by researchers at Sandia National Laboratories indicates that there may be a chemical combination of cesium and silicon oxides on the leadscrew surfaces. During the next year, examinations of the TMI leadscrews will be made to determine if this cesium-silicon oxide interaction occurred at TMI-2.

Also during 1983, an effort was instituted at ORNL to investigate cesium behavior at TMI-2. This effort includes correlating all the TMI-2 radiocesium data and determining if the observed behavior is consistent with accepted cesium behavior knowledge.

The mass balance data base was broadened to include data through the end of 1979. Figure 24 shows the summary of release fractions based on samples obtained during that period. Additional sampling of the core, reactor vessel, and RCS surfaces are needed to determine the distribution of those fission products not yet found and to lend increased confidence to those measurements already made.

Computer predictions were made on the TMI-2 accident sequence with MIMAS (Multifield Integrated Meltdown Analyses System) and the TRAC-PF1 (Transient Reactor Analysis Code) at Los Alamos National Laboratory to provide a prediction of core condition and fission product release.

The MIMAS and TRAC results were used as input to a calculation of TMI-2 fission product transport and depositions with the RETAIN (fission product behavior) computer code. Results of these calculations will be reported during 1984. Also, the SCDAP (Severe Core Damage Analysis Package) was used to provide a prediction of the TMI-2 core condition, hydrogen generation, and fission product release. These results will be reported during the next year.

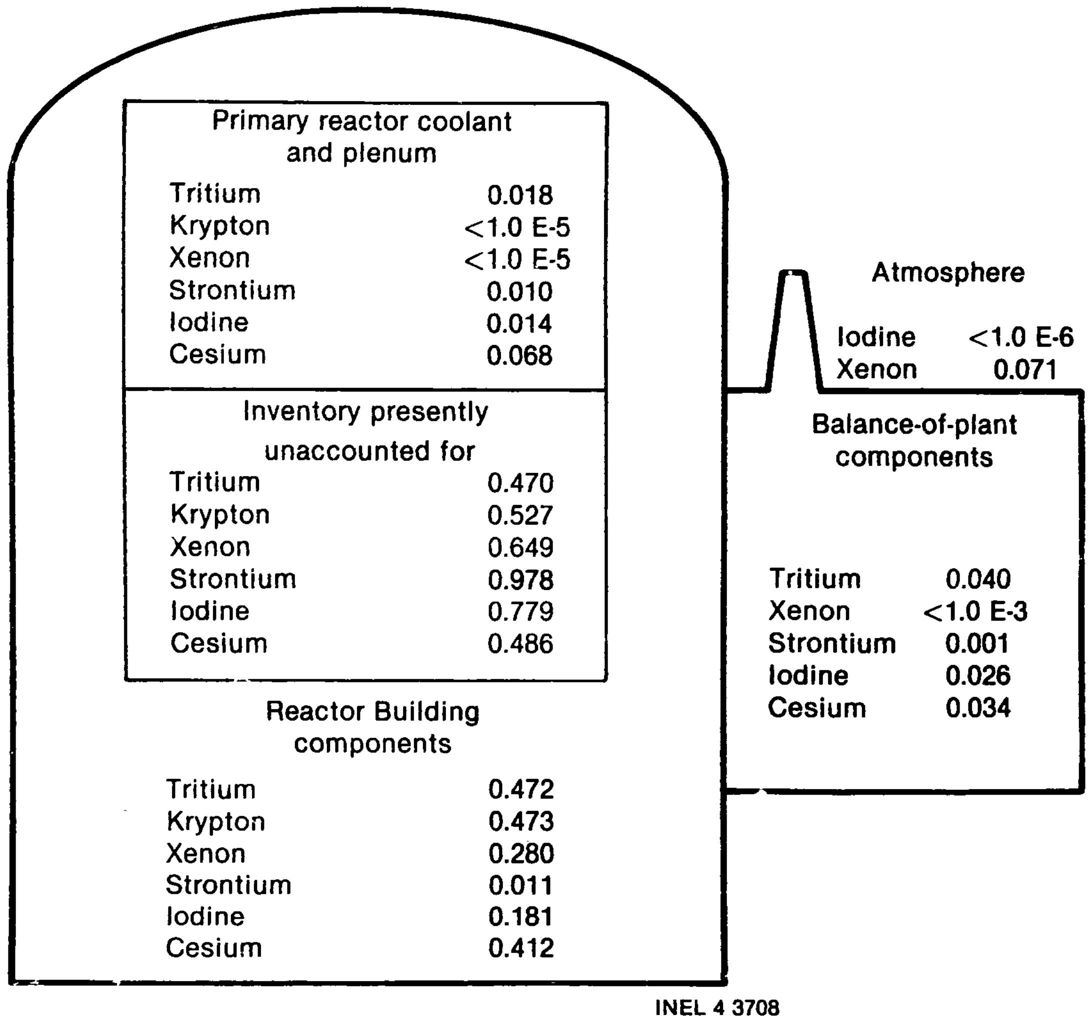

Figure 24. Distribution of key radionuclides resulting from a preliminary mass balance (μ ci/cc).

INFORMATION AND INDUSTRY COORDINATION

At the outset of the DOE program in 1980, it was recognized that data related to several areas of interest to the nuclear industry and regulatory agencies would be developed by virtue of DOE activities at TMI. Accordingly, information transfer mechanisms were established to ensure that valuable information derived from the DOE TMI program would reach all concerned parties in a timely manner. In 1982, DOE established the Information and Industry Coordination (I&IC) group at its TMI-2 office to coordinate technology transfer activities.

I&IC serves as a link between TI&EP work and the nuclear power industry. The I&IC group keeps the industry informed of TI&EP developments with computer conferencing networks and technical meetings. I&IC also provides a channel for feedback from the industry, using not only computerized technology transfer, but also discussion at technical meetings and contacts with individual companies. During 1983, I&IC continued its activities in technology transfer, application of TI&EP work, and investigation of industry issues.

Technology Transfer Mechanisms

Technological developments from DOE programs have been transferred to the nuclear industry, and have altered industry perspectives on many of the issues confronting the nuclear community today. Each technology transfer mechanism is designed to reach a specific audience. Research results are evaluated to determine technical content, applicability to the industry, and the target audience.

Nuclear Operation and Maintenance Information Service. Maintenance and operations personnel at commercial nuclear facilities comprise the audience using the Nuclear Operation and Maintenance Information Service (NOMIS). Approximately 80% of U.S. nuclear generating capacity and 50% of the free world nuclear generating capacity participate in the program. The format allows a user to query members on maintenance, operations, and design problems and their solutions. The information must be of sufficient importance to warrant alerting such a large audience. A unique feature of this system is that receivers of information are required to provide feedback to the originator on how they used the information. TMI has transmitted 10 messages to date and 145 responses have been received.

Electric Power Research Institute. I&IC is working with a utility-funded research organization called the Electric Power Research Institute (EPRI) to distribute TMI program results. EPRI has several technology transfer systems which are appropriate for use by the DOE TMI-2 programs.

The EPRI Technical Briefs are short reports which summarize research results by briefly describing an industry problem and then outlining a solution. Normally, 4,000 copies are printed and distributed to responsible utility project managers. I&IC has prepared technical briefs on radiation monitoring operational deficiencies and installation problems, gas generation in ion exchange resin waste containers, loose parts monitoring system failure mechanisms, and the pressure transmitter examination program.

Nuclear Notes is a newsletter that is published quarterly and has a distribution approaching 700. Recipients include vendors, equipment manufacturers, and universities, as well as utilities. Information has been published on TMI program investigations of pressure transmitters, flow transmitters, Reactor Building polar crane refurbishment, control rod drive lead screws, equipment damage mechanisms, and preventive measures. TMI contact is provided for participants who desire more detailed information.

Institute of Nuclear Power Operations. The Institute of Nuclear Power Operations, or INPO, is a utility-sponsored organization formed in response to a recommendation made after the accident by the President's Commission on Three Mile Island. Two mechanisms are used to transfer technology to the industry.

INPO is responsible for the administration of a world-wide computer conferencing network called Nuclear Network, formerly called Nuclear Notepad. The system is designed to support the exchange of information on matters related to nuclear plants. The I&IC works through GPU Nuclear, a Nuclear Network member, and has written network messages on heat stress studies, improper electrical equipment mounting instructions, control rod drive in situ surveillance test methods, and equipment failures caused by moisture ingress.

INPO's Good Practice Program is designed to pass along procedures developed at certain nuclear plants that might be worth adopting at other plants. I&IC has sent many summary reports and recommendations to INPO for consideration in the Good Practice Program. Topics include instrument installation and location practices, equipment surveillance requirements, preventive maintenance, equipment selection, purchasing procedures, and safety-related equipment environmental qualification practices. A report on ion exchange waste handling and disposal was sent which has application in the areas of radioactive waste procedures; programs to classify, sample, and test radioactive ion exchange waste; radioactive waste documentation and shipping records; and personnel training and staffing.

Technical Information and Examination Program *Update*. The DOE TMI-2 program publishes the TI&EP *Update* newsletter semiannually to describe research program accomplishments and recovery actions of interest to the nuclear industry. The *Update* is distributed world-wide and has a circulation of approximately 2000. Recent topics include body cooling suits for heat stress reduction and increased worker efficiency, recommendations for improved equipment survivability, and brief summaries of GEND documents. The *Update* has stimulated responses and requests for information from parties not previously aware of activities at TMI.

Technical Evaluation Groups. The TEGs are composed of leading members of the nuclear industry. They are experts in their fields, and bring their expertise to bear on TMI work. These groups evaluate how DOE TMI program data are being used and make recommendations for additional data acquisition, based upon industry needs. The TEG members themselves help effect technology transfer as they take information about TMI back to their home organizations and technical society meetings.

I&IC organizes TEG meetings and, based upon TEG recommendations, aids in tailoring the research programs to be responsive to industry needs. Recent Core Activity/Source Term TEG meetings have assisted the TMI program in forming a plan for future RCS and Reactor Building sampling activities. A list of prioritized samples was produced that would provide the most useful fission product behavior and temperature benchmark data. This information will aid the nuclear community by providing data to be used in the industry's severe accident analysis and source term programs.

The Instrumentation and Electrical TEG is working with the DOE TMI-2 program on cable evaluation, Resistance Temperature Detector work, and instrumentation programs. This TEG has recommended that more papers and reports be published dealing with the installation and maintenance inadequacies discovered as a result of TMI work.

Standards Development and Regulatory Activities. I&IC has provided data and recommendations to standards committee working groups which have influenced the contents of regulatory guides. Several examples are given below demonstrating how DOE TMI results can assist the industry in re-evaluating standards and regulatory guides.

Core Exit Thermocouples. Regulatory Guide 1.97 Rev. 2 required thermocouples to be able to measure core exit temperatures under accident conditions. TMI data show that these thermocouples can not survive accident conditions. Examination of TMI and other data led to a revision of the Rev. 2 requirements: the thermocouple itself no longer has to survive longterm accident conditions.

Response Time Testing. Reg. Guide 1.97 Rev. 2 also states that response time characteristics are required variables for nuclear plant operation. The TMI program serves as a source of information for assessing if response times are adequate to give valid data. A pilot program for in situ response time tests is being developed to provide a new basis for showing acceptable response time characteristics.

Area Radiation Monitors. I&IC investigations prompted by TMI program results have uncovered design, standard, and regulatory guide discrepancies with area radiation monitors. Relaxed regulatory requirements qualify existing equipment for use in the Reactor Building but this existing equipment is not designed to withstand the Reactor Building environment. The equipment does not meet design criteria outlined in the Area Radiation Monitor Standard (ANSI/ANS 6.8.1-1981). If utilities were to comply with the stricter guidelines of the ANSI standard, the monitors would better withstand Reactor Building environments, but then most existing equipment would be rendered obsolete. I&IC is working to resolve this problem through improved installation and maintenance procedures, using the INPO Good Practice program outlined above and participation in standards groups such as the Extreme Environment Standards Group described below.

Extreme Environment Standards Group. I&IC's active role in standards activities prompted an invitation to participate in the formation of a new standards group that will address extreme environments, such as nuclear plant Reactor Buildings. An assessment of damage and failure mechanisms of equipment examined by the DOE TMI program concluded that the dominant degradation mode is wetting of electrical components. Other damage and failure mechanisms include radiation effects and overpressure sensitivity.

This kind of damage could be avoided through modifications in accord with the environment in which Reactor Building equipment must operate. TMI will provide a data base that will address equipment selection, installation procedures and considerations, design criteria, and preventive maintenance procedures.

Resistance Temperature Detectors Standard. I&IC will participate in the formation of a committee to produce a standard on performance characteristics and response time testing of resistance temperature detectors (RTD). This standard is a direct result of the DOE TMI-2 Program Instrumentation and Electrical Group's RTD program. DOE TMI findings will form a significant portion of the standards data base.

Regulatory Code Assessments. One of the most valuable uses of information recovered on the sequence of events during the TMI-2 accident is for benchmarking of computer codes. To evaluate the consequences of comparable events occurring during severe core damage, the Nuclear Regulatory Commission uses computer codes, many of which are still in developmental stages. Usually these codes are benchmarked with experimental data which are intended to duplicate the parameters which are important in large nuclear reactors. The use of TMI-2 data for such benchmarking for codes on hydrogen burn, core damage, and fission product release and deposition can be one of the most important transfers of technology from the TMI-2 programs. TMI data can verify the performance and applicability of the codes to real events.

Research Results Transferred to Industry and Government

The DOE program has been exploring several key areas of interest to the nuclear industry. Research has yielded important information on nuclear instrumentation and electrical component performance and survivability during an accident. The DOE programs are improving understanding of not only the TMI event, but accidents in general, including such things as hydrogen burns and the behavior of radioactive fission products during an accident. The work at TMI has directly benefitted industry waste management capabilities, and has taught important accident recovery lessons with application at other nuclear plants for routine plant activities.

Instrumentation and Electrical Program. The DOE-sponsored Instrumentation and Electrical Program at TMI-2 is being conducted to assess the survivability of instrumentation and electrical components within the Reactor Building and to analyze the effects of the accident. A variety of electrical equipment has been removed from the Reactor Building and examined in laboratories. In situ tests performed outside the Reactor Building indicate that some of the electrical equipment had failed or its performance had degraded. Moisture and water ingress seem to have caused the greatest problems.

The examination of this equipment has shown important things about the design and application of specific equipment selected for use at TMI. What have also been revealed are certain factors which go beyond TMI-2 and do not appear to be dependent on the make, model, or manufacturer. These factors include installation and maintenance procedures, failure modes leading to faulty instrument readouts, and selection of nonqualified instrumentation for Reactor Building use. Ongoing research in the Instrumentation and Electrical Program will continue to yield results of benefit to the nuclear industry.

Installation and Maintenance Practices. The examinations of TMI-2 instrumentation and electrical equipment have repeatedly revealed the vulnerability of this equipment to poor installation and maintenance practices. TMI work illustrates that all instrumentation regardless of qualification is vulnerable to the same problems.

All of the radiation detectors located in the Reactor Building failed subsequent to the accident. Two have been removed and examined in the laboratory. One of the detectors, HP-R-211, was installed in the plant upside down, with the electrical connector at the top. This was probably the result of ambiguous installation instructions in the manufacturer's instruction manual. The installation manual states, "mount the detector with its small diameter side facing down." However, the manual refers to a figure which shows the detector mounted with its small diameter side facing up—the proper installation orientation.

The examination of detector HP-R-211 also revealed that the cable connector was screwed on incorrectly; electrically sound but not mechanically correct. This connector mismatch provided a path for moisture to enter the cable connector and cause a short circuit in the cable connector pins. HP-R-211 is a prime example of how a combination of ambiguous installation instructions and poor installation practices can create the failure mechanism. This detector had the potential to fail before the accident as the result of any simple spill which could cause the same water ingress.

In situ tests have revealed apparent failures of some of the pressure transmitters located in the Reactor Building. Two of these have been removed and examined in the laboratory. One had failed as a result of water which ran down the conduit, leaked past the seal, and flooded the electronics. The corrosion damage was extensive and precluded the possibility of examining for degradation due to other effects such as radiation or temperature. The other pressure transmitter examined also had indications of water in the conduit but the seal was effective in preventing water ingress and subsequent failure. The pressure transmitters were designed by different manufacturers, and the failure mechanism was not unique to either. Design testing by the manufacturer does not ensure that electrical equipment will be installed correctly or that periodic maintenance will not compromise the integrity of correctly installed equipment.

Susceptibility to Failure Modes. The loose parts monitoring system detects loose metallic parts in the RCS of commercial nuclear power plants. The system is composed of piezoelectric transducers mounted at strategic locations on the reactor vessel and related reactor coolant components. Charge converters convert the electrical charge generated by the transducers to a proportional voltage signal. A remote readout station monitors the signals generated by the transducers. This remote station provides an audio output, alarm, and recording functions to alert control room personnel of possible problems.

For weeks following the accident, the loose parts monitoring system was carefully monitored to determine if accident-generated debris was causing damage to the internals or blocking coolant flow in the RCS. The subsequent removal and examination of the charge converters revealed that the components had failed within days of the accident.

The cause of failure was found in electronic components called metal oxide silicon transistors, which are known to be radiation sensitive. The transistors' output degraded as they accumulated radiation dosage. The failure was not readily apparent because the decreased output came across simply as low noise levels at the readout station. Consequently, although it was carefully monitored, the loose parts monitoring system was incapable of detecting damaging debris in the RCS and it was not apparent that the equipment had failed.

Area radiation detectors drive control room ratemeters scaled from 0.1 to 10^4 mR/h. The cables connecting the detectors to the ratemeters are so long that they create an impedance mismatch between the detector and the control room ratemeter. This impedance mismatch is of little consequence when the detector is exposed to low levels of radiation and consequently operates at a low frequency. However, as the radiation level increases and the circuit operates at a higher frequency, the impedance mismatch is characterized by what is known as a "foldover effect." This unsafe failure mode is evidenced by a decreasing radiation level indication on the control room ratemeter even though levels may actually be increasing. The foldover effect occurs at radiation levels as low as 500 R/h. The problem illustrates the need to address the failure modes of nonqualified instrumentation. While not all Reactor Building instrumentation needs to survive extreme environmental conditions, it is important that instruments whose failure status is not apparent do not provide conflicting information during accidents or off-normal situations that could lead to inappropriate operator action.

Nonqualified Instrumentation Unsuitable for Reactor Building Use. The following examples show that when nonsafety related or non-LOCA qualified equipment is selected for use in the Reactor Building the question of its ability to perform satisfactorily in a normal operating environment is not always adequately assessed. The high temperatures, humidity, radiation levels, and occasional test pressures form a harsh environment capable of affecting the operation of electronic instrumentation. The careful selection of equipment with appropriate design parameters in the Reactor Building will prevent the type of problems identified here.

The area radiation monitors in the TMI-2 Reactor Building are identical to others used throughout the facility, but the environment they encounter during normal plant operations is quite different. They are exposed to containment temperatures ranging as high as 52°C, relatively higher humidity and radiation levels, and the periodic Integrated Leak Rate Test (ILRT). The ILRT can expose sealed equipment to differential pressures as high as 69 psig, a pressure that is well above the recommended 30 psig environmental limitations specified by the radiation monitor manufacturer. Consequently, the detectors have to be removed from the Reactor Building prior to the ILRT and reinstalled after the tests. This removal and installation requires additional handling, which could have contributed to the poor installation that led to the failures of two radiation detectors.

The ILRT posed special handling problems for more electrical equipment then just the radiation monitors. Differential pressure transmitters, RTDs, pressure transmitters, and rotometers must all be protected from the effects of differential pressures. This protection includes taking pressure transmitters out of service, and removing covers from RTDs and rotometers. Obviously there is a significant potential to damage or violate the integrity of the affected instrumentation through mishandling. The problem could have been avoided if the design engineer had specified to the supplier that the equipment must be capable of withstanding an external pressure of 70 psig.

A laboratory examination of the failed radiation monitors also revealed that certain electronic components had degraded. These components, metal oxide silicon transistors, are known to be radiation sensitive and degrade as a function of accumulated radiation exposure. The examination confirmed that the transistors' current gain had decreased and saturation voltage had increased. The damaged electronics tend to compound the foldover effect, a situation described earlier, where the detector readout can actually decrease as radiation levels increase. These detectors are unsuitable for use in the harsh environments that can be encountered in Reactor Buildings.

The laboratory examination of the failed loose parts monitoring (LPM) system charge converters and subsequent tests performed on the metal oxide silicon transistors they contain, revealed that their output degrades after 10^5 rads accumulated dose. The TMI-2 charge converters accumulated dose was the result of the accident released radioactive gases. These levels are not so high that they couldn't be encountered in a normally operating reactor environment. In fact, the Sequoyah Unit 1 nuclear facility suffered similar failures of their LPM charge converters after less than 160 effective full power days of operation. This was because their charge converters were installed underneath the reactor vessel where radiation levels exceeded 10 R/h. The loose parts monitoring systems used at TMI-2 and Sequoyah I were not designed for use in radiation areas nor does the manufacturer suggest that they could be used in radiation environments. Nevertheless, they are commonly used in nuclear facilities and their deficiencies deserve careful examination.

Cable and Connection In Situ Testing Program. The cable and connections in situ test program is determining which cables in the building suffered the greatest degradation during the accident and its aftermath. The cables have been exposed to elevated temperatures, radiation levels, and humidity, which all can have an effect on insulation properties. In situ tests are being used at TMI in an attempt to determine which cables suffered the most damage. The actual extent of degradation can be determined by laboratory examination.

In situ tests should prove to be a successful diagnostic technique at TMI, and they could be used in other plants to assess the "health" of cables in less severe environments. This in turn could lead to a technical basis for the safe continued use of qualified cables beyond their predicted 40-year life. When the development of techniques and the confirming laboratory work at TMI are complete, reports will be published on the accuracy of in situ testing as a diagnostic tool.

Resistance Temperature Detectors Research Program. The Instrumentation and Electrical Program has been studying the effect of the TMI-2 accident on the response time and calibration of the RTDs in the RCS. Since February 1983, 13 RTDs have been tested at TMI-2 by ORNL and Analysis and Measurements Service.

The preliminary report from these analyses indicates that the accident did not effect detector calibration, but that the insulation resistance has degraded. The report also shows that the measured response time of TMI-2 RTDs under low temperature and low flow conditions ranged from 23.0 to 35.9 s. From this information, the response time of the RTDs under operating conditions (288°C and 12 mR/s) is estimated to be between 12.4 and 13.1 s. This estimated response time is 2.3 to 2.4 times longer than the maximum 5.4 s required by the *TMI-2 Final Safety Analysis Report*. The exact cause of the longer response time is not known, but explanations range from speculation that the response time never was 5.4 s to a theory that the degradation was caused by the degradation of Never-Seez, used in the thermowell to initially improve response time. When research is complete and conclusive findings are available, I&IC will release the information to the industry.

I&IC contacted Weed Instrument Company, Inc., a manufacturer of qualified RTDs. Weed explained that plants designed by Combustion Engineering have tapered thermowells for their RTDs, whereas B&W plants like TMI-2 have straight thermowells. Weed has designed an RTD to fit into the Rosemount thermowell used in Combustion Engineering plants which has a response time of approximately 3 s at 6 to 13 m/s. I&IC does not know if this response time degrades with time in an operating plant and if it does, how far it degrades. As for replacements for the RTDs with straight thermowells (B&W), Weed has not been able to improve the response time and recommends replacing the entire unit using a tapered thermowell.

Combustion Engineering has been aware of the RTD response time problem for over a year and has been working on solutions. They tested all commercially available thermal coupling compounds in search of one which would improve response time and not degrade like Never-Seez. This testing did not locate any useable compounds. Combustion Engineering is using gold and silver plating on the tips of their RTDs to provide maximum contact with the thermowell. The gold plating must be done for each individual thermowell and the problem of spares for this type has not been resolved. The silver plated RTDs are more versatile but do not provide as much contact as the gold.

At the Arkansas Nuclear One Unit 2 plant, Combustion Engineering convinced the Nuclear Regulatory Commission to relax the response time requirements for the RTDs. They changed the technical specification requirement by 6 to 8 s by decreasing the reactor trip setpoints. This is the only occurrence known of relaxation of the response time requirements. Combustion Engineering feels that they have solved their RTD problem by the combination of gold plating and relaxed requirements.

Severe Accident Analysis. One of the most problematic safety issues associated with nuclear energy generation is the production and control of fission products, and preventing their release to the environment. Presently, the phenomena controlling the release and transport behavior of fission products during severe core damage accidents are not adequately understood. An understanding of the core degradation process is very important if the release and reactor system transport of fission products are to be adequately predicted. TMI-2 examinations will be used to characterize and confirm our understanding of core damage mechanisms and fission product release for a full size reactor system transient.

Many experiments being conducted around the world are assessing the effects of small-scale separate-effects accidents, but the TMI-2 accident is the only source of data on actual fuel damage processes and fission product release, transport, and deposition in a large pressurized water reactor. The severe accident analysis work will investigate the following issues:

- Core coolability
- Core recriticality
- Core degradation phenomena
- Fission product release, transport, and deposition mechanisms
- Fuel rod failure mechanisms
- Metal/water reactions.

Core cooling processes during severe accidents prior to loss of core integrity can be well predicted, but knowledge of core geometry changes and shattering, liquifaction, melting, and other fuel damage phenomena is insufficient to predict core behavior after severe core disruption. At TMI-2, core degradation processes and fission product behavior phenomena were certainly typical of those which would occur in a wide range of severe core damage accidents. A significant feature of the TMI-2 core damage and fission product release data generated thus far is that the degraded core apparently contains fuel which has undergone nearly the entire spectrum of damage from slight to severe, including fuel liquifaction but apparently not melting.

The TMI program has completed an analysis of the hydrogen burn event that occurred approximately 9-1/2 h after the accident began. Assessment of the amount of hydrogen burn, the amount of the fuel cladding that oxidized to produce the hydrogen, and the resultant damage in the Reactor Building are completed. The Nuclear Regulatory Commission (NRC) is currently using these results to validate hydrogen burn models they have under development.

TMI-2 Radioactive Waste Management. The TMI-2 accident resulted in large amounts of highly radioactive water which was contaminated with fission products and, in some cases, fuel debris. The contaminated water has been processed using the EPICOR II mixed resin and zeolite ion exchange system and the Submerged Demineralizer zeolite ion exchange system. These systems removed contaminants from the water and loaded them onto resin or zeolite beds in system containers, creating a solid waste form for disposition. The abundant use of ion exchange resins in the TMI-2 Auxiliary and Fuel Handling Buildings led to a number of technological developments associated with the storage, shipment, and disposal of this type of waste.

Safely Shipping Ion Exchange Wastes. Prior to the shipment of an EPICOR II prefilter liner containing highly loaded ion exchange waste, significant concentrations of hydrogen gas were detected. This resulted in the need to develop a prototype gas sampler with a remotely operated support facility to safely remove the gases from the ion exchange waste container for shipping the waste. Routinely generated high activity wastes with entrained water, such as resins, can similarly require consideration of combustible gases. This information has been transmitted to nuclear industry personnel through EPRI information transfer mechanisms and through the TI&EP *Update*.

Safely Disposing of Ion Exchange Waste. A high integrity container (HIC) was developed by the DOE to safely dispose of radioactive ion exchange waste. The primary design goal for the HIC was to maintain structural integrity for safe and reliable commercial shallow-land disposal for a minimum of 300 years. Three hundred years is equal to 10 half-lives of the predominant hazardous radioisotope to be buried in the HIC. Other design considerations included:

- A passive vent system enabling gases generated during internal radioactive decomposition and/or chemical activity to be removed from the container during its lifetime.

- Eliminating all potential radiation hazards to both operating personnel and the public during transport and disposal of the container.

- Minimizing costs for fabricating, loading, transporting, and disposing of containers in a commercial facility.

The structural and mechanical integrities of the HIC were tested. Lift, drop, and penetration pin-drop tests were performed which demonstrated the design criteria were satisfied under accident conditions. A drop test demonstrated that the container met the requirements for a Type "A" package as specified in 49 CFR Part 173 and 10 CFR Part 71. As a result of DOE's demonstration of this successful technique, commercial enterprises are pursuing HICs as an alternative to solidification to meet the 10 CFR 61 regulation.

Ion Exchange Waste Volume Reduction. Vitrification is a single-step process that incinerates the resins and incorporates the radionuclides and residual ash into glass. It was demonstrated as a result of TMI research. The glass product was only 9% of the original resin volume. Converting the resins into glass significantly reduces the volume and could prove to be a cost effective method for safe disposal of this type of waste. An evaluation of this processing technique also resulted in its selection for the high level waste sludge at the West Valley waste disposal site. An economic analysis will be suggested for disposition and repository sites.

Impact on Industry Waste Management Practices. The majority of NRC radioactive waste requirements are contained in 10 CFR Part 61, "Licensing Requirements For Land Disposal of Radioactive Waste," and 10 CFR Part 20, "Standards For

Protection Against Radiation." The NRC has also prepared Regulatory Guides on both waste classification and waste form.

The technical rationale to support the NRC's draft Branch Technical Position Paper on "Maximum Permissible Radionuclide Loading For Organic Ion Exchange Resins," included data from work done in connection with the TMI-2 accident cleanup activities. NRC's Branch Technical Position Paper "Waste Classification and Waste Form" offers guidance on acceptable procedures to determine radionuclide concentrations for waste classification, methods for processing waste in acceptable waste forms, design criteria for HICs, and limits on resin loading to reduce the effects of radiation on organic resins.

In the future, consideration will be given to evaluate present waste management programs for aspects of TMI-2 work which could assist industry in compliance with radioactive waste regulatory requirements. Areas which could be directly affected in nuclear power plant waste management programs include: radioactive waste procedures, waste classification and sampling capabilities, waste documentation and reporting requirements, plant equipment and facilities upgrades, personnel training and staffing, plant procedures, and ALARA programs.

Accident Recovery Developments. I&IC has assembled and distributed accident recovery-related data packages and reports in several areas. The technology transfer mechanisms discussed in an earlier section have been used to distribute accident recovery information on several topics.

A report that outlines the activities required to certify the polar crane has been published by EPRI. I&IC has sent an abbreviated summary of the report to *Power* magazine, an industry periodical, for publication. The document describes the accident-caused damage to the crane and steps taken to refurbish, test, and qualify it for reactor vessel head lift. A section on recommendations suggests some relatively inexpensive measures that industry might take to protect other polar cranes from damage. A description of polar crane activities also appears in the December 1983 issue of the TI&EP *Update*.

The DOE TMI program in conjunction with the Diamond Power Specialty Co. has developed in situ test methods that can be used to determine the health of the control rod drive system. These procedures may aid nuclear plants in diagnosing control rod movement problems. I&IC has provided industry with a description of this new technique through the NOMIS computer conferencing service.

The severe environmental conditions in the TMI-2 Reactor Building prompted GPU Nuclear to perform a worker heat stress study. Heat stress occurs when body heat gets trapped in multilayered protective clothing worn by workers to shield them from external contamination. GPU Nuclear has developed two systems to address the problem. The first is the ice vest, a passive system used when mobility over large areas is required. The second system uses an air-fed vortex suit, used when workers will remain relatively stationary over long periods. I&IC notified the industry of the development through the NOMIS computer conferencing network and distributed information packages to many interested utilities. A description of the systems appeared in the TI&EP *Update*, December 1983.

DOE TMI programs working with Pacific Northwest Laboratories has developed a multielement thermoluminescent dosimeter (TLD) capable of correcting for energy distribution in beta radiation fields. These devices have been used in the TMI recovery program to characterize beta and gamma fields in the Reactor Building while minimizing worker exposure to ionizing radiation. This new TLD system may aid the industry in refueling practices, personnel protection, and spent fuel pool characterization concerning identification of leaks. I&IC is preparing a technical brief on area TLDs to be distributed through EPRI.

CORE ACTIVITIES

The Core Activities program supports analysis, characterization, and storage of fuel, core debris, and nonfuel samples from the TMI-2 accident and cleanup. This analysis work contributes to understanding the extent of the TMI-2 accident and its effects on fuel and core components. The DOE selected the INEL to provide facilities and manage the storage and examination of fuel and core debris samples from TMI-2. Program work will be accomplished in appropriate facilities at the INEL. The program will also develop requirements for shipping, packaging, and handling TMI materials.

During 1983, several important documents and studies were completed as part of the planning for removal and shipment of the core from TMI-2 to the INEL. Documents, studies, and facility upgrades are summarized in the following paragraphs.

Documentation and Critical Planning

The TMI-2 Core Examination Plan was released for general distribution. This plan reviews critical nuclear safety issues facing the U.S. light water reactor industry and addresses how TMI-2 core examination can provide the information required to resolve those issues. The document serves as a baseline from which examination plans, schedules, and funding of the Core Activities program can be developed. It is intended to be a living document, accommodating regular reviews and revisions to reflect improved knowledge of the detailed condition of the TMI-2 core. Planned examinations are divided into three phases: predefueling examinations, examinations during defueling, and offsite examinations.

Predefueling examinations will provide early documentation of the postaccident condition of the core. The recommended examinations are compatible with existing reactor recovery tasks and schedules. Core examinations during reactor defueling are confined largely to selection of representative samples of core debris for offsite examination and analysis. Guidelines for sample selection and recommendations for sample preservation during handling and shipping are also presented. Finally, the document recommends that specific hot cell examinations be performed offsite on selected categories of TMI-2 core debris.

The draft, *Plan for Shipment, Storage, and Examination of TMI-2 Fuel,* was prepared for DOE review. This plan prescribes preparations for shipment of TMI-2 fuel canisters, shipment of the canisters from TMI to the INEL, and canister receipt, storage, and examination. The TI&EP is responsible for developing and implementing TMI-2 core shipment plans. This portion of the program coordinates fuel shipments from TMI, ensuring that fuel handling systems are designed, constructed, or modified to establish and maintain system compatibility. The Technical Support Branch at the INEL is responsible for developing and implementing the fuel handling activities at the INEL. This program encompasses the handling, storage, and examination activities at the INEL and also defines core-related examination activities planned at other laboratories.

Data Acquisition Activities

During 1983, four data acquisition activities collected information from a variety of sources in the effort to characterize the accident effects throughout the plant and the current condition of the core. Analysis of purification filters ended this year. In addition, preliminary results of studies on reactor leadscrew components helped to determine estimates of accident temperatures. Also, the first samples of the debris inside the core are being analyzed. Finally, a new data acquisition tool gathered information used to build a model of the upper portion of the damaged core.

TMI-2 Makeup and Purification System Filters. The Makeup and Purification System at TMI was in use for about 18 hours on the day of the accident. Reactor coolant water flowed through the system until fluctuating flow rates and overpressure indications from the filters and demineralizers led operators to bypass the system. Filters were eventually removed from the plant and shipped to offsite laboratories for evaluation of filter condition, the elemental chemistry of debris particles on the filters, debris particle size and shape, and debris radioisotope content.

The examination and analysis work on the TMI-2 Makeup and Purification System filters was completed in 1983 and EG&G Idaho plans to issue the final report by May 1984. The examination task was conducted at three DOE Laboratories: INEL, Los Alamos National Laboratory, and Argonne National Laboratory East. Examination and analysis concluded that debris on the filters consisted of fragments from fuel, cladding, stainless steel components, and control materials. Stainless steel and Inconel alloying elements were found in 67% of the particles, and control materials were found in 60% of the particles. Particle size ranged from 1 to 40 μm; mean particle diameter was 6 μm and the spheral mean was 0.6 μm. There was a higher accumulation of cesium, antimony, ruthenium, iodine, and strontium than could be accounted for by the UO_2 content. Reasons for these and other findings are still being investigated. The final report on the Makeup and Purification System filter analysis will explore all results in depth.

Leadscrews. Three control rod drive leadscrews have been removed from the TMI-2 reactor vessel, and two of these leadscrews are being examined at the INEL. These studies are examining leadscrews H-8 from the reactor vessel centerline and B-8 from the outermost control position toward the A loop. Examination of the leadscrews should yield data on the plateout of fission products and on peak temperatures experienced during the accident. This information will assist in assessing fission product transport codes, estimating plenum temperatures, and calculating core exit steam temperatures.

Preliminary analyses of the lower 3.2-m segment of the H-8 leadscrew indicate that the portion of the leadscrew closest to the core experienced temperatures in the range of 1090 to 1470 K and that the portion in the upper plenum reached temperatures between 750 and 825 K. Figure 25 shows the location of the leadscrew in the vessel, and the approximate region on the leadscrew for

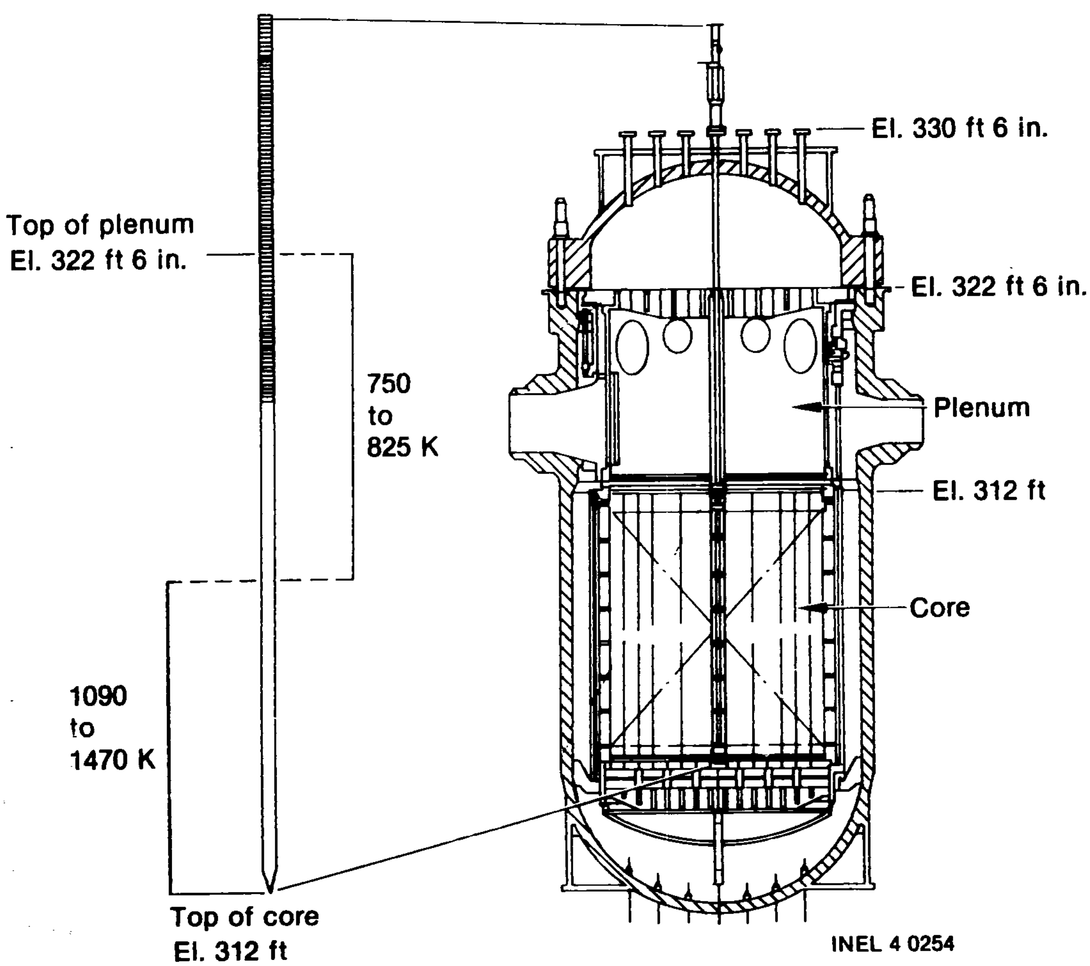

Figure 25. Approximate temperature ranges experienced by lower portion of the H-8 leadscrew during the accident.

each estimated temperature range. UO_2 and zirconium were deposited on the portion of the leadscrew closest to the core. Significant radioactivity was associated with particle sizes less than 0.45 μm. (Aerosol or airborne transport of radionuclides may account for this apparent association between significant radioactivity and small particles.) Most of the radioactivity was contained in a surface layer that was soluble only in a strong acid ($HNO_3 + HF$).

Transmission electron microscopy annealing studies, and tellurium, iodine, and fissile material analyses are underway on samples from the top of the H-8 leadscrew. Analyses of samples from the B-8 leadscrew are continuing. A final report summarizing analytical results of H-8 and B-8 leadscrew studies is expected before August 1984.

Core Debris Grab Samples. In 1983, EG&G Idaho designed, built, tested, and deployed equipment which acquired the first samples of core debris from the rubble bed inside the TMI-2 reactor. The apparatus used to obtain these samples, shown schematically in Figure 26, consisted of a 12-m long multipiece deployment boom with a particulate debris sampler attached to the lower end. Two different samplers were actuated by the operating team working from the reactor service structure. Each sampler was lowered into the debris bed through an opening left by the removal of the leadscrew. Working by feel and the amplified sound of the sampler contacting the core rubble, the operating team acquired six distinct samples of core debris. These samples were collected at two locations in the TMI-2 core: midcore and midradius. Each set of three grab samples contained core material collected from the surface of the rubble bed, and 8 cm and 56 cm down into the rubble bed.

Once the samples were obtained, five were shipped to the INEL and one went to B&W. Both hot cell facilities are conducting extensive analytical examinations of the samples. Each of the five samples at the INEL were weighed and photographed, and underwent gamma-ray spectroscopy.

All the core debris samples together weigh approximately 430 g and range in size from less than 1 to about 15 mm. The debris appears to consist primarily of fuel pieces, although once-molten pieces and metal shards—which could possibly be fuel rod cladding—are also present. Figure 27 shows photographs of each sample and where in the rubble bed each sample originated.

A preliminary analysis of gamma-ray spectroscopy measurements of the five samples indicates the presence of the following isotopes: ^{60}Co, ^{106}Ru, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{154}Eu, and ^{155}Eu. Based on this preliminary analysis, engineers have concluded that spectra from the five samples are similar, with some being almost identical. A more detailed analysis will follow as examination and analysis of the TMI-2 core debris samples continue in 1984.

Core Topography System. A major accomplishment of the TMI-2 Core Activities task in 1983 was the successful installation and operation of the Core Topography System (CTS) at TMI-2. This system was designed to acoustically map the size and shape of the cavity in the upper region of the damaged TMI-2 core. The data gathered with the CTS were used to reconstruct the plexiglass model of the cavity seen in Figure 28.

An instrument column with twelve acoustical sensors, pointing at six angles from straight down to nearly straight up, was inserted into the center of the core cavity, down to the rubble bed marking the floor of the cavity. The column sensors, similar to an underwater sonar depth finder, sent out ultrasonic signals which reflected off any barriers and then returned to the instrument column. The time required for the signal to return indicated the distance between the reflecting surface and the instrument. The instrument column rotated 360 degrees while recording acoustical measurements. It then was raised in 2.5-cm increments, each time to record the 360-degree radial measurements, until the instrument column reached the top of the cavity and entered the plenum assembly. All data were stored in a dedicated computer. An off-line software program was then used to reduce the data and reconstruct the size, shape, and surface contours of the core cavity.

Three processes were used to convert the data into topographic maps. First, the Cyber computer system in use at the INEL processed the raw data into three-dimensional coordinates and printed them as "data slices" through the core cavity region. Second, these data slices were manually interpreted to develop horizontal contour plots in 5-cm intervals. Interpretation of the data was complicated by the presence of material hanging

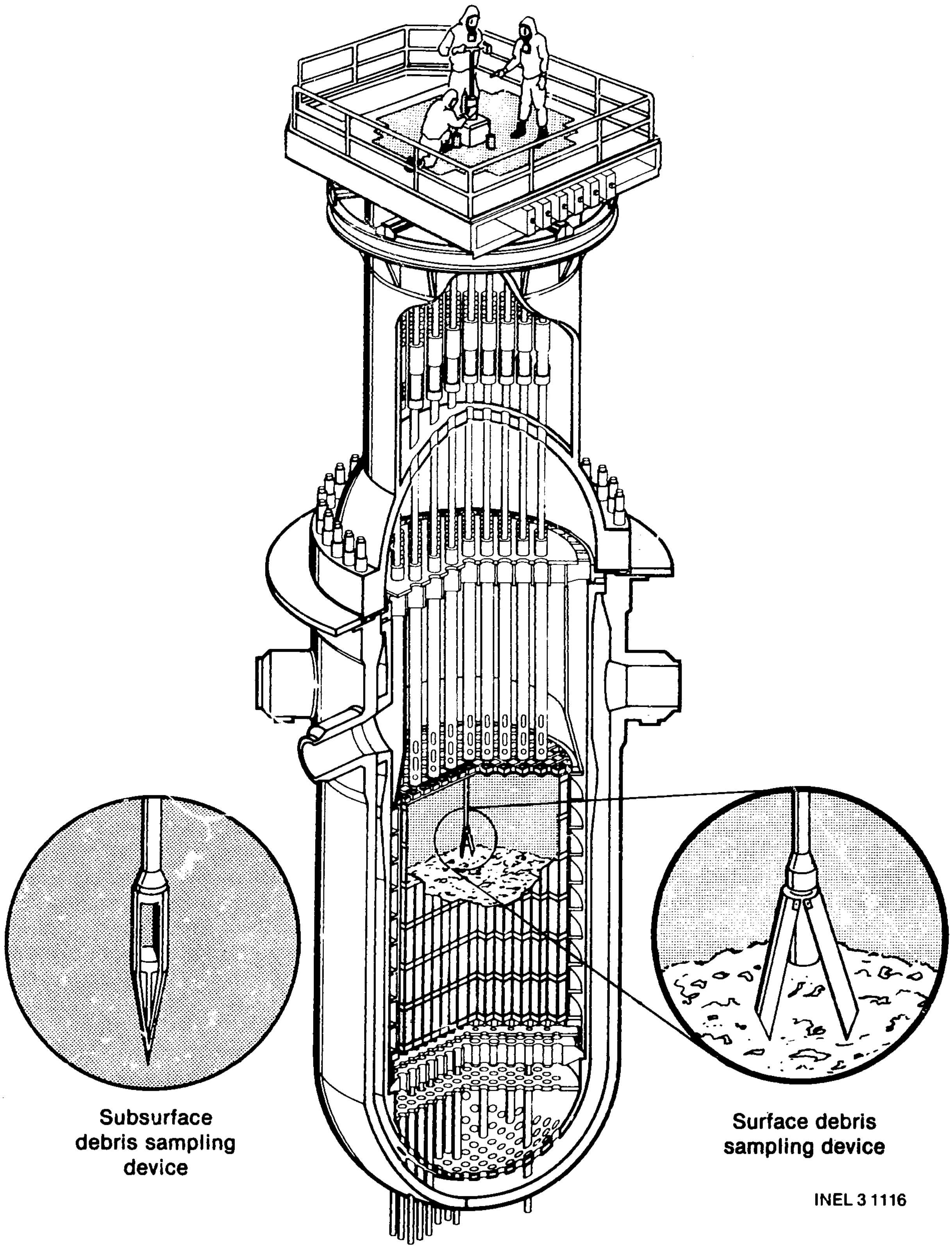

Figure 26. Core debris sampling at TMI-2 used two different sampling devices.

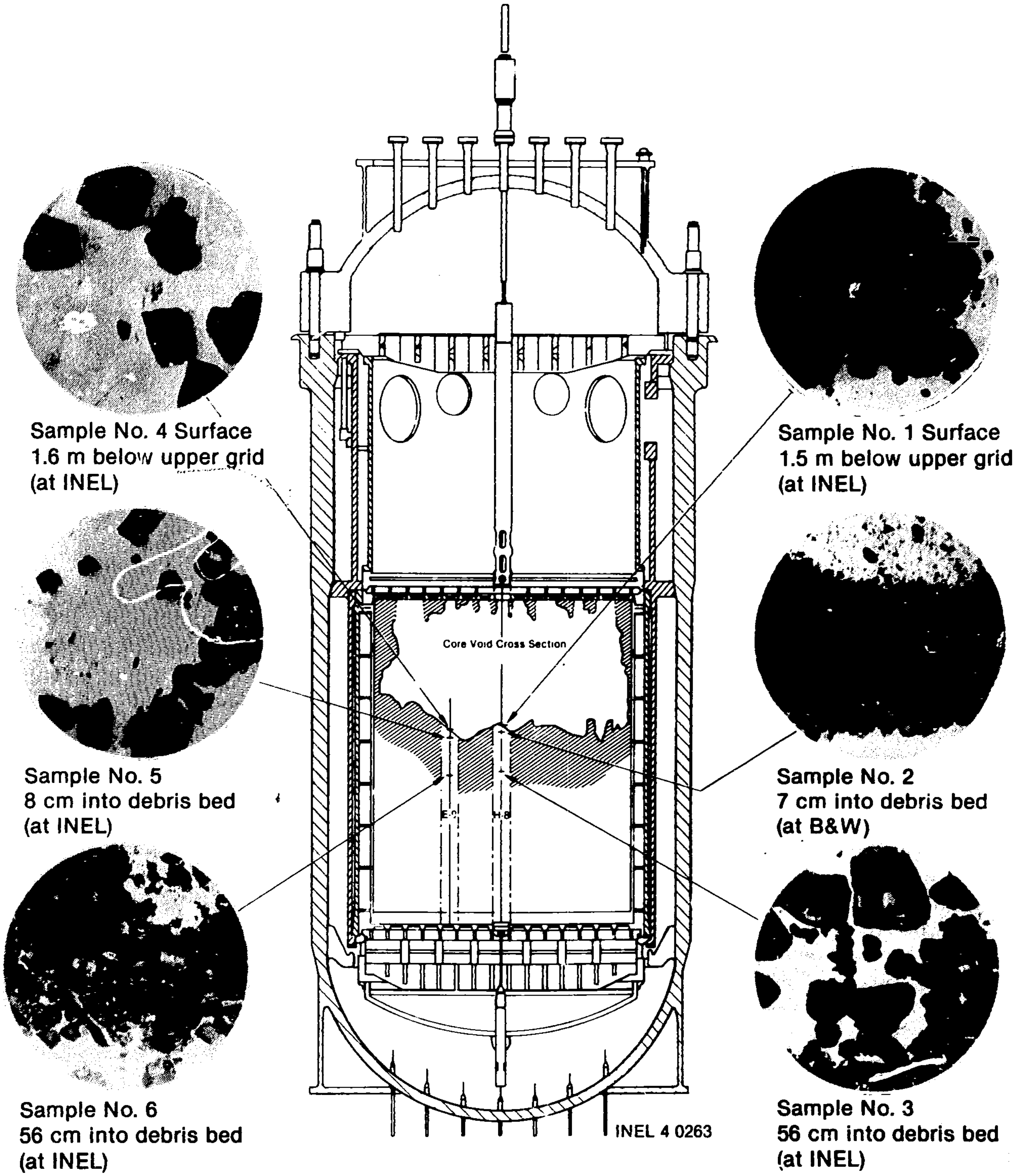

Figure 27. Appearance and location of TMI-2 core debris samples.

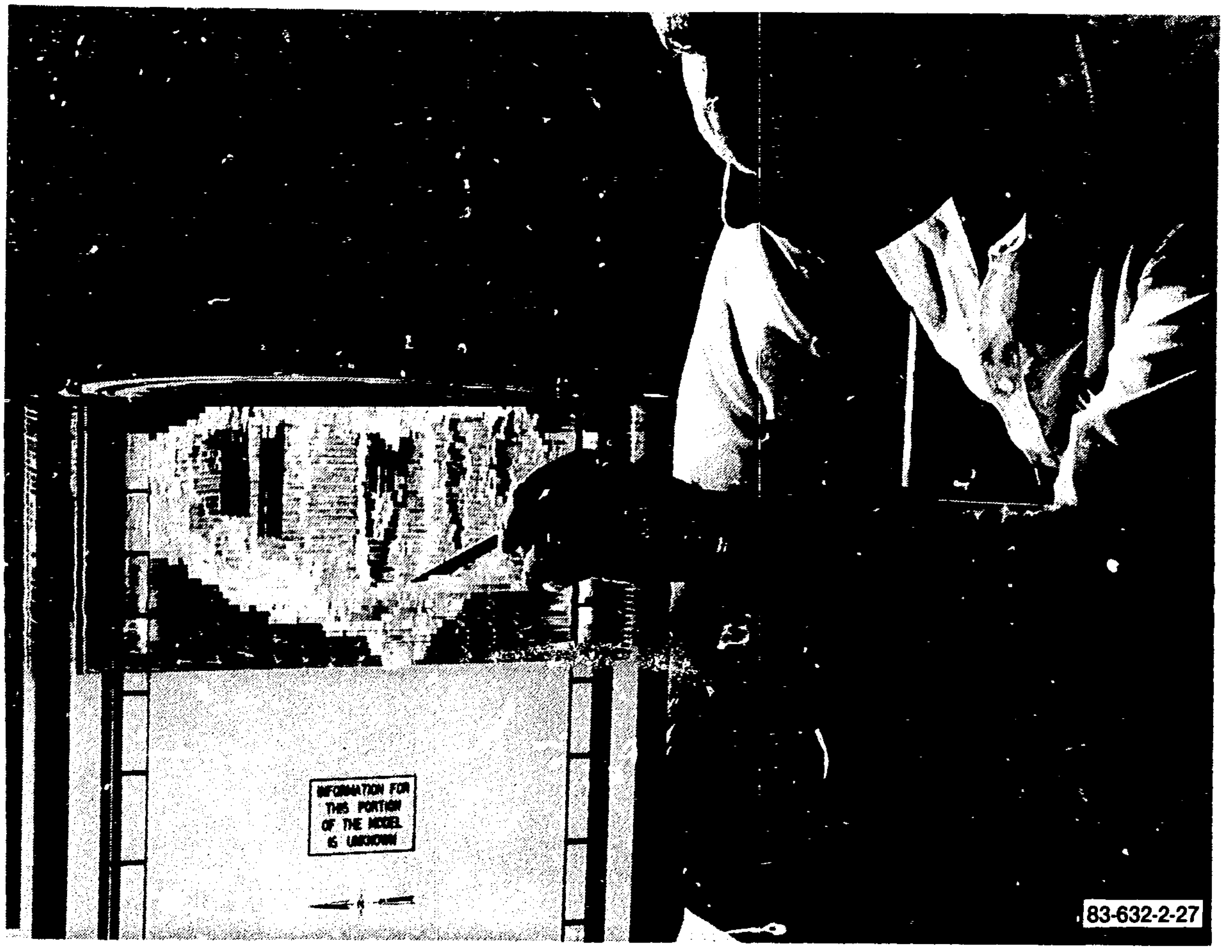

Figure 28. Information gathered with Core Topography System was used to construct this plexiglass model of the top portion of the TMI-2 core.

from the plenum and by accumulations of rods at the margins of the cavity. Those structural interferences caused shadows and multiple imaging during the interpretive process. Finally, the resulting information was input into a computer-aided design system. This system added the orientation and reference information to the data and assembled the individual contours into contour maps. The plexiglass model of the core cavity, shown earlier in Figure 28, was fabricated based upon final interpretation of all data.

Facility Upgrades

Equipment in an INEL hot shop is being modified to receive TMI-2 core material and debris. Included in the task are reconditioning the 100-ton crane system, and upgrading the handling equipment for the water pit. The remote controlled manipulators are being rebuilt and tested, and the radiation monitoring systems are being upgraded. Facility preparation will continue into 1984.

The EPICOR and Waste Research and Disposition Program is responsible for the storage and disposal of special and abnormal wastes generated during cleanup of the TMI-2 nuclear power station. The program encompasses tasks required for the receipt, storage, and disposition of those wastes. The scope of the program includes receipt and interim storage of the EPICOR II liners and development and use of the HIC for disposal of those liners. The program is also responsible for resin and liner research and interim storage operations for abnormal wastes. All these key activities are summarized and described in the following paragraphs.

High Integrity Container Development for Liner Disposal

High Integrity Containers were developed for safely containing the high activity of ion exchange media liners for 300 years in commercial disposal facilities. The HIC is designed to maintain its integrity during varying environmental conditions of temperature, moisture, soil composition, and disposal depth. HICs are an alternative to solidifying or otherwise immobilizing filtration resins obtained from operating commercial power reactors.

Development of the HIC was completed in 1983, while independent evaluation tests continued into 1984. Handling and vent tests successfully demonstrated remote handling, fit checking, and grouting of the lid to the HIC. The vent rate into soil was approximately 15 mL/h at standard temperature and pressure. The soil was compacted to 28 m of lithostatic pressure and approximately 70 kPa driving pressure. The leach rate was adequately above the 10-mL/h generation rate observed earlier in EPICOR II liner PF-3. Vibration tests on the HIC filter assembly were completed successfully and indicate that no adverse effects should occur while transporting HICs. Leak testing of the HIC lid seals was also performed.

EG&G Idaho is coordinating efforts for a Certificate of Compliance with the State of Washington to dispose one EPICOR liner enclosed in an HIC in commercial burial grounds near Richland, WA. That agreement constitutes an approval to use HICs for disposal of EPICOR II liners. U.S. Ecology, Inc., the contractor operating the Richland disposal facility, was asked to prepare the Certificate of Compliance for the Richland commercial low-level waste facility. U.S. Ecology, Inc. reviewed the EG&G Idaho report, *High Integrity Container Design Analysis*, evaluated the design, and recommended attaching a lifting lanyard to the HIC lifting eyes at loading time. U.S. Ecology also recommended a drop test in accord with the State of Washington's concern over whether the HIC could withstand the impact of a drop onto soil similar to that at Richland.

In accordance with these recommendations, an HIC loaded with sand was drop-tested from a height of 7.6 m onto a soil surface at the INEL similar to that found at the commercial disposal facility near Richland. Following the drop, the HIC was photographed and inspected carefully. Figure 29 is a photograph taken after the test. The container successfully satisfied Washington State requirements by passing the test without loss of contents or even visible damage.

The State of Washington reviewed U.S. Ecology's recommendations and requested that NRC review and comment on the HIC design analysis report. EG&G Idaho discussed the design and transportation of the HIC and EPICOR liner with the NRC. NRC commented on the HIC's design and usability. EG&G Idaho, U.S. Ecology, and Nuclear Packaging, Inc., the designer and builder of the HIC, collectively prepared written responses to those comments. Once the NRC comments have been resolved, the Certificate of Compliance should be finalized, probably by mid-1984.

Two new HICs were delivered to the INEL on schedule during 1983. In addition, the new CNSI-14-190 transportation cask and trailer were received according to plan. The remaining 33 EPICOR II liners were shipped from TMI and arrived at the INEL two months ahead of schedule. As they arrived, the liners were placed in the Test Area North (TAN) hotshop until that storage area was filled; the remaining liners were placed in temporary storage casks outside the hotshop.

In September, liner PF-18 was retrieved from temporary storage, dewatered using equipment developed in the liner integrity examination study, and sealed in one of the HICs. The loaded HIC was

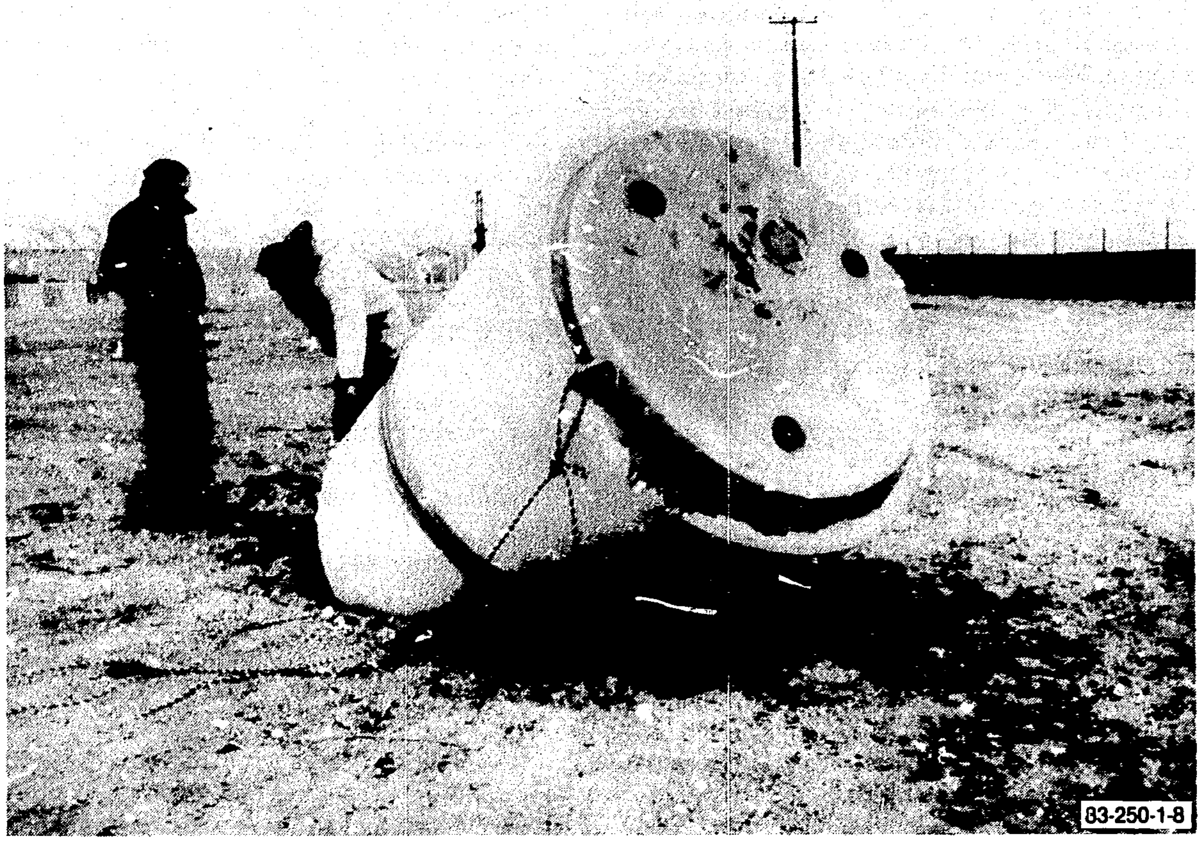

Figure 29. High Integrity Container shortly after drop test from 7.6 m, showing deep soil penetration and no damage to container.

placed in the transportation cask and moved outside the TAN-607 hotshop facility. It will remain there until the State of Washington receives final NRC comments and authorizes disposal at the commercial facility near Richland.

Resin and Liner Research

Several of the 50 EPICOR II liners received for interim storage were selected for resin and liner research. Liner sections from PF-16 and -3 were dedicated to liner integrity studies; full length core samples from PF-8 and -20 were used for resin degradation studies; and resin samples from PF-7 and -24 were used to provide solidified waste forms for resin solidification and field tests.

Liner Integrity Studies. The epoxy coated steel liners from PF-16 and -3 were inspected, using photographic and metallurgical methods, to measure the degree of deterioration or corrosion. First, the contents of PF-16 and -3 were moved to new unused liners using a vacuum transfer system. Outlet headers on the original liners were removed to reduce contamination and improve viewing of the liner bottoms. Both liners had radiation readings of 1 to 2 rads at contact. Next, the original liners were decontaminated and sections were removed for metallurgical examination. Sections from four locations in liner PF-16 and three locations in liner PF-3 were examined in detail. The interior coating of PF-16, shown in Figure 30, appeared blistered with moderate corrosion of the base metal. Liner PF-3, shown in Figure 31, appeared to be in better condition. However, metallurgical results later identified minimal corrosion on both liners.

The examinations showed that anticipated lifetime of EPICOR II liners is much longer than their planned storage time at the INEL. Metallurgical results suggest that liners can be disposed safely using an HIC for more than 300 years, without any threat to the environment.

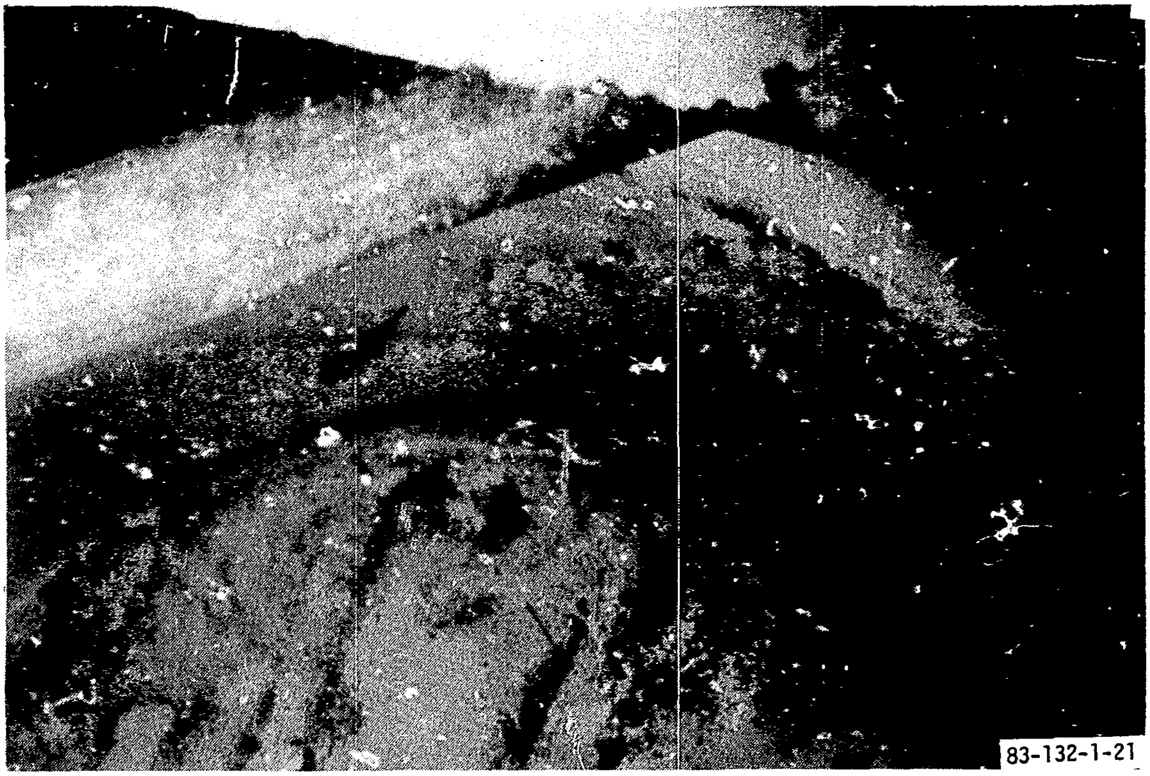

Figure 30. Inside view of PF-16 showing the manway support structure and outlet header (top), flaking of protective coating (right center), resins and flakes of protective coating (liner bottom), and arrows showing where metallurgical samples were removed.

During preparation of liner PF-16 for liner integrity examination, a sludge sample was removed from near the liner bottom and examined to explain possible difficulties encountered during resin transfer. The sample was divided and the aliquots were analyzed. Sodium and sulfur were the major inorganics identified. Energy dispersive X-ray spectrometry determined that small amounts of iron, chlorine, potassium, and silicon were also present. Studies showed no unusual characteristics of the ion exchange material which would explain the transfer problems. The examination did reveal that the transfer equipment was not designed properly. The equipment was redesigned and no further resin transfer problems were encountered.

Resin Degradation Studies. Resin core samples from PF-8 and -20 were gamma-scanned at the Test Reactor Area hotshops at the INEL. The full length gross scans of each sample provide axial profiles of radionuclide distributions. Isotopic gamma measurements were made at locations of highest radionuclide concentration. Preliminary results indicate that highest radionuclide concentrations were at or near the upper surface of the ion exchange material. This agrees with earlier in situ measurements made at TAN hotshops. Final results will be available in early 1984.

Resin Solidification. Resin solidification studies explore using commercial formulations of either Portland cement or Dow polymer as media for solidifying ion exchange resins. A total of 267 samples, called coupons, were made from resin core samples taken from two EPICOR liners (PF-7 and -24). Those coupons were gamma scanned at Auxiliary Reactor Area (ARA) hotshops to determine resin consistency within each batch. Gamma readings for cement batches were consistent, but the batches of Dow immobilized resins varied in resin content from coupon to coupon. This inconsistency was thought to occur from resin settling in the injection device used to fill the molds. Only coupons of proper consistency were used in the testing program.

Coupons selected for compression testing were placed in the Engstron Tensile-Compression Test machine located at the ARA hotshop loading area. Flat surfaced end caps were attached to the coupons

Figure 31. Inside view of PF-3 showing the exposed liner surface of the conductivity level probe (upper right), and resins remaining at the bottom of the liner (lower left). Note corrosion products adhering to exposed surface.

before they were compression tested to failure. Both the concrete and Dow polymer coupons broke at a stress of approximately 20,700 kPa with the exception of one specimen that failed at 10,300 kPa because of poor alignment in the testing fixture. Future testing in resin solidification will be directed in accordance with the NRC Technical Position on Waste Form and will include the areas of leachability, biodegradation, and thermal and radiation degradation.

Field testing experiments involve subjecting solidified resin coupons to long term leach tests in lysimeters. The lysimeters are 2.3-m^3 stainless steel tanks, containing immobilized resin waste forms surrounded by local soil. They are buried to experience full environmental effects. Two sites were chosen for field testing: ORNL, and Argonne National Laboratory (ANL-East). Each will have five lysimeters; four containing the standard waste types, organic and inorganic resins in both cement and Dow polymer in soil; and one unit holding a waste form in an inert medium. The ANL-East lysimeters will contain local soil, while the ORNL units will be filled with soil similar to that at the Barnwell, SC burial site. The lysimeters will remain buried for 20 years to complete the field tests.

Abnormal Wastes

TMI abnormal wastes are from plant filters, sludges, ion exchange resins, and contaminated equipment which contain TRU contamination in excess of 100 nCi/g, and have high beta/gamma activity. As a result, they cannot be disposed in existing commercial disposal facilities, but must be held in interim storage until a final repository is developed. The EPICOR and Waste Research and Disposition Program is in the process of defining activities in the abnormal wastes area, including: waste acceptance criteria, disposal scenarios, and cost estimates. Early in 1983, the report, *TMI Abnormal Waste Disposal Option Study* (EGG-TMI-6271), was issued. It discusses abnormal waste volumes and forms, a disposal sequence for TMI wastes, options for temporary storage of those wastes at the INEL, and final disposal options.

Work is in progress to define acceptance criteria and cost estimates for ion exchange resins from the TMI Makeup and Purification Demineralizer as discussed in the Waste Immobilization Program section of this report. In addition, INEL storage methods for these resins are presently under development.

Disposal options are also being investigated for a "4 ft x 4 ft" liner currently at TMI which contains 17 plant filters. Disposal scenarios and cost estimates are being developed for this package. Preliminary plans prescribe immobilization of liner contents with concrete, and shipment to a commercial disposal facility.

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