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TMI-2 TECHNICAL INFORMATION AND EXAMINATION PROGRAM 1982 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 seven major areas: instrumentation and electrical components, radiation and environment, core activities, information and industry coordination, configuration and document control, waste immobilization, and reactor evaluation.

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 of the maximum amount of information possible.

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PROGRAM HISTORY AND PURPOSE

The Department of Energy (DOE) Technical Information and Examination Program (TI&EP) at Three Mile Island Unit 2 (TMI-2) completed almost three years of operation with the close of 1982. The Program has seen considerable progress since the GEND group,[a] concerned about the lessons to be learned from the recovery following the March 28, 1979 TMI-2 accident, created the TI&EP in March 1980. The GEND group established the TI&EP to obtain and distribute information which will contribute to the nuclear power industry's knowledge of accident sequences and their effects, and to coordinate research on ways to respond to the technical challenges associated with an effort of this magnitude.

The plant at Three Mile Island continues to be a focus of national and international attention. Nearly every system and component in the Reactor Containment Building was affected in some way by the accident. Cleanup activities involve every major building, including the Turbine, Auxiliary, and Fuel Handling Buildings, as well as the Reactor Building. In Figure 1, the reader will find a diagram of this pressurized water reactor system, showing the arrangement of the main systems and components which have been the focus of key TI&EP activities.

Since its inception, the TI&EP 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 its Data Acquisition Program, comprised of the Instrumentation and Electrical, Radiation and Environment, Core Activities, Information and Industry Coordination, and Configuration and Document Control Programs. Beginning in October 1981, the TI&EP also initiated research and development programs based on the technical challenges related to the recovery and cleanup. Two programs represent the TI&EP's research and development focus: the Waste Immobilization Program and the Reactor Evaluation Program.

During its nearly three years of operation, the TI&EP has played an important role in the progress of the cleanup at TMI-2 while pursuing its goals of obtaining, developing, and distributing vital information to its target audience. With TI&EP assistance, GPUNC 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, GPUNC had decontaminated water which had collected in the Auxiliary and Fuel Handling Buildings as a result of the accident. Fifty of the ion exchange canisters used to clean the water posed a waste disposal problem for GPUNC because of the canisters' unacceptability for burial at commercial burial sites. Through agreements between the NRC and DOE, the TI&EP agreed to take the waste canisters as resource material in their ongoing research and disposition programs in radioactive waste management.

While Auxiliary Building water processing was going on, GPUNC, its contractors, and several DOE contractor experts developed an effective technique for processing the more highly contaminated water in the Reactor Building basement. The DOE TI&EP 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 GPUNC to begin processing. Cleanup of this basement water, as well as processing of 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 in this

a. The acronym GEND identifies the following four organizations: General Public Utilities, the plant owner (since January 1982, General Public Utilities Nuclear Corporation [GPUNC] has operated the plant); the Electric Power Research Institute (EPRI), the research arm of the nuclear power industry; the U.S. Nuclear Regulatory Commission (NRC); and the U.S. Department of Energy (DOE).

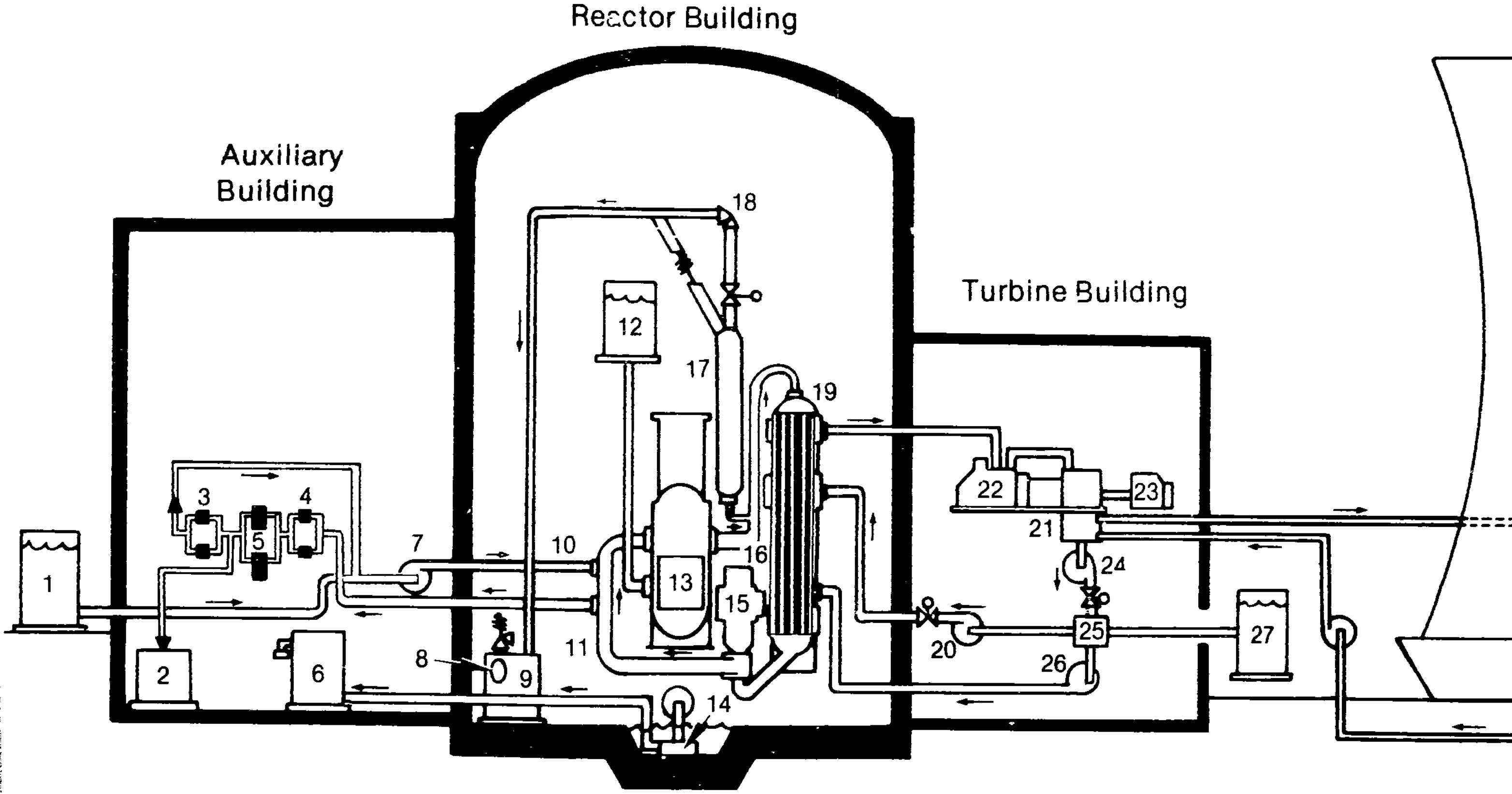
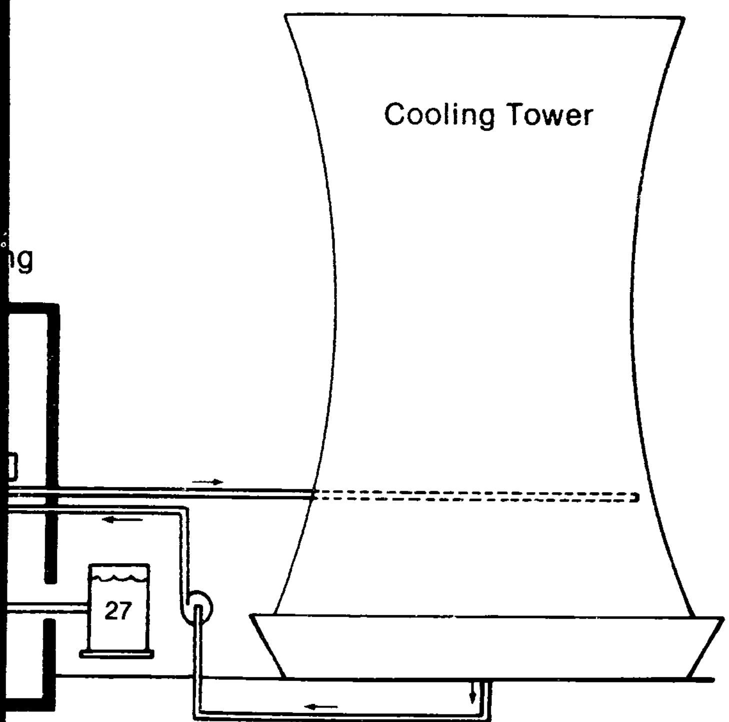


Figure 1. Arrangement of reactor and main components at the TMI-2 nuclea



1. Borated water storage tank
2. Reactor coolant bleed tanks
3. Makeup filters
4. Letdown filters
5. Demineralizers
6. Radiation waste storage tank
7. High pressure injection pump
8. Rupture disk
9. Reactor coolant drain tank
10. Makeup line
11. Cold leg
12. Core flood tank
13. Reactor core
14. Sump pump
15. Reactor coolant pump
16. Hot leg
17. Pressurizer
18. Power-operated relief valve
19. Steam generator
20. Emergency feedwater pump
21. Condenser
22. Turbine
23. Generator
24. Condensate pump
25. Demineralizer
26. Main feedwater pump
27. Condensate storage tank

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main components at the TMI-2 nuclear plant.

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, investigation, and ultimate removal of the reactor core and fuel. TI&EP scientists and engineers have been actively involved in evaluating ways to handle a reactor under such conditions. Much of the research and development work has centered around the actual condition of the core, resulting in new data on reactor component response in the event of an accident such as the one at TMI-2.

Many other TI&EP activities have contributed to the progress of the cleanup while obtaining information of benefit to research and development 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 which occurred during the accident, 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. All major TI&EP activities are documented in technical reports distributed through an established report publication mechanism. Many program findings are communicated directly to the industry via computer conferencing and information networks. And as often as possible, TI&EP researchers present their ongoing work at engineering society meetings and nuclear industry gatherings. Overall, the past three years have been marked by goals met, objectives achieved, and progress made. The year 1982 in particular saw considerable progress in TI&EP programs.

SIGNIFICANT ACCOMPLISHMENTS IN 1982

While pursuing its goals of obtaining and distributing information and furthering beneficial research and development, the TI&EP successfully met several important milestones. The Data Acquisition, Waste Immobilization, and Reactor Evaluation Programs all saw progress in their major activities.

Data Acquisition Program

Instrumentation and Electrical Program. The Instrumentation and Electrical (I&E) Program contributed to the ongoing progress of the recovery while gathering data and developing techniques of benefit to the industry at large. In the incore instrumentation studies, extensive in situ testing determined that instrumentation in the lower regions of the core sustained damage. This finding was important because it gave information on a region of the core which is visually inaccessible. The studies have also been instructive in learning about failure mechanisms and durability of such incore instruments.

The refurbishment of the Polar Crane, a prerequisite to head removal, was assisted by program testing of the electrical equipment on the crane. This testing demonstrated that most of the electrical equipment was functional but also helped indicate equipment needing replacement.

Another important milestone occurred when I&E engineers successfully moved all eight Axial Power Shaping Rods (APSRs) to a hard stop position. Technicians measured the distance each rod traveled, which provided the first estimates of core damage. The insertions were also one of the first steps toward head removal since they allowed the APSRs to be safely uncoupled and parked.

In the other I&E tasks, including in situ testing and some off-site examinations of pressure transmitters; radiation monitors; cables, connections, and penetrations; and electrical components and discrete devices, engineers analyzed instruments for common failure modes and survivability. Where information might impact industry design, installation, or maintenance standards, results were distributed through technical reports, meeting presentations, and computer conferencing networks.

Radiation and Environment. Major accomplishments of the research efforts in the Radiation and Environment Program include ongoing analysis of liquids and solid materials from the primary system and its components. Results indicate these systems are repositories for concentrations of fission products and core and fuel debris that were deposited as a result of the accident.

Development of a radionuclide accounting system is progressing with the assistance of a computer program developed as a result of the Mass Balance Program. This system incorporates measurements of the fission product concentrations to track quantities and locations of all the repositories called source terms. Source term information will eventually result in a radionuclide accident scenario. A preliminary radioiodine mass balance has concluded that while 25% of the core iodine was released, only 0.2% was released to the building atmosphere. Most of the released radioiodine remained in the reactor coolant and other liquids.

Upon removal of water from the Reactor Building basement, work was initiated to characterize the solids on the basement floor. Using a specially designed and equipped closed circuit television camera, several areas of the basement, including the Reactor Coolant Drain Tank cubicle, were explored. This drain tank is believed to be a large repository of radionuclide fission products and core debris. Liquid and solid samples were obtained from the floor area and analyzed.

Greater access to the Reactor Building has resulted from another major 1982 effort, the Gross Decontamination Experiment. Several techniques were tested and evaluated and each proved effective in its specific area of application in the Reactor Building from the polar crane to the 305-ft elevation.

Sufficient information relating to the hydrogen burn event has been accumulated and evaluated to develop a description of the sequence of events accompanied by estimates of temperatures reached.

Core Activities. Several notable achievements occurred in the Core Activities Program which support analysis, characterization, archiving, and storage of TMI-2 fuel and core debris materials. DOE and a broad segment of the nuclear power industry reviewed a plan for examination of the core and its materials. This broad review helped sharpen and redefine the focus of core examination activities so that these activities would more closely address industry concerns.

A special data-gathering tool called the Core Topography System was developed in 1982. Using ultrasonic sensors, this system will gather information on the internal configuration of the core. The data generated will be used to produce topographic maps representing the locations and shapes of existing internal core structures.

Planning for facilities to be used in archiving and storing fuel and core debris and nonfuel and reactor internal components also got under way. The archive facilities will receive, inspect, document, and store TMI-2 samples.

Waste Immobilization and Reactor Evaluation Programs

Waste Immobilization. Research and development in 1982 complemented ongoing cleanup tasks and contributed to successes in several areas. The Waste Immobilization Program accomplished several tasks which benefited the cleanup while also making significant advances in radioactive waste management.

In March 1982, DOE and the NRC reached an agreement about handling TMI-2 radioactive waste. The Memorandum of Understanding signed by the two agencies states that DOE may use special TMI-2 wastes in research and development work with benefits applicable to the nuclear industry. Under the terms of the agreement, DOE performed research and development which facilitated safe waste shipment and waste characterization during 1982.

The TI&EP oversaw design and implementation of a specially designed gas sampling device which ensured safe shipment of 17 EPICOR II water processing liners to DOE laboratories, where all these liners will eventually undergo extensive characterization testing. Two liners were examined in 1982 to determine their response to high levels of radiation and to evaluate possible disposal methods for these and other EPICOR liners. In 1982, a special High Integrity Container (HIC), capable of immobilizing high activity wastes for 300 years, successfully underwent extensive impact and container integrity testing. The HIC will be part of a disposal demonstration for safe, efficient disposition of EPICOR II wastes.

The TI&EP played an important role in facilitating shipment of canisters from another water processing system called the Submerged Demineralizer System (SDS). The first highly loaded SDS liners were safely shipped to a DOE laboratory after a catalyst recombiner system was developed to successfully control generation of radiolytic gases. When the liners arrived at the DOE laboratory, they were used in waste immobilization experiments. In these experiments, DOE proved that the zeolite ion exchange media in the SDS liners could be mixed with glass formers and vitrified into a glass log capable of trapping the radioactive contaminants. Further tests will be performed on SDS liners as they continue to leave TMI in 1983.

TI&EP characterization of resins in the purification demineralizers offered researchers an opportunity to characterize such resins in place in the plant while assisting in the necessary identification of fissile material in these resins. Several national laboratories assisted in characterizing the resins and all results indicated that fuel content in the resins was well below criticality levels. Once this was determined, scientists began identifying and analyzing methods to further sample and then remove the resins in 1983.

Reactor Evaluation. The techniques and equipment developed and applied under the Reactor Evaluation Program during 1982 significantly influenced the course of the entire TMI-2 cleanup effort while acquiring one-of-a-kind data essential for evaluating accident prediction codes and reactor design, operation, and recovery standards. The Quick Look camera inspection of the TMI-2 core provided the first direct evidence of actual core conditions and served as an important point of reference for evaluating accident damage estimates and core and fuel removal plans. The information acquired in the Quick Look was obtained when a closed circuit television camera was lowered into the core to reveal that a 1.5-m-deep void exists in the upper central portion of the core. The Quick Look also provided information on the plenum and various reactor internal components. All of this information is being integrated into plans for vessel head lift and future core activities.

In preparation for reactor vessel head removal, all 61 nuclear control rods, as well as the 8 APSRs, were successfully uncoupled. The Head Removal Task Group formed and began conducting technical evaluations which will help them recommend the best way to remove the reactor vessel head. Radiation surveys were conducted of areas just under the head and down to the top of the plenum which revealed radiation levels somewhat higher than originally anticipated. The data gathered were integrated into head removal plans. Head removal is the next major milestone in the ongoing TMI-2 cleanup; it is presently scheduled to occur in the summer of 1983.

Planning for ultimate removal of the plenum and TMI-2 fuel also began in earnest in 1982. The following two task groups completed and issued technical plans: the Reactor Disassembly and Defueling Task Group and the Plenum Removal Task Group. Review and refinement of plans for these major tasks continued into 1983.

INSTRUMENTATION AND ELECTRICAL COMPONENTS

The proper functioning of instrumentation and electrical equipment is especially critical for the safe operation and control of a nuclear plant in the event of an accident. The Technical Information and Examination Program (TI&EP) Instrumentation and Electrical Program (I&E) evaluates the effects on this hardware of actual accident and postaccident environmental conditions, including wide ranges of temperature, pressure, radiation, and moisture.

Equipment failures did occur during and shortly after the accident, and have continued to occur during the cold shutdown period following the accident. The I&E effort is using a variety of examination techniques including in situ tests conducted from outside the Reactor Building, physical in situ assessment inside the Reactor Building, and component and sample removals for subsequent off-site detailed examinations. 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 within the nuclear power industry. The data and information being obtained from the TMI-2 facility continue to be examined and are providing valuable insight into the adequacy of the instruments and equipment, the adequacy of current instrument standards and qualification procedures, and the importance of plant construction controls and maintenance and operating procedures.

While the primary objective of the I&E Program continues to be to assess the ability of specific safety related instruments, electrical systems, and hardware to perform their intended functions during and after an accident, secondary I&E Program objectives have emerged as a result of the testing and examination. These include participation in the assessment of electrical equipment damage in the Reactor Building polar crane recovery effort and an expansion of the electrical cables and connections evaluation task into an effort to assess the condition of the Reactor Building electrical systems. Both of these latter objectives are slanted towards the TMI-2 recovery efforts; however, insight into the adequacy of the current construction controls and equipment standards is being and will continue to be accumulated. This information will also be passed on to the nuclear industry.

Administrative and technical coordination of the I&E Program is accomplished at the TMI Technical Integration Office by EG&G Idaho. The program scope has been divided into several major work areas, with the lead role for each area assigned to an organization having expertise, facilities, special capabilities, or a combination of these attributes. EG&G Idaho maintains an I&E technical staff at the Island to organize the on-island work, generate the procedures and work packages required to initiate the work, and participate in the work as it is accomplished.

Several other organizations participate in the program at the TI&EP's request. These include United Engineers and Constructors, Sandia National Laboratories, the Idaho National Engineering Laboratory, the Hanford Engineering Development Laboratory, and the Oak Ridge National Laboratory. All of these facilities and organizations have played an active role in the I&E Program during this reporting period.

Accomplishments

The I&E program conducted numerous in situ tests during 1982 in pursuit of its goal to assess instrumentation and equipment survivability. In three areas, incore instrumentation, axial power shaping rods, and Polar Crane work, I&E work supports the progress of the TMI-2 cleanup by gaining information which aids plans for reactor vessel head lift. In the other areas, I&E engineers have identified failure modes which may lead to improvements in instrumentation and electrical equipment design, installation, and maintenance.

Incore Instrumentation. Incore instruments are an integral part of the TMI-2 core. Damage patterns which they sustained may strongly correlate to core damage. The TI&EP undertook to test all the incore instrumentation in an effort to determine the extent of damage in the lower, visually inaccessible regions of the core.

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 each terminates at a seal table in the Reactor Building at the 347-ft elevation, above the vessel head, but below the tops of the hot leg piping.

The I&E program conducted in situ tests on all the incore instrumentation, including 364 SPNDs, 52 background detectors, and 52 thermocouples. These in situ tests were designed to evaluate the general condition of the incore detectors, identify possible failure modes, and determine how much of the original 39 m of each instrument string was still intact. The in situ test measurements included loop resistance, detector insulation resistance, thermocouple voltage output and temperature indication, and time domain reflectrometry data on the cabling and detectors. Scientists compared the in situ test data to installation records, postinstallation data, and the results of laboratory tests to help them determine the extent of the damage and to generally interpret the in situ test data.

The evaluation of the in situ test data indicated the incore instrumentation was significantly damaged and that the most severe damage probably occurred in the upper central region of the core although damage also occurred in the lower core region. Figure 2 maps the location of incore instrumentation. The figure shows the number of detectors at each location that survived the accident.

Present data indicate that all 52 thermocouples failed. Twenty-six had open junctions with moisture in the insulation and two had open junctions with dry insulation. Although 24 thermocouples appeared to have junctions, the loop resistance data indicated that they apparently had formed new junctions as a result of accident or postaccident conditions. Additional in situ tests will refine the estimates of actual incore thermocouple length and will concentrate on determining where these new junctions formed. Knowing where along the 39-m length of the instrument string the new junctions formed will help indicate how far down into the core the high damage zone extends.

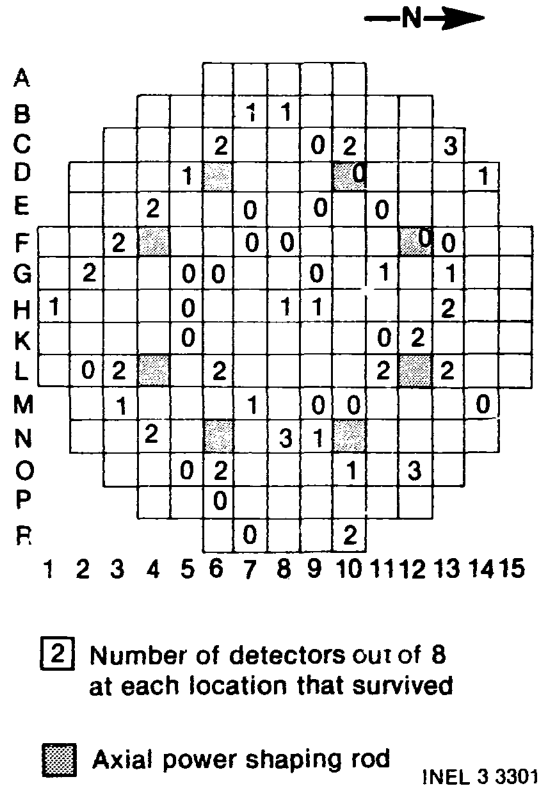

Figure 2. Map of TMI-2 core showing the number of incore instruments which survived at each location.

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 located mainly 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 in these 331 so their output could not be considered reliable. 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. As a group, the majority of the detectors failed under the core accident or postaccident conditions. Their failure indicates that the lower regions of the core, not yet visually accessible, also sustained damage during the accident.

Axial Power Shaping Rod Insertion Tests. In preparation for the reactor vessel head removal phase of the recovery operation, all eight Axial Power Shaping Rods (APSRs) had to be fully inserted to a hard stop position in the TMI-2 core. The I&E program played an important part in this phase of the recovery by devising and implementing procedures which inserted the APSRs as far into the core as they would travel. This rod movement was accomplished using the normal electrical systems aided by sophisticated monitoring techniques to detect resistance to travel. Significantly, the I&E-developed procedures allowed the insertion work to be conducted remotely from outside the Reactor Building, thus considerably reducing radiation exposure levels for workers.

The TMI-2 reactor contains eight APSRs along with 61 nuclear control rods. The APSRs are located in a symmetrical matrix within the reactor, forming a ring about mid-radius in the cross section of the core configuration. Figure 2 shows their locations. In normal plant operation, the eight APSRs are used to flatten the power profile within the core and provide more uniform and efficient use of the nuclear fuel. They are not considered part of the reactor safety or control system and therefore they did not automatically insert into the TMI-2 core when the reactor scrammed at the time of the accident.

When the TMI-2 accident occurred, the APSRs were operating in an approximately 25% withdrawn position, that is, about 1 ft from their fully inserted position. No attempt was made to insert the APSRs either during or subsequent to the accident. Then, in preparation for the reactor vessel head removal phase of the recovery operations, the APSRs had to be inserted to their full down position. Following insertion, technicians would then have to uncouple the drive mechanisms from the APSR control elements and retract the drive mechanism to a parked position.

A special procedure was developed to instrument and move each APSR drive mechanism, one at a time under controlled and recorded conditions. By monitoring the instrumentation and recorder outputs, the drive mechanism response to rod movement attempts could be accurately followed. The equipment and procedures were demonstrated on an undamaged drive mechanism in the manufacturer's plant, the Diamond Power Specialty Company, prior to use at TMI-2. The Diamond Power demonstration confirmed that the procedure could, in fact, accurately determine if the drive mechanism was functional, if the rod drive was moving, or if the shim rod assembly or rod drive was jammed.

The TMI-2 APSRs were then instrumented and the control and recording equipment was set up in the cable spreading room outside of the Reactor Building. Figure 3 shows an engineer operating the special equipment developed for the insertion tests. Attempts to insert each rod resulted in full insertion of two rods, insertion of two rods to about the 5% withdrawn position, insertion of one rod to about the 18% withdrawn position, and, for all practical purposes, no movement of the remaining three. The final position of each rod is listed in Table 1.

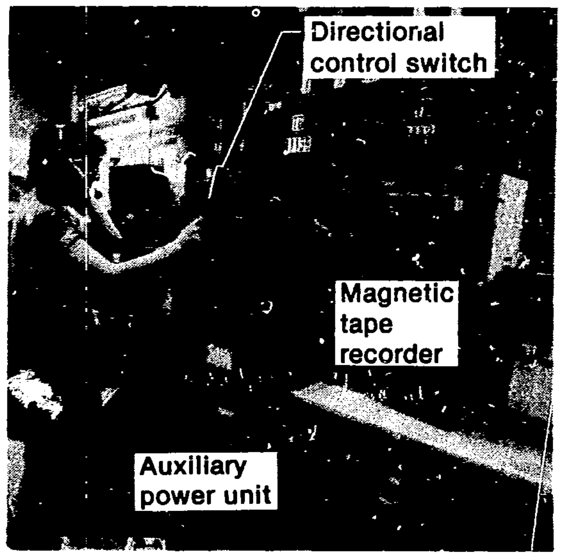

Figure 3. Engineer operating auxiliary power supply unit for APSR test.

Acoustical data generated during the APSR insertion test were analyzed and several preliminary conclusions on the location of damage were reached. These conclusions were examined in the light of the closed circuit television "Quick Look" examination of the core damage and the results of incore instrument failure analysis. Both the Quick Look examination, discussed in the Reactor Evaluation section of this report, and the incore instrumentation work performed by the I&E group indicate that the core is most severely damaged in the upper central region. Engineers

Table 1. Positions of APSRs after insertion testing

Location in Core[a]	APSR Position[b] (%)
F-4	5.2
L-4	18.8
N-6	25.0
N-10	0.0[c]
L-12	4.2
F-12	1.1[c]
D-10	22.9
D-6	26.1

a. See Figure 2 for a map of the core.

b. Percentage of rod not inserted into core.

c. Full in position.

postulated that the apparent full movement of some APSRs probably meant that these APSRs were in fact severely damaged, and that it was the rod drive mechanism, not the APSR shim rod itself, that was moving.

While analysis of the APSR movement only allowed tentative conclusions on core damage to be reached, the test itself was a success because each APSR mechanism was inserted to a hard stop position, thereby allowing the APSR drive to be uncoupled from the APSR control element. Subsequent to APSR insertion efforts, the test procedure and equipment were used in the initial stages of APSR uncoupling and parking in preparation for reactor head removal.

Polar Crane. The I&E Program has played an active part in the recovery and requalification of the Reactor Building Polar Crane. Assessment of the polar crane electrical system and components has followed a series of steps designed to progressively recover the crane functions. The four-part crane program included the following steps: inspection, assessment, refurbishment, and requalification.

Initial investigations concluded that the installed electrical rail systems busing power to both the bridge portion of the crane and the trolley on the bridge of the crane were damaged as a result of the heat associated with the hydrogen burn. It was also concluded that the local control pendant cabling had been damaged by the hydrogen burn and needed replacement. The electrical rail busing systems were abandoned in favor of cable loops designed to supply power to the bridge and from the bridge to the trolley. The control pendant cable was replaced with a unit similar to the originally installed cable.

With power to the crane and local control over the crane functions restored, the balance of the crane electrical system was checked out, one step at a time. This work was accomplished by first performing visual inspections, then nondestructive in situ tests, followed by low power and subsequently full power tests. Any anomalies discovered in this sequence were corrected prior to taking the next step. The net result of this effort has been that very little additional damage has been detected and the recovery of the polar crane has progressed on schedule, with completion anticipated in la · March 1983.

In addition to the pendant control cable shipped to Sandia National Laboratories (SNL) for evaluation, the only other devices placed in storage for subsequent evaluation have been some electromechanical relays and electrowound rotor motor resistor banks. When crane recovery is completed in 1983, a thorough analysis of the damage the crane sustained during the accident will be summarized in a report covering both the damage and subsequent crane recovery.

Radiation Instrumentation and Effects. Four radiation monitors have been removed from the Reactor Building to determine the effects of the accident environment on these important personnel safety devices. SNL conducted extensive tests on these monitors to determine failure modes and analyze possible design characteristics which might alleviate such failures in the future.

SNL has completed laboratory examinations of all four area radiation monitors removed from the building. Three of these, HP-R-211, HP-R-212, and HP-R-213, were geiger-muller (GM) tube personnel safety monitors capable of measuring radiation levels of up to 10 R/h. The dome monitor, HP-R-214, was a dual ion chamber gamma monitor used to measure radiation rates as high as 10^4 R/h which might occur during a loss of coolant acciden It was housed inside a 5-cm thick stainless steel and lead container designed to be hermetically sealed. All these monitors failed either during the first days of the accident or sometime thereafter. Table 2 gives a summary of the estimated failure times and causes. Although the three GM tube detectors were not intended to survive a loss of coolant accident (LOCA), they could have reasonably been expected to survive the TMI-2 environment, at least for a time.

All the GM tube detectors demonstrated a multivalued response characteristic where, at very high radiation levels, the detector can actually indicate low levels. The failure of HP-R-211 was caused by high voltage overstress of one of the output driver transistors as a result of steam and moisture entering an improperly mated connector backshell. HP-R-213 failed at the time of the hydrogen burn when the already weakened or scratched GM tube fractured. Detector HP-R-212 was turned off early in the accident and thus survived until later when it was again powered and experienced quench gas depletion due to the total absorbed radiation dose.

Perhaps one of the most important instruments inside TMI-2 to be examined is the HP-R-214 dome monitor. This instrument, shown in place in its container in Figure 4, was designed to survive a LOCA and was relied upon during the accident to declare various states of emergency. Early in the accident it indicated radiation levels as high as 10^6 mR/h inside the container. While the device eventually registered improperly high radiation levels and finally became erratic and failed, it is believed to have been accurate during at least the first 20 hours after the start of the accident.

Investigators had at first suspected that noble gas leaks into the container were causing inaccurate readings during the early part of the accident. However, after conducting noble gas transport calculations, SNL concluded that gas levels outside the container were close to what the strip chart recorded during the first 20 hours of the accident. Although gas did probably leak slowly into the container starting at about 13 hours into the accident, grossly incorrect readings did not begin to occur until about 20 hours into the accident. SNL concludes that at 20 hours into the accident, moisture which had been slowly seeping into the gasket seal finally leaked slowly through the seal and then through two unsealed screw holes in the detector housing. Due to a switched dc feedback circuit in the high impedance ion chamber-electronics interface, moisture entering the housing is believed to have caused high radiation levels to be registered when in fact the levels had greatly decreased. Figure 5 shows the dome monitor strip chart and the various events the instrument probably experienced.

Table 2. Radiation detector failures

Detector	Type	Failure Time	Failure Cause
HP-R-211	GM[a]	1 to 10 hours	Transistor high voltage overstress due to unsealed connector backshell
HP-R-212[b]	GM	218 days	Total dose quench gas depletion in GM tube
HP-R-213	GM	10 hours	Fractured GM tube
HP-R-214	Ion	100 to 1000 hours	Humidity caused electrode resistive path
		100 to 1000 hours	Metal Oxide Semiconductor transistor radiation degradation
		$>$10,000 hours	Electrolytic capacitor failure

a. All geiger-muller (GM) detectors had multivalued outputs with radiation degradation impedance mismatch.

b. Unit turned off early in accident and activated 88 days later.

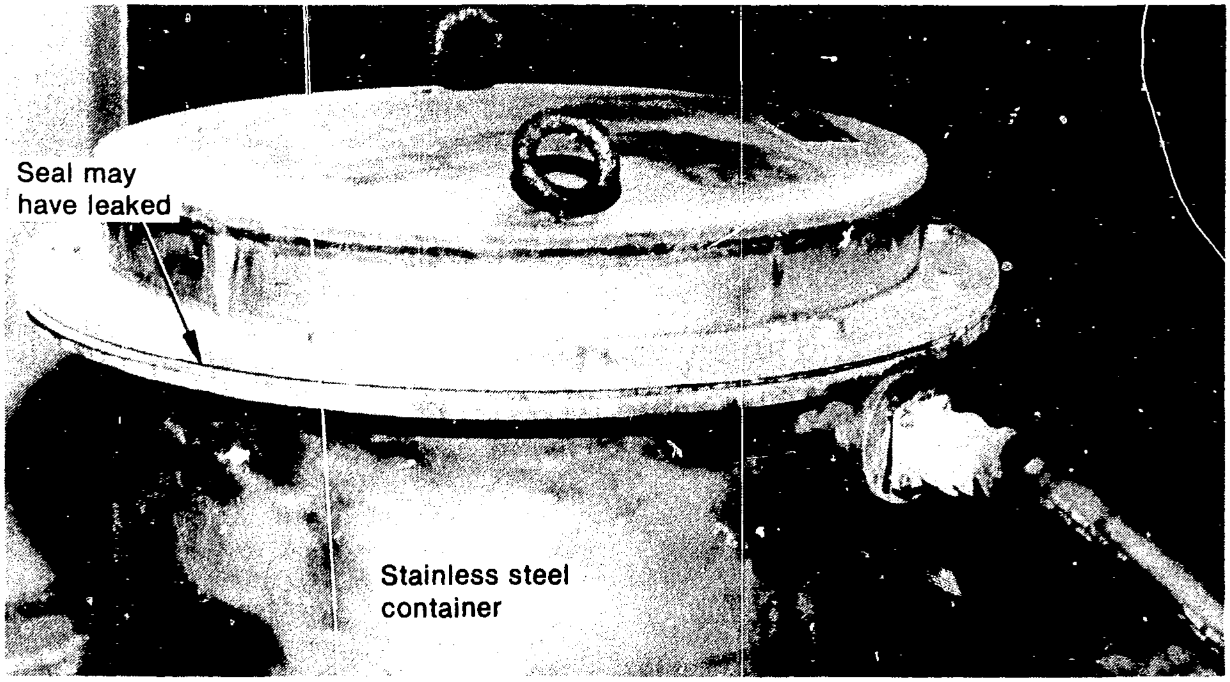

Figure 4. Close-up of hermetically sealed stainless steel canister containing dome monitor HP-R-214 in place in the Reactor Building.

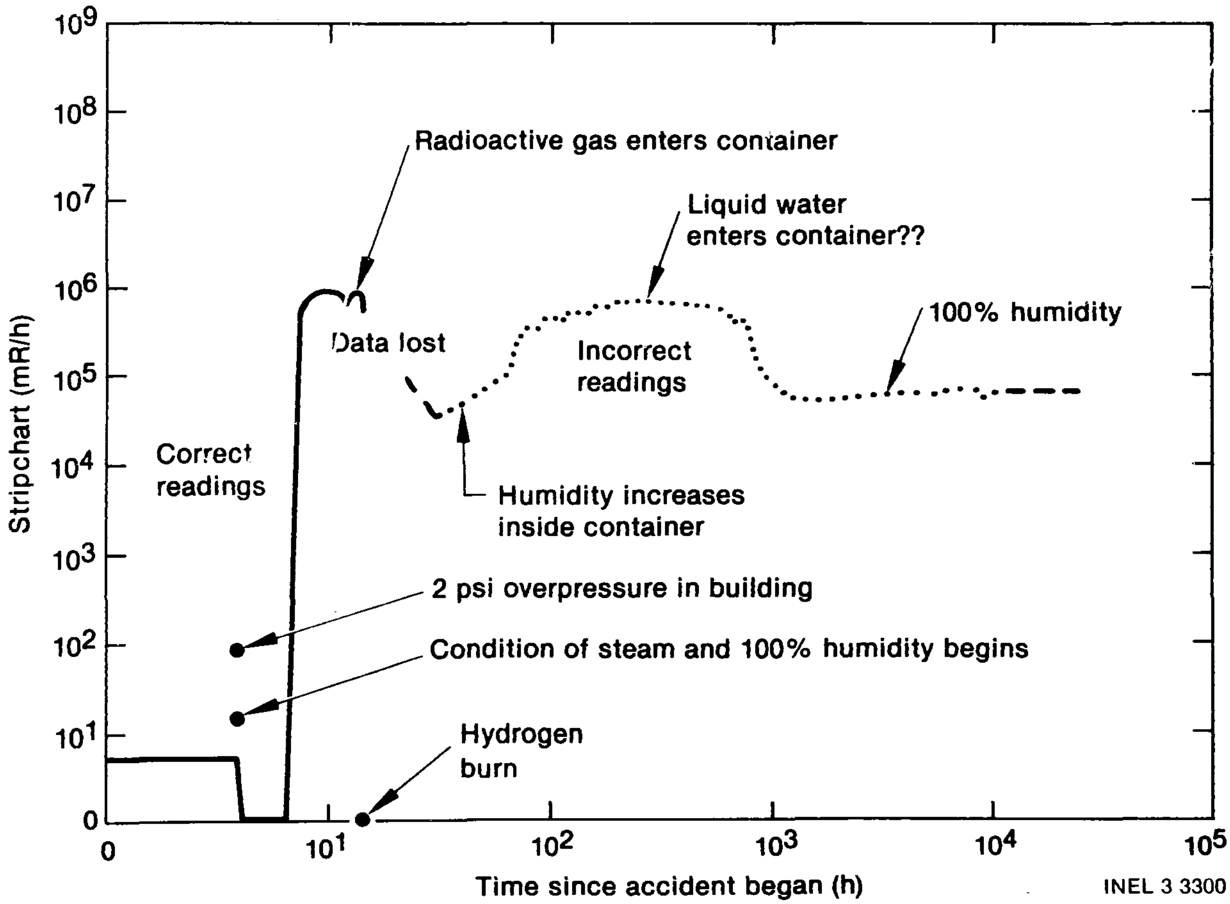

Figure 5. Dome monitor strip chart showing when incorrect readings began and when significant events occurred.

In further investigations at SNL, a metal oxide semiconductor (MOS) transistor in the dome monitor input circuit was found to be severely degraded and erratic, but it is not clear whether this device had much bearing on the detector response. A standard commercial grade electrolytic capacitor was found to have leaked electrolyte onto the circuit board, causing a transistor lead to dissolve in half. Low output readings at the time of removal were caused by the severed lead and shorted capacitor. Both the MOS transistor degradation and capacitor seal failure were radiation dose induced.

In addition to examining instruments for failure modes, SNL has also made estimates of radiation total doses received by the equipment which has been examined. Table 3 gives a summary of the gamma radiation total dose estimates based on exposing similar transistors and elastomers to known radiation levels and then comparing the observed degradation to that in the TMI-2 samples. Transport calculations are also under way to use the dome monitor strip chart and estimates of radionuclide content in the Reactor Building to estimate radiation dose rates outside the dome monitor container. Preliminary results using various noble gas concentrations show gamma radiation levels outside the container to peak at 10^4 R/h approximately 4 hours after the accident began. The total gamma readings peaked at 4.5 x 10^5 rads.

From a design point of view, radiation detector failures such as those experienced at TMI-2 might be avoided or minimized by: (a) performing more extensive environmental testing of equipment prior to installation, (b) eliminating MOS transistors from designs, and (c) potting connectors and conformally coating printed wiring boards. These possible design changes, identified as a result of I&E program examinations, could greatly extend the usefulness of these instruments during a loss of coolant accident.

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

Evaluation began on some of the transmitters located in the TMI-2 Reactor Building to help establish the operational characteristics of these instruments and to determine failure modes where evidence of failure existed. Three transmitters were removed from the building during the reporting period and two of these were examined at the Idaho National Engineering Laboratory (INEL). The third will be examined during 1983 along with other pressure transmitters slated for removal. Figure 6 is a photograph of transmitters to be removed in 1983. The transmitters examined in 1982 are identical to those in the figure.

The two transmitters examined at the INEL were not themselves designed and fabricated to withstand a loss of coolant accident. Nevertheless, investigators determined that they were similar

Table 3. Preliminary gamma dose estimates for radiation monitors

Instrument	Building Elevation (ft)	Material Analyzed	Dose Estimate (10^5 rads)	Range (10^5 rads)
HP-R-211	305	6 transistors	2.5	0.85 to 5.1
		2 Teflon sleeves	2.0	0.7 to 6.0
HP-R-212	305	6 transistors	4.5	1.5 to 11.0
HP-R-213	347	6 transistors	9.9	3.9 to 18.5
HP-R-214	372	10 transistors	2.83	2.1 to 4.5

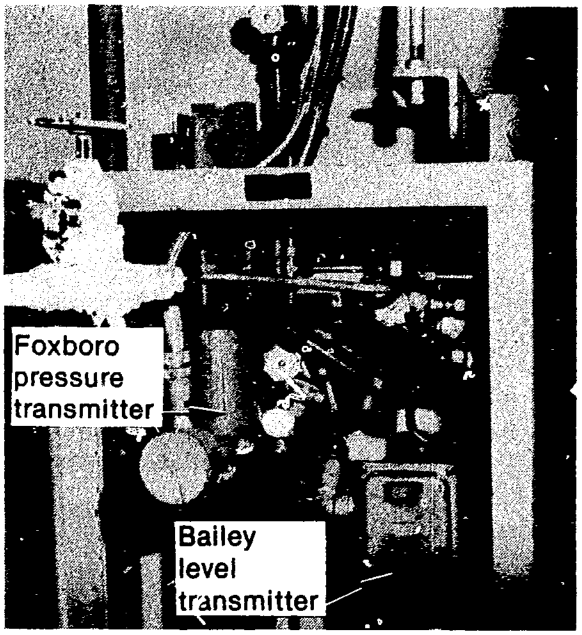

Figure 6. Two pressure transmitters in place in the TMI-2 Core Flood Tank A.

enough to the LOCA-qualified transmitters (which are in an inaccessible location in the Reactor Building) to provide information on common failure modes.

The first instrument to be examined was CF-1-PT3, a Foxboro pressure transmitter designed to monitor pressure in Core Flood Tank B, located at the 324-ft elevation. CF-1-PT3 appeared to have survived the accident and the subsequent Reactor Building environment. Investigators performed a pressure calibration of this transmitter and found it operational and in calibration. The internals of this transmitter appeared to be well sealed from any moisture.

The second instrument to be examined was CF-2-LT3, a Bailey level transmitter designed to monitor water levels in Core Flood Tank B. CF-2-LT3 was taken out of service one year after the accident and was not operational upon examination at the INEL. A detailed examination of this transmitter revealed that extensive damage of the internal components had occurred as a result of moisture. The transmitter still had 30 ml of water in its housing when the cover was removed for examination.

Figure 7 shows the general condition of the internal components of the transmitter. Some electronic components were badly corroded, with some of the leads actually corroded away. A transformer appeared to have been badly burned even though the input power fuse was still good. A power transistor was rusted so badly that it crumbled when touched.

The pressure ports and gaskets of the corroded transmitter 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 transmitter housing appeared to be the most likely sources for moisture in the transmitter.

While neither unit was LOCA-qualified, it appears that the seals in pressure transmitter CF-1-PT3 were adequate to withstand the pressures, temperatures, and steam encountered during the accident, while those in level transmitter CF-2-LT3 were not. Examination of two other transmitters, CF-2-LT1 and CF-2-LT2, will provide an opportunity to determine if a common failure mode exists between these and the corroded CF-2-LT3 transmitter. All the examinations will be considered together in an evaluation of why some of these safety-related pressure transmitters failed.

Electrical Components and Discrete Devices. The I&E program undertook to examine the many electrical components and discrete devices in the TMI-2 Reactor Building to evaluate the effects of the accident on such hardware and to assess its survivability under LOCA conditions. While some of this hardware is safety-related and therefore considered vital to safe management of the plant, much of it is of the "nuts and bolts" variety, without which the plant simply cannot operate. This hardware includes switches, solenoids, valves, motors, control rod drive mechanisms, and resistance temperature devices. By studying accident effects on electrical hardware, the program is contributing to the overall understanding of the accident while discovering electrical failure modes and analyzing ways to avoid such failures in the future.

An aggressive in situ testing program continued through 1982, furthering evaluation of the condition of electrical components in the Reactor Building. In addition to the original testing program, many items were retested subsequent to the Gross Decontamination Experiment. The highlights of the program are as follows:

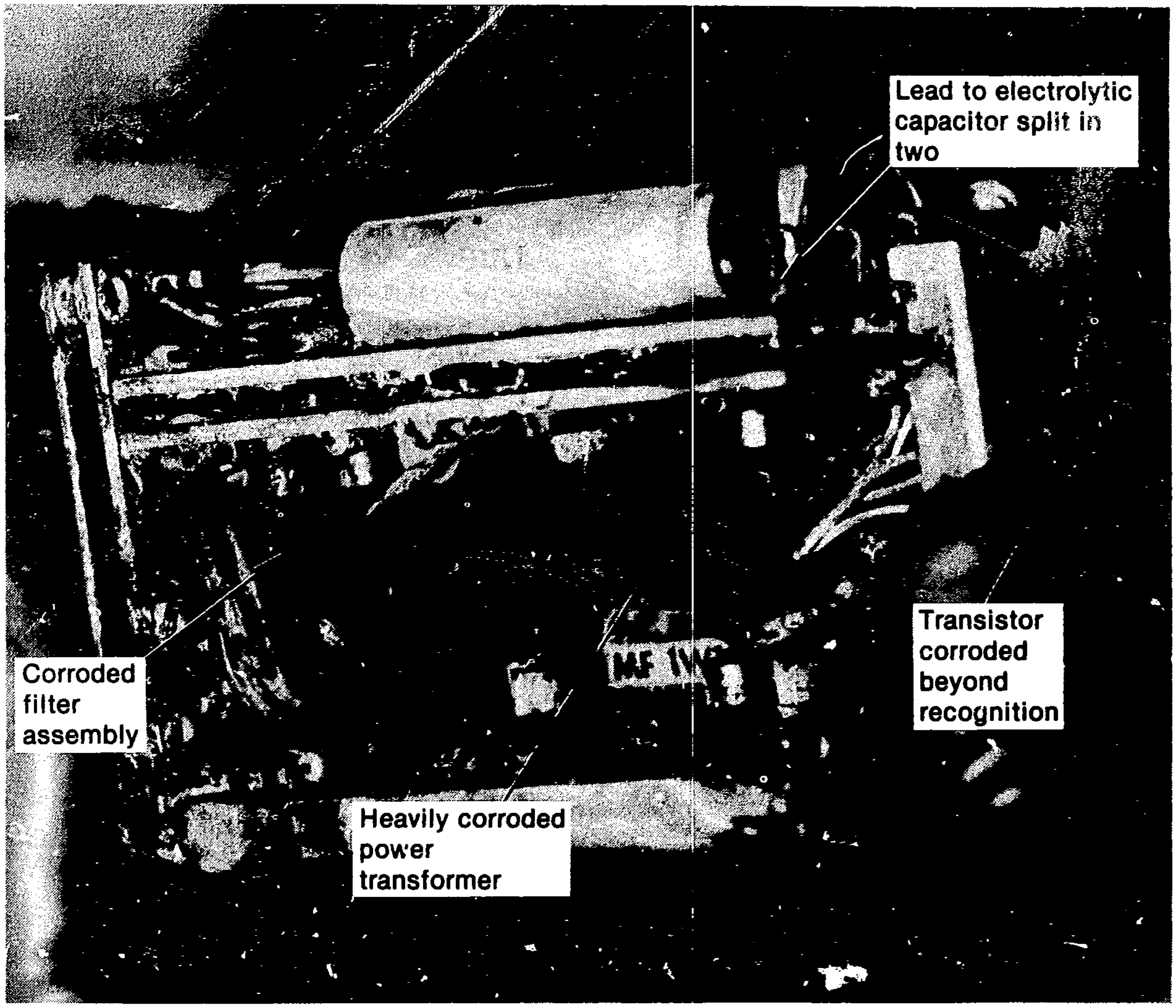

Figure 7. Badly corroded internal components of TMI-2 level transmitter damaged by moisture.

- Nearly 70 electrical components have been tested, of which 23 have been retested to determine the effects of gross decontamination.

- Twelve electrical components have anomalies considered to be the result of the accident. Table 4 summarizes the anomalies observed in these 12 components.

- Four electrical components have been removed from the Reactor Building for detailed off-site examination and five additional pieces have been approved for removal.

Cables, Connections, and Penetrations. The I&E program undertook to study the effect of the TMI-2 environment on cables, connections, and penetrations in an effort to evaluate their survivability. In order to do anything in the future with these components, recovery and requalification planners must know how well they survived and what condition they are in. As the I&E program samples and examines these cables, connections, and penetrations, common failure modes or areas of degradation might give rise to changes in installation requirements, design requirements, or nuclear qualification requirements.

The TMI-2 Reactor Building contains approximately 152 km of cables and connections essential to the safe operation of the plant. Many of these cables lay in trays and travel through conduits such as those seen in Figure 8. Cables and connections were defined as all components in a

Table 4. Devices and components that exhibited anomalies

Identification Number	Type	Anomaly Observed	Disposition
AH-V6 (1E)	Valcor solenoid valve	Reed limit switch is welded closed	Removed for off-site examination
RC60-LS2	Static-o-ring pressure switch	Failed to respond to signal	To be removed for examination
NM-PS-4174	Static-o-ring pressure switch	Failed to respond to signal	Removed for off-site examination
AH-LS-5006	Gems level switch	Very high dc resistance	To be removed for examination
CRD-ST50	Thermocouple	Open circuit	No further testing planned.
RC-P-1A	Reactor coolant pump motor 9000 hp, 6900 Vac	High resistance on differential current transformer circuit at penetration joint	Check penetration R405 inner box when accessible
RC-P-1A Backup Oil Lift Pmp Motor	10 hp, 250 Vdc motor	Open circuit at the armature presumably due to brush-commutator corrosion	To be removed for examination
Cable 1DC110P	Power cable to RC-P-1A backup oil lift pump motor	Circuit break at the penetration inner box R400 and jumpers missing	To be examined closely when penetration is accessible
Cables H2911 and H3011	Instrument cables	Gross impedance discontinuity at the midpoint	Visually examine faulted section when accessible
Penetration R607	Electrical penetration	Crosstalk of several balance of plant computer circuits	Examine penetration when accessible
RC67-VS1	Robertshaw vibration switch	Circuit break on reset coil circuit	To be removed for examination
RC67-VS2	Robertshaw vibration switch	Very high resistance on reset coil circuit	To be removed for examination

given electrical channel or circuit from the Reactor Building penetration assembly up to but not including the instrument or unit 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.

SNL assumed management of the cable and connections task for the TI&EP. They worked together with a team of experts from the nuclear power industry, GPUNC, DOE, NRC, and other national laboratories to ensure that representative cable and connection samples would be obtained from environmentally representative regions of the TMI-2 Reactor Building.

During several Reactor Building entries, such as the one seen in Figure 9, SNL studied cable layout, cable conditions, and general building conditions. Then, in 1982, two cable samples were removed from the building for study. The first sample was exposed cable from radiation dome monitor HP-R-214 pictured in Figure 10 in place on top of the elevator shaft in the southeast quadrant of the building at the 370-ft elevation. The cable was subjected to mechanical testing and the results showed that it did survive the accident.

The polar crane pendant cable was the other cable sample removed during 1982. The control cable was an interesting sample because it was exposed to the accident environment, and the

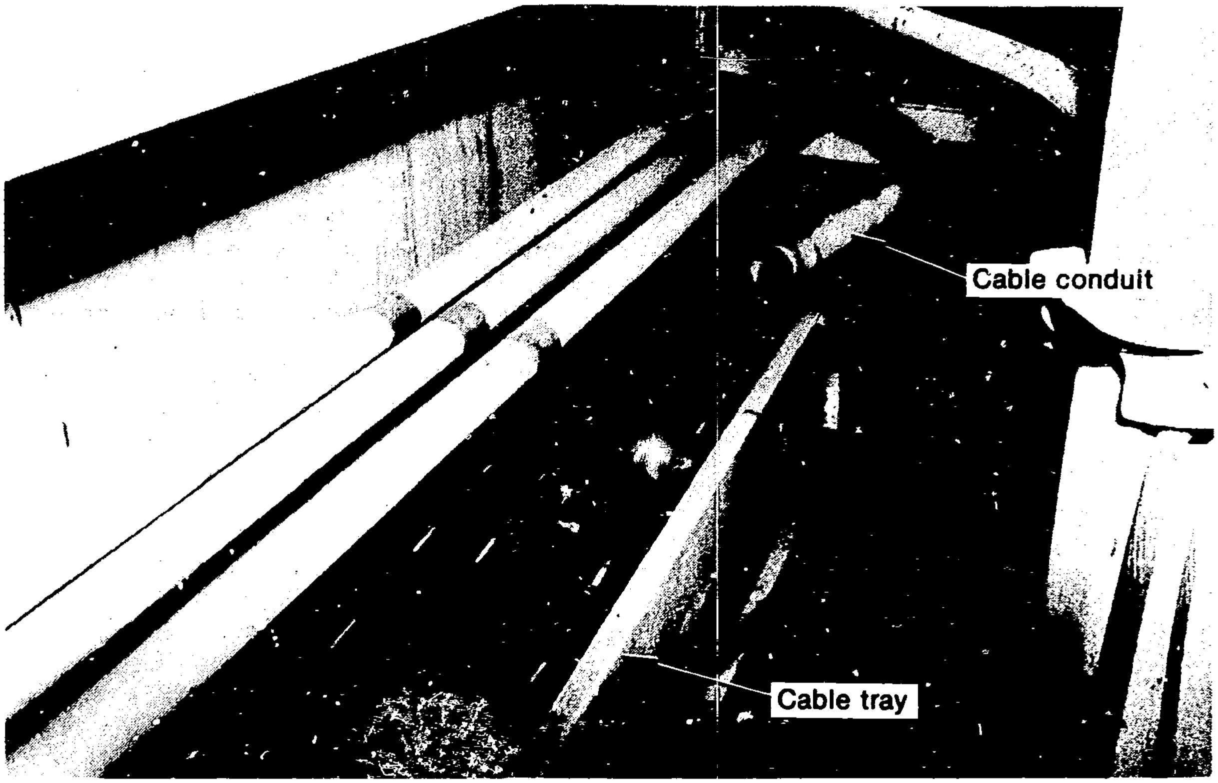

Figure 8. Cable in conduits and cable tray in the Reactor Building.

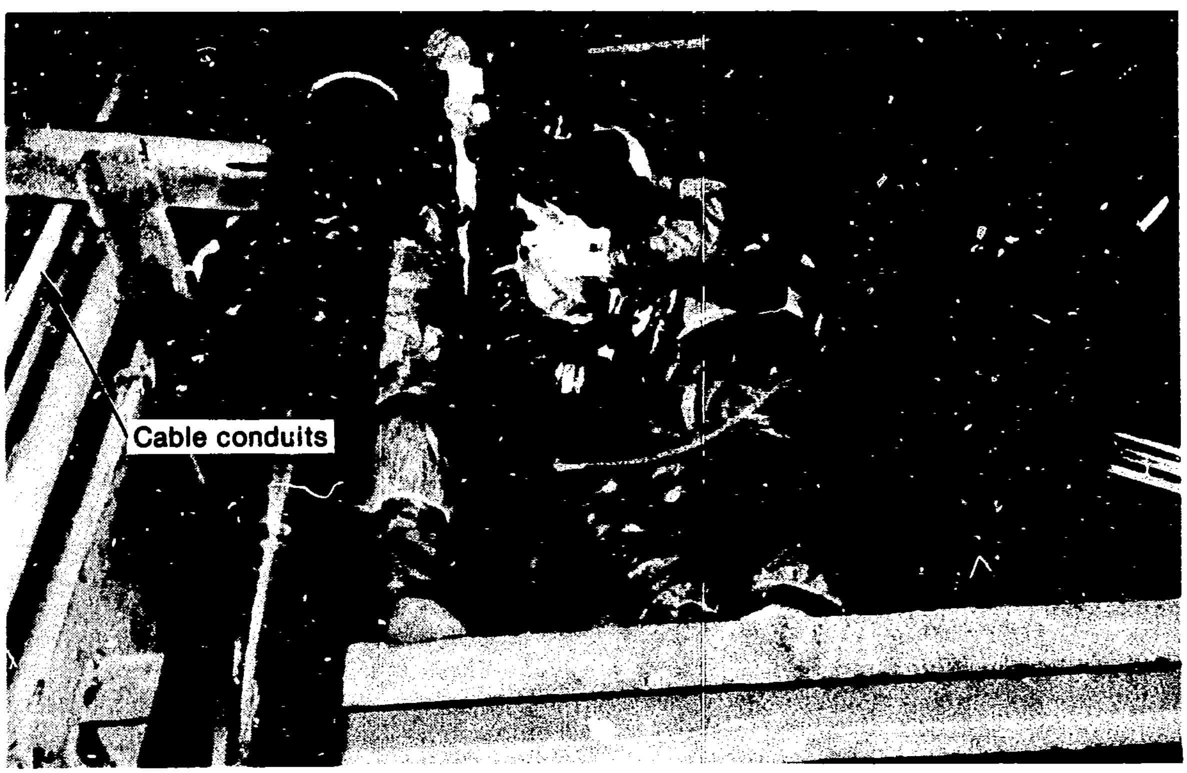

Figure 9. Engineers inspecting cable conditions inside the TMI-2 Reactor Building.

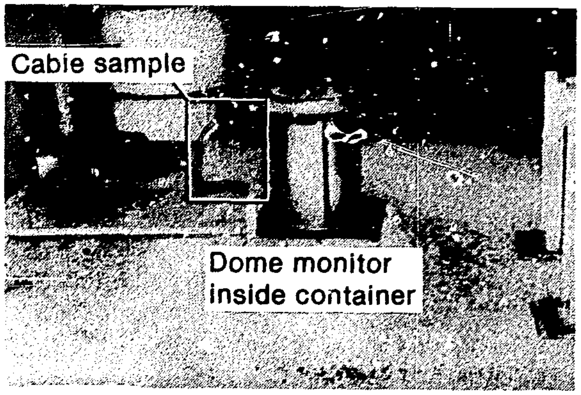

Figure 10. Dome monitor HP-R-214 in place in the Reactor Building showing cable which was later removed for analysis.

hydrogen burn in particular, along an 18-m vertical plane in the building. The cable was hanging down from the polar crane 427-ft elevation to the top of the west D-ring at the 367-ft elevation. Analysis of the burn pattern on the 18-m vertical sample indicates that the hydrogen burn at TMI-2 did indeed begin on the west side of the building and moved east, as was concluded by investigations discussed in the Radiation and Environment section of this report. Patterns on the cable indicate that the burn may have propagated up the open stairwell.

Mechanical testing showed that the cable was serviceable. Although the cable was exposed to a burn sufficient to char the cable jacket, the insulation around the individual conductors in this 40-conductor cable were affected very little; they did survive the accident.

Future samples to be studied include control as well as Reactor Building samples. The control samples are important in establishing baseline physical and electrical data with which to compare the properties of Reactor Building samples exposed to the TMI-2 accident. Finally, the importance of a sophisticated in situ test plan was clearly identified in 1982 and the support of personnel at two national laboratories has been engaged to implement this effort in 1983.

Resistance Temperature Detectors. The data recorded by the Reactor Building Resistance Temperature Detectors (RTDs) during and after the accident provide temperature profiles which are one key to understanding the accident and subsequent plant conditions. Given this consideration the I&E program conducted examinations and in situ tests on the RTDs during 1982. The RTDs are located throughout the Reactor Building. They contain platinum resistance elements sensitive to temperature change. As their temperatures increase, their resistances increase in exact and repeatable proportions. The changes in resistance are then shown as changes in temperature on strip chart recorders in the control room, giving plant operators an indication of rising or dropping temperatures.

Specific tests on RTDs in the Reactor Building Air Handling System showed that all 16 of the devices in that system are functional, based on preliminary testing and data. During in situ testing of these RTDs, the integrity of each recorded data channel was traced and verified, the data gathered were reduced, plotted, and analyzed, and RTD readings obtained during the accident were plotted in graph form. The RTD tests confirmed that accident data recorded by the instruments during the accident were probably accurate. One RTD accident-data plot, shown in Figure 11, represents the kind of activity some of the 16 Air Handling System RTDs recorded during the accident. This RTD, located on the 353-ft elevation, nearly 2 m above the upper working level in the plant, recorded an increase in temperatures beginning shortly after 4:00 a.m. on March 28, 1979, when the accident began. Temperatures increased sharply at about 10 hours into the accident, the estimated time of the hydrogen burn.

Once the relative accuracy of the 16 RTDs had been established, INEL scientists conducted tests to determine the time response characteristics of the devices. The data from the time response analyses were then applied to the accident-recorded data in an attempt to estimate the actual Reactor Building temperatures as a function of time. Scientists were particulary interested in the temperatures occurring at the time of the hydrogen burn.

During 1982, the I&E Program arranged for Oak Ridge National Laboratories (ORNL) to perform a series of special in situ tests on Reactor Coolant System RTDs during the first quarter of 1983. These tests, called Loop Current Step Response Tests, are designed to determine the response times and conditions of the RTDs. The test results, when compared with RTD manufacturer data and data from laboratory tests and other nuclear plants, will provide an indication of the RTD status and will indicate whether or not degradation has occurred. All RTD testing aims to test the reliability of the instruments; data gained may be useful in creating a time-based history of Reactor Building temperatures during the accident.

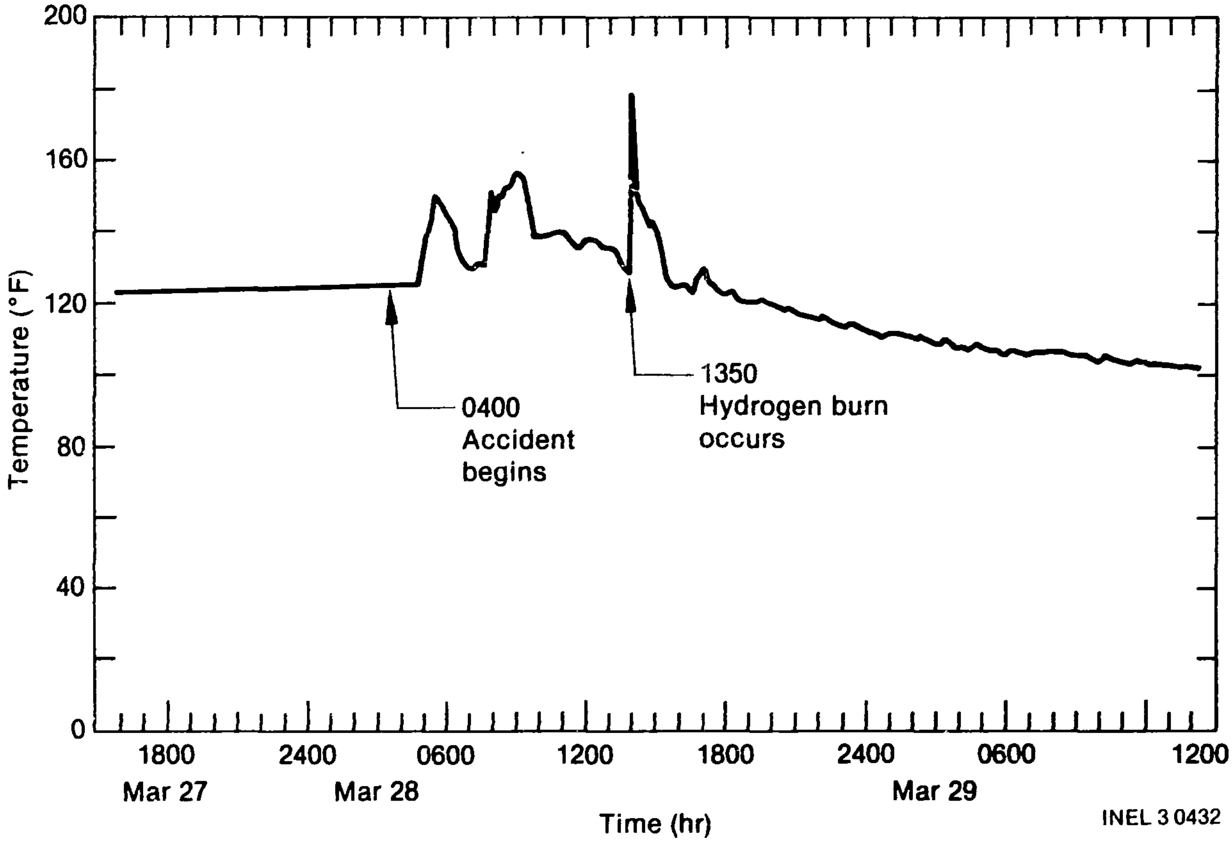

Figure 11. Plot of Resistance Temperature Detector data compiled from investigation of top-ceiling RTD at the 353-ft elevation.

RADIATION AND ENVIRONMENT

The Radiation and Environment program consists of three major areas of interest, Fission Product Transport and Deposition, Reactor Building Gross Decontamination Experiment, and Accident Evaluation. Program objectives, defined by these areas of interest, include measurement of transport and deposition of fission products from the reactor core to the plant environment, evaluation of various decontamination techniques, and assessment of the accident effects upon the physical 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.

Accomplishments

In tracking the transport and deposition of fission products from the core throughout the Reactor Building, the Radiation and Environment Program has divided release paths into three major areas, each representing a discrete fraction of the plant accident environment. These areas, Primary Systems, Reactor Building and Support Systems, and Surface Deposition and Environment, are made up of smaller areas and systems that present the major fission product repositories—source terms—for the plant. Each smaller area quantifies an individual source term and its contribution to fission product transport or deposition. The overall fission product interest area made considerable progress in 1982 quantifying fission product releases and in relating the individual source terms to each other and to the accident.

In March 1982, the Gross Decontamination Experiment successfully tested several decontamination techniques which contributed to overall efforts to reduce contamination levels in the building. Data gathered during 1982 also contributed to a more detailed understanding of the accident hydrogen burn. Taken together, the three major interest areas in the Radiation and Environment program made significant progress toward enhancing understanding of the accident environment and associated phenomena.

Primary Systems. During 1982, work was initiated to obtain information on the transport and deposition of core fuel debris, fission products, and activated corrosion products in the primary coolant system and its associated components. Fission product identification and quantification are important for completing a radionuclide mass balance and providing information on radionuclide distribution to support planning of reactor vessel head removal operations.

The release boundaries and pathways for fission product transport from the reactor core to the reactor coolant system (RCS), reactor auxiliary components, and ultimately the Reactor and Auxiliary Buildings are the major source terms for core debris and fission products outside the reactor vessel. Material transported to and through these primary systems during the accident was the source of all fission products released to the plant environment.

Reactor Coolant System Liquid Samples—RCS liquid samples have been obtained periodically and analyzed extensively since the time of the accident to measure concentrations of radionuclides released to the reactor coolant from the degraded core, and to observe changes in radionuclide concentrations in the coolant over time. These data are necessary for assessment of radionulide releases that occurred during the accident, and they also provide long-term information about other release mechanisms such as leaching. The 1982 liquid sample from the RCS was taken in March, before the RCS water was decontaminated.

The principal point of interest in the RCS sample is the concentration of ^{90}Sr. This radionuclide began increasing in the RCS after the accident and has remained high (approximately 15 μCi/ml) despite continuous coolant dilution due to leakage. It appears that this usually insoluble radionuclide is leaching from the core or is in the form of a slightly soluble compound. High concentrations of carbonate in the reactor coolant may indicate that strontium carbonate is the species responsible for maintaining the ^{90}Sr concentration in the coolant. Increases of ^{90}Sr in the coolant detected after completion of RCS decontamination indicate ^{90}Sr sources still exist in either the RCS, or the core, or both.

Reactor Coolant Bleed Tank Sludge—Measurement of the types and quantities of fission products in the three Reactor Coolant Bleed Tanks (RCBTs) is important in evaluating radionuclide releases to the TMI-2 Auxiliary Building as a result of core degradation. During the course of the accident, reactor coolant was transferred through the letdown line to the RCBTs (see Figure 1). Liquid samples from these tanks contained appreciable quantities of radionuclides. Analyses show that the RCBTs were the principal source terms in the Auxiliary Building.

In 1980 and 1981, when radiation levels near the tanks permitted, samples were taken of all the RCBTs. Although the water in the tanks was recirculated for several hours to thoroughly mix liquids and solids together before the samples were taken, sample constituencies were mostly liquid with only a few milligrams of solids in each sample. Analysis of these preliminary liquid and solid samples revealed high levels of fission products in both sample fractions. The results indicate between 3 and 5% of the total core inventory of soluble radionuclides may have been contained in the RCBTs. However, too few solids were available to perform all the analyses required to provide definitive answers on the quantities of radionuclides contained in the tanks.

When RCBT A was opened to observe tank solids and determine the most effective way to remove contamination from the inner tank surfaces, a sludge sample weighing about 60 g was removed. The sample was shipped to Westinghouse Hanford Engineering Development Laboratory (WHEDL). A sieving for particle size analysis performed on the sludge samples yielded a mean particle diameter of 4.3 μm indicating a transfer of very small particulate material out of the RCS to the RCBTs. Figure 12 is a photograph (under 10x magnification) of one of the larger sludge particles from RCBT A. WHEDL's analysis demonstrated that solids did contain quantities of fission products, fuel, and core material that were not apparent in the liquid analysis.

Science Applications, Inc. performed a gamma spectrometer scan on the tank in August 1981 which showed that RCBT A contained 4.1 Ci of ^{144}Ce. The WHEDL analytical results indicated RCBT A contained a ^{144}Ce concentration of 2.5 μCi/g. Combining these analytical results with the gamma spectrometer measurements resulted in a rough approximation of 5 kg of solids in RCBT A—a tiny fraction of the approximately 180 m.t. of core material. However, the much

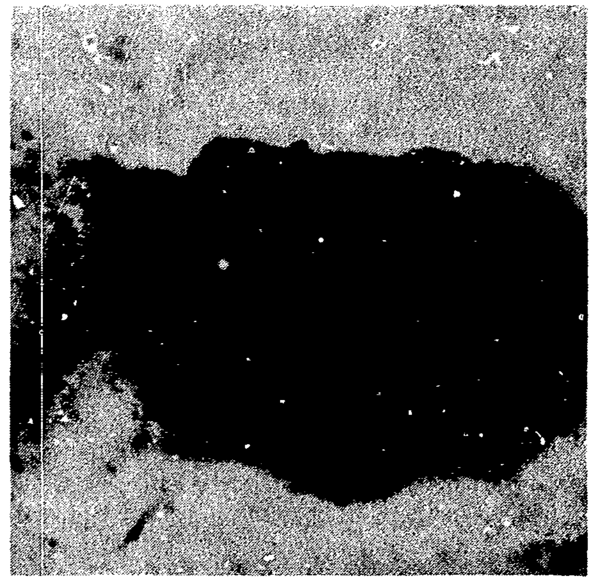

Figure 12. RCBT A sludge particle under 10x magnification.

larger fraction of fission products measured in the tanks represents a source term larger than the quantity of solid materials transferred. This out-of-proportion quantity of core fission products to core mass indicates that the fission product release and the transfer of solids are probably the result of different transport mechanisms.

Purification System Samples—The Makeup and Purification System at TMI-2 contains RCS sampling points, supplies reactor coolant makeup water, and allows for reactor coolant chemical adjustment. Using filters and demineralizers, the system provides a means of controlling RCS fission product inventories. On the day of the accident, the system, usually in continual operation, was shut down because of fluctuating flow rates and overpressure indications from the filters and demineralizers. When returned to service, system flows had been reduced and the filters and demineralizers bypassed, thus overriding the system's purification capabilities.

Radiation surveys performed in 1981 and 1982 near the system filters, demineralizers, and system components showed radiation levels ranging from 1 to 1500 R/h. A 1981 qualitative gamma spectrometer measurement showed that the material in makeup filter MUF-2B consisted of soluble and insoluble mixed fission products. Of special interest was ^{144}Ce, observed in the spectra of the gamma measurement. Because of its chemical characteristics, ^{144}Ce usually follows and is associated with both uranium and plutonium. Also visible in the spectra was ^{110m}Ag, which is an activation product of silver (Ag) control rod material. The presence of these two nuclides along with soluble nuclides such as ^{134}Cs and ^{137}Cs indicates that the filters probably plugged with solid core material that contained mixed fission products, fuel, and probably some activated control rod material.

On April 1, 1982, filters MUF-2A, -2B, -4A, -4B, and -5B from the Makeup and Purification System were shipped to the INEL along with vacuum filter samples obtained from each filter housing. In the Test Reactor Area hot cell, the samples were gamma scanned. The degraded makeup filter MUF-5B is shown in Figure 13. Filter material and solids on MUF-5B had fallen off the filter housing when it was received in the hot cell. Material from MUF-5B was subjected to additional chemical and radiochemical analysis.

Selected data from the analysis of MUF-5B are presented in Table 5 where they are compared to a small "grab" sample obtained in 1981 during an unsuccessful attempt to remove the filter. The analytical differences occur because the grab sample represents only a fraction of the total materials which were heterogeneously distributed on the filter. The material from the filter consists of fuel, core structural material, control rod material, and fuel cladding. The MUF-5B filter, as received, contained approximately 210 g of these solids. Using the uranium weight fractions from Table 5, a weight of approximately 15.3 g of uranium is estimated in MUF-5B.

Particle size distributions show that greater than 80% of the solids are smaller than 5 μm. Larger particles up to 50 μm were observed, but due to the slight magnetic properties exhibited by the material, these larger particles may be agglomerations of smaller particles. A microscopic examination of the particulate material showed no evidence of melting. The solids are principally material which appears to have fractured along grain boundaries.

Zirconium oxide (ZrO_2) particles discovered in the filter sample collected from MUF-5B have been analyzed on the scanning electron microscope (SEM). The ZrO_2 could have originated in either the ceramic spacers or oxidized fuel rod cladding. The ceramic spacers in the TMI-2 core contained calcium to stabilize the ZrO_2. The presence of calcium in the filter sample would indicate that the ZrO_2 particles are fragments of ceramic spacers. No calcium was detected by the SEM on any of the ZrO_2 particles, suggesting that they came from oxidized fuel rod cladding.

The spectra of some ZrO_2 particles from samples were similar to spectra of pure ZrO_2 raised to 1200°C under laboratory conditions. This fact indicates that the ZrO_2 fuel rod cladding was oxidized and heated to at least 1200°C. At this temperature, ZrO_2 undergoes transition to the tetragonal phase which at lower temperatures is unstable. However, the ZrO_2 in the samples had been locked into the tetragonal phase and had not become unstable as it cooled. Two possible conditions may explain what locked the ZrO_2 into the tetragonal phase. It could be locked by either rapid cooling (quenching) or the presence of impurities. Further analysis of the ZrO_2 particles from the filters will determine if impurities which could cause the phase lock are present.

The gamma scans of the remaining Makeup and Purification System filters and vacuum filter samples show that the predominant gamma-emitting nuclides were fission products, ^{137}Cs, ^{134}Cs, ^{125}Sb, ^{106}Ru, and ^{144}Ce, and activation products, ^{60}Co, ^{110m}Ag, and ^{54}Mn. The total uranium inventory of the filters was estimated to be less than 12 g. The radionuclide inventories of the filters segregated the filters into two distinct groups. The B filters (MUF-4B, VAC-4B, VAC-2B, and VAC-5B) comprised one group having just ^{137}Cs, ^{134}Cs, and smaller or nondetectable amounts of other nuclides. The A filters (MUF-4A, VAC-4A, VAC-2A-1, VAC-2A-2, VAC-2A-3, and VAC-5A), which were operating when the accident occurred, comprised the second group, having comparable relative amounts of all 8 radionuclides. The explanation for the different nuclide distributions is that the A- and B-train filters were in service at different times following the start of the accident.

Reactor Coolant Drain Tank—The Reactor Coolant Drain Tank (RCDT) is believed to be a repository for a large mass of core material. A component of the Reactor Coolant Leakage Recovery System, the RCDT collects, cools, and stores leakage and/or discharge from the Power Operated Relief Valve (PORV), pressure relief valves, reactor coolant pump seals, and reactor vessel closure

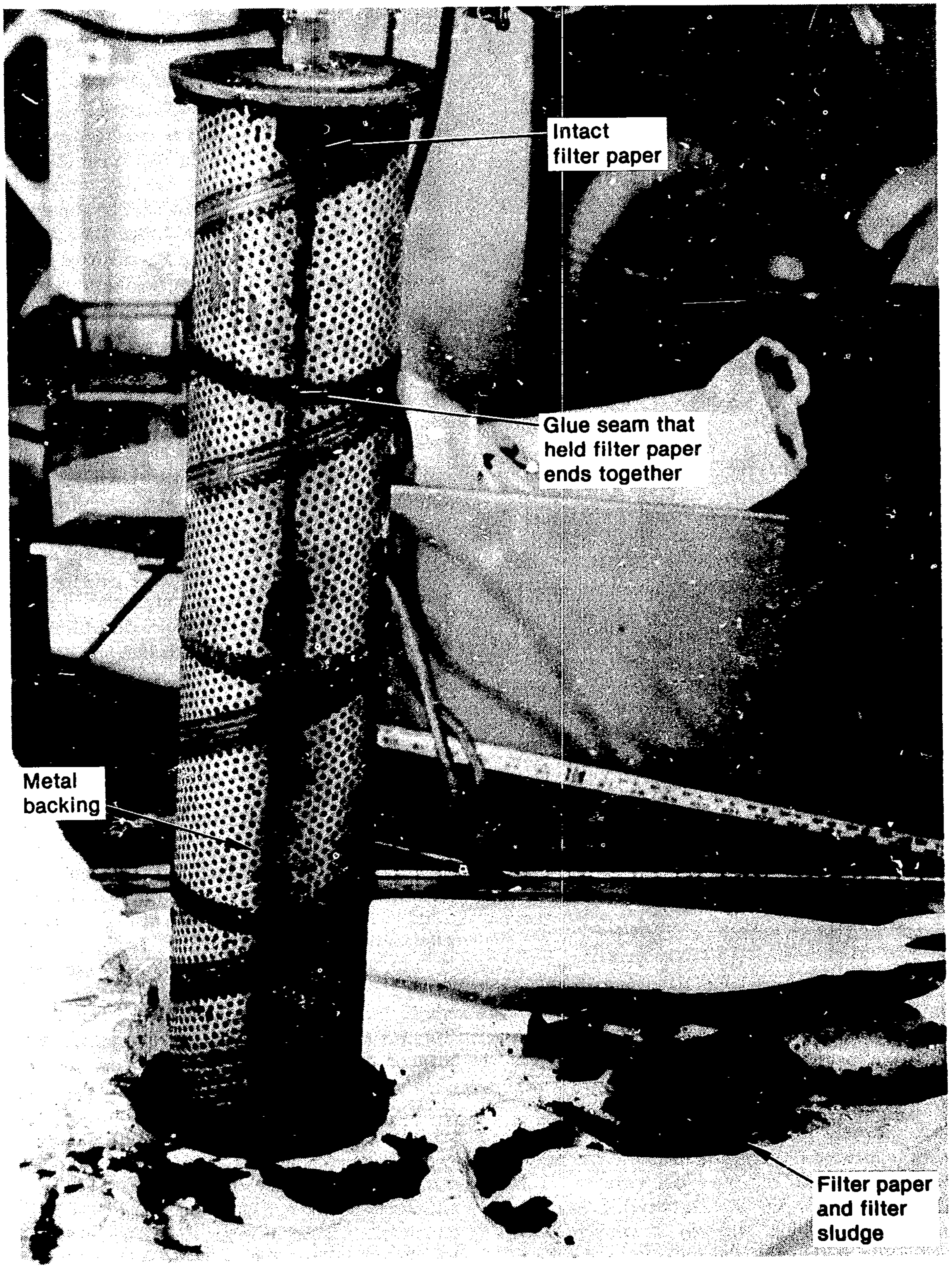

Figure 13. Makeup filter MUF-5B in hot cell for analysis.

Table 5. Uranium and elemental analysis results for makeup and purification system filter samples (wt%)[a]

Nuclide	MUF-5B Sample Fraction A	MUF-5B Sample Fraction B	
Total U	7.29	7.25	
^{234}U	0.21	0.10	
^{235}U	2.86	2.75	
^{236}U	0.24	0.08	
^{238}U	96.69	97.07	
Element	**MUF-5B**	**MUF-5B Grab Sample**	
B	~0.1	0.62	0.64
Mg	~0.01	0.02	0.02
Al	~0.5	0.55	0.43
Si	~0.16	<0.3	<0.3
K	4 to 5	—	—
Cr	2.1	1.0	0.8
Mn	~0.06	0.1	0.096
Fe	3.9	5.7	5.2
Ni	~0.9	4.9	4.5
Cu	~0.5	0.22	0.23
[illegible]	12.6	5.4	5.7
Mo	~0.08	0.82	0.86
Ag	12.2	11.1	13.0
Cd	—	11.4	11.2
In	4.5	5.7	5.5

a. Activities decay corrected to time of analysis, September 21, 1982.

head gasket. Its approximate location is shown in Figure 1. When the reactor shut down, RCS pressure increased and the PORV opened to relieve pressure. However, the PORV then malfunctioned, stuck open, and discharged water through the designed flow path to the RCDT. Approximately 12 min after reactor trip, RCDT pressure reached 190 psig and a rupture disk in the tank's 45-cm vent line blew out reducing tank pressure to 0 to 4 psig and discharging water to the Reactor Building. Analysis of samples from the Reactor Building surfaces and basement floor have shown measurable levels of fission products, fuel, fuel cladding, and core debris associated with insolubles and solids samples. The only flow path from the core known for this material was the RCDT.

In June 1982, a string of thermoluminescent dosimeter devices (TLDs) was lowered into the RCDT cubicle located on the 282-ft elevation to measure beta and gamma radiation doses at distances of 0.8, 2.3, 3.9, and 5.3 m above the cubicle floor. The TLD at 2.3 m showed the highest beta and gamma dose. Results from TLDs placed in other areas of the basement show a similar effect indicating that the dose readings may be associated with the "bathtub ring" left on the walls after the water was removed from the basement and are not necessarily associated with the RCDT.

Beta and gamma readings in the RCDT cubicle, as shown in Table 6, are comparable to readings in other areas of the basement and show the same trends in dose rate change with elevation. It was expected that large concentrations of solids in the RCDT would be indicated by high gamma dose rates. However, if there are large quantities of solids in the tank, they were not indicated by the cubicle dose profile. It is possible that many of the gamma-emitting fission products may have leached out of the solids over time.

A visual inspection of the external surface of the RCDT and its cubicle, performed in October 1982 with a video camera system, revealed no apparent external damage to the tank, its attached piping systems, or the cubicle. Future planning calls for a visual inspection of the tank interior and samples of the tank liquid and any solids. Internal observations will determine the nature and quantity of solids discharged to the tank from the core, which, in turn, gives insight to core damage and core material dispersal.

Reactor Building and Support Systems. The Reactor Building and Support Systems are one plant environment to which fission products were released from the RCS, primarily through the stuck open PORV into the RCDT, the collection vessel for RCS liquid drainage. The support systems were the principal barrier to fission product release to the outside environment when large quantities of accident water were released to the Reactor Building. The analysis of samples to determine the location, quantity, and radioactive nature of fission products released to each system

Table 6. TLD measurements of Reactor Building basement environment (rad/h)

	Position[a]							
	1		2		3		4	
TLD Location	Beta	Gamma	Beta	Gamma	Beta	Gamma	Beta	Gamma
East wall of refueling canal	0.10	1.13	3.45	3.35	73.62	17.17	107.46	18.51
Beneath basement equipment hatch	2.34	10.48	6.52	25.78	12.03	53.72	74.38	75.65
North area of RCDT room	0.72	6.48	6.31	9.80	339.38	21.02	104.20	16.57
In the cable chase area	0.19	1.29	1.81	2.98	53.75	10.00	96.77	18.58

a. Position 1 denotes a TLD hanging 5.3 m above the basement floor (282-ft 6-in. elevation); Position 2 = 3.9 m above; Position 3 = 2.3 m above; Position 4 = 0.8 m above.

provides information about the radionuclide release deposition and mitigation capabilities within the building environment.

The Air Cooling System circulates air through the Reactor Building. Operating continuously, the system circulates 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 date from the time of 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. Samples from the cooling coils, plenum internal surfaces, and drip pans will be obtained when the air coolers are shut down for maintenance.

Reactor Building Basement Samples—Sampling the approximately 2.5 m 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. Water samples from the basement floor included a solids or sludge fraction that contained greater than 90% of the insoluble fission products and core materials found in all the basement water samples.

Using water level measurements, data from Submerged Demineralizer System (SDS) basement water processing, and measured radionuclide concentrations, it was possible to quantify the dissolved fission products and core materials that were contained in the water. However, the radionuclide and elemental composition of all solid materials on the basement floor could not be generalized from the samples taken in 1982. These samples only reflect the condition of the sludge at the specific sampling locations and do not define the mass and distribution of all the sludge.

After removing and processing the 2.5 m of basement water through the SDS, the first sample of the sludge was taken by direct collection using a small sample scoop. Figure 14 shows the location in the basement sludge from which the scoop sample was taken.

Results of the basement samples characterization analysis performed by WHEDL, Oak Ridge National Laboratory (ORNL), and the INEL are

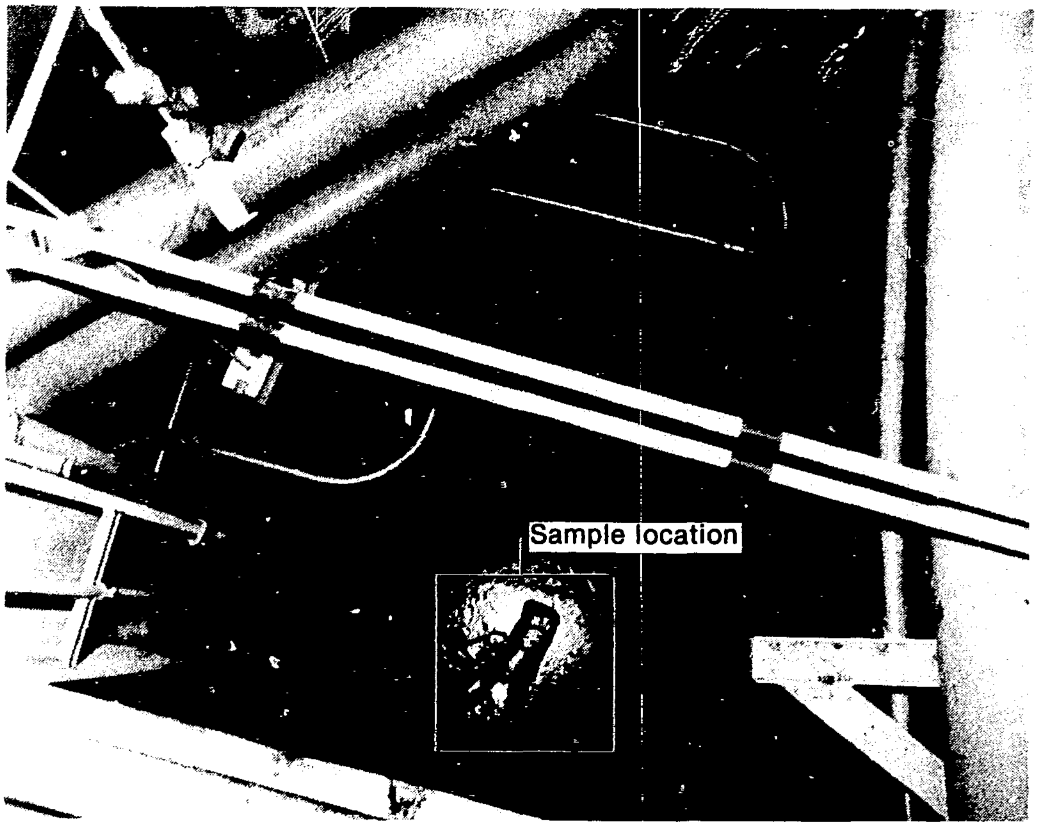

Figure 14. Location on basement floor where technician obtained direct scoop sample of sludge.

presented in Tables 7 and 8. Additional analysis results performed by EG&G Idaho on previously obtained multilevel basement liquid and sludge samples are also presented in the tables for comparison. The data are indicative of the complex chemical and radiochemical nature of the samples, and particularly the solids. In general, the large disparities between the basement samples may be due, in part, to heterogeneous solids, and in part to different sampling, laboratory handling, and analysis techniques employed. Note that the radionuclide concentration for the open stairwell samples vary. For example, ^{137}Cs concentrations shown in Table 7 for September 24, 1981 and June 23, 1982 differ by a factor of six. The variances may imply changes of composition of the sludge samples due to SDS water processing, dilution by water washed down to the basement during the Gross Decontamination Experiment, or dissolution of the sludge material.

Members of a Reactor Building sampling team described the basement sludge as loosely consolidated material which tended to swirl away from the sample scoop. The sample team indicated that the sludge layer is thin, less than 0.6 cm deep. These same observations were later supported by a closed circuit television (CCTV) camera survey of the Reactor Building basement area.

Later in 1982, three additional samples, more representative of sludge throughout the basement, were taken from different basement locations using EG&G Idaho-designed solenoid-operated samplers, such as the one displayed in Figure 15. These samples taken from three different and widely spaced areas of the basement yielded three different kinds of material: a fine red sediment, brown "mudlike" material, and black flakey material. The physical appearances of the basement material suggest solids from several possible sources such as silt, dust, rust, and construction dirt. The fact that some dilution occurred when gross decontamination water washed down to the basement suggests that a part of the solids may be slightly soluble in water. Data from the debris samples indicate the material on the basement floor is heterogeneous. The radionuclide fraction is simply a particulate or insoluble association and not a chemical association with other solids on the basement floor.

Table 7. Gamma spectrometry measurements, uranium, plutonium, and elemental analysis results for Reactor Building basement sample insolubles[a]

	Multilevel Sample: Covered Hatch (May 14, 1981)	Multilevel Sample: Open Stairwell (September 24, 1981)	Open Stairwell Scoop Sample (June 23, 1982)	
	INEL	INEL	ORNL	WHEDL
Total Insolubles (mg/ml)	0.9	0.21	21.6	26.1
Activity (μCi/g)				
^{60}Co	12.1	21	11.1	135.4
^{125}Sb	487	13	177	189
^{134}Cs	107	44	108	267
^{137}Cs	808	327	823	2093
^{144}Ce	66	130	—	139
^{106}Ru	104	75	—	87.5
^{90}Sr	800	2219	2440	5054
^{54}Mn	2.5	<1	—	1.51
^{110m}Ag	7	<4	—	2.9
^{113}Sn	7	<9	—	2.42
U (mg/g)	3.9	0.39	2.97	3.9
Pu (μg/g)	2.9	—	4.41	6.1
^{235}U (atom %)	2.7	<4	2.37	2.4
Composition (ppm)				
Mg	2.0E+3	4.0E+3	2.0E+3	5.0E+3
Al	1.0E+4	5.0E+4	~3.0E+3	3.0E+4
Si	7.0E+4	3.0E+4	2.0E+4	7.0E+3
Ca	2.0E+4	4.0E+4	2.0E+3	3.0E+3
Fe	3.0E+4	1.2E+4	1.0E+4	3.0E+3
Ni	3.0E+4	25	1.0E+4	8.0E+3
Cu	2.2E+5	3	~4.0E+3	2.0E+4
In	1.0E+5	—	3.5E+3	3.0E+2
Zr	—	—	3.0E+3	2.0E+2
Sn	—	—	1.5E+3	2.0E+3
Ag	—	—	9.0E+3	—
Cd	1.0E+3	—	1.0E+4	1.0E+3

a. Activities for all samples corrected to May 14, 1981 for comparison with the Covered Hatch multilevel sample.

Table 8. Gamma spectrometry measurements, uranium, plutonium, and elemental analysis results for Reactor Building basement liquid samples[a]

	Multilevel Sample		Open Stairwell Scoop Sample (June 23, 1982)	
	Covered Hatch (May 14, 1981)	Open Stairwell (September 24, 1981)		
	INEL	INEL	ORNL	WHEDL
Activity (μCi/ml)				
^{134}Cs	19.3	18.3	20.3	20.9
^{137}Cs	144	138	155	163
^{90}Sr	5.3	4.8	7.16	6.0
Fuel (μg/ml)				
U	—[b]	<3E-2	1.6E-2	5E-3
Pu	2.2E-4	—	5.6E-5	<1E-4
Composition (ppm)				
B	2.3E+3	2.3E+3	3.0E+3	8.0E+3
Ca	—	4.1E+1	3.0E+1	2.0E+1
K	1.6E+1	2.0E+1	7.0E+1	2.0E+2
Na	1.2E+3	1.2E+3	~3.0E+3	6.0E+3
Si	3.3E+0	6.8E+0	2.0E+1	9.0E+1
Mg	5.2E+0	7.3E+0	5.0E+0	8.0E+1
Fe	<1.0E-1	7.0E-1	6.0E-1	3.0E+0
Cu	<1.0E+1	<1.0E+2	5.0E+0	5.0E+0
Al	1.2E+0	1.8E+0	3.0E+0	9.0E+0
Ag	<1.0E+0	<1.0E-1	3.0E-1	—
Zr	1.9	1.4	<4.0E-1	—

a. Activities for all samples corrected to May 14, 1981 for comparison with the Covered Hatch multilevel sample.

b. Not detected.

Surface Deposition and Environment—All of 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 to the much cooler building atmosphere, probably as steam initially.

Entrained fission products were released and distributed by a number of mechanisms depending on the chemical nature or molecular and ionic associations of the discrete 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 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 cesium, or core products such as uranium and silver, were carried over 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 surface deposition sampling emphasis has been on characterization of the three most prominant TMI-2 surface types—steel, painted

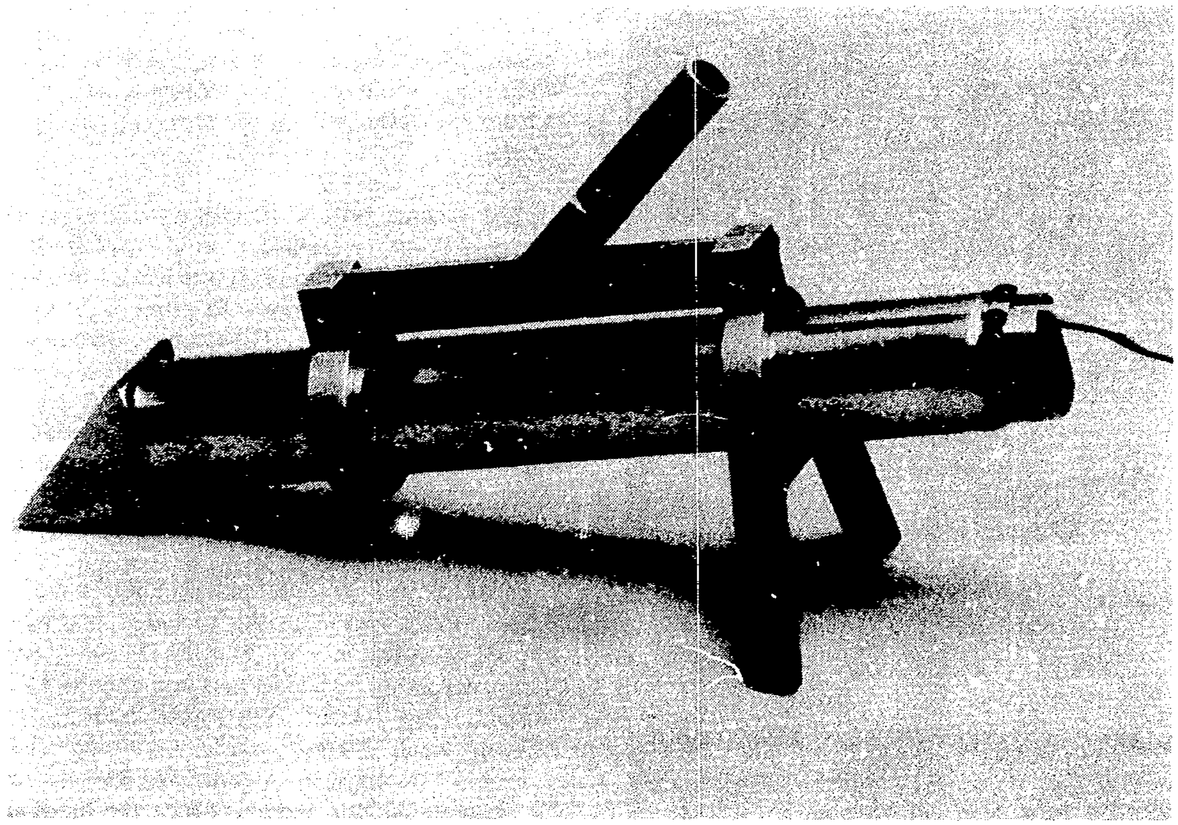

Figure 15. Solenoid-operated sampler used to obtain sludge samples from basement floor.

concrete, and unpainted concrete—to see how deposition occurs on each of the major surfaces. Because of the nature of deposition, some surfaces serve to reduce the spread of radionuclides by chemically or physically interacting with the radionuclides to prevent their release. Other surfaces may increase radionuclide dispersion, and with it the seriousness of the accident, by converting radionuclides to a form that enhances their release potential. Still other surfaces neither reduce nor increase the spreading of radionuclides. Surface deposition studies during 1983 will characterize how deposition occurs on each of the major TMI-2 surfaces.

Surface Deposition—Because of different environmental effects and building area access, the Reactor Building surface deposition characterization is occurring in two regions: 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. This region was measured before and after the Gross Decontamination Experiment at selected sites to quantify and characterize surface deposition, and to measure the effectiveness of the gross decontamination techniques used.

An EG&G Idaho-designed sampler collected loose particulate samples from a surface area and milled subsurface samples from a measured depth. Surfaces sampled were selected to show variations of deposition phenomena on different types of vertical and horizontal surfaces. Sample collection from vertical and horizontal surfaces is shown in Figures 16 and 17, respectively. In addition, special beta and gamma TLDs were used to measure long-term (over a period of hours) radiation fields at the sampling sites. The surface and TLD measurements were supplemented with portable beta and gamma instrument surveys and in situ gamma spectral measurements to provide differentiation between general radiation fields and dose rates attributable to surface contamination. Each measurement technique was chosen to complement and serve as a cross-comparison to the other techniques.

Figure 16. Workers obtain vertical surface deposition sample from Reactor Building wall.

Figure 17. Workers obtain horizontal surface deposition sample from Reactor Building floor.

The average data from surface deposition samples for ^{137}Cs, a soluble isotope, are shown in Table 9. These data display the general trends observed: contamination levels in the building were heterogeneous, contamination levels were comparable for similar measurements on each elevation, and the decontamination experiment was reasonably effective for horizontal surfaces but ineffective for the few vertical surfaces which were decontaminated. Most vertical measurements showed higher levels after decontamination indicating recontamination, probably as a result of water splashing from the spraying techniques used.

The second surface deposition region is the Reactor Building basement, at the 282-ft elevation. The basement was subjected to an evaporation and condensation cycle like that in the upper part of the building. However, contaminated water which covered the floor remained in contact with the basement walls before being processed through the SDS in September 1981. The effect of the standing water and the sludge layer on the floor created special conditions for the study of surface deposition.

Measurement of beta and gamma radiation began in August 1982, prior to water flushing of the 282-ft elevation walls. Four TLD "trees" were fabricated using high level beta and gamma TLDs supplied by Pacific Northwest Laboratories (PNL). Each "tree," containing four TLDs spaced 1.5 m apart on a cord, was lowered into the Reactor Building basement from the 305-ft elevation. The preliminary TLD data (see Table 6) indicate the basement walls, up to approximately 2.3 m, and the floor area are the principal sources of gross beta and gamma radiation. The TLD preliminary radiation profiles of rad/h versus distance from source indicated that the basement walls and the floor are about equal in dose contribution.

When the basement radiation levels are reduced and personnel access is possible, additional surface and subsurface measurements will be made. The degree of radionuclide penetration into the concrete as a result of the standing water is a major area of interest to the recovery project.

Reactor Building Damage—Visual surveys, taken with closed circuit television cameras (CCTV) and reported on by Reactor Building work crews during task debriefing sessions, are helping researchers to provide a graphic record of building damage. In 1982, emphasis on Reactor Building damage has shifted to the 282-ft elevation. Because of the high basement dose rates, a CCTV system was constructed to provide visual information at greatly reduced man-rem exposures. This system is a color CCTV with remotely operated functions for focus, zoom, iris, and pan-tilt operations. In October, the entire

Table 9. Average pre- and postdecontamination surface deposition readings for ^{137}Cs[a]

Sample	Activity ($\mu Ci/cm^2$) Predecontamination	Postdecontamination	Decontamination Factors
305-ft elevation, vertical, 50 mil	1.71 ± 0.81E-1	1.82 + 0.82E-1	0.9
305-ft elevation, horizontal, vacuum	5.78 + 1.81E-2	1.18 + 0.49E-3	49.0
305-ft elevation, horizontal, 10 mil	4.23 + 0.79E-0	1.52 + 0.59E-1	27.8
347-ft elevation, vertical, 50 mil	3.58 + 0.05E-2	8.01 + 2.00E-2	0.5
347-ft elevation, horizontal, vacuum	5.45 + 1.68E-2	1.42 + 0.23E-3	38.4
347-ft elevation, horizontal, 10 mil	3.93 + 1.73E-0	5.29 + 1.60E-1	7.4
367-ft elevation, horizontal, vacuum	1.74 + 0.94E-2	1.31 + 0.46E-2	1.3
367-ft elevation, horizontal, 10 mil	7.56 + 7.14E-1	4.77 + 1.99E-2	15.9

a. Data corrected for collection efficiencies.

system was lowered into the Reactor Building basement. During several entries, the camera surveyed the RCDT, the area below Core Flood Tank A, and the area below the equipment hatch.

The camera was also used to monitor operation of a malfunctioning motor-operated valve located on the sampling line from Steam Generator B. This motor-operated valve must be opened to drain the steam generator, a necessary operation prior to head lift. Past attempts to operate this valve have been unsuccessful, even though valve circuitry in the control room indicates the valve is operating. The special capabilities of the camera system allowed observations, otherwise unavailable, which showed that the motor operator and valve stem were functioning normally. A camera-eye view of the RCDT and sampling line valve are shown in Figure 18. Following the camera inspection, engineers concluded that the pin connecting the valve motor stem to the valve was broken and that the valve must be bypassed in order to drain the steam generator prior to reactor vessel head removal. The best points in the sampling line for cutting and installing the valve bypass were selected using the camera.

These camera surveys showed no visible signs of physical damage resulting from the accident. Some corrosion of carbon steel was observed. All systems appeared intact; however, further quantitative testing may reveal internal damage. The "bathtub rings" left by changes in level of post-accident basement water are evident as shown in Figure 19. Sludge on the basement floor appears evenly distributed, thin, and loosely settled in the small area the camera surveyed.

Reactor Building Gross Decontamination Experiment. The first significant effort to test techniques to reduce contamination levels in the TMI-2 Reactor Building, the Gross Decontamination Experiment, removed radioactive contamination from the vertical and horizontal surfaces on the 305-, 347-, and 367-ft elevations, thus permitting greater access for workers. The decontamination experiment, conducted in March 1982, involved a variety of techniques which could be applied to large areas of the building without physically or chemically affecting the water processing systems for the Reactor Building basement water.

Water sprays, the principal decontamination technique, used accident-generated water previously processed with the EPICOR II ion exchange system. A pressure spray probe called a hydrolaser, shown in Figure 20, was used to apply water sprays to the floor and walls of the Reactor Building. A portable high pressure pump and water heating system located outside the Reactor Building was capable of supplying water heated to 60°C at flow rates up to 25 gpm with pressures up to 10,000 psi.

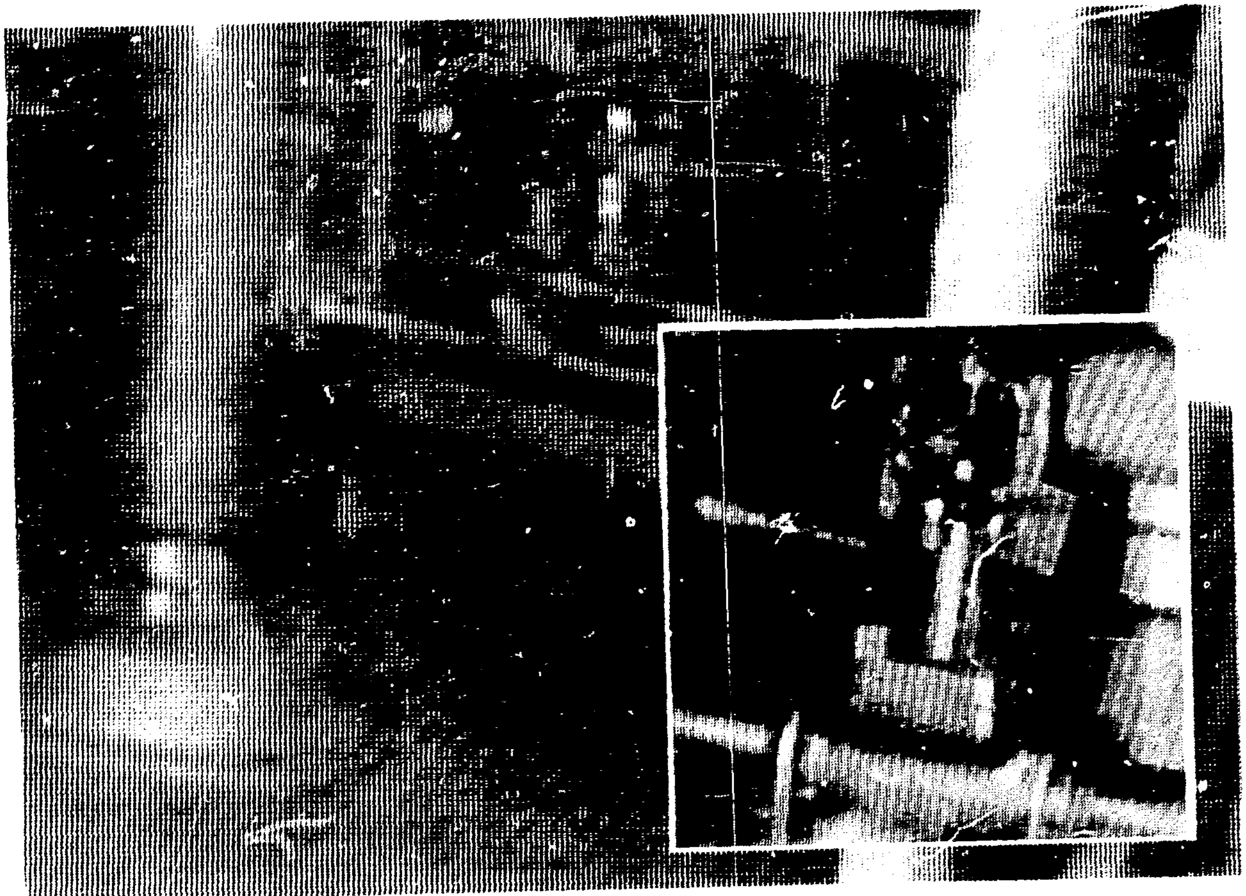

Figure 18. CCTV view of top of RCDT. Insert is a close-up of the steam generator secondary side sampling line valve.

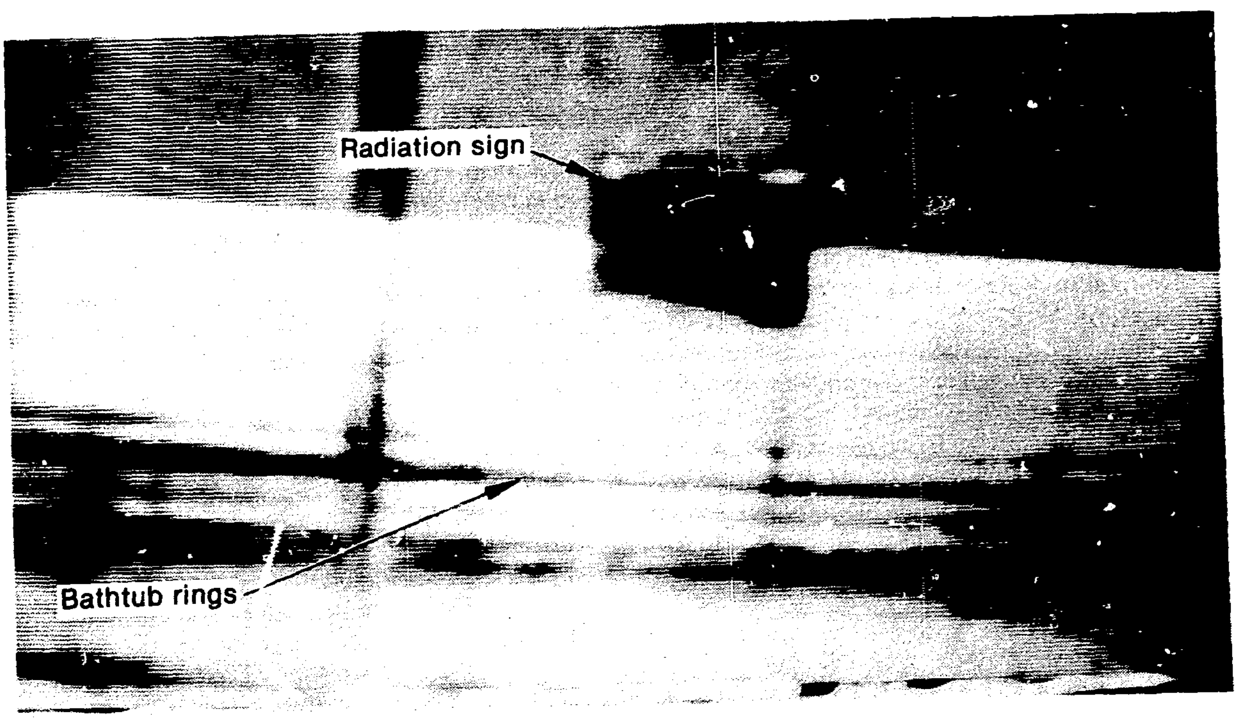

Figure 19. Wall of the Reactor Building basement shows bathtub rings left by changes in water level. Note the deteriorated radiation sign on the wall.

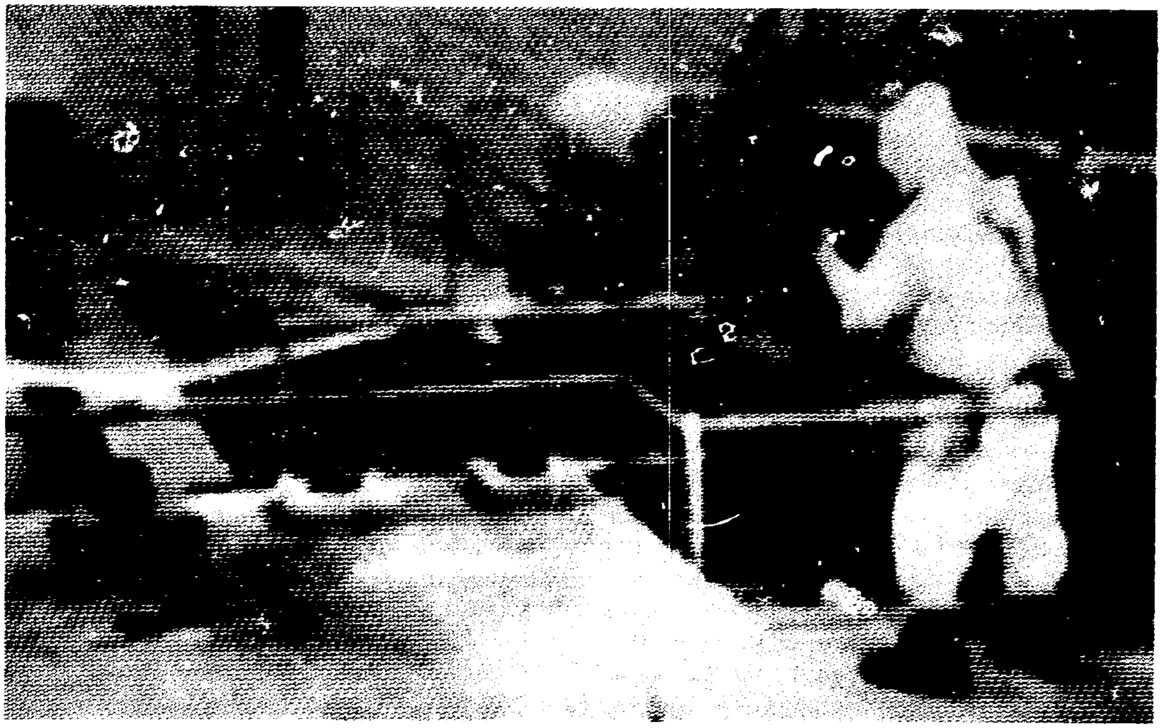

Figure 20. Worker uses hydrolaser to decontaminate surfaces in the Reactor Building during the Gross Decontamination Experiment.

Low pressure spraying, 2000 psi, began on the polar crane, refueling canal, and top of the D-rings to remove bulk quantities of loose contamination. On the walls and floors of the 347- and 305-ft elevations, low pressure sprays were followed by high pressure, 6000 psi, sprays using the same top-to-bottom technique to remove more tightly bound contaminants. The high pressure, high temperature sprays proved to be the most effective decontamination technique; however, these sprays also possessed the largest potential for recontamination of areas by splashing or overspraying areas previously washed. The decontamination spray water drained to the Reactor Building basement and from there it could be reprocessed through the SDS water processing.

Three other decontamination techniques tested in the experiment included spinjet spraying, floor scrubbing, and strippable coatings. Wheel-mounted spinjets with rotating water jets applied even water sprays to the floor before mechanical scrubbers equipped with abrasive pads cleaned the floor areas. Using special wet and dry vacuums, the liquids and detergents used in the floor scrubbing operation were transferred to barrels so that the detergents would not wash into the basement and interfere with the efficiency of the SDS. Strippable silicon coatings removed contaminants from pores, cracks, and otherwise inaccessible areas in the floor. Applied as a thick liquid, the coatings were removed after they had dried.

The Gross Decontamination Experiment was generally successful for removing smearable contaminants. For ^{137}Cs (see Table 9), contamination was reduced more than 90% to levels of about 10^5/dpm 100 cm^2. Because of dose rate variability on different elevations, and contributions from sources and contamination remaining in the basement, overall dose reduction is difficult to assess. The removal of 2.6 m of highly contaminated water from the basement occurred just prior to the experiment. The dose reduction which did occur was a function not only of the experiment but also of the removal of this basement water. Depending upon the source location and the original radiation dose, average rates on the 305-ft elevation were reduced about 26% or by 80 to 120 mR/h. On the 347-ft elevation, dose rates were reduced about 38% or by 40 to 97 mR/h, and near the polar crane, dose rates were reduced about 47% or by 40 mR/h.

Accident Evaluation. The two special accident evaluation tasks, Radionuclide Mass Balance and Hydrogen Burn, are intended to fully evaluate two major occurrences relating to the TMI-2 accident. Mass Balance is an analysis of fission product releases to the plant systems and environment. The Hydrogen Burn task analyzes the extent and nature of core cladding reaction, hydrogen release quantities and pathways and the resulting Reactor Building hydrogen burn. Data from each of these tasks are intended to provide input for calculating current plant regulations, analytical code verification, and possible plant design modifications.

Mass Balance—Performed in two parts this year, the Mass Balance task consisted of a Radionuclide Mass Balance and a complementary Radioiodine Mass Balance. The Radionuclide Mass Balance is a computer based accounting system that currently includes all data through April 1979. When complete, the system will give locations of fuel and fission products at all times after the beginning of reactor core damage. A special Radioiodine Mass Balance determined release quantities, locations, and characteristics of fission product radioiodine releases.

The Radionuclide Mass Balance task is the focal point for all fission product source terms as well as release and deposition data related to the accident. Information on plant operations, accident sequences, recovery operations, and mass transfer of gases, liquids, and solids is being compiled. These data define the relationship between measured fission product source terms, allow quantification of fission product releases and transfers, provide a fission product accident scenario, and will result in an accounting of fission product core inventory. To accomplish these tasks, a computerized data base management system developed by NUS Corporation, the Mass Balance subcontractor, was implemented. Elements of this system are shown in Figure 21 to illustrate the system's required input and capabilities for calculations.

By the end of 1982, 1961 radionuclide concentration measurements from 653 gas and liquid Reactor Building samples had been entered into the data base. Complete sample analysis and 335 mass transfer events have been entered for the period through April 30, 1979. Using the computerized data base manipulation techniques illustrated in Figure 21, a mass balance was calculated for April 30, 1979. Results of the calculated mass balance are shown in Table 10.

The mass balance results illustrate data "holes" for both specific radionuclides and general systems. Some data may never be acquired because measurements were not taken. This is especially true for airborne concentrations of noble gases. Some data may be approximated from other nuclide measurements, or from direct measurements presently outside the time of the data base. In other areas the measurement points do not represent a total mass balance. For instance, the RCBT fraction is for liquid analysis only, and later samples have shown radionuclide levels in RCBT sludge. This type of information will be added when it is available to refine the data base and reduce the margin of error. Use of the mass balance data base in this way provides guidance to the ongoing sampling plans to ensure that a complete set of accident-related fission product transport information is available for a fission product accident analysis.

Certain radionuclide levels, such as those of ^{90}Sr and ^{131}I, are lower than the fractions of others, like radiocesium. Selective deposition in the RCS, reactor upper internals, or basement sludge may have mitigated release of these nuclides which are not yet accounted for in the data base. Continuing efforts will be focused on expanding the analytical concentration measurements and mass transfer chronology (time reference for estimating quantities) data bases to include all of the 1979 data. Efforts will enhance the capability for treating fission product transport within individual reactor plant components, and will develop realistic error bounds on the mass balance radionuclide amounts calculated in all reactor plant locations.

During the accident, fuel failure resulted in the release of radioiodine from the reactor core. Releases of radioiodine are of special concern because they can exist as soluble and volatile chemical species and have the chemical capability to concentrate in the human thyroid. Because of these characteristics, radioiodine is the limiting radionuclide for power plant siting and emergency planning.

During this year radioiodine measurements were made of samples taken from radionuclide repositories called source terms in the RCS, Reactor Building, Auxiliary Building, and the Auxiliary and Fuel Handling Buildings exhaust and ventilation system for comparison with incore shutdown concentrations of radioiodines. The shutdown concentrations were calculated using

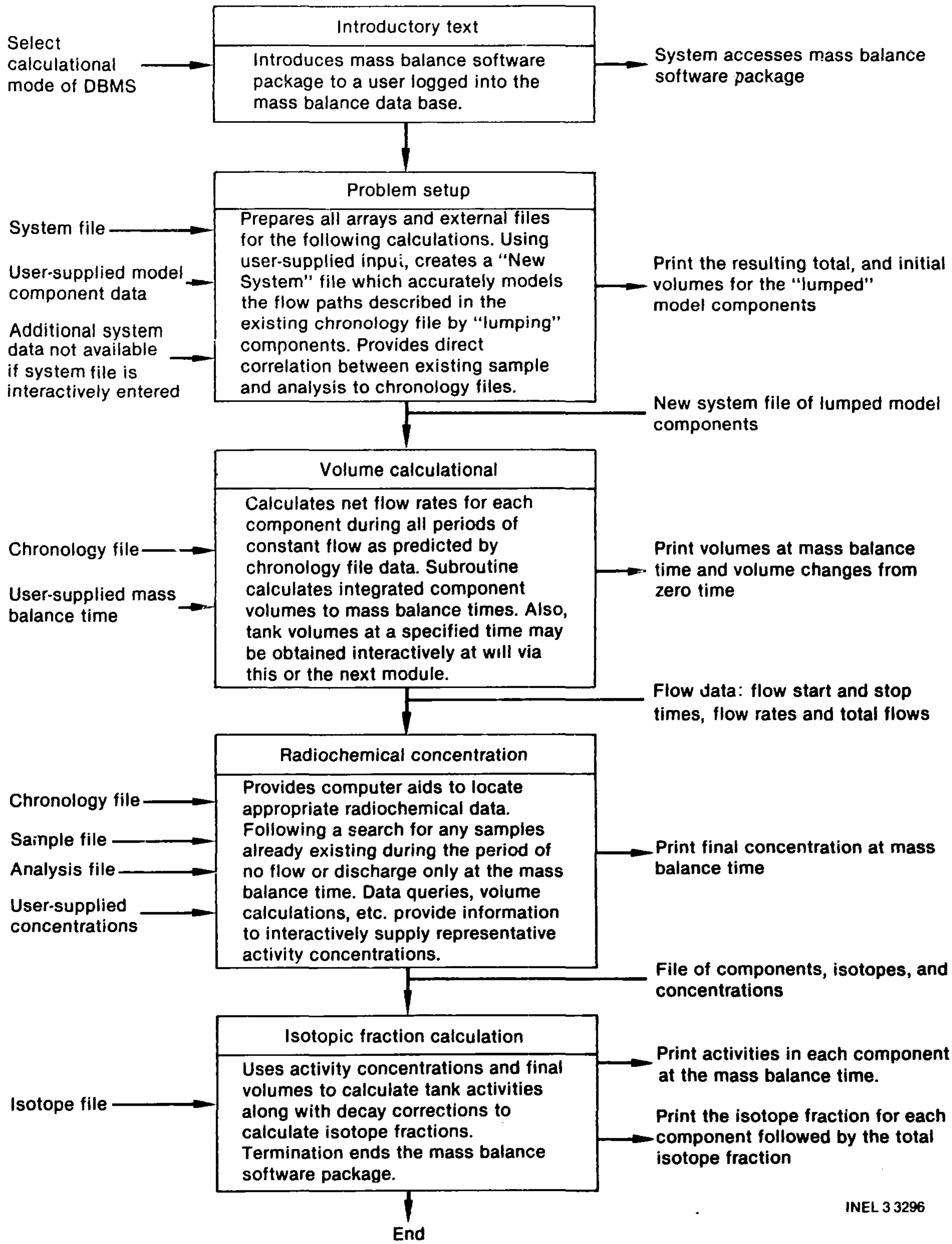

Figure 21. Elements of the computerized data base management system used to calculate the TMI-2 Mass Balance.

Table 10. Preliminary mass balance for April 30, 1979

Location	Percent of Total Isotope Inventory					
	^{89}Sr	^{90}Sr	^{131}I	^{133}Xe	^{134}Cs	^{137}Cs
Liquids						
RCS	0.58	03.0	7.5	—	08.1	7.1
RCDT	0.015	0.010	0.0027	—	0.77	0.60
RB basement	0.88	0.61	16.4	—	46.2	36.0
RCBT	—	—	0.95	—	1.0	0.90
Makeup Tank Sludge	—	—	0.023	—	0.025	0.022
Gases						
RB Air	—	—	0.006	38.4	—	—
RCS	—	—	4.0E-8	0.00054	—	—
Totals	1.5	3.6	25.1	38.4	56.1	44.6

the ORIGEN-2 computer code developed by the Nuclear Safety Analysis Center and the Electric Power Research Institute. All radioiodine measurements have been radioactive-decay corrected to the time of reactor shutdown and are expressed as a fraction of the shutdown inventory of the measured radionuclide. Two isotopes of radioiodine, ^{129}I, with a 1.7×10^7-year half-life and ^{131}I with an 8.04-day half-life, were used to determine radioiodine releases from the core. The easily measured gamma emissions from ^{131}I, available at measurable levels up to five months after the accident, represent the levels of radioiodine during that period. Since then, ^{129}I has been used to measure radioiodine concentrations because of its long half-life.

Table 11. Summary of measured radioiodine inventories on August 28, 1979

Location	Percent of Initial Core Iodine Inventory
Reactor Coolant System liquid	4.0
Reactor Building basement	18.0 to 23.0
Reactor Building atmosphere	0.03
Reactor Building surfaces	0.7
Auxiliary Building liquids	3.3
Total	26 to 31

A summary of results is presented in Table 11. Data currently show that about one-fourth of the core inventory of radioiodine was released. Except for about 3% in the RCBTs, the radioiodine released from the core remained in the RCS and Reactor Building liquids with very little being released to the atmosphere or surfaces. Data showed that approximately 25% of ^{131}I core inventory was released to the Reactor Building and Auxiliary Building over a period of months. In comparison, radiocesium which is also a soluble fission product, showed a release of about 55% of its total core inventory, implying that a fraction of the radioiodine released from the core has not been identified in the release liquids. These data do not include possible source terms such as the Makeup and Purification System filters and demineralizers, RCDT, and Reactor Building basement walls. Data from these source terms, when available, should increase the radioiodine release fraction. If not, the possibility exists that radioiodine is still in the RCS, or the fuel, as an insoluble compound. The Radioiodine Mass Balance information is presently being used as a source document to a draft NRC document, NUREG-0956, relating to pressurized water reactor source terms and release rates.

Hydrogen Burn—During 1982, the hydrogen burn task focused on conditions that existed inside the Reactor Building prior to and during the hydrogen burn, to determine the quantity of hydrogen involved and to analyze and characterize burn damage to organic materials taken from the Reactor Building.

Lawrence Livermore National Laboratory examined wood, paper, plastic, and rubber materials taken from the Reactor Building to determine probable temperature excursions, time of exposure, heat fluxes, and integrated time-at-temperature burn paths, as well as the existence of any localized effects. Reactor Building photographs shown as Figures 22, 23, and 24 display damage resulting from the hydrogen burn. Based on the examination, local differences in flame or thermal radiation exposure damage may be explained in terms of shielding, heat sinks, or selective moisture absorption. The investigation suggests that a high localized H_2 concentration in the enclosed stairwell and elevator shaft could have been responsible for damage in that area.

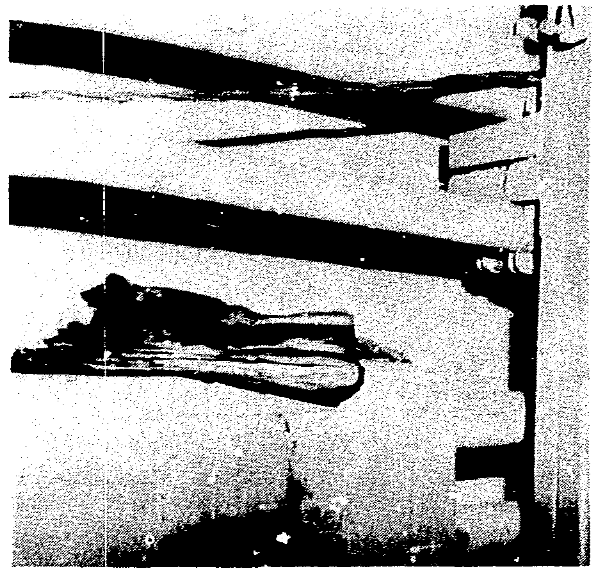

Figure 22. Charred manual lying on top of electrical box.

Figure 23. Softened buttons on the auxiliary fuel handling bridge control panels.

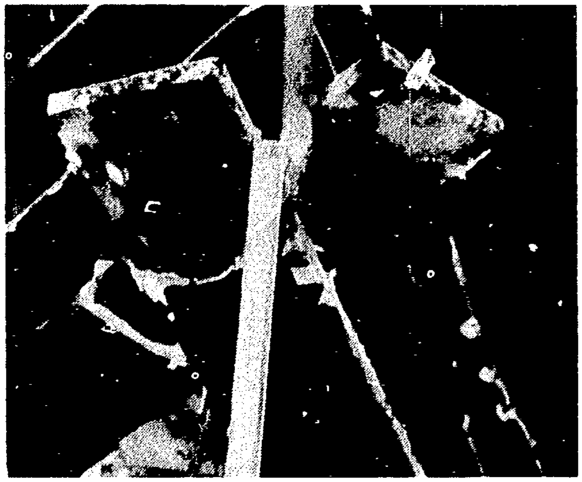

Figure 24. Darkened boards which had been covered with a polyethylene sheet that burned away as a result of the hydrogen burn.

Data from various temperature and pressure detection instruments and gas content analysis of the post-hydrogen-burn Reactor Building atmosphere were examined at the INEL to determine which data were reliable for characterizing the hydrogen burn. The INEL studies determined that the steam generator steam pressure transducers and the Reactor Building Engineered-Safety-Feature pressure switches provided reliable indications of Reactor Building conditions during the burn.

Using the data analyzed at the INEL, Rockwell Hanford Corporation was able to reconstruct the progression of events during the hydrogen burn:

- Prior to the burn, the gases in the Reactor Building were well mixed. The building atmosphere contained 3.5% water vapor and 7.9% hydrogen at a temperature of about 53.3°C.
- The burn apparently started somewhere in the west side of the Reactor Building basement, 282-ft elevation, and spread both horizontally and vertically from the point of origin.
- The burn spread at low velocities, probably less than 10 ft/s vertically and much less than 10 ft/s horizontally.
- The burn lasted about 15 s; however, nearly all the burning occurred during the last 6 s and over half the burn occurred during the last 3 s.
- Temperatures in the flame front were about 760°C. The average Reactor Building gas temperature immediately after the burn was probably near 660°C.
- A pressure pulse of about 29.5 psig resulted in actuation of the Reactor Building spray systems. The spray rapidly cooled the building gases to approximately 230°C within one minute and to approximately 120°C within two minutes after the burn.
- Burn damage was observed primarily in the east and south quadrants on the upper levels of the Reactor Building. The distribution of burn damage resulted from the presence of numerous heat sinks, cool surfaces at lower elevations, and more complete combustion in the upper regions of the building. The lack of burn damage on the west side of the Reactor Building is probably due to condensation of steam being discharged from the RCDT rupture line, which is located near the open stairwell. On the building's north side, the D-rings are relatively close to the building walls presenting a large surface-area-to-gas-volume ratio which resulted in localized rapid cooling.
- About 319 kg of H_2 were burned. About 51.6 kg of hydrogen or 1.1% of the total generated remained unburned in the Reactor Building atmosphere. Approximately 88 kg is calculated to have remained in the RCS. A total of 459 kg of H_2 has been accounted for. Assuming that 90% of the H_2 was generated by a zirconium water reaction, then 9300 kg, which is 45.6% of the core zirconium inventory, reacted.

Mobile Laboratories. Two mobile laboratories were placed on-site to reduce the time and capital expenditures required to analyze samples from TMI-2. These laboratories are staffed and operated by PNL.

The radiochemistry laboratory, located in the Unit 2 Fuel Handling Building, is capable of receiving and handling solids and liquid samples with activities up to 5 R/h. The inside of the laboratory is shown in Figure 25. This laboratory, which has a sample hood, glove box, and analytical chemistry facilities, performs radiochemical

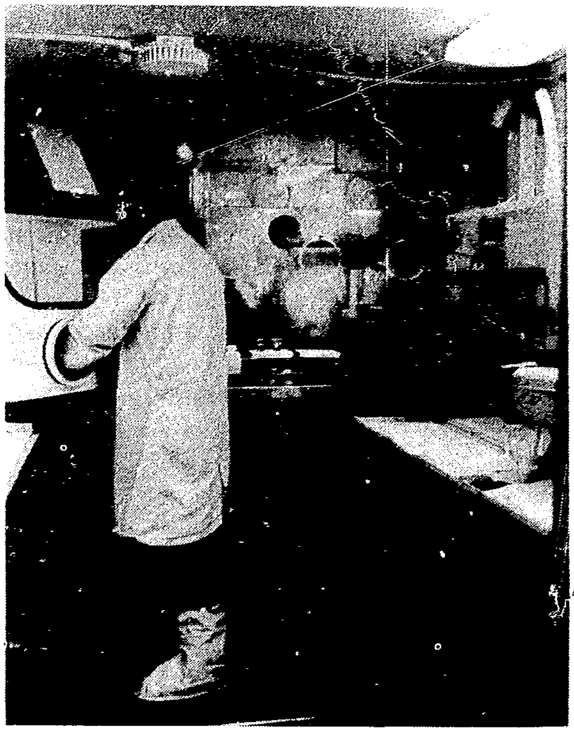

Figure 25. Radiochemist manipulates radioiodine sample in hood in mobile radiochemistry laboratory.

separations and preparation of alpha, beta, gamma, and x-ray emitting radionuclides for quantification by instrumental analysis. In addition, the laboratory performs traditional wet chemical elemental analysis.

The counting laboratory, pictured in Figure 26, is located in the Unit 2 Turbine Building. This laboratory receives alpha, beta, gamma, x-ray, and elemental samples produced by the chemical separations in the radiochemistry lab. This lab handles very small quantities of material with activities of less than 1 mR/h. The samples received are analyzed instrumentally by spectroscopic techniques to produce quantitative analysis data.

The combined capabilities of the two laboratories make it possible to analyze samples of fission products, fuel, transuranics, and elemental core debris, greatly enhancing on-site analytical capabilities. Since installation, the laboratories have performed analysis on RCS water, and on SDS liquid samples from "feed and bleed" decontamination of the RCS. They have also analyzed a 31-cm leadscrew section and performed decontamination studies on scrapings from the leadscrew.

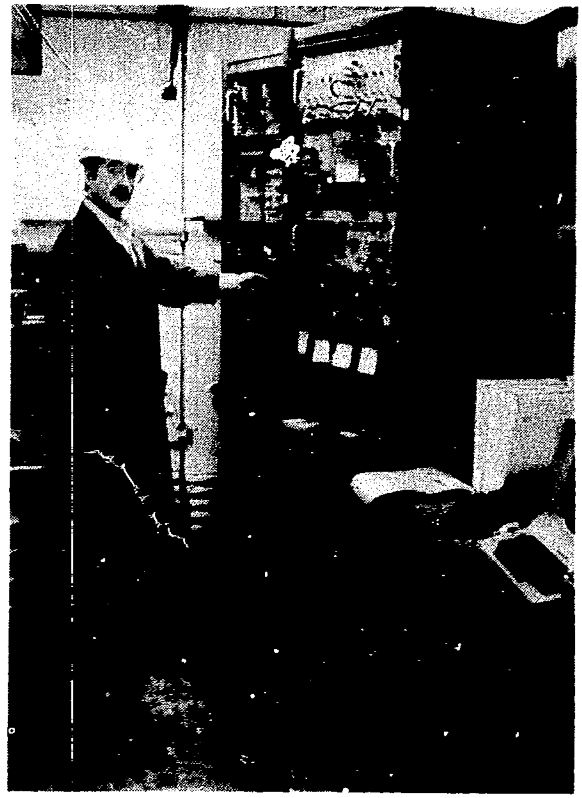

Figure 26. Radiochemist makes adjustment on the multichannel analyzer in the mobile radiocounting laboratory.

Portable Gamma Spectrometer. EG&G Idaho designed and built a portable cart-mounted germanium-lithium (GeLi) gamma spectrometer, shown in Figure 27, for use at TMI-2. The

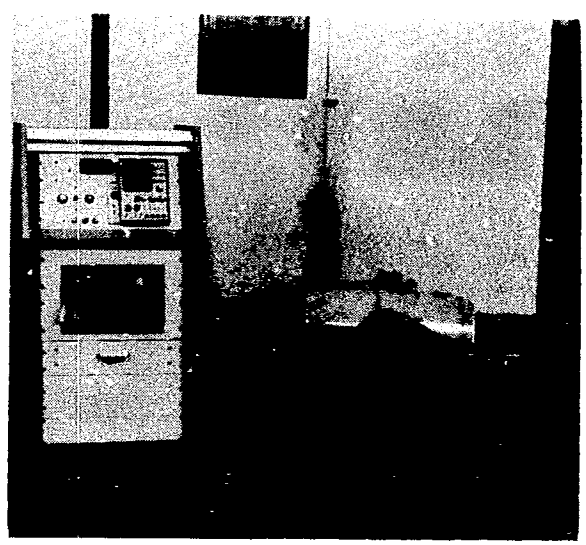

Figure 27. Mobile gamma spectrometer on cart. Control panel on left permits remote operation.

spectrometer will quantitatively characterize the fractions of core mass and fission products transported to system piping, valves, and other components inaccessible to generally routine sampling methods.

This spectrometer has several unique features that reduce man-rem exposures while obtaining extensive gamma spectral information. The system only requires an operator to position it near a source. Once placed, the spectrometer can be operated remotely from a cabinet containing controls for an automated collimator and a tilt motor that allows the shield to be tilted above and below the horizontal plane. A multichannel analyzer in the control cabinet allows remote initiation and termination of gamma spectra which are displayed on a CRT. Accumulated spectra are transferred to a cassette tape through an attached tape drive and deck. The spectrometer will be used in 1983 to scan the RCDT for the presence of core mass and fission products.

CORE ACTIVITIES

The Core Activities program supports analysis, characterization, archiving, and storage of fuel and core debris materials, as well as nonfuel samples, from the TMI-2 accident and cleanup. Analysis contributes to understanding the extent of the TMI-2 accident and its effects on fuel and core components.

The Department of Energy (DOE) selected the Idaho National Engineering Laboratory (INEL) to provide facilities and to manage the archiving, repackaging, and examining of fuel and fuel debris samples from TMI-2. The program will select an appropriate facility at the INEL where program work can be accomplished. The program will also develop procedures and quality assurance requirements for shipping, packaging, handling, and storing those samples.

Accomplishments

During the reporting period, several important documents were prepared and studies completed as part of the planning for removal of the core from TMI-2 and its transport to INEL. These documents, studies, and other program accomplishments are summarized in the following paragraphs.

Planning Documents. A draft *TMI-2 Core Examination Plan* was published in August. The plan was reviewed by DOE and a broad segment of the nuclear power industry. Based on these reviews, a revised TMI-2 Core Examination Plan was prepared and was submitted to DOE for final review and acceptance. The plan presents the objectives of the TMI-2 core examination task by first reviewing the critical nuclear safety issues facing the U.S. light water reactor industry, and then by addressing how the TMI-2 core examination can provide the information required to resolve these issues. The recommended TMI-2 core examinations are divided into three phases:

- Predefueling examinations
- Examinations during defueling
- Off-site examinations.

The first phase, predefueling examinations, is intended to provide early documentation of the postaccident condition of the core. The recommended examinations are compatible with existing reactor recovery tasks and schedules. The second phase, core examinations during reactor defueling, is largely confined to the selection of representative samples of core debris for off-site examinations and analysis. Guidelines for sample selection and recommendations for sample preservation during handling and shipping are also presented. The third examination phase, off-site core examinations, recommends specific hot cell examinations on selected categories of TMI-2 core debris.

The *TMI-2 Core Examination Plan* is intended to be a working document to guide the advanced planning of the examination tasks. The document will be updated periodically to reflect improved knowledge of the detailed condition of the TMI-2 core and examination accomplishments.

Core Topography System. During the reporting period, a TMI-2 Core Topography System (CTS) was designed and fabricated. This system will be used to gather phase one and two data as outlined in the *TMI-2 Core Examination Plan* (summarized above). The CTS consists of a mechanical deployment system and a sensing head equipped with an array of ultrasonic transducers and detectors. The orientation of the CTS in the reactor will be as is shown in Figure 28. The representation of core damage in this figure is largely hypothetical; the CTS will help determine a more accurate assessment of the damage.

The CTS is designed to be mounted on a control rod drive motor tube from which the control rod lead screw has been removed. Using remote control from outside the Reactor Building, the sensing head will then be lowered into the cavity that exists in the damaged TMI-2 reactor core. The transducers and detectors will be energized and rotated 360 degrees. During rotation, the data acquisition system will acquire thousands of data points that represent the distance from the sensing head to the various core damage features. A computer program will then be used to reduce these data and produce a series of graphic images of the core cavity, damaged fuel assemblies hanging from the upper plenum, and other features. These graphic images, accurate to within a few centimeters, include topographic maps, and horizontal "slices" through the core cavity.

The resulting data will be used to help plan reactor defueling and will also generate data on certain technical aspects of severe core damage, such as

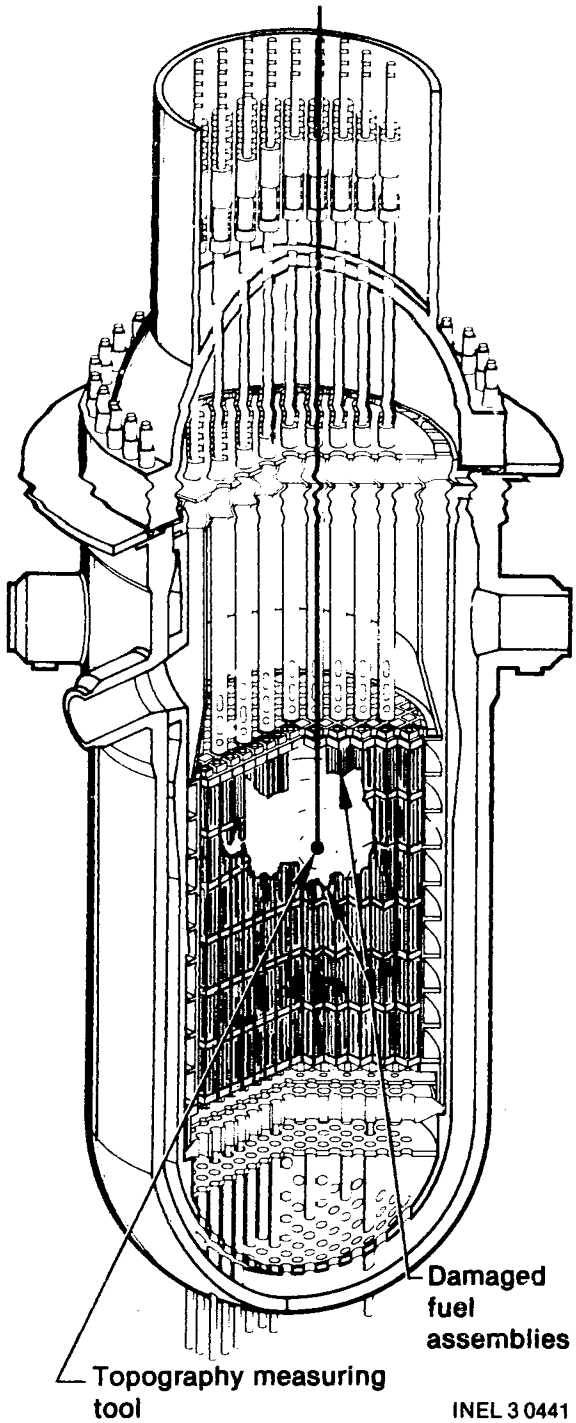

Figure 28. Core topography system measuring tool lowered through a control rod leadscrew opening into the core debris cavity.

damage symmetry. Because the CTS will be remotely operated from outside the Reactor Building, worker radiation exposure will be minimized. The CTS is presently undergoing acceptance testing and is scheduled to be used at TMI in the spring of 1983.

Archive Facilities. Other Core Activities tasks initiated in 1982 included planning for implementation of archive operations and selection of INEL facilities to temporarily archive TMI-2 nonfuel samples and components. This archive will serve as a centralized point that will receive, inspect, document, and store liquid and solid samples and components removed from the TMI-2 reactor plant. The archive will also provide specialized functions such as maintaining, decontaminating, and repackaging (as required) of items, as well as routine retrieval of these items for examination and analysis at the INEL or other laboratories. It is anticipated that the archive will operate for 5 to 10 years, or perhaps longer, as warranted by future programmatic requirements.

Facility selection will be based on the following requirements:

- Secure, safe, and dry storage
- Handling capability for the bulk of nonfuel samples and components
- Access for drum handling equipment
- Radiation and contamination control and accountability measures.

Accommodation of very large or highly contaminated items is not considered a facility requirement. These items will be addressed on a case-by-case basis.

In addition to the archive facility selection, equipment in an INEL hot shop is being refurbished and modified to receive TMI-2 core material and debris presently scheduled to begin arriving in 1985. Some examples of these activities include: reconditioning the 100-ton crane system, cleaning and refinishing the fuel storage pool, cleaning and reconditioning the galley windows, and rebuilding and testing the remote-controlled manipulators. Additional equipment reconditioning is planned during 1983.

INFORMATION AND INDUSTRY COORDINATION

The success of the Technical Information and Examination Program (TI&EP) Data Acquisition Program is largely measured by the usefulness of the information generated. The thrust of the Information and Industry Coordination effort has been to maximize the application of this information. This maximization is being accomplished in two ways.

The first application of TI&EP information is to keep the nuclear industry abreast of program developments. Three computer conferencing services are being used to notify the industry of developments at TMI that have interest for utilities, manufacturers, and architect and engineering firms. These three services are Notepad and the Significant Events Evaluation Information Network (SEE-IN), both managed by the Institute of Nuclear Power Operations (INPO), and Nuclear Operations and Maintenance Information Services (NOMIS), managed by NUS Corporation. The Coordination program has also used presentations to technical societies and industry groups to distribute information on TI&EP accomplishments.

The second way the Coordination program has maximized usefulness of TI&EP data is perhaps more important than the first. The Coordination program provides a channel for feedback from the industry as a result of TI&EP information output. From this feedback, the TI&EP has evaluated how research data are being used and what the real needs of the industry are, and can then tailor the research programs to be responsive to those needs. Technical Evaluation Groups (TEGs), composed of leading members of the nuclear community, present a good portion of this feedback and have made recommendations during periodic reviews of TI&EP work. Presentations have been structured to solicit audience opinions. NOMIS, Notepad, and SEE-IN requests for additional information have given the Coordination program and the TI&EP insight into areas that need more investigation.

Accomplishments

During 1982, the Coordination program provided a mechanism for information exchange in two instances which had direct results on the design of nuclear plant instrumentation. The Loose Parts Monitoring (LPM) System at TMI-2 degraded as a result of extended exposure to high levels of radiation. An effort to inform users and the manufacturers of the problem began in 1981. The Director of Nuclear Products, Rockwell International, wrote that Rockwell's tests had confirmed TI&EP findings that LPMs fail as a result of high exposure. Rockwell reported that a new charge converter that would alleviate the problem had been designed and was being tested.

As reported in the Instrumentation and Electrical section of this report, the area radiation monitors used in the TMI-2 Reactor Building exhibited a multivalued effect when exposed to radiation levels far above their designed operating range. Using Notepad and NOMIS, this fact was brought to the attention of users and of the monitor manufacturer, Victoreen, Inc. Requests for additional information were received from users. Victoreen, Inc. is presently testing an instrument design change in a high radiation environment and has requested that all future inquiries about the detectors be referred to their systems marketing branch.

The Coordination task has responded to requests for presentations on TMI-2 information being gathered under the TI&EP. During 1982, audiences have included the Nuclear Power Engineering Committee, the nuclear arm of the Institute of Electrical and Electronics Engineers; and the Nuclear Power Plant Instrument Engineers Committee, a subdivision of the Instrumentation Society of America. In each case, the questions posed by the audience gave additional insight into the areas of concern to the nuclear industry.

CONFIGURATION AND DOCUMENT CONTROL

Since the Technical Information and Examination Program (TI&EP) was established, the mission of the Configuration Document Control (CDC) section has been to obtain, store, and make available data and information pertinent to the TMI-2 accident and recovery. The CDC maintains the TI&EP data bank, a computerized records storage and retrieval system to which TI&EP participants have access. Requests for information pertaining to the TMI-2 accident, recovery effort, and research and development activities are processed by CDC personnel.

Publications activities are growing concurrently with the growth of the TI&EP. The CDC publications function manages technical and administrative reports as well as presentation materials used by TI&EP personnel. During the reporting period the publications function expanded to include video documentation of research and recovery efforts, and production of DOE-approved video programs. The addition of a videotape library to the TI&EP data bank provides an accurate, timely, easily distributed visual record of pertinent research and recovery activities.

Accomplishments

The TMI-2 accident is the basis for extensive data that can contribute to increases in nuclear power plant safety and reliability. Accident analyses and technological advances associated with recovery of the damaged power plant continue to generate significant data. Accomplishments in CDC operations have facilitated the collection and distribution of that information to interested members of the nuclear power industry.

Document Control. The CDC devoted efforts to enhancing overall document control system performance. At the end of the previous reporting period in December 1981, the data bank contained more than 7000 records. At the end of this reporting period in December 1982, the data bank contained more than 20,000 records.

Data bank operations continue to emphasize simplification and streamlining of the data bank system, reducing costs without degrading data integrity and retrievability. Redesign of the system to enhance routine operations and to facilitate transfer of relevant records to a long-term TMI-2 Recovery data base is planned for 1983.

Publications. GEND reports, the documentation of the DOE research program, were published during the reporting period. Two categories of GEND reports were issued:

- Formal reports—These reports communicate information of lasting importance to the broad technical and scientific community, and generally include conclusions and recommendations based on a completed research project.

- Informal reports—These reports communicate information of a preliminary nature to selected program participants and a small government and industry audience; they generally document only part of an overall project.

By the end of the reporting period, 11 formal reports and 14 informal reports had been published. All are listed in Table 12. DOE contractors may obtain these reports from the DOE Technical Information Center at Oak Ridge National Laboratory, P.O. Box E, Oak Ridge, Tennessee 37830. Members of the general public may obtain them from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161.

Another major publications effort continued to be the TI&EP *Update*, a technical newsletter. The *Update* covers a variety of technical topics. During the reporting period, two issues were published. Other 1982 work included preparation of numerous visual aids used by TI&EP presenters and publication of the *Proceedings* of the TMI-2 Special Sessions at the American Nuclear Society's Winter Annual Meeting. Both publications are available from the Technical Integration Office, EG&G Idaho, Inc., P.O. Box 88, Middletown, Pennsylvania 17057.

CDC video documentation efforts yielded valuable records of a wide range of TI&EP activities, including setup and testing of the Prototype Gas Sampler, shipment of the first Submerged Demineralizer System liner off TMI, and footage of the testing of the Surveillance and Inservice Inspection transporter. Personnel assigned to video documentation also began production of a planned series of video programs that will communicate significant results of TI&EP work to a broad industry audience.

Table 12. TI&EP GEND publications January 1, 1982 to December 30, 1982

Report Number	Title	
GEND-002	Vol. II	Facility Decontamination Technology Workshop—Discussion Groups
GEND-010	Vol. III	In-Vessel Inspection Before Head Removal: TMI-2, Phase III: Tooling and System Design and Verification
GEND-012	The Feasibility of Vitrifying EPICOR II Organic Resins	
GEND-015	Characterization of EPICOR II Prefilter Liner 16	
GEND-019	Examination Results of Three Mile Island Radiation Detector HP-R-213	
GEND-020	Examination Results on TMI-2 LPM Charge Converters YM-AMP-7023 and YM-AMP-7025	
GEND-021	Controlled Air Incinerator Conceptual Design Study	
GEND-022	TMI-2 Information and Examination Program 1981 Annual Report	
GEND-023	A Vitrification Process for the Volume Reduction and Stabilization of Organic Resins	
GEND-024	Zeolite Vitrification Demonstration Program Nonradioactive Process Operations Summary	
GEND-025	Zeolite Vitrification Demonstration Program Characterization of Nonradioactive Demonstration Product	
GEND-INF-009	Pre-Decontamination Gamma-Ray Surface Scans in TMI-2 Containment Building 305-ft Elevation	
GEND-INF-011	Vol. II	Reactor Building Basement Radionuclide Distribution Studies
GEND-INF-017	Field Measurements and Interpretation of TMI-2 Instrumentation	
	Vol. III	HP-R-211
	Vol. IV	CF-2-LT4
	Vol. V	CF-2-LT2
	Vol. VI	IC-10-dPT
	Vol. VII	YM-AMP-7023 and YM-AMP-7025
	Vol. VIII	HP-R-212
	Vol. IX	HP-R-213

Table 12. (continued)

Report Number	Title	
	Vol. X	HP-R-214
	Vol. XI	NI-AMP-2
	Vol. XII	Recommendations for TMI-2 Instrumentation Surveillance Program
GEND-INF-019	Estimated Source Terms for Radionuclides and Suspended Particulates During TMI-2 Defueling Operations Report on Phase I	
GEND-INF-021	Analysis Data on Samples from the TMI-2 Reactor Coolant System and the Reactor Coolant Bleed Tank	
GEND-INF-022	Status of TMI-2 Instruments and Electrical Components	
GEND-INF-023	Vol. I	Investigation of Hydrogen Burn Damage in the Three Mile Island Unit 2 Reactor Building
GEND-INF-024	Review of TMI-2 Resistance Temperature Detectors Accident Data and In Situ Testing	
GEND-INF-025	Development of a Prototype Gas Sampler for EPICOR II Prefilter Liners	
GEND-INF-026	Static In Situ Test of the Axial Power Shaping Rod and Shim Safety Control Rod Mechanisms	
GEND-INF-028	Development of In Situ Test Procedures for TMI-2 Axial Power Shaping Rod Drive Mechanisms	
GEND-INF-031	Preliminary Report on TMI-2 Incore Instrument Damage	
GEND-INF-034	Testing and Examination of TMI-2 Electrical Components and Discrete Devices	
GEND-INF-036	Task Plan for the U.S. Department of Energy TMI-2 Programs	

WASTE IMMOBILIZATION

Radioactive waste handling at Three Mile Island has required adaptation and development of advanced waste processing technology to surmount the problems associated with the special wastes generated as a result of the March 28, 1979 accident at the Unit 2 nuclear plant. Commercial industry practices for normal radioactive waste handling proved inadequate when applied to the large quantity of high specific activity waste generated at TMI-2.

In March 1982, the Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) reached an agreement about handling TMI-2 radioactive waste. The Memorandum of Understanding signed by the two agencies states that DOE will use special TMI-2 wastes in research and development (R&D) work with benefits applicable to the nuclear industry. Under the terms of the agreement, DOE performed R&D which facilitated safe waste shipment and waste characterization during 1982.

The DOE Technical Information and Examination Program (TI&EP) Waste Immobilization Program includes three major areas of interest. These three are the Zeolite Disposition program for shipping and examining ways to dispose of Submerged Demineralizer System zeolite ion exchange beds, the Abnormal Waste Technology program for characterizing special wastes at TMI and developing methods for removing them, and the EPICOR II Disposition program for characterizing and examining ways to dispose of resins from the EPICOR II water processing system.

Accomplishments

Planning and design work initiated in 1981 saw much of its fulfillment during 1982. Significant accomplishments include shipping the first Submerged Demineralizer System (SDS) waste liners off the Island after finding a way to successfully control generation of gases and shipping 17 EPICOR waste liners off the Island after implementing a specially designed gas sampling system to ensure safe transport. The TI&EP also directed DOE laboratories in characterization of the purification demineralizers and in experiments to vitrify radioactive waste.

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 General Public Utilities Nuclear Corporation (GPUNC) with technical assistance from TI&EP-coordinated DOE laboratory personnel. The SDS uses inorganic zeolites to remove radionuclides from the contaminated water and concentrates them in a form suitable for safe shipment and disposition.

SDS operation began in 1981 and continued throughout 1982. By the end of 1982, the SDS had processed over 5000 m^3 of contaminated water from the Reactor Coolant Bleed Tanks (RCBTs), Reactor Building basement (RBB), and Reactor Coolant System (RCS). Table 13 shows the water volumes processed by SDS for each of these water sources. RBB water includes both accident-generated water and water added to the basement from the Reactor Building Gross Decontamination Experiment.

Table 13. Water volumes processed by SDS

Water Source	Volume Processed (m^3)
Reactor Coolant Bleed Tanks	1142
Reactor Building Basement	2827
Reactor Coolant System	1132
Total	5101

SDS processing resulted in generation of expended zeolite ion exchange media liners. SDS liners, one of which is pictured in Figure 29, are 1.4 m high and 0.6 m in diameter and contain approximately 0.23 m^3 of zeolite. The zeolite material in the liners is a homogeneous mix of two zeolites, Linde IE-96 and A-51, products of the Union Carbide Corp, Linde Division. The 3:2 mix ratio, IE-96 to A-51, for these two zeolites was established in 1981 in DOE-sponsored testing at the Oak Ridge National Laboratory (ORNL). Tests showed that IE-96 and A-51 have superior selectivity for cesium and strontium ions, the primary contaminants in TMI-2 water, and are effective in water with high sodium ion concentrations, which is also a characteristic of TMI-2 water.

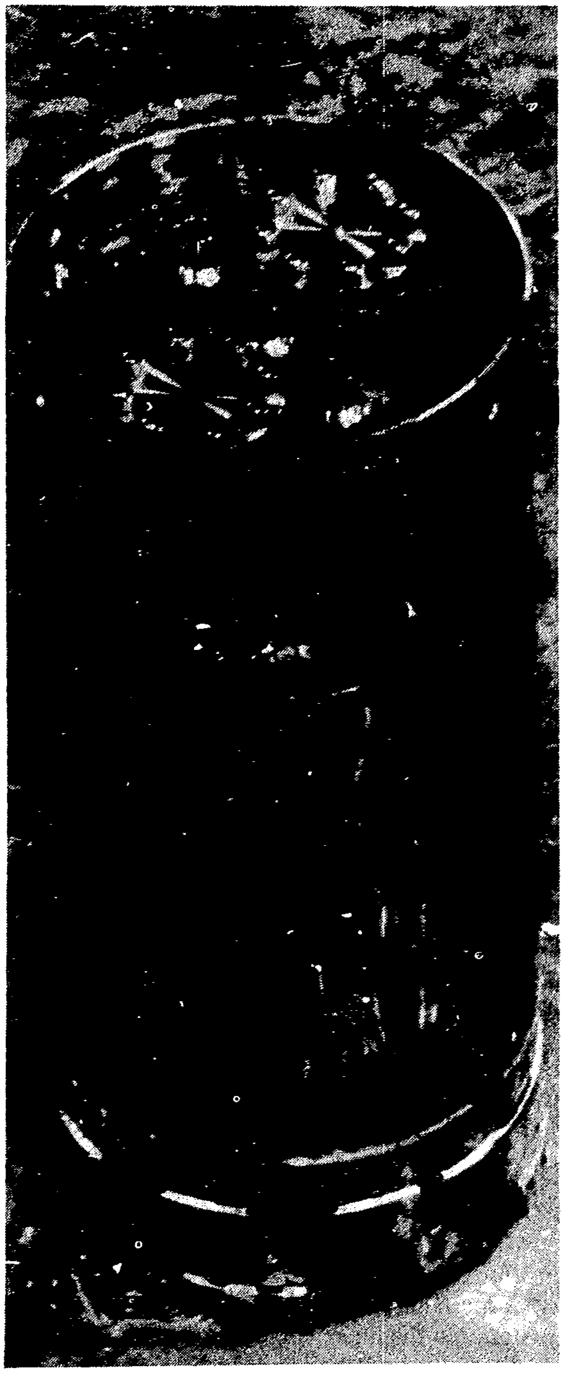

Figure 29. Empty SDS liner awaiting use in the water processing system at TMI-2.

Expended zeolite liners will be used in Waste Immobilization research and development programs. Table 14 lists expended SDS zeolite liners and the radionuclide content estimated for each.

Preparations to Ship SDS Liners—While GPUNC and the TI&EP were preparing SDS liners for shipment, technicians determined that the highly loaded zeolite vessels were generating hydrogen and oxygen radiolytic gases at a rate which would produce unsafe concentrations in the liners during shipment. The TI&EP called in technical experts to develop a solution to the problem. Because the gas generation rate depended on the residual water content in the liner as well as the curie loading, a vacuum outgassing system was developed to remove water from the liners, thereby reducing the radiolytic gas generation rate. The radiolytic gases generated by whatever water was left in the liners after outgassing could then be recombined into water using a catalyst. The use of the two techniques of vacuum outgassing and catalytic recombination proved an effective solution to the gas generation problem.

Westinghouse Hanford Company developed the vacuum outgassing system for the two-step process. Aided by the effect of radioactive decay heat in the liners on vapor pressure, the vacuum outgassing process removes water from the zeolite beds by reducing the vessel pressure below the vapor pressure of water. The residual water then boils off at room temperature.

Rockwell Hanford Operations (RHO) conducted tests which clearly demonstrated that platinum-palladium coated alumina oxide catalyst pellets would safely recombine the gases. Hydrogen and oxygen recombined to H_2O under test conditions using gas generation rates twice those expected from the highest gas-production-rate liner at TMI-2.

RHO also developed the tool shown in Figure 30 which allows operators to first vacuum outgas and then add pellets to the liners at TMI. After successful testing at RHO the vacuum outgassing-catalyst recombiner system was installed at TMI. In a demonstration on a nonradioactive liner, vacuum outgassing removed 4.5 kg of water per day from the liner and catalyst pellets were successfully added to the liner.

Table 14. **Expended SDS vessel curie loadings as of September 1982**

Vessel Number	Water Source[a]	Cs (134,137)	Sr (90)	Total Cs + Sr	Total w/Daughters
D10011	RBB	44,317	2,061	46,378	88,158
D10012	RBB	57,176	2,003	59,179	112,635
D10013	RBB	49,281	1,974	51,255	97,151
D10015	RCBT	5,767	1,012	6,779	12,896
D10016	RBB	57,156	1,869	59,025	112,622
D10017	RBB	30,312	1,021	31,333	59,542
D20027	RBB/RCS	4,289	5,096	9,385	18,380
D20028	RBB	43,333	1,660	44,993	86,334

a. Reactor Building Basement (RBB), Reactor Coolant Bleed Tanks (RCBT), and Reactor Coolant System (RCS).

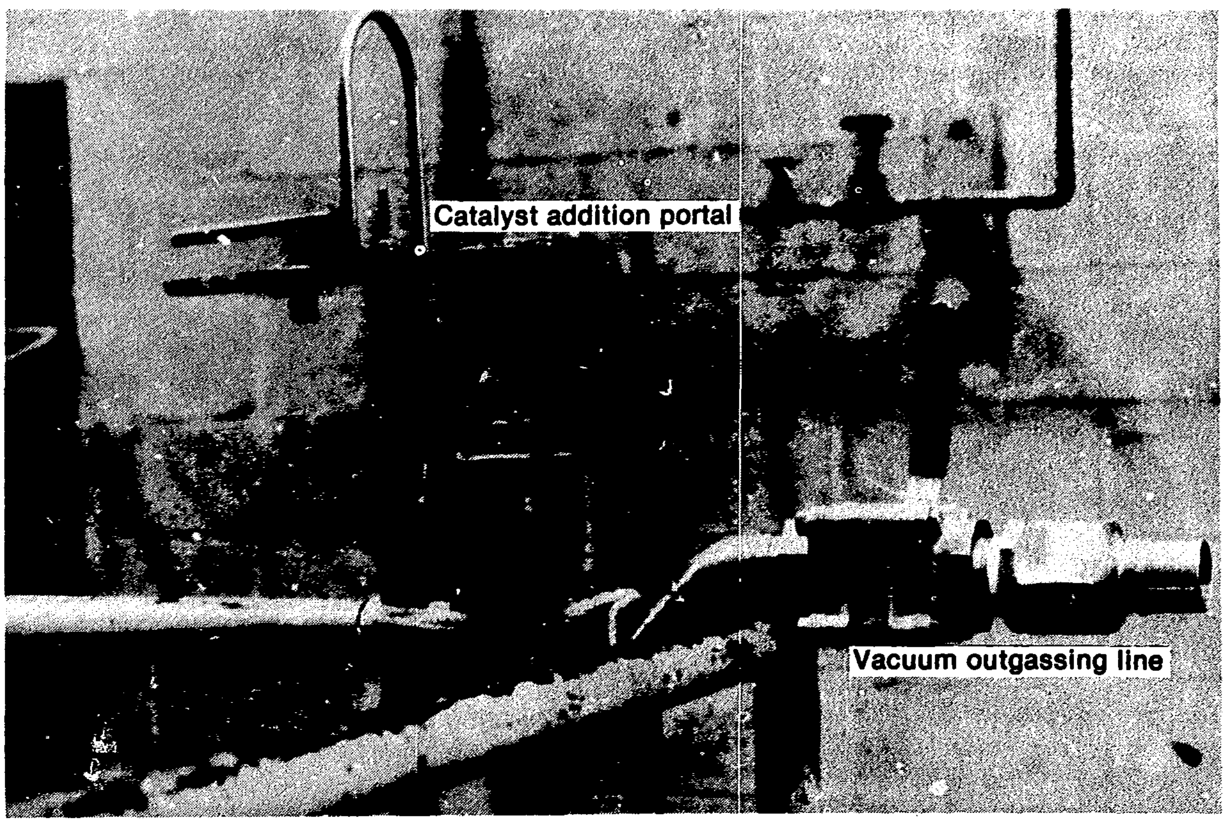

Figure 30. Catalyst addition tool and vacuum outgassing line which are part of the system which controlled radiolytic gas generation in SDS liners.

A demonstration on the highest-loaded radioactive liner completed the test program. Liner D10012 had a loading of 112,635 Ci. A total of 71 kg of water was removed from the liner during the demonstration. Figure 31 shows D10012 underwater with the vacuum outgassing dewatering tool attached.

Catalyst pellets were added to the liner using the catalyst addition tool, as shown in Figure 32. The pellets entered the liner through the liner vent port and collected in a filter called the Johnson screen assembly, positioned above the zeolite media as shown in a cutaway view of an SDS liner in Figure 33. The catalyst successfully recombined radiolytic gases during the two-week monitoring period.

At the end of the demonstration, GPUNC and the TI&EP concluded that catalytic recombination of radiolytically generated hydrogen and oxygen in SDS liners after vacuum outgassing is an effective method of liner preparation for safe shipment. SDS liner D10012 was safely shipped to Pacific Northwest Laboratories (PNL) for research and development work.

Zeolite Vitrification Demonstration Program—On May 21, 1982 DOE shipped the first radioactivity-bearing ion exchange media liner, number D10015, from the SDS at TMI-2. The liner went to PNL at Richland, Washington for testing and disposition research.

At PNL, the liner, loaded with approximately 13,000 Ci of radioactive fission products, underwent studies 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 liner zeolites were vitrified and preliminary results indicate that the radioactive zeolites are being successfully contained in the glass log.

A higher-activity liner, D10012, shipped in December with catalyst recombiners added, and another liner to be shipped in 1983 will also

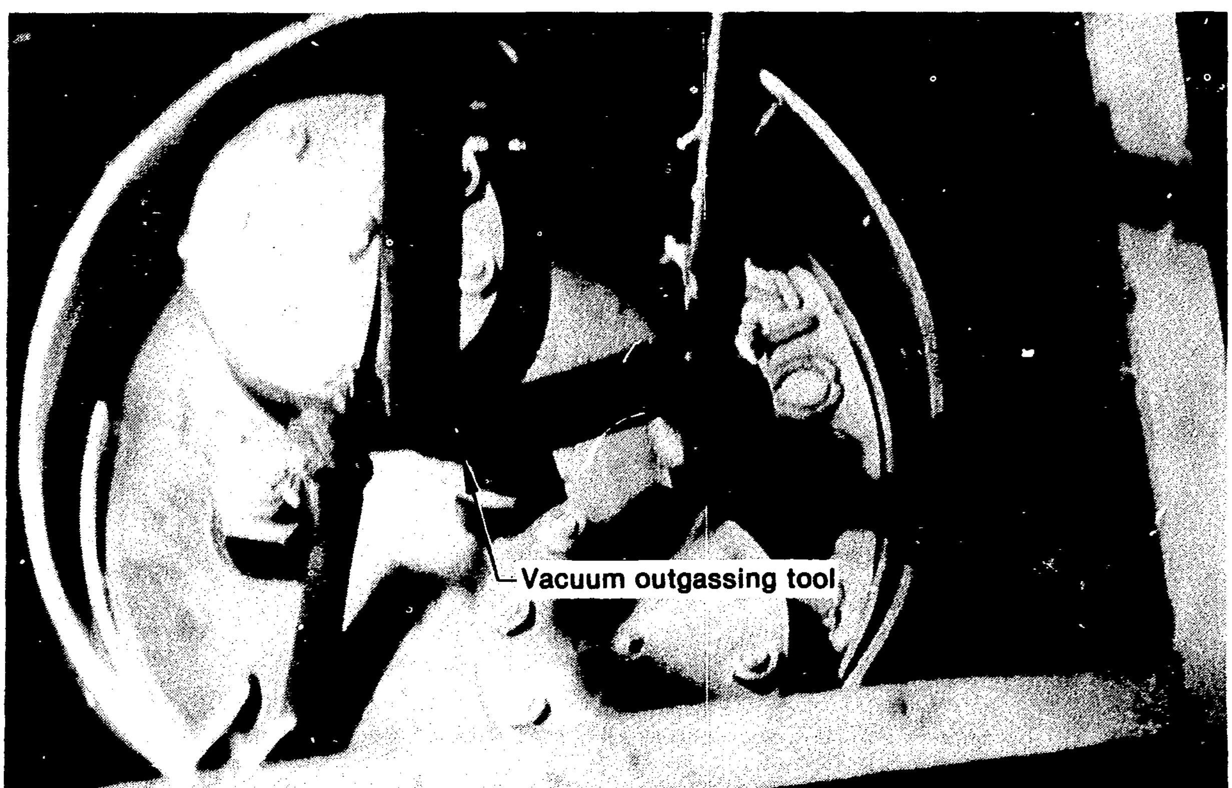

Figure 31. SDS liner D10012 underwater with the vacuum outgassing dewatering tool attached.

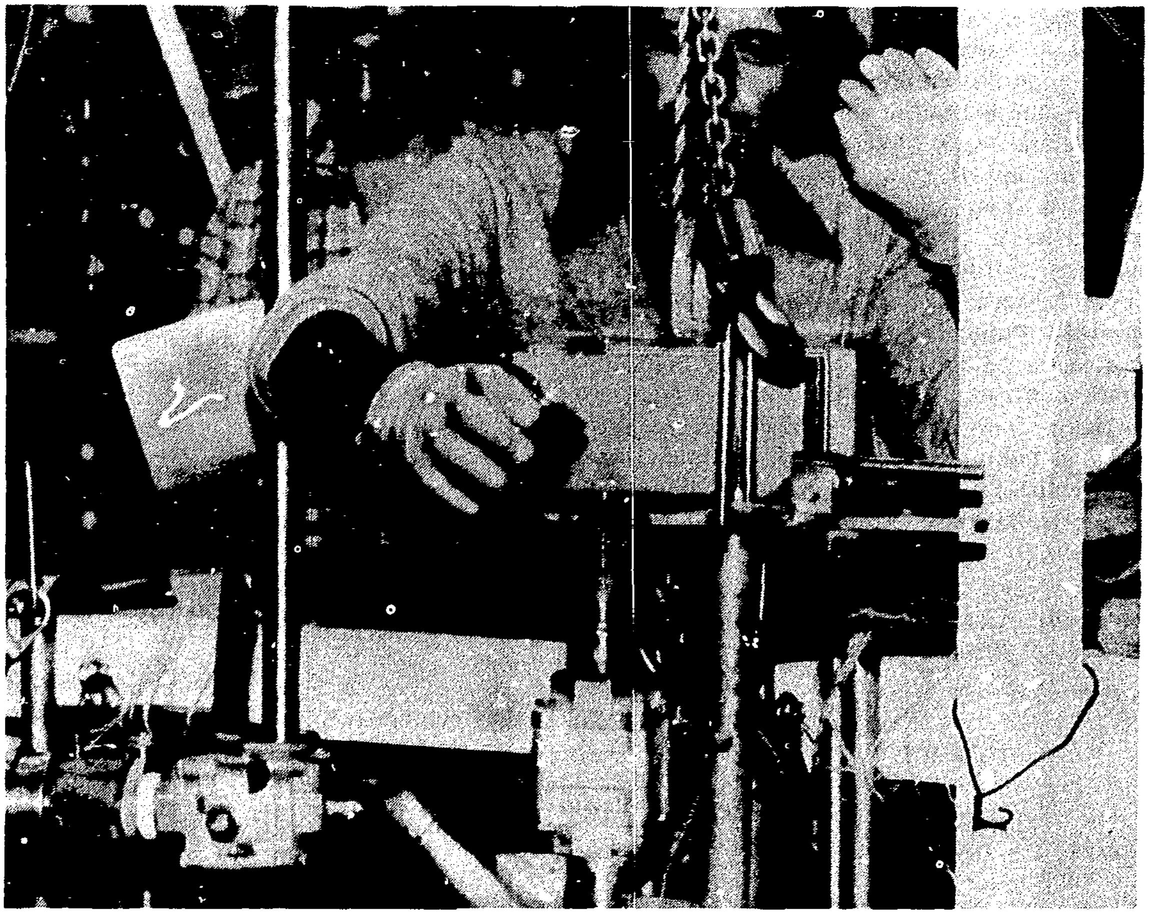

Figure 32. Technician adds catalyst pellets to SDS liner through catalyst addition portal.

undergo vitrification in 1983. The three radioactive demonstrations will further establish vitrification's technical feasibility as a disposition option for TMI's highly loaded radioactive wastes.

Characterization of Liner D10015—Liner D10015 was a relatively low-level radioactive shipment compared to other SDS liners (see Table 14). The zeolites in D10015 were loaded to 13,000 Ci of cesium, strontium, and daughters from having processed 720 m^3 of water from the Reactor Coolant Bleed Tanks. At PNL zeolites from liner D10015 underwent a series of tests to fully characterize them and determine the ratio of zeolites to glass formers needed for vitrification.

During unloading shortly after arrival at PNL, the gas content of the liner was sampled. The results, shown in Table 15, indicate that gas levels were safely below limits established to prevent potential combustible gas mixtures. Technicians then removed about 700 ml of free water from the liner's interior.

The characterization studies began with a liner gamma scan to obtain a rough estimate of radionuclide distribution. The scan showed that the cesium was concentrated in the top 0.23 m c. the zeolite. This was expected, since, during processing, the contaminated water enters the top of the liner and flows down through the zeolite bed. The scan did indicate that some cesium had penetrated the zeolite bed to a depth of 0.5 m.

The primary data base for characterizing the zeolites was analysis of core samples extracted from the media bed. A 3-cm diameter core sample of the zeolite bed was taken to determine water content and cesium, strontium, and transuranic (TRU) isotope distributions. Table 16 shows results of core sample analyses for water content by weight percentage, and cesium, strontium, and TRU isotope distributions at each of four intervals. Cesium and strontium concentrations were indeed higher near the top of the liner as indicated by the preliminary gamma scan, and were also distributed from top to bottom in a manner expected of a downflow system. The cesium loading is distributed with a factor of 400 difference between the top and bottom intervals, while the strontium loading is more uniform, having a factor of only 4 difference. This uniform strontium distribution was expected: liner D10015 had been removed from service because the zeolites had absorbed their limit of strontium ions as indicated by the presence of strontium in downstream samples.

Table 15. Gas sampling results of SDS liner D10015

Sample	Volume Percent: Upon Sealing the Liner Prior to Shipment (May 17, 1982)	Volume Percent: Upon Completion of Shipment (May 26, 1982)
H	0	5.6
N	97	93.8
O	0	0.05
CO_2	Not analyzed	0.05
Gas Pressure	2 psig	3.7 psig

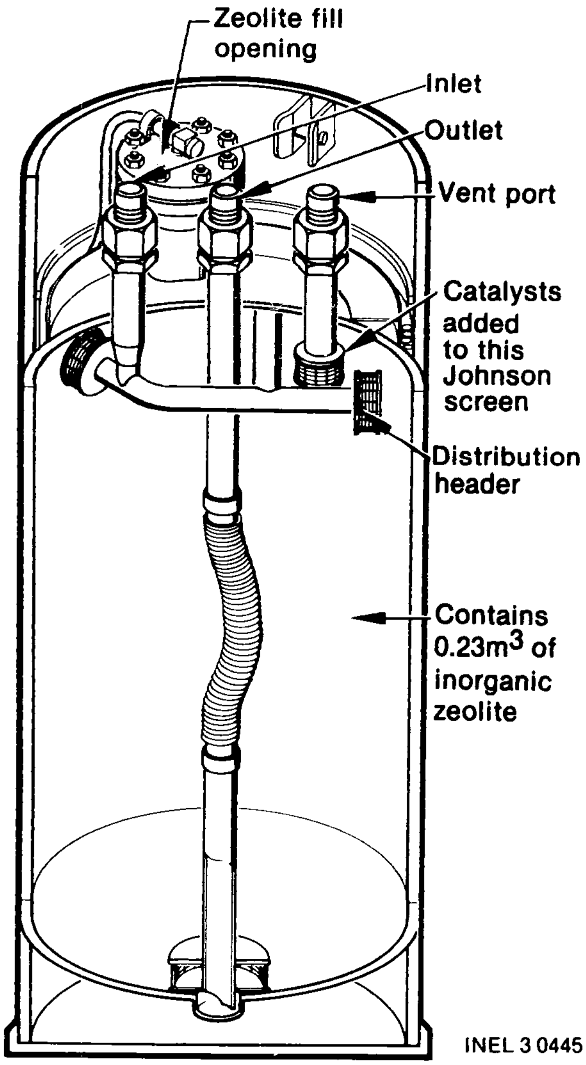

Figure 33. Cutaway view of an SDS liner showing Johnson Screen to which catalysts are added.

Table 16. Cesium, strontium, TRU, and H_2O distributions in SDS liner D10015[a]

Sample Interval From Top of Zeolite (cm)	^{134}Cs (mg/g)	^{137}Cs (mg/g)	^{90}Sr (mg/g)	TRU (Pu) (mg/g)	H_2O (wt%)
0 to 5	3.8E-3	0.6E-0	4.6E-2	1E-4	33.6
5 to 20	3.0E-3	0.5E-0	4.4E-2	1E-4	33.2
20 to 45	4.1E-4	6.4E-2	2.6E-2	8E-5	34.2
45 to 63	8.7E-6	1.5E-3	1.3E-2	2E-4	37.8

a. Values shown are averages for each interval, as the core samples were thoroughly mixed prior to analysis.

Vitrification Demonstration—Following careful analysis of zeolite-to-glass-former ratios, PNL proceeded with vitrification of the liner contents. Over a 44-hour period, a total of 210 kg of feed (zeolite from liner D10015 plus glass formers) were fed to the in-can melter system pictured in Figure 34. This operation resulted in a glass column about 2.1 m tall, weighing 177 kg, in an 0.2-m diameter canister.

During vitrification, a gamma-ray detection system was used to monitor the glass level in the canister. The cesium loading of liner D10015 was high enough for the gamma-ray detection system to monitor the level of the glass in the container and also to diagnose other process-related conditions such as foaming and zeolite buildup in the canister.

Vitrification of liner zeolites 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, and subjected to leach rate tests. The results of these tests are shown in Table 17 along with the results that were previously obtained from a nonradioactive vitrified zeolite glass log. The leach rates show that the glass is successfully trapping the contaminants; the rates are comparable with existing standards for vitrified nuclear wastes. These leach rate test results will serve as points of comparison for future leach rate tests on other vitrified glass logs, in order to evaluate vitrification as an effective long-term means for waste storage.

Table 17. Low activity and nonradioactive glass leach test results (g/cm^2/day)

	Low Activity Glass	Nonradioactive Glass
Top	4.6E-5	3.0E-5
Middle	4.6E-5	3.9E-5
Bottom	1.3E-4	3.6E-5

The canister of glass was gamma-scanned after removal from the furnace to check for uniformity of the cesium in the vitrified zeolite. A 15% decrease in activity was observed while scanning from the bottom to the top of the canister. This decrease, at least in part, is due to the decreased density of the glass in the upper portions of the canister. (Density measurements on the glass taken at the core sample locations indicate that the bottom and middle samples had a density of 2.5 gm/cm^3, while the top sample had a density of 2.3 g/cm^3.)

The process components were also analyzed to determine how well they performed during vitrification. Analytical results of the off-gas system samples indicate that cesium did not get past the sintered metal filters during processing of the low-activity liner. Measurements of the canister wall thickness indicated a slight decrease of about 0.05 cm during the vitrification process, mostly due to oxidation of the canister surface. Other measurements, including such things as canister roundness, indicate very little, if any, change. Overall, the canister appears to have maintained its integrity, as predicted.

Abnormal Waste Technology. The current objectives of the Waste Immobilization Program at TMI reflect the Memorandum of Understanding between the NRC and the DOE dated March 15, 1982. Under the terms of the understanding, DOE will perform R&D work on waste from TMI-2 which is categorized as Abnormal Waste.

Categories of Abnormal Waste—In 1982, DOE's TI&EP initiated an effort to provide technical support to GPUNC in removal, packaging, and shipment of abnormal waste from TMI. The term abnormal waste designates that waste which is not routinely generated at nuclear power plants and for which GPUNC does not have a commercial disposal option. At TMI-2, this abnormal waste includes wastes in filters, tanks, equipment, and sumps suspected of containing greater than 10 nCi/g of TRU isotopes, the current concentration limit for commercial shallow land burial. The DOE TI&EP informed all DOE offices about the special abnormal waste and requested that the field offices examine their ongoing waste management programs to see if they might benefit from including TMI-2 waste in their studies.

GPUNC categorized the abnormal waste at the TMI-2 plant to determine which will be acceptable for commercial low level waste disposal and which will be candidates for acceptance by DOE as abnormal waste. DOE acceptance of abnormal waste for R&D or reimbursable disposition will require characterization activities, including

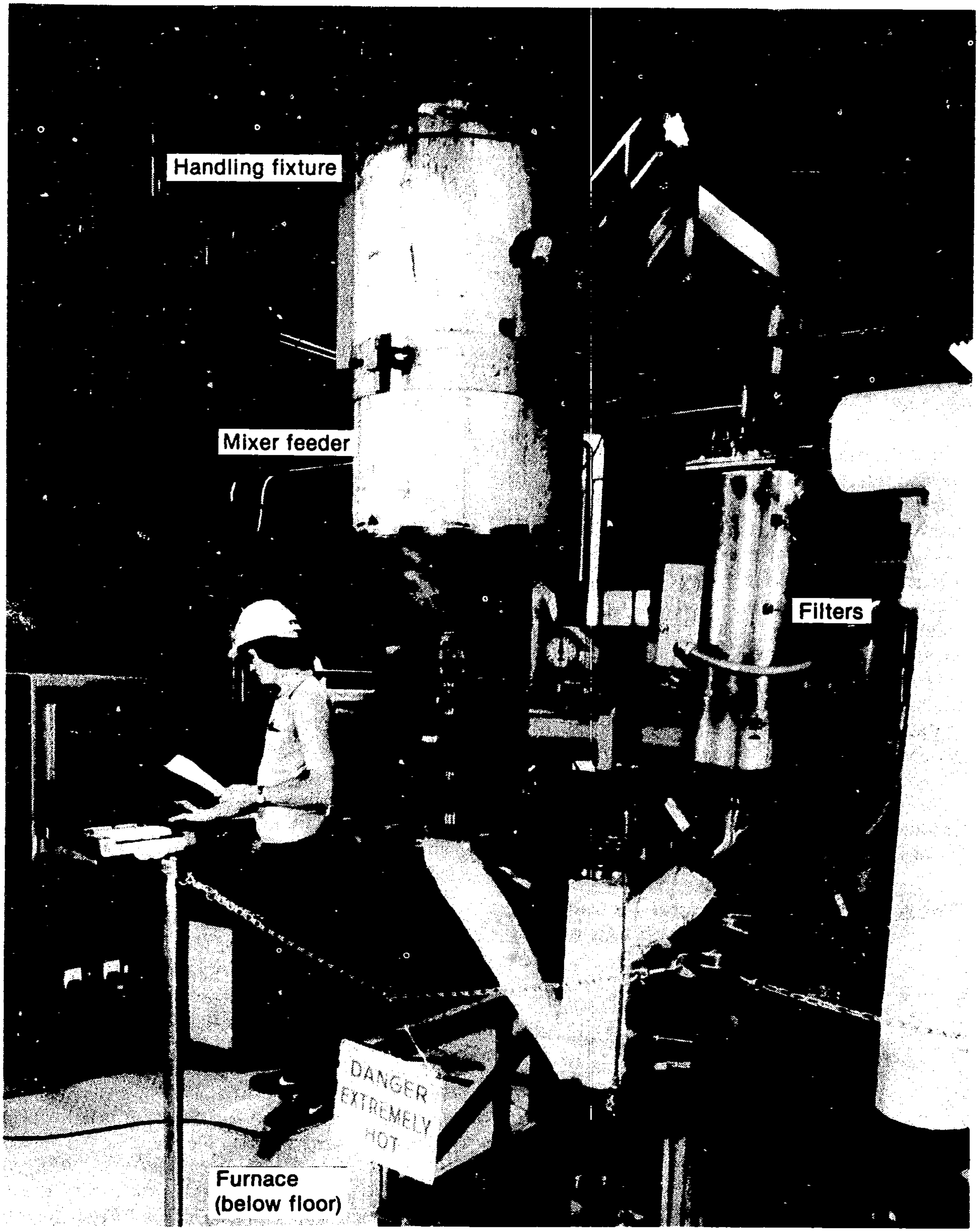

Figure 34. In-can melter system in use at Pacific Northwest Laboratories to vitrify SDS zeolites.

analysis of suspected abnormal waste for curie content, TRU contamination level, dose rate, quantity, and physical form. The criteria for acceptability as low level waste are for now those requirements in effect at the Washington State burial site, i.e. not greater than 10 nCi/g of TRUs. Current Washington State requirements will be used to classify abnormal waste until another site opens to TMI waste or until Washington State regulations change. A Northeast regional burial site is scheduled to open later in the decade and the NRC has proposed new regulations in 10 CFR 61 which may be implemented by Washington State.

Purification System Abnormal Waste—The Makeup and Purification System contains abnormal waste of interest to the TI&EP. The system, sketched in Figure 35, maintains Reactor Coolant System quality and chemical limits while the plant is operating. After initiation of the TMI-2 accident, the letdown flow from the Reactor Coolant System was directed through the Makeup and Purification System filters and demineralizers for at least 18 hours and 35 minutes on March 28, 1979, after which time flow stopped. The filters and demineralizers have since been isolated.

The Makeup and Purification System filters were removed on March 24, 1982 and sent to the the Idaho National Engineering Laboratory (INEL) for analysis. These filters were of particular interest to the TI&EP Radiation and Environment program in their studies of fuel and fission product transfer in the primary coolant system. While the filter analysis is discussed in more detail in the Radiation and Environment section, a visual inspection showed the paper filter media to be degraded and stripped from the filters. This degraded condition led researchers to conclude that the filters would have been ineffectual in preventing fuel particles from reaching the demineralizers.

Since the likelihood of fuel in the demineralizers was great, the resins in these demineralizers are identified in the Memorandum of Understanding between DOE and the NRC as abnormal wastes. GPUNC undertook to characterize the demineralizer resins with TI&EP assistance. They had to determine if fuel levels were at or near the criticality level of 70 kg. Using demineralizer drawings, the accident operating history, and GPUNC's analyses on the condition of the demineralizers, characterization information was developed on the demineralizer cubicles and the contents of each vessel. Videotapes, radiation surveys, and contamination swipes were also made in each cubicle. The TI&EP coordinated technical assistance provided by Westinghouse Hanford Company, Rockwell Hanford Operations, and Pacific Northwest Laboratories. All studies to date indicate that fuel levels in the demineralizers were well below criticality levels.

Mechanical Transporter at Work—In support of the characterization effort, Westinghouse Hanford Company (WHC) supplied a mechanical transporter, pictured in Figure 36, called

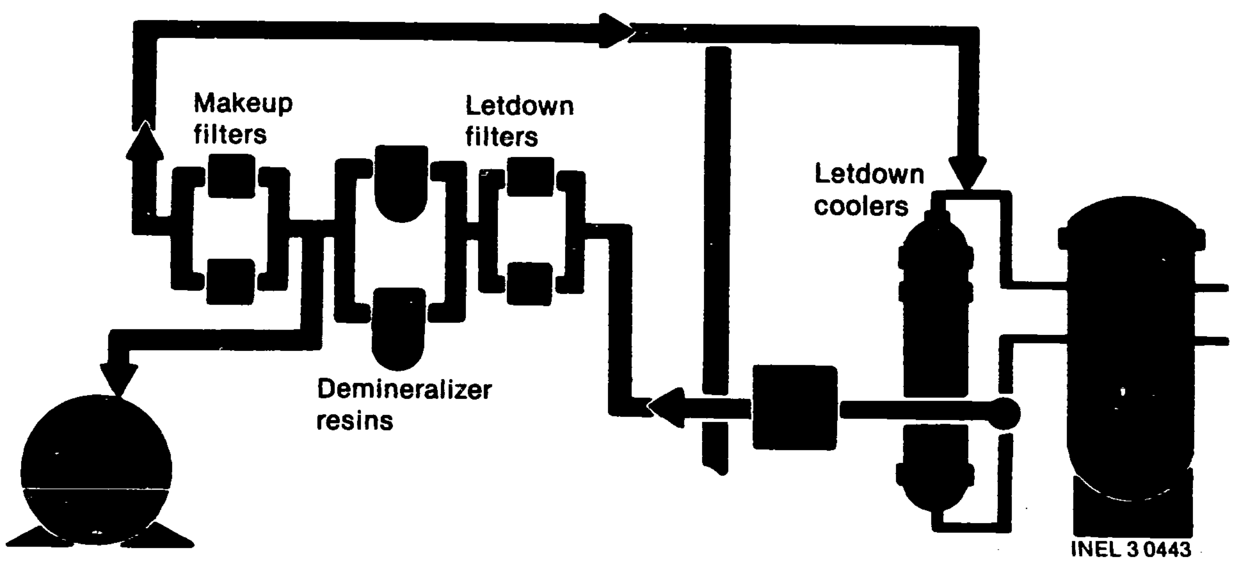

Figure 35. Makeup and Purification System which maintains RCS quality and chemical limits.

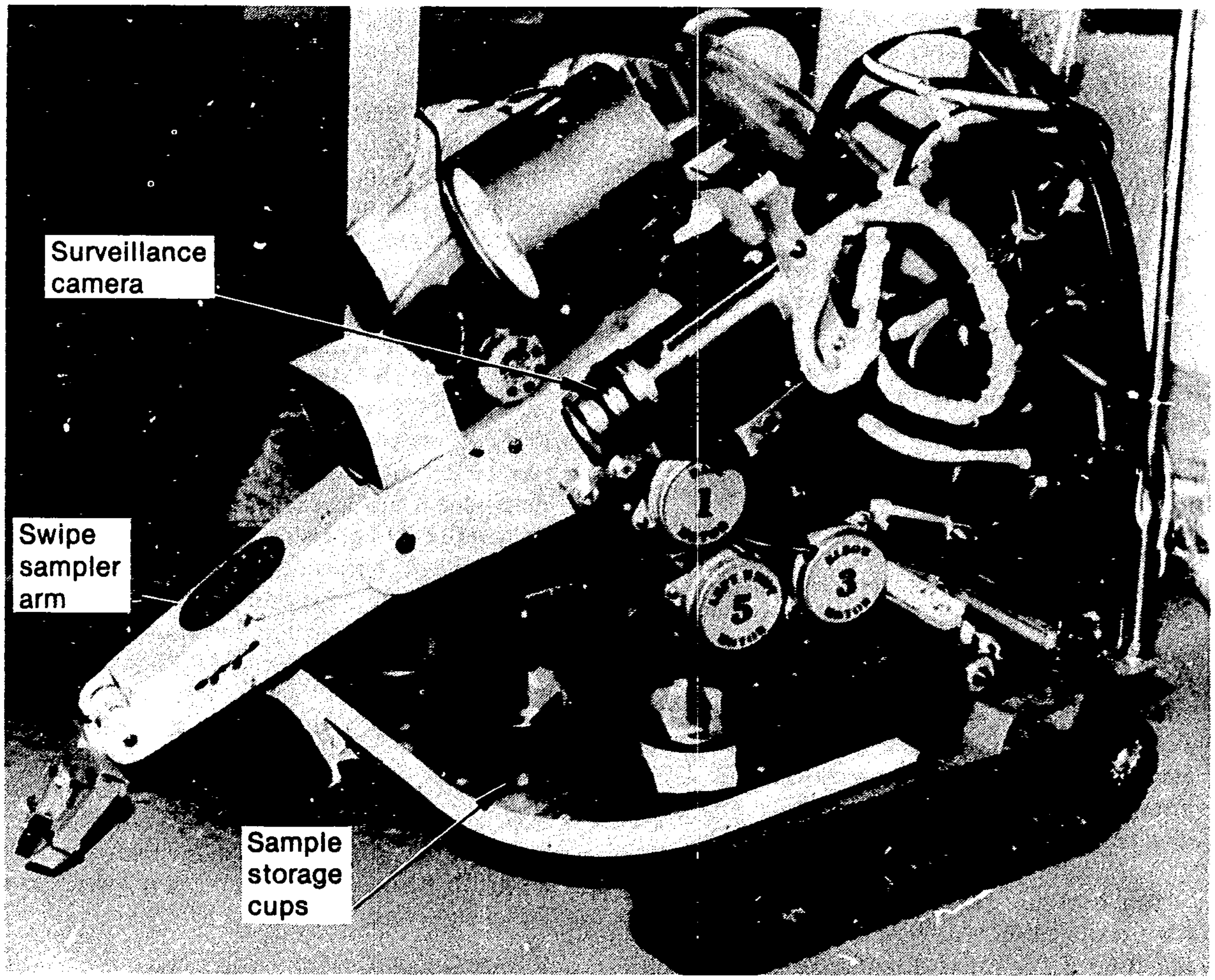

Figure 36. Mechanical transporter used to inspect the demineralizer cubicles and take samples.

Surveillance and Inservice Inspection Robot (SISI) to videotape the interiors of the demineralizer vessel cubicles and to take radiation readings and contamination swipes. High radiation levels in the cubicles necessitated the remote survey capabilities which SISI could provide to reduce man-rem exposure levels. SISI followed the path shown in Figure 37 in its characterization studies.

The readings, samples, and measurements contributed to preliminary characterization of the cubicles and their contents. A radiation detector attached to the SISI transporter obtained a radiation dose rate measurement of 1150 R/h from the B vessel about 0.3 m from the bottom surface. In the A vessel cubicle, interference from a pipe prevented the transporter from getting any closer than 1 m from the vessel surface, where a reading of 320 R/h was obtained. Radiological swipes of SISI's cable analyzed after the A cubicle entry showed 4000 disintegrations per minute (dpm) and the treads showed 1000 dpm. (Engineers could not eliminate the possibility that some of this contamination may have been picked up in the adjoining rooms as SISI moved to the cubicles.) The SISI videotapes showed the as-built condition of the equipment in the cubicles and assisted in verifying piping isometrics developed from construction drawings.

Resin Characterization Studies—Gamma spectroscopy of the demineralizer vessels was performed to determine fuel and fission product content. WHC employed a gamma spectroscopy system which used a silicon-lithium (SiLi) crystal detector and a 15-cm diameter by 23-cm long lead shield with variously sized collimators. Figure 38 pictures the remotely operated boom used to lower

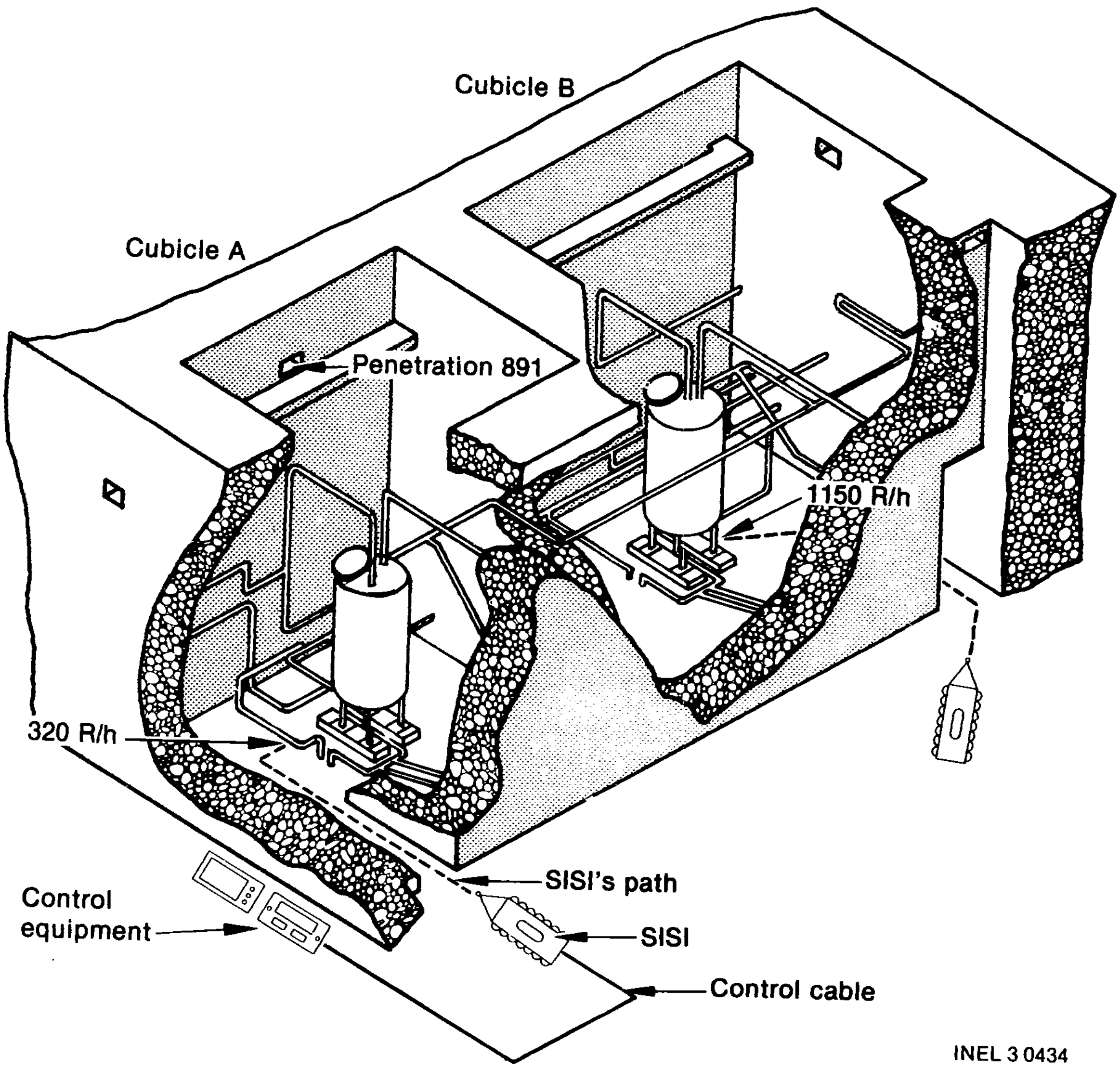

Figure 37. SISI mechanical transporter path into the demineralizer cubicles.

the detector through cubicle penetration 891 (see Figure 37) into various positions around the demineralizer vessel. The SiLi detector traveled from the top to the bottom of opposite sides of the A vessel, and also gathered data from positions directly above the vessel.

Data obtained with the SiLi detector on the amount of $^{144}Ce/Pr$ present in the resin were used to calculate a fuel content for the A vessel. Based on the fact that $^{144}Ce/Pr$ and UO_2 fuel have similar chemical behavior and also on $^{144}Ce/Pr$-to-fuel ratios determined from samples taken of different locations in the TMI-2 plant, scientists determined a fuel content of 1.3 ± 0.6 kg in the A vessel, significantly less than the criticality level of 70 kg. Although no quantitative fuel estimate could be made for the B vessel on the basis of the SiLi data, the one data point obtained indicated less fuel but more fission products than the A vessel.

The A vessel is estimated to contain about 6000 Ci of ^{137}Cs. A combination of data indicates there is no water above the top of the resin bed in the A vessel. The top of the resin bed is estimated to be at the 309-ft elevation, which would mean the resin volume is one-half the originally installed resin volume. This finding is consistent with laboratory resin irradiation tests

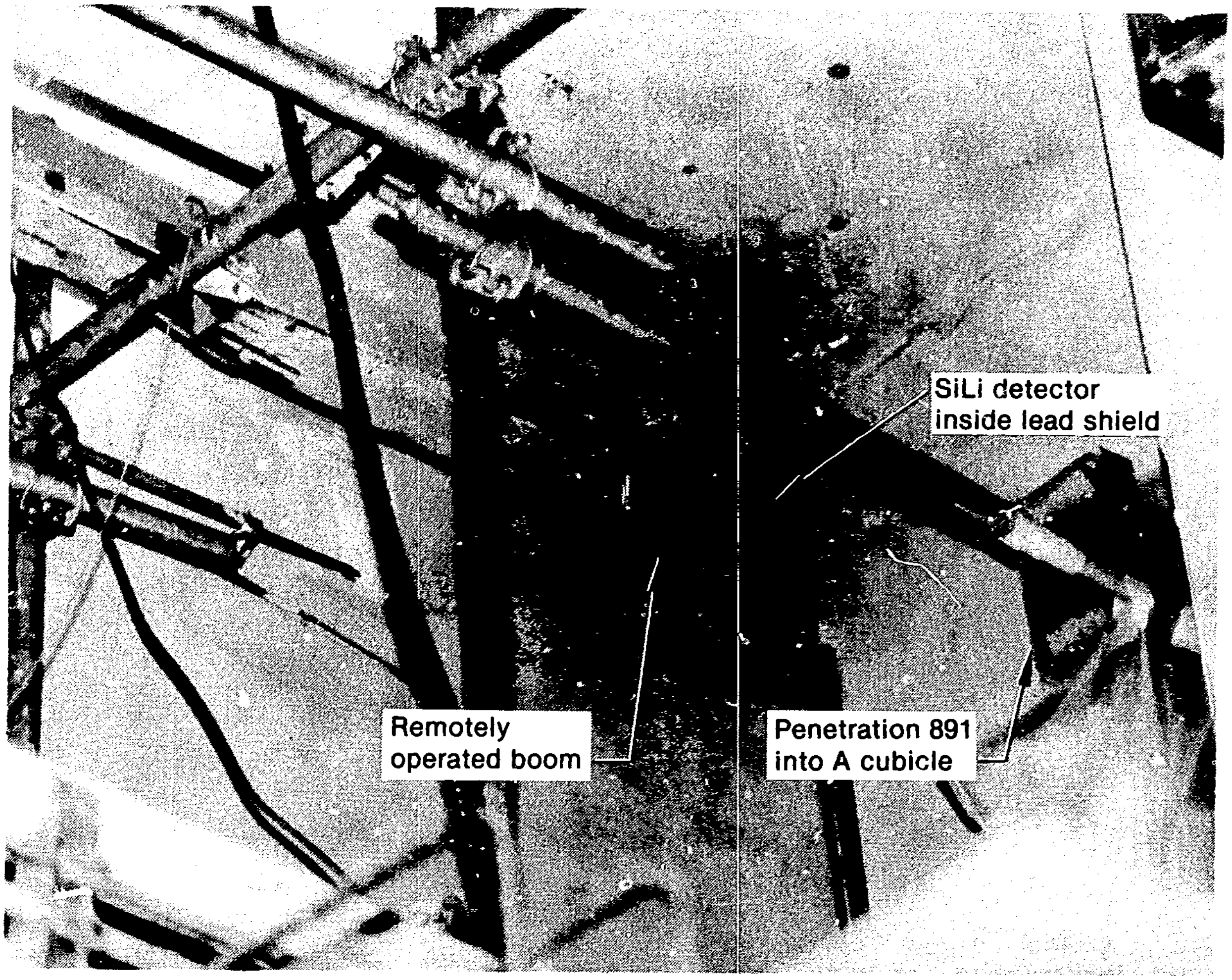

Figure 38. SiLi detector about to be lowered into demineralizer cubicle A to perform gamma spectroscopy.

that showed a similar volume reduction for resin exposed to 1.7×10^9 rads, the dose GPUNC estimated the resins received as a result of the accident.

In another resin characterization study, Solid State Track Recorders (SSTRs) were placed in the A cubicle for 29 days to determine the presence of any fuel in the vessels. The SSTRs were positioned above the A vessel and along one of its sides from top to bottom. A gross indication of the relative integrated gamma dose to each SSTR was evident from the yellowing of the lucite in each SSTR. The SSTRs work by providing a record of tracks of fission products generated by neutron-initiated fissions in the ^{235}U contained in the SSTR. The neutrons initiating such fission would originate from spontaneous fissions occurring in any fuel in the vessel. The fission tracks in the SSTR were proportional to the calculated neutron flux; they indicated the presence of an estimated 1.7 ± 0.6 kg of fuel in the demineralizer.

In an additional characterization effort, Los Alamos National Laboratory used a technique based on the reaction which occurs when beryllium interacts with gamma and releases a neutron. This reaction has a gamma threshold energy of 1.6 MeV. A block of beryllium, 28 x 28 x 46 cm, coated with cadmium, has a fission chamber located at its center. Thus the fission chamber sees a neutron flux proportional to a gamma flux greater than 1.6 MeV. This high energy gamma flux is predominantly due to $^{144}Ce/Pr$, a radionuclide which indicates the presence of fuel in amounts proportional to the amount of $^{144}Ce/Pr$ present. Using this method, fuel content in the A vessel was estimated at 11 ± 6 kg; the estimate for the B vessel was 3.9 ± 1.5 kg. Both estimates were well below the assigned criticality levels for resins in the demineralizers.

Resin Removal Studies—WHC identified and analyzed alternate methods for resin removal. In this work, WHC evaluated the viability of the resin sluicing method with and without chemical pretreatment, considered demineralizer vessel removal, analyzed packaging and shipping requirements, and studied system tie-in points and space envelopes for location of processing or packaging equipment.

As part of the conceptual study of resin sluicing, WHC coordinated experiments by PNL on the irradiation of Rohm and Haas resins similar to the IRN-217 resins used in demineralizers. An integrated dose of 1.7×10^9 rads was applied to the resins, which is the dose that GPUNC estimated the resins received during the accident. The resins were irradiated both wet and dry and subjected to elevated temperatures. In all cases, the resins remained as discrete particles without agglomeration or bonding. Should this be the case with TMI-2 resins, sluicing would remain a viable removal option.

Laboratory scale studies performed by PNL were made on cesium retention of resins irradiated to 1.7×10^9 rads. Cesium removal was accomplished using $NaBO_3$ as a chemical reagent. Irradiated resins were actually sluiced with greater ease than new resins which had not been irradiated.

The resin irradiation test results provide reasonable confidence that the preferred alternative for resin removal is to discharge the resins by normal plant sluicing techniques. However, resin sluicing will not be attempted until after a demineralizer resin sample has been acquired and the feasibility of the sluicing approach is confirmed by laboratory tests.

GPUNC has developed plans to obtain gas and liquid samples through the differential pressure lines to each vessel and to obtain resin samples through the resin fill lines. These samples will be used to confirm the sluicing approach for resin removal and waste handling.

Resin Disposition. The TMI-2 accident resulted in the transfer of more than 1900 m^3 of contaminated water to the Auxiliary and Fuel Handling Buildings. This water was processed through a three-stage ion exchange cleanup system called EPICOR II. The first stage uses liners called prefilters, while the second and third stages are called demineralizers. The processing of the entire batch of water resulted in 50 prefilters highly loaded with radioactivity and 22 second- and third-stage demineralizers with lower radioactivity levels. The 22 second- and third-stage liners have been shipped off for commercial shallow land burial. Because prefilters constitute an abnormal waste form covered under the Memorandum of Understanding, the DOE agreed to take possession of the liners for characterization, packaging for safe shipment to a DOE laboratory, research and development, and reimbursable disposition.

By the end of 1982, 17 prefilter liners had left the Island for characterization studies at DOE laboratories. All 17 shipments used transportation casks mounted on low-boy trailers similar to the one shown in Figure 39. Continuation of the DOE Resin Disposition Program will ensure that TMI-2 will not become a long term storage site for EPICOR II wastes. The Resin Disposition Program included work in laboratory characterization of two of the prefilter liners, on-island preparations to ship liners, and coordination of the actual shipping. The program also coordinates liner receipt and interim storage at the INEL and the EPICOR II Research and Disposition Program which includes high integrity container development.

Liner Characterization—The Resin Disposition program began in 1981 when DOE undertook the characterization of EPICOR Prefilter Liner 16 (PF-16). PF-16 was chosen for characterization because of its high loading with radioactivity and the low pH of its residual water. These characteristics led researchers to conclude that PF-16 would likely be the most susceptible to deterioration of any of the liners in its class (PF-12 to 50). In 1982 the characterization was completed, and the liner was shipped to the INEL in April 1982.

At the outset of planning for the resin characterization program the Waste Immobilization Program decided that two liners would be characterized: a liner containing both resin and zeolite ion exchange media, and a liner containing only resin. PF-16 was of the first type. PF-3, an all-organic resin liner of the second type, was chosen for characterization because its high loading with radioactivity and other characteristics made it the most likely of its class (PF-1

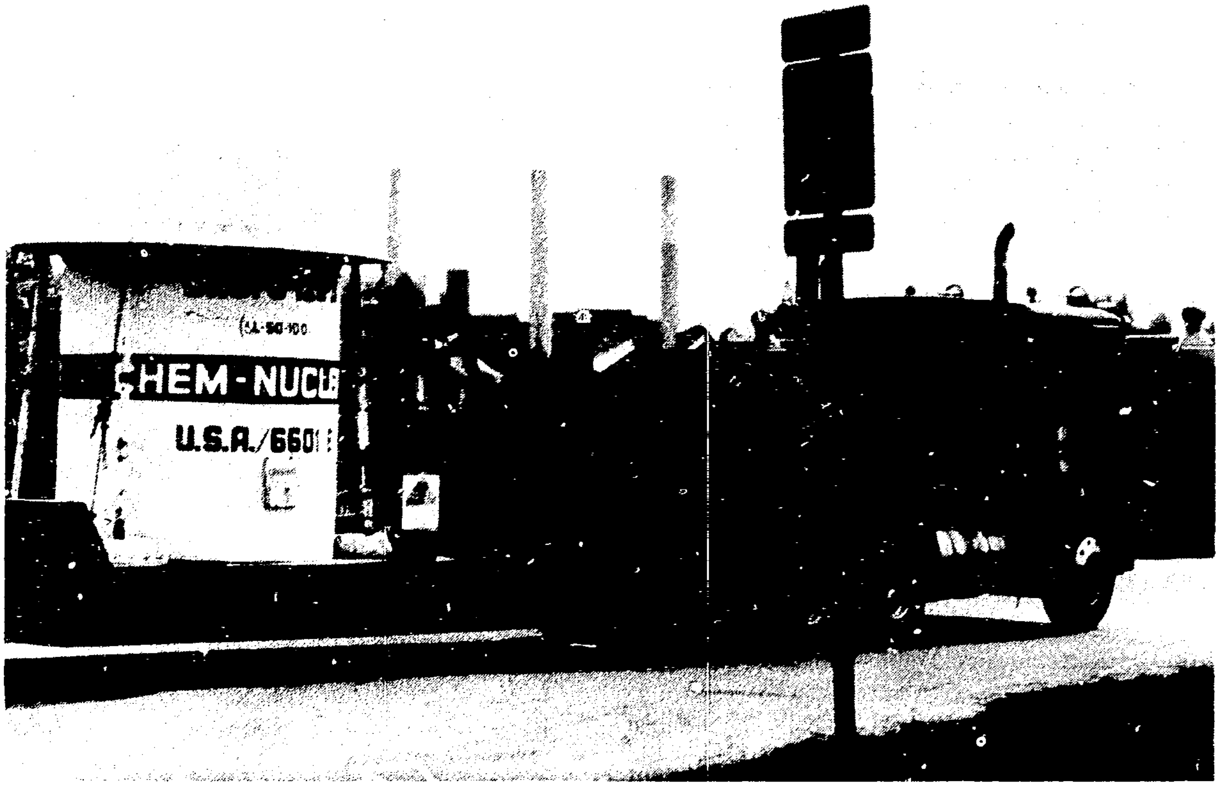

Figure 39. EPICOR II prefilter liner in shielded transportation cask enroute to DOE laboratory for characterization.

through 11) to possibly degrade with time. PF-3 underwent a characterization program in 1982 similar to what PF-16 underwent in 1981 and 1982. Selected results of both examinations are presented below.

Liner Integrity—Both PF-16 and PF-3 exhibited some minor, localized surface corrosion on the interior and exterior surfaces. In both cases, however, there was no evidence that the liner integrity had been compromised. Detailed interior visual examinations, and if warranted, destructive examinations, will be performed on the two liners at the INEL.

Gas Analysis—When PF-16 was being prepared for shipment, technicians vented it for one full hour. When it arrived at BCL 12 days later scientists sampled and analyzed the gases in the liner again. Results of this BCL sampling are reported in Table 18. By way of comparison, Table 18 also reports BCL's analysis of the gases in PF-3. PF-3 was also sampled at TMI, but unlike PF-16 was purged and inerted with nitrogen before it was shipped to BCL. The table shows that the hydrogen content in PF-3 was much lower upon arrival at BCL (0.8 ± 0.1 vol%) than the hydrogen content in PF-16 upon its arrival (12.4 ± 0.2 vol%). Purging and inerting with nitrogen explains the difference in the gas content of the two liners. The section entitled "Preparations to Ship Liners" further discusses the purging and venting process.

Liquid Analysis—Some free-standing residual water was found in the bottom of both liners. The complete results of the water analyses are shown in Table 19. Analyses indicated that the water was fairly clean, and with a pH of 5.3, not very acidic.

Ion Exchange Media—PF-16 had what appeared to be a zeolite-like material as the very top media layer, followed by ion exchange resins. The top layer of PF-3 appeared to be a mixture of resin and small black angular particles. None of the ion exchange media appeared to have suffered any significant physical degradation as a result of their exposure to TMI-2 waste water. Radiochemical analyses of PF-16 ion exchange media are shown in Table 20; results for PF-3 will be available in 1983.

Table 18. EPICOR prefilter liner gas analysis

	PF-16 Sample at BCL[a]	PF-3 Sample Before Venting[b]	PF-3 Sample After Venting[b]	PF-3 Sample at BCL[c]
Volume Percent				
Nitrogen	80.6 ± 0.4	87.5	98.2	94.7 ± 0.4
Carbon Dioxide	5.52 ± 0.06	—[d]	1.2	3.47 ± 0.04
Hydrogen	12.4 ± 0.2	9.9	0.4	0.8 ± 0.1
Argon	0.96 ± 0.05	—	—	0.17 ± 0.02
Oxygen	0.20 ± 0.02	N/D[e]	0.1	0.86 ± 0.05
Carbon Monoxide	0.2 ± 0.02	—	—	0.05 ± 0.01
Parts per Million by Volume				
Methane	500.0 ± 2.5	—	—	82 ± 10
Ethane	42.0 ± 4	—	—	101 ± 10
Propane	6.0 ± 1	—	—	2 ± 0.5
Iso-butane	0.6 ± 0.1	—	—	0.7 ± 0.1

a. Twelve days after an one-hour venting.

b. Taken at TMI.

c. Two weeks after it was inerted at TMI.

d. Dashes indicate items for which no samples were obtained.

e. Sampled for but not detected.

Preparations to Ship Liners—After BCL determined the level of hydrogen present in PF-16 twelve days after venting, the balance of the liners prepared for shipment were required to undergo remote purging with an inert gas. In 1981 EG&G Idaho, Inc. developed a Prototype Gas Sampler (PGS) to remotely sample and purge the TMI-2 liners; it was delivered to TMI in February 1982. The requirements for this device were that it be able to remotely remove a vent plug from the liner, capture a gas sample, purge the liner with an inert gas, and then reinsert the vent plug. The tool had to be capable of removing any of the three or four penetration plugs in the liner.

In conjunction with the PGS, GPUNC designed, constructed, and tested the facilities required at TMI to sample, vent, and purge the EPICOR II liners. In addition to the PGS tool, these facilities consisted of a portable shielding "blockhouse," and a portable support facility. GPUNC located the entire system at the TMI-2 solid waste staging facility, where EPICOR liners are stored in individual concrete storage modules. Figure 40 shows the relationship of all the system components.

The shielding blockhouse, shown in Figure 41, is a reinforced concrete structure that serves both as a radiation shield and as a barrier to contain any forces in the unlikely event of an explosion. The blockhouse has viewing windows so that operators can observe conditions inside. The support facility can also be seen in Figure 41 and contains the gas circulator for purging with inert gas, a gas chromatograph for the on-line analysis of gas samples, and the controls for the PGS, including TV monitors. The inside of the support facility is shown in Figure 42.

The sequence of operations for preparing the liners for shipment is as follows: The lid to the

Table 19. EPICOR prefilter liner residual water chemistry and radiochemistry analysis

	PF-16	PF-3
pH	5.3 ± 0.1 at 27°C	5.3 ± 0.1 at 20°C
Conductivity	30 μmho/cm at 27°C	16 μmho/cm at 20°C
Acidity	1.2 meq/ml at pH 7.0	4.2 meq/ml at pH 7.0
Total residue upon evaporation	3.0 ± 0.1 mg/ml	7.5 ± 0.1 mg/ml
	Parts per million	
Na	<2.0	5.7
Fe	0.034	3.5
P	<0.11	0.1
Zn	0.088	0.19
Mg	<0.02	0.095
Ca	0.10	0.53
Al	0.11	0.05
B	1120	2200
NH_4	0.8	0.09
SO_4	5.2	1.44
NO_3	<0.3	0.13
Cl	3	1.25
Total organic carbon	65	6.45
Total Kjeldahl nitrogen (TKN)	0.48	0.5
	μCi/ml	
Gross beta/gamma	1.77 ± 0.01E-2	1.36 ± 0.01E-2
Gross alpha	5.9 ± 0.01E-4	7.5 ± 0.1E-4
$^{89,90}Sr$	5.2 ± 0.1E-4	1.04 ± 0.05E-3
^{125}Sb	7.94 ± 0.42E-4	—[a]
^{134}Cs	1.32 ± 0.02E-3	—
^{137}Cs	1.308 ± 0.005E-2	—
$^{238, 238, 240}Pu$	<1.0E-4	—
^{238}U	<1.0E-4	—

a. Calculations not completed at time of publication.

Table 20. EPICOR PF-16 ion exchange media radionuclide content (Ci/g)

Section	^{235}U	$^{239,240,242}Pu$	$^{89,90}Sr$	^{137}Cs	^{134}Cs	^{60}Co
Top (0 to 12 cm)	<5E-12	3.1E-10	1.6E-6	4.9E-3	6.4E-4	4.8E-6
Middle (12 to 33 cm)	<5E-12	1.1E-10	1.4E-6	7.3E-4	1.3E-4	1.4E-6
Bottom (33 to 40 cm)	<5E-12	1.2E-10	9.9E-7	5.5E-4	4.4E-5	5.5E-7

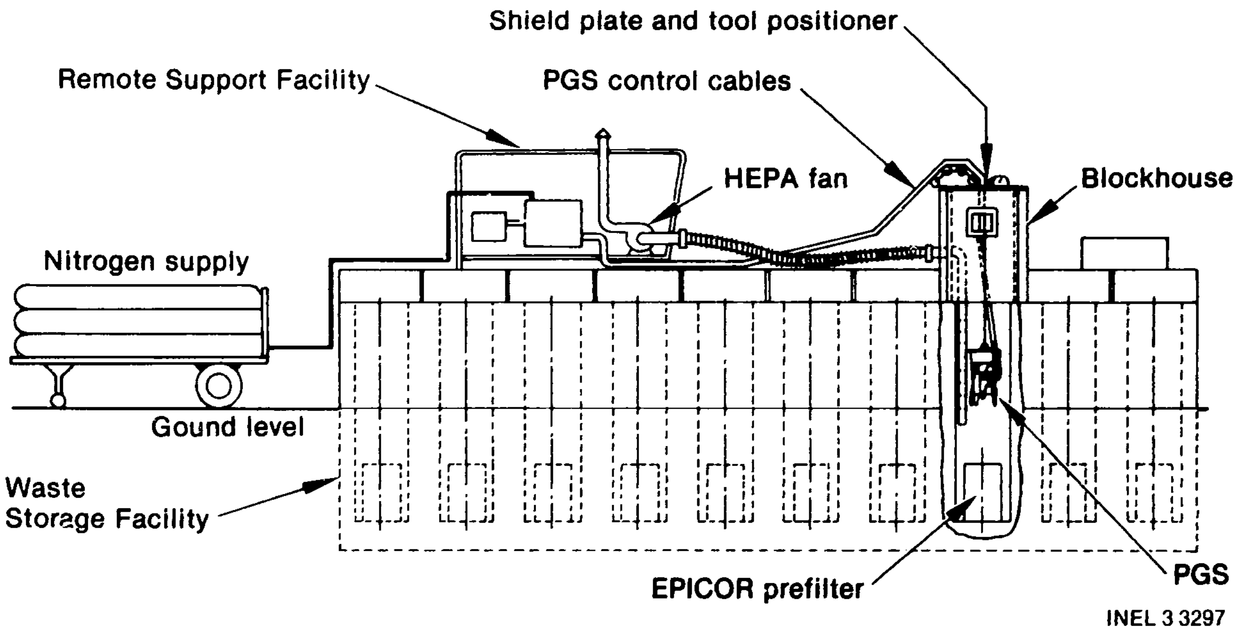

Figure 40. Arrangement of Prototype Gas Sampler system components at the TMI-2 solid waste staging facility.

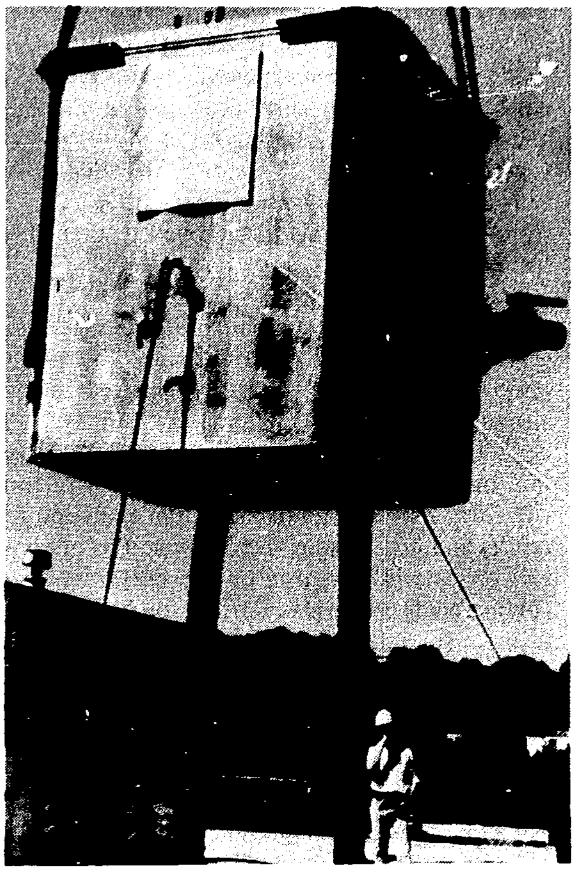

Figure 41. Shielded blockhouse, near trailer-based remote support facility, being lowered into place for use with the Prototype Gas Sampler.

Figure 42. Inside Prototype Gas Sampler support facility showing control panels in right foreground and TV monitors in right background.

concrete storage module is removed. The PGS is suspended from the support plate as shown in Figure 43. The support plate, which acts as the top of the blockhouse, and the blockhouse are then placed over the module. The blockhouse and module are then purged with an inert gas, and the PGS is lowered from the support plate to the liner. Using the TV monitors, operators in the support facility precisely position the PGS tool tip on the vent plug with the air-driven positioning motors on the PGS. The plug is then removed and a gas sample taken and analyzed with the on-line gas chromatograph. The remaining gases in the liner are then purged through a filtering system and after careful monitoring, are released to the atmosphere.

Figure 43. Prototype Gas Sampler being lowered into place onto a liner located in the storage module beneath the pictured blockhouse.

Selected liners are monitored for a two-week period to determine the rate at which a combustible gas mixture is generated. The liner is then reinerted and when engineers determine that it can be shipped safely, the liner leaves the island in a shielded shipping cask.

Shipping—By July 1982, GPUNC and the TI&EP had completed integrated functional tests of the entire liner venting and inerting system and were ready to begin shipping. The first liner to be shipped was PF-3. It was opened, sampled, inerted, monitored for two weeks, and then shipped to BCL in August 1982. After that, 15 more shipments were made in 1982, bringing the total shipped to 17. Table 21 summarizes the shipments through the end of 1982, including the results of the gas sampling. All 50 EPICOR prefilter liners are scheduled to be shipped by the end of September 1983.

EPICOR II Research and Disposition at the INEL—The EPICOR II prefilter liners generated in Auxiliary and Fuel Handling Building water processing are being sent to the INEL as part of the TMI Waste Immobilization Program, where they will be stored in special facilities, and researched as special wastes.

The EPICOR II Research and Disposition program at the INEL is contributing to the goal of removing abnormal wastes produced during cleanup of the TMI-2 plant. The program will conduct research on the effects of ionizing radiation on various types of ion exchange media and will form a bridge between external radiation test results and actual resin degradation. The program will provide information on the performance of commercially available matrices for immobilizing resins. It will also conduct leach tests of radionuclides from those matrices and will provide leach test results on resins stored for long periods of time. The program is also providing a High Integrity Container option for disposal of radioactive wastes that heretofore have not been acceptable at commercial disposal facilities without first being immobilized in some matrix.

The significant accomplishments of the program during 1982 are related to completion of

Table 21. EPICOR prefilter liner shipments through December 1982[a]

Shipment Number	Liner	Date Shipped/ Destination	Curies	Liner Gas Composition Volume % Prior to Inerting: Hydrogen	Nitrogen	Oxygen
1	PF-16	05/19/81-BCL	2270	—[b]	—	—
		04/05/82-INEL	2270	—	—	—
2	PF-3	08/17/82-BCL	1880	9.9	87.5	<0.01
3	PF-1	08/25/82-INEL	1500	3.6	94.7	<0.01
4	PF-2	10/07/82-INEL	1050	4.5	92.8	0.5
5	PF-7	10/21/82-INEL	1400	3.7	91.4	<0.1
6	PF-8	10/23/82-INEL	1370	8.0	89.4	<0.1
7	PF-9	10/28/82-INEL	1350	12.9	85.3	<0.1
8	PF-45	11/02/82-INEL	2040	0.7	99.0	0.7
9	PF-46	11/03/82-INEL	2180	0.07	98.0	0.2
10	PF-20	11/17/82-INEL	1950	22.4	72.7	1.2
11	PF-47	11/29/82-INEL	1940	12.8	82.0	<0.1
12	PF-27	12/01/82-INEL	1950	7.3	88.4	1.5
13	PF-48	12/06/82-INEL	1940	0.08	94.8	1.2
14	PF-6	12/13/82-INEL	170	<0.01	97.0	<0.1
15	PF-44	12/13/82-INEL	1850	10.4	86.7	<0.2
16	PF-18	12/14/82-INEL	2030	26.0	66.8	0.2
17	PF-49	12/29/82-INEL	1780	2.7	89.0	<0.2

a. The remaining 33 prefilter liners are scheduled for shipment in 1983.

b. PF-16 was not inerted.

program planning and preparation of needed equipment and facilities. Substantial progress was achieved in several areas.

Facility Preparation—Several refurbishment and maintenance activities were performed on equipment associated with the INEL Test Area North (TAN) 607 Hot Shop in order to provide a safe environment for handling and interim storage of EPICOR II liners. Fulfillment of those actions permitted the Hot Shop to safely handle the liners.

Figure 44 shows the TAN 607 Hot Shop and some of the equipment built or refurbished for use in the program. Technicians performed major equipment refurbishment work on the overhead crane, overhead manipulator, wall mount manipulators, and turntable. A venting tool, similar to the Prototype Gas Sampler being used at TMI, was built for reventing the liners at the INEL. Two special storage silos with lids were built for use on the Hot Shop turntables. In Figure 45 a diagrammatic sketch of the internals of a storage silo shows the arrangement of various components.

Liner and Resin Research Operations—Planning for actual resin and liner research operations began in 1982. The planned operations were divided into three parts: liner integrity operations, resin degradation studies, and resin solidification studies.

Liner integrity examinations will use metallographic methods to determine the condition of the steel and epoxy coatings in the liners. Six locations in each liner will be sampled and examined in detail. The liner integrity examination study started in September 1982. Part of the ion exchange media of PF-16 was transferred to a new liner (PF-16A) so that the internal surfaces of the original liner could be examined. That transfer required development and fabrication of some special tooling, which performed successfully during the operation.

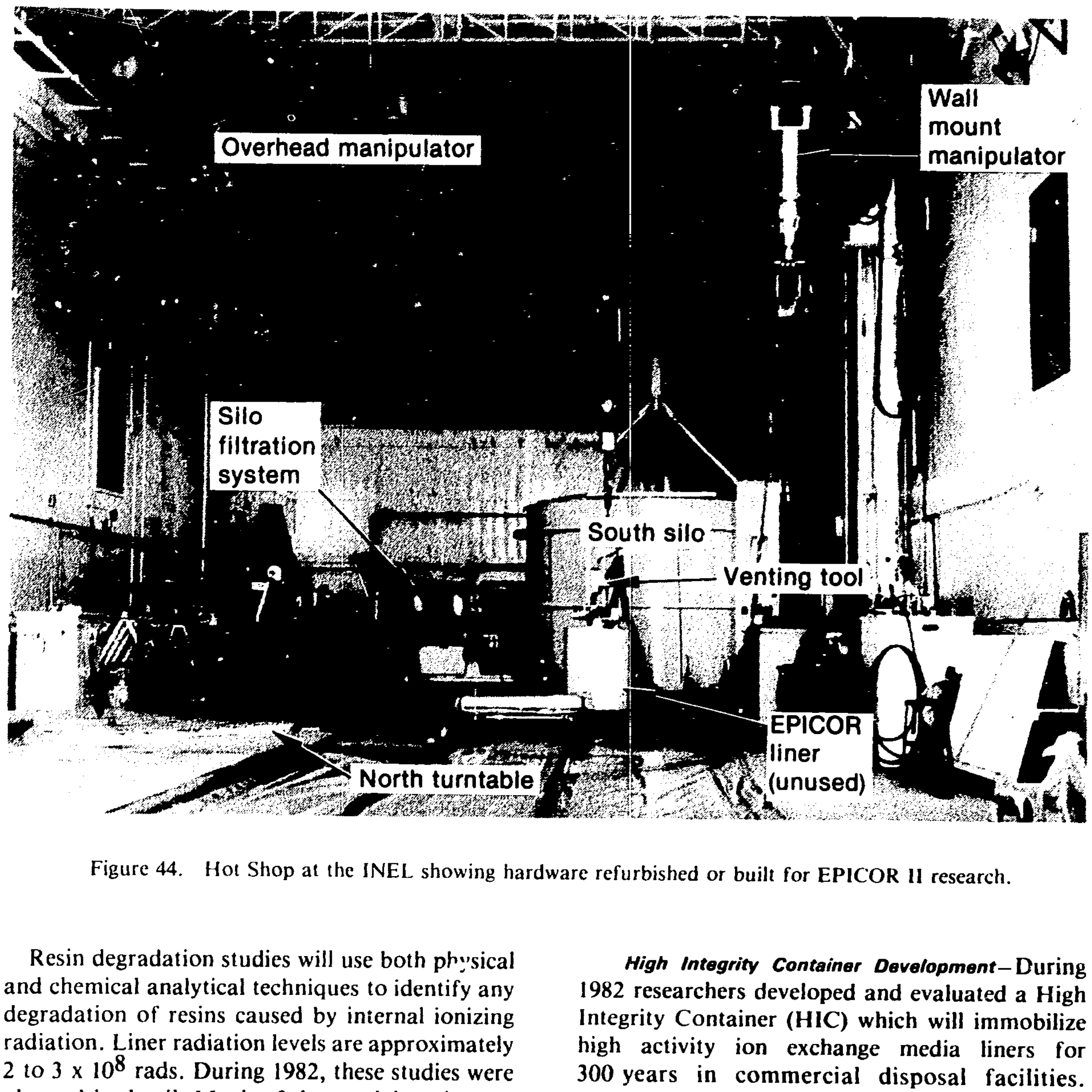

Figure 44. Hot Shop at the INEL showing hardware refurbished or built for EPICOR II research.

Resin degradation studies will use both physical and chemical analytical techniques to identify any degradation of resins caused by internal ionizing radiation. Liner radiation levels are approximately 2 to 3 x 10^8 rads. During 1982, these studies were planned in detail. Much of the special equipment needed for collecting and manipulating samples was designed and procured. Further equipment procurement will occur in 1983.

The resin solidification studies will explore the use of polymer or cement as media for solidifying ion exchange media collected from the EPICOR II liners. INEL scientists will immobilize ion exchange media samples in formulations of a Portland cement or Dow polymer, will subject the solidified samples to compression and leach testing experiments, and will subject pieces of selected samples to long-term leach tests in lysimeters. Detailed planning for these studies occurred in 1982. Most of the equipment to be used in the studies was designed and procured; the rest will be obtained in 1983.

High Integrity Container Development—During 1982 researchers developed and evaluated a High Integrity Container (HIC) which will immobilize high activity ion exchange media liners for 300 years in commercial disposal facilities. Development and use of a HIC is an alternative to the immobilization of filtration resins from operating commercial power reactors. One of the two HICs developed in the reporting period will undergo stringent testing at the INEL. This testing will support documentation for obtaining an agreement to use the HIC at a commercial burial site in the state of Washington.

The HIC is designed to maintain its integrity in both eastern and western disposal facilities. It is capable of withstanding conditions of moisture, soil composition, and disposal maximum depths encountered at either site location. The HIC will permit disposal of filtration resins at a commercial disposal facility as Class "C" wastes, because the overpack is an efficient barrier to intruders as

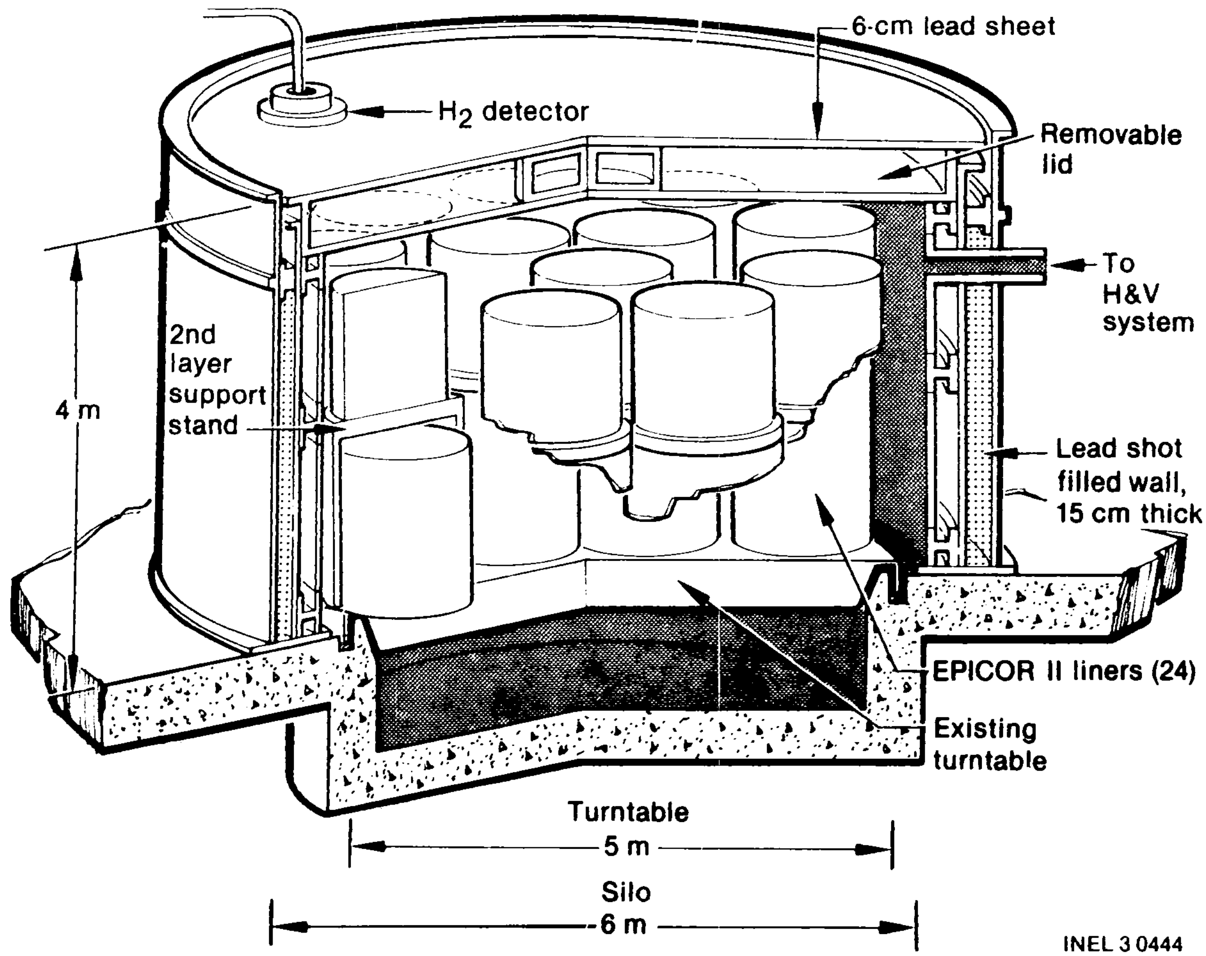

Figure 45. Cutaway view of an EPICOR storage silo in the INEL Hot Shop.

discussed in proposed regulation 10 CFR 61 (Fed. Regist. 46 [142], 24 July 1981 [Draft]). It will effectively provide for waste stability for 300 years, eclipsing the 150-year minimum in 10 CFR 61. A storage period of 300 years spans about 10 half-lives of ^{90}Sr or ^{137}Cs. After 10 half-lives, the wastes no longer need additional shielding.

HIC Development—Two HICs were designed and fabricated by Nuclear Packaging, Inc. under contract with EG&G Idaho, Inc. The first HIC underwent extensive impact and container integrity testing and performed well. The HIC was drop-tested 0.9 m in compliance with the requirements of 10 CFR Part 71 Appendix A for a free drop onto a flat unyielding surface. It was determined by analysis that the worst test condition would be a corner drop with the container tipping over onto its side after the initial impact. A lifting bar was employed which rotated the container to a 42 degree off-vertical angle, thus ensuring that the container would tip onto its side after contact. The unyielding surface consisted of a block of concrete weighing about 91 m.t. and topped with a 2.86-cm-thick grouted steel plate.

The container was attached to the hoisting hook of a mobile crane with a quick release latch. The container was raised 0.9 m off the pad and then dropped. Figure 46 shows the container at the moment of impact. There was some superficial damage at the point of impact; about 3.8 cm of concrete chipped off and exposed some of the internal rebar cage, as shown in Figure 47. Otherwise, the damage was limited to some hairline cracks in the HIC concrete. Some concrete adjacent to the lifting eye on the opposite side of the HIC shook loose from the impact. The impact loads of the lifting bar used to raise the HIC off the ground before the drop test probably weakened the concrete in this area. Although there was some damage, there was no breach of containment. The container successfully passed the test.

Figure 46. High Integrity Container undergoing impact testing shown here as it hits the concrete test pad.

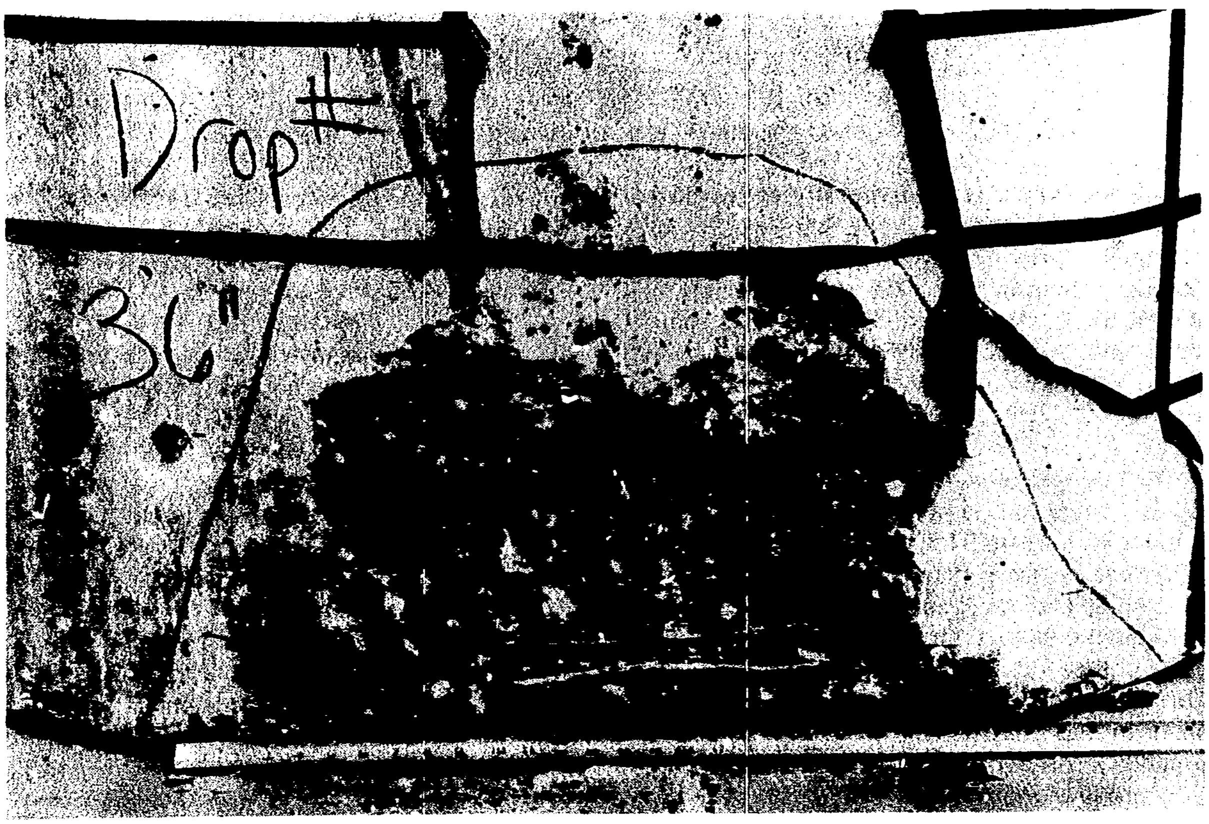

Figure 47. Superficial damage to High Integrity Container at point of impact following drop test.

The second prototype HIC, specifically designed to house an EPICOR II liner, was delivered to EG&G Idaho, Inc. in September 1982. The HIC is diagrammed in Figure 48. The HIC is 1.59 m in diameter by 2.13 m tall, and it is constructed of high-strength reinforced concrete, with a 6.35 mm internal carbon steel liner. Both the internal and external surfaces are coated with several layers of epoxy. The epoxy coats and steel inner jacket serve as internal corrosion barriers. Studs welded to the steel jacket ensure that there will be no separation of the concrete from the steel liner while the concrete is curing. A special positive-pressure hydrophobic gas ventilation system is in ..lled in the lid. The lid projects above the lifting lugs so that HICs can be easily stacked to a maximum of six high, with possible disposal to depths of 27 m.

Disposal Demonstration—The HIC will undergo extensive evaluation at the INEL during 1983 to determine its capability to safely contain an EPICOR II liner for 300 years. Results of this evaluation testing will support negotiations for approval to bury an EPICOR II liner in a HIC at a commercial waste burial site. Such a burial demonstration would include retrieving an EPICOR II liner from storage, sealing it in the HIC, transporting the container with the liner to the commercial burial facility, and disposing of the container with its contents.

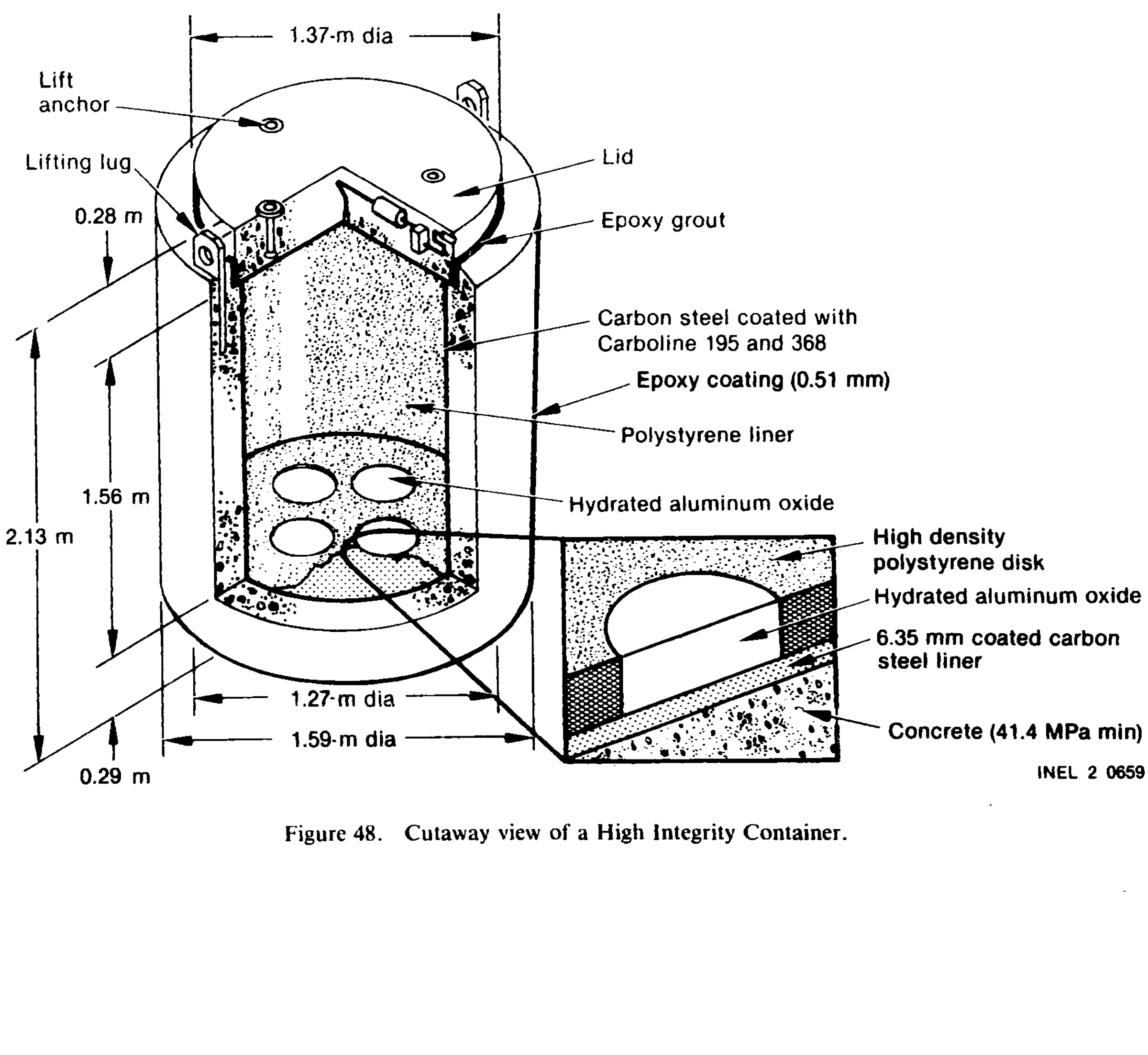

Figure 48. Cutaway view of a High Integrity Container.

REACTOR EVALUATION

Throughout 1982, significant milestones in instrument studies, accident assessment, and waste management all brought the progress of the TMI-2 recovery closer to its goal: characterization, disassembly, defueling, and removal of the damaged reactor internals and core. The Reactor Evaluation Program focuses on acquiring data on the TMI-2 core and reactor internals to develop technology for reactor disassembly and defueling so that the core can be removed for off-site examination. Data on the reactor core and internals are essential for developing and assessing accident prediction codes, providing a basis for rulemaking evaluations, and aiding in evaluations of reactor design and operational safety. The techniques and equipment developed under the Reactor Evaluation Program also provide basic technology for recovery of a damaged reactor core.

The Reactor Evaluation Program is divided into three major tasks: Pre-Head Removal Core Damage Assessment, Reactor Vessel Head Removal, and Plenum and Fuel Removal. Under the Pre-Head Removal Core Damage Assessment task, engineers used a closed circuit television camera to gain the first visual access to the reactor vessel and conducted preliminary examinations of the upper reactor internals and core. The information acquired through these examinations provided the first direct evidence of the actual core condition and helped researchers evaluate the various accident damage assessments which preceded visual access. Information gained from the Core Damage Assessment task also supported reactor vessel head removal planning and future core damage research and tooling development by establishing what reactor internals conditions will be encountered once the reactor vessel head is removed.

The Reactor Vessel Head Removal task began performing the engineering and technical evaluations necessary for recommending the best method for accomplishing safe reactor vessel head removal. Under this task, engineers planned to conduct prelift photo-visual inspections, radiation surveys, and prelift flushing operations in and around the vessel head area. Equipment and procedure modifications that may be required to safely move the vessel head to the head storage stand were also investigated.

The Plenum and Fuel Removal task concentrated on developing and evaluating the methods and equipment required for removing the plenum assembly. The task also began studies to evaluate the safest methods for defueling the reactor and storing, handling, and transporting damaged as well as intact fuel assemblies. Preliminary plans for acquiring invessel data during and after plenum and fuel removal were also evaluated in 1982.

Accomplishments

During the reporting period, Reactor Evaluation Program tasks have made significant progress in characterizing the conditions inside the damaged Unit 2 reactor. These accomplishments, highlighted in the following paragraphs, provide a sound basis for conducting future defueling operations and also contribute to development of technology for defueling severely damaged light-water reactor cores.

Pre-Head Removal Core Damage Assessment. The preliminary project of this task, special tooling required for invessel inspection before head removal, was completed early in 1982. This tooling is needed for Control Rod Drive Mechanism (CRDM) removal, invessel inspection, and contingency leadscrew separation. Some of this equipment was used during the first visual examination, or Quick Look, inside the TMI-2 reactor.

Quick Look Examination—In one of the most significant accomplishments of 1982 engineers obtained initial visual access to the TMI-2 reactor vessel during a series of camera inspections called the Quick Look examination. The Quick Look examination inspected control rod guide tubes, portions of the reactor plenum and upper grid, the tops of fuel assemblies, and, where the fuel assembly upper end fittings were missing, the reactor core. This inspection provided the first visual information on the condition of the core and upper reactor internals and was of significant interest to both researchers and recovery planners. The postulated core damage, based on camera inspections and information collected from core probes, is illustrated in Figure 49. The results of the Quick Look established a significant point of

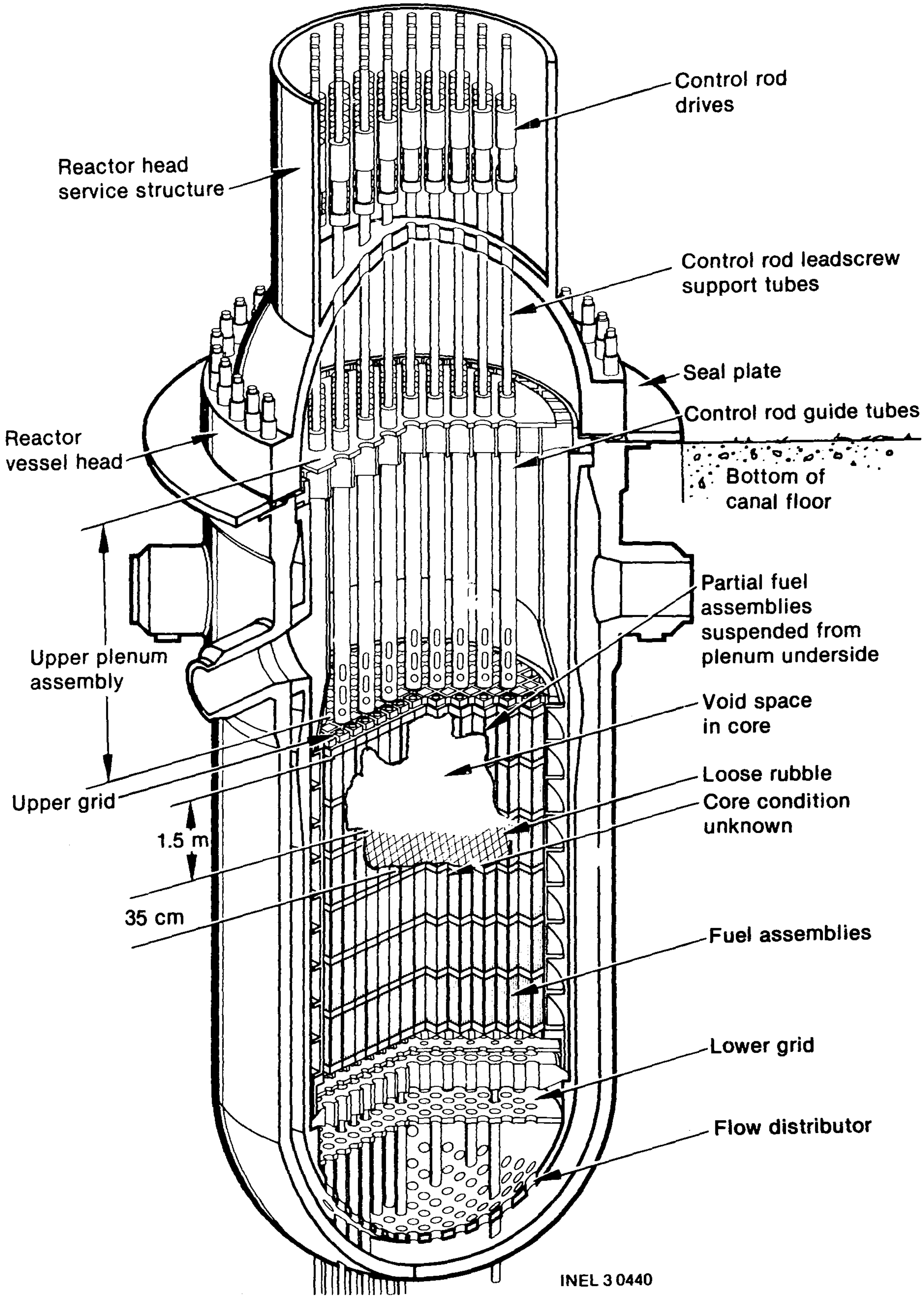

Figure 49. Structural components of TMI-2 core showing postulated damage area.

reference against which the range of core condition and accident sequence estimates can be compared.

The Quick Look examinations were conducted over a three-week period and examined reactor internals at the three core locations shown in Figure 50. These locations are identified by their grid coordinates, B-8, H-8, and E-9. While each Quick Look involved slightly different procedures because of the varying internals conditions at the three locations, the Quick Look examinations as a whole provided a composite picture of the TMI-2 core and internals. The results of the individual examinations are not treated separately, but rather are compiled into one overall discussion on what the Quick Look examinations revealed about the damaged reactor.

During the Quick Looks, a radiation-tolerant, underwater closed circuit television (CCTV) camera, pictured in Figure 51, was inserted into the reactor through an opening created by the removal of a CRDM leadscrew. The camera access

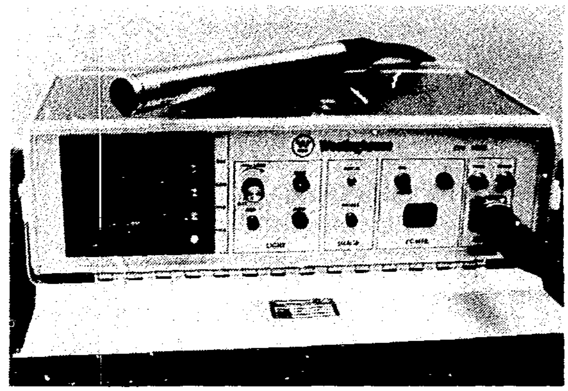

Figure 51. Quick Look inspection camera and control unit.

route is shown in Figure 52. The camera was manipulated using the camera power cable and a separate articulating cable attached to the camera tip.

The camera inspection started in the reactor plenum (see Figure 49) in the general region of the

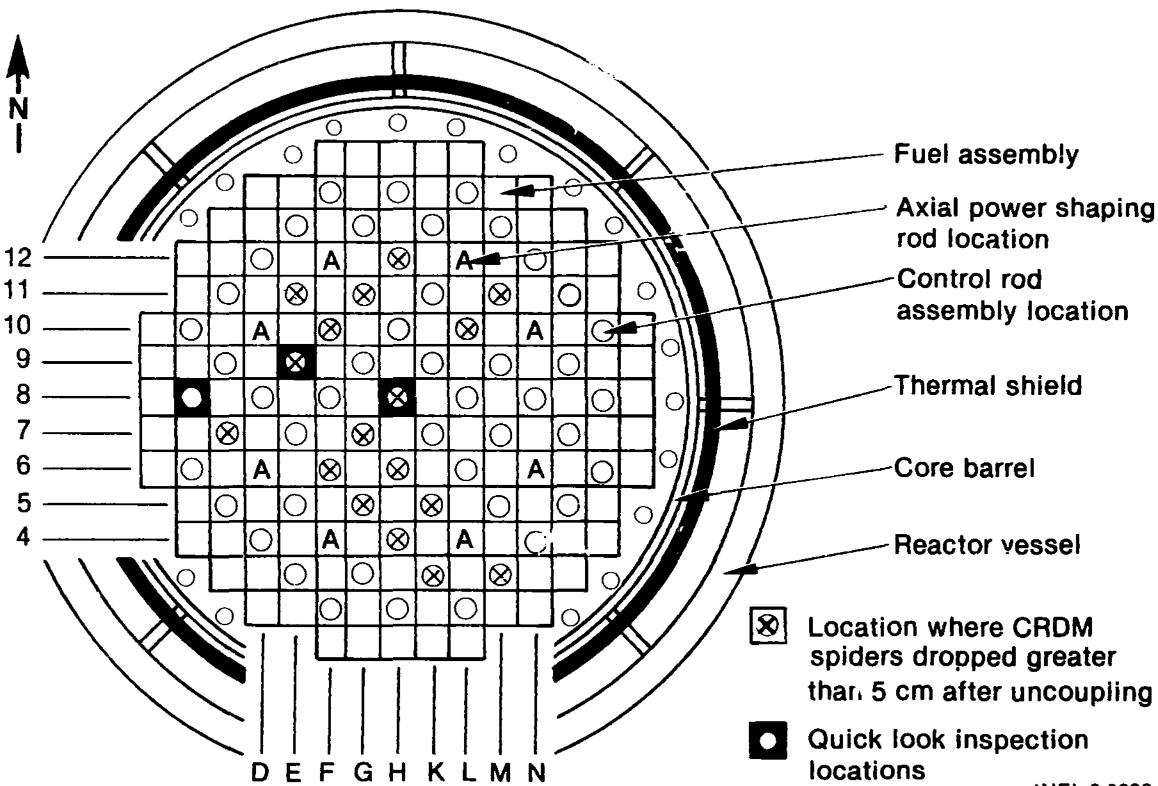

Figure 50. Unit 2 core cross section showing location of Quick Looks and CRDMs.

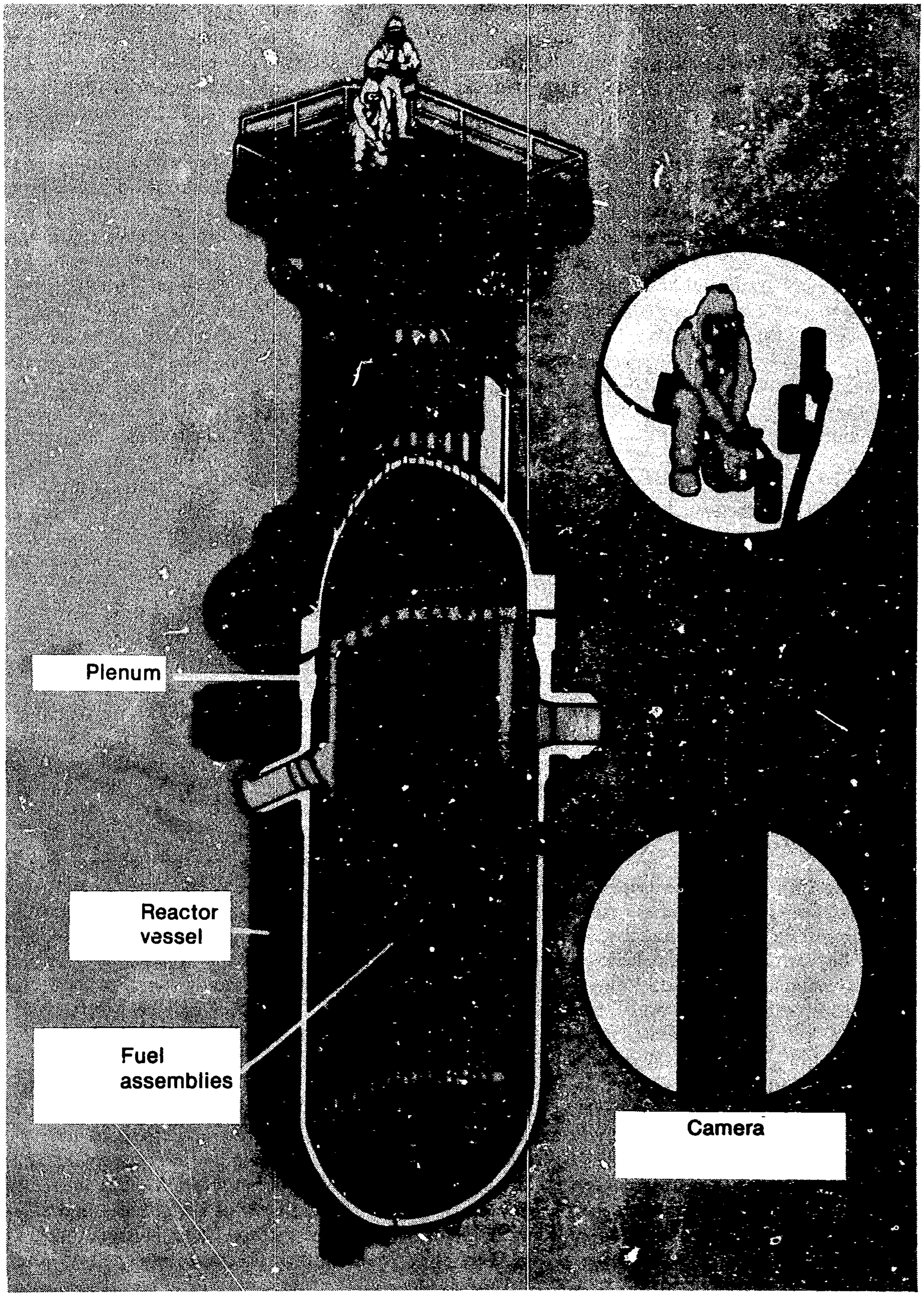

Figure 52. Quick Look camera access path into reactor internals and core.

tenth control rod guide tube support plate, which is shown in a detail of a control rod guide tube in Figure 53. Depending on specific conditions at each location, the inspection proceeded as described below.

Following inspection of the tenth support plate, the camera examined the upper end fitting of the fuel assembly directly below the leadscrew opening of the control rod guide tube assembly. (The location of an upper end fitting relative to a fuel rod assembly is shown in Figure 54.) Once the camera inspected the upper end fitting area at the core location through which it had been inserted, technicians manipulated the camera out through the guide tube flow holes (see Figure 53) to inspect the upper end fittings of the four adjacent fuel assemblies.

These inspections were followed by detailed examinations of the end fitting directly below the leadscrew opening and the tenth support plate and its guide tubes. If the inspection indicated this region had been subjected to high temperatures, the two support plate brazements directly above the tenth support plate (see Figure 53) were also inspected.

If, during these examinations, an end fitting was missing, technicians lowered the camera through the end fitting opening to determine the location of the top of the core. The camera then was used to inspect the core region and examine actual conditions there.

During the first camera inspection, at center-core location H-8, the water was quite cloudy, causing visibility to be limited to approximately 8 cm beyond the camera lens. A significant improvement in water clarity during the second camera inspection, at B-8 and E-9, increased the visibility to about 23 cm beyond the lens; however, the intensity of the available light continued to limit the visibility range, as was the case during the third camera inspection, also at location E-9. This lighting condition caused objects to be indistinct, but still discernible, at distances of 30 to 60 cm from the camera lens. All the photographs obtained from the Quick Look exhibit this visibility problem. In addition, as a result of the extreme magnification provided by the camera, objects shown in Quick Look photographs often appear much larger than their actual size.

Plenum and Upper End Fitting Examination—The first part of the Quick Look involved a detailed examination of the reactor plenum assembly. Overall, the plenum appeared to be intact and relatively undamaged. The interior surfaces of the CRDM guide tubes at locations B-8, H-8, and E-9 were thoroughly examined and appeared to be in good condition. Flakes of debris, such as shown in Figure 55, were observed on the top of nearly every horizontal surface; these flakes measured approximately 0.3 cm in diameter or less. The thickest layer of flakes seen was not more than 0.15 cm in depth. These layers were loosely deposited, since the motion of the camera in the water often disturbed the flakes. The thickness of the layers increased on surfaces closer to the core. The undersides of horizontal surfaces and the faces of vertical surfaces were clean and free of loose debris. The vertical surfaces of the CRDM guide tubes, split-tubes, and C-tubes were relatively free of debris near the top of the plenum, but had some slight deposit of material in the lower portion.

The camera saw many setscrews, which fasten the support plate brazements to the guide tubes (see Figure 53). All screw threads were clearly visible and intact. Varying amounts of fine surface deposits, as well as some small flakes, were observed on the top surfaces of the screws, as can be seen in Figure 56. All support plates inspected were unbroken, free of distortion, and generally undamaged.

The junctions where the C-tubes and split-tubes are brazed to the support plates appeared to be normal, with no visible evidence of melting. The bottom end of one of the split tubes at the E-9 location, shown in Figure 57, appeared to have evidence of minor metal removal. However, some of the C-tubes only centimeters away were undamaged, as shown in Figure 58.

Since markings on the camera cable allowed the depth of the camera lens to be known to within 2.5 cm, it was possible to determine that the plenum grid plate had not sagged at any of the three examination locations. The brazement support plates, ends of the C-tubes and split-tubes, grid plate surfaces, and pressure pads were all found in their normal locations.

At core-center position H-8 the entire upper end fitting was missing, as were all adjacent ones;

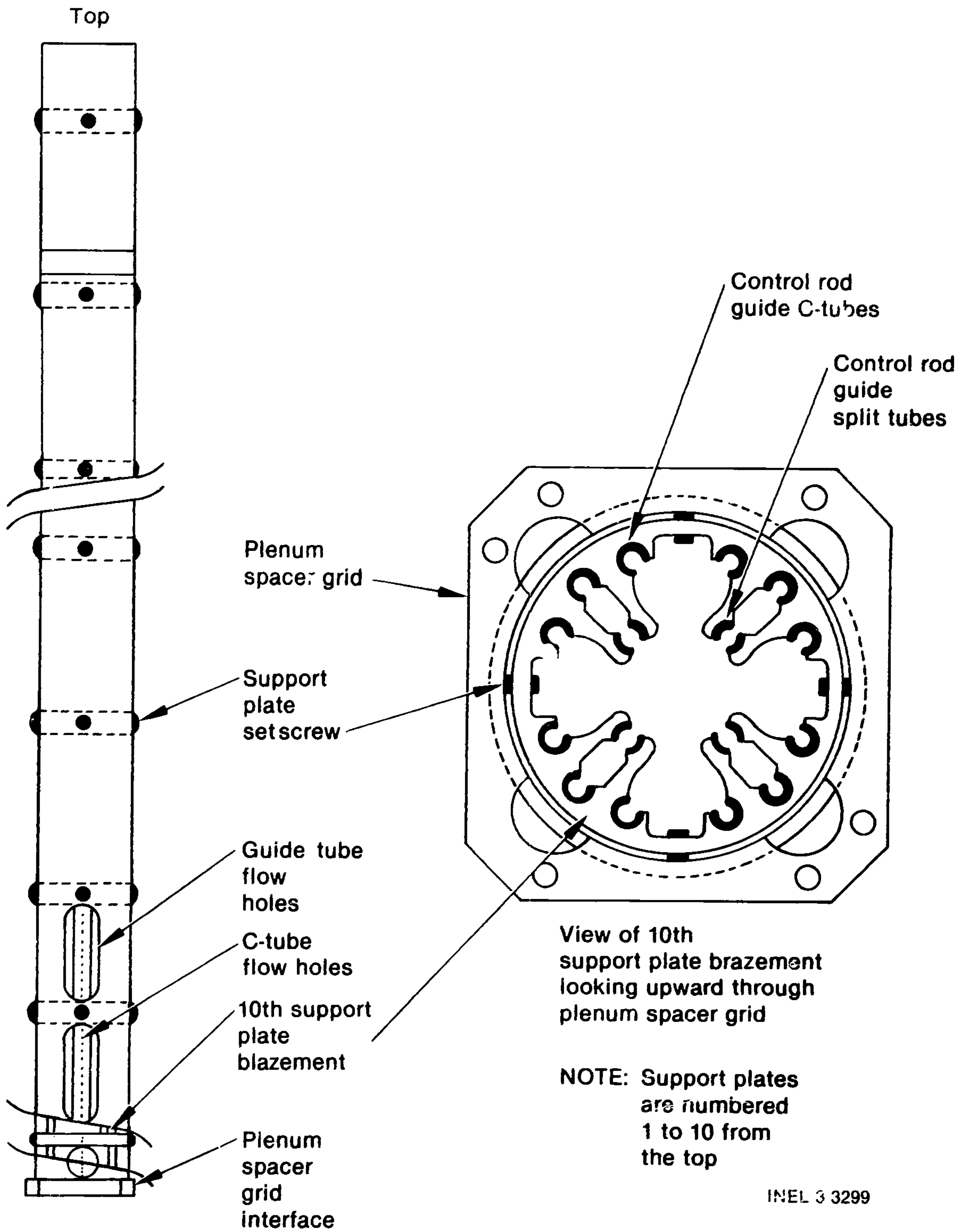

Figure 53. Control rod guide tube assembly with detail of tenth support plate.

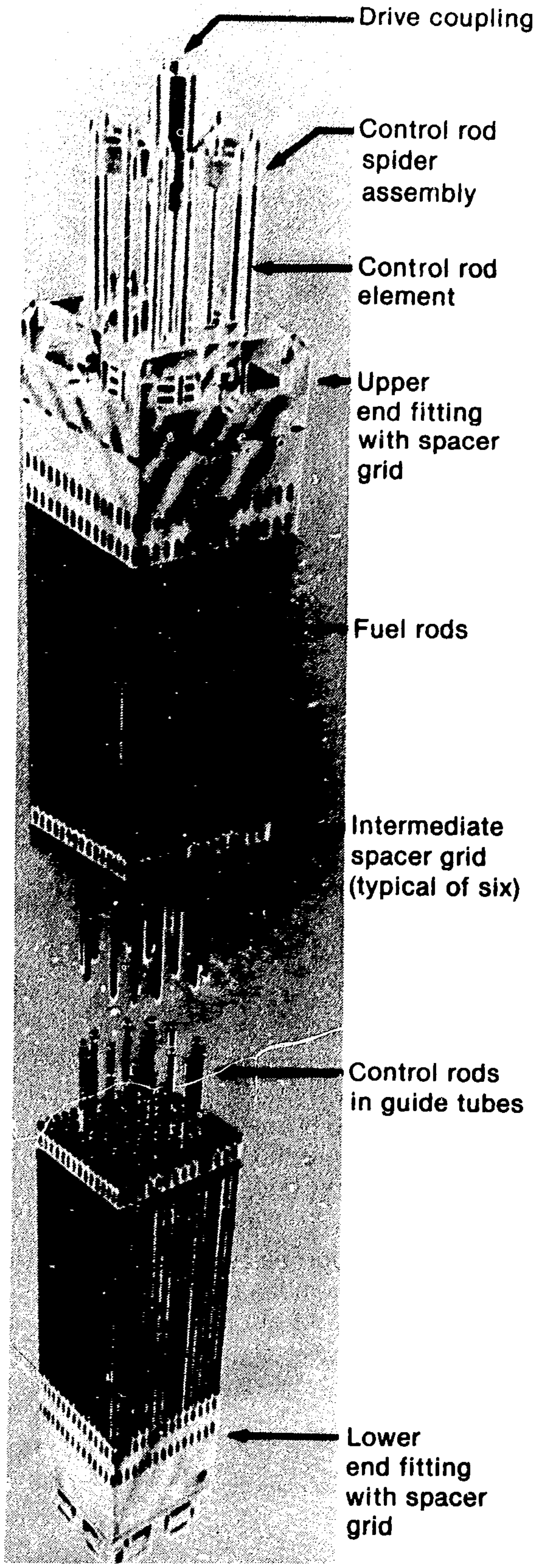

Figure 54. Components of typical fuel assembly.

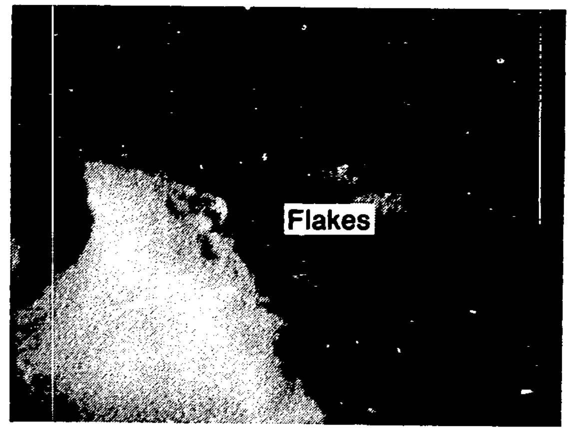

Figure 55. Layers of flakes on top of horizontal surface inside the Unit 2 reactor.

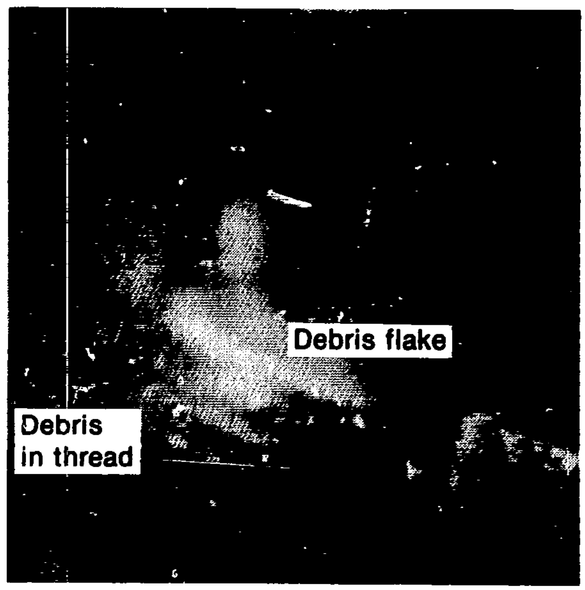

Figure 56. Support plate setscrew with debris in threads.

however at positions D-8, E-8, D-9, and E-9, the upper end fittings were still suspended from the plenum grid plate. The grill work from the E-9 upper end fitting was completely missing. The control rod spider, spring, and spring retainer were also missing. The grill work on each of the other three upper end fittings visible from location E-9 was present but partially melted as shown in Figure 59. The upper end fitting at position D-8 (visible from E-9) had other identifiable components, such as a spacer grid, stubs of control elements, and partial fuel rods,

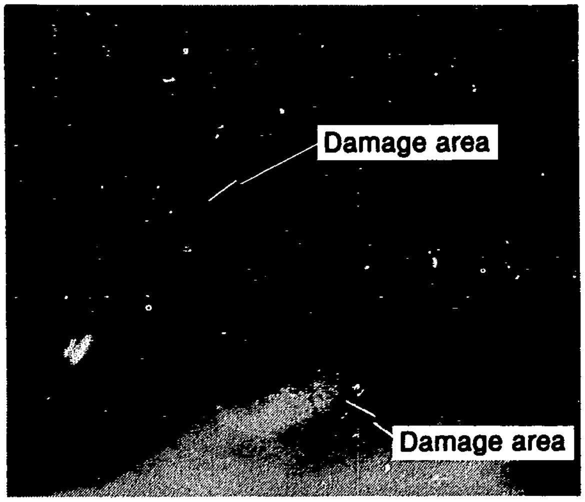

Figure 57. Bottom end of split tube at location E-9 showing areas where metal appears to be missing.

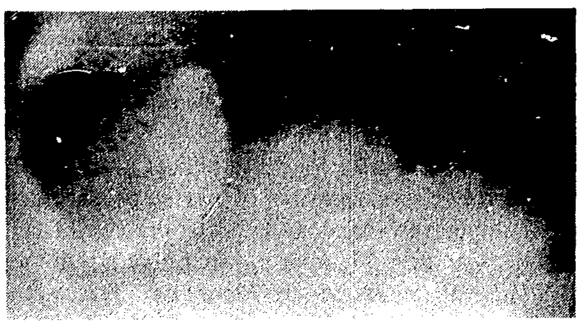

Figure 58. C-tubes adjacent to split tube at location E-9.

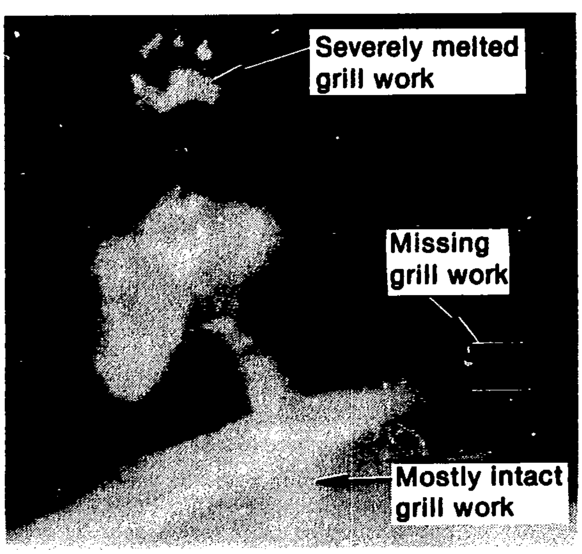

Figure 59. Damage to upper end fitting grill work.

suspended from it. A control rod element stub, showing evidence of melting, and partial fuel rods are shown in Figures 60 and 61, respectively.

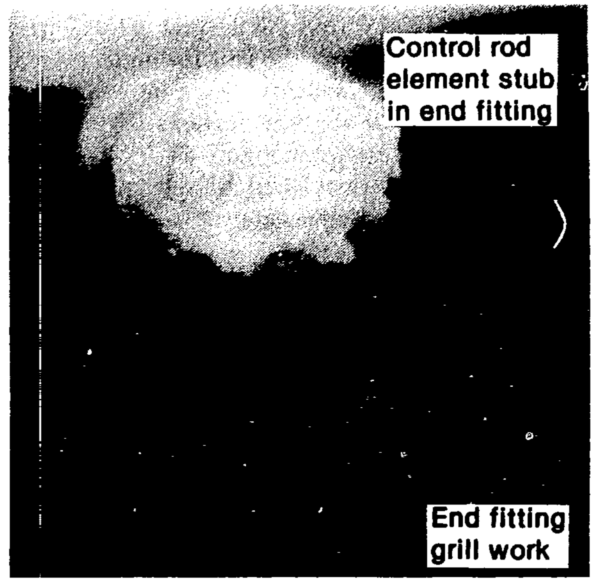

Figure 60. Control rod element stub in upper end fitting grill work.

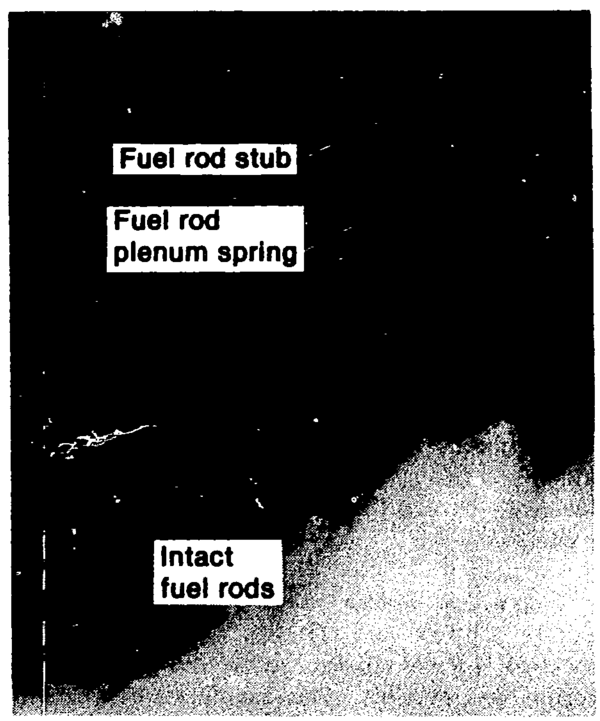

Figure 61. Partial and intact fuel rods suspended from upper end fitting grill work.

The insides of the E-9 mid-radius position upper end fitting were scanned using the camera's right angle lens. The end fitting appeared to be in its normal position with respect to the grid structure. Metal chips and debris, shown in Figure 62, were found in the small space between the centering tabs on the end fitting and the grid. This debris may be what is holding the end fitting in place, since the remainder of the fuel element was missing. Some areas of the top portions of the E-9 upper end fitting, shown in Figure 63, look melted and shorn as metal does when it has been cut by a torch, while adjacent areas appear to be in the as-manufactured condition.

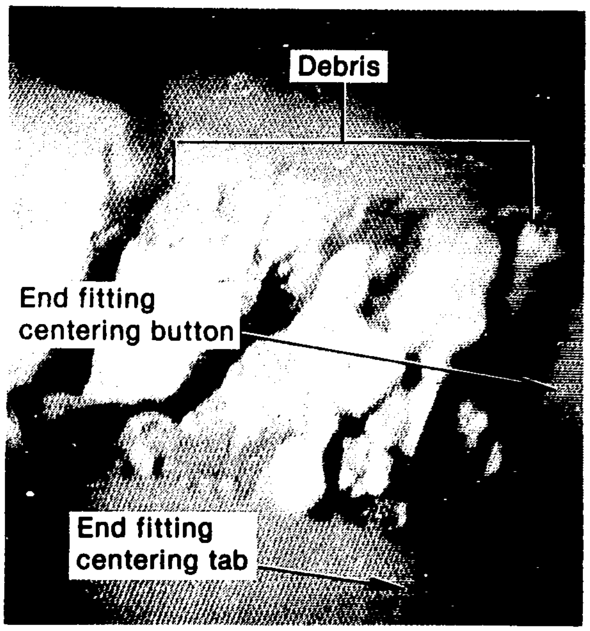

Figure 62. Metal chips and debris between centering tabs of an upper end fitting.

The upper end fitting spider assemblies were encountered in their normal positions at B-8 and one adjacent location. This indicates that the upper end fittings and the fuel assemblies in these locations were sufficiently intact to support the spiders.

Core and Rubble Examination—Because the entire upper end fitting at core-center location H-8 was missing, access to the active core region was possible. This examination revealed that a void exists in the upper central portion of the core. At position H-8, the center of the core,

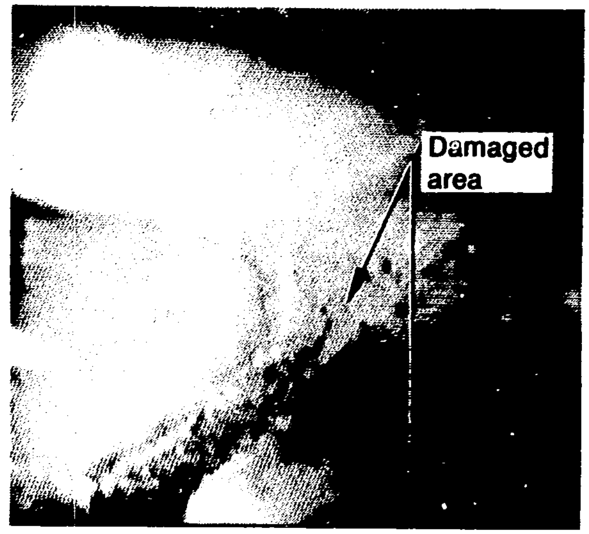

Figure 63. Damage to E-9 upper end fitting looks like metal after it has been cut by a torch.

the void extends from the bottom of the plenum to the top surface of a rubble bed, approximately 1.5 m below the bottom of the plenum. This void was formed by the redistribution of fuel from at least the nine central fuel assemblies. The rubble bed in this region consists of fine granular particles, angular in shape, and approximately 0.3 cm in size. An unusual feature of the rubble bed at the H-8 location is shown in Figure 64. The large potato-shaped object in the center of the figure is

Figure 64. General appearance of the rubble bed at core-center location H-8. Potato-shaped object in center of picture is actually only 0.32-cm in diameter.

actually only about 0.32-cm in diameter and is surrounded by the granular particles characteristic of the rubble bed at this location. During this examination no recognizable shapes could be identified except for the control rod spider assembly shown in Figures 65 and 66. It is believed that this is the H-8 spider assembly which fell into the rubble bed when its leadscrew was uncoupled to provide access for the Quick Look camera.

Since the end fitting grill work at mid-radius location E-9 was also missing, a second access to the active core region was obtained at that location. The void in this mid-radius region also extends approximately 1.5 m below the bottom of the plenum. The general appearance of the rubble bed in this region was considerably different than what was observed at the H-8 core-center location. The rubble bed was comprised of much larger pieces and numerous recognizable shapes, such as those shown in Figures 67 through 69. Rotation of the camera 360 degrees, as was done at position H-8, was not possible since the obstructions prevented the camera from turning on every attempt. Stubs of fuel rods, such as the

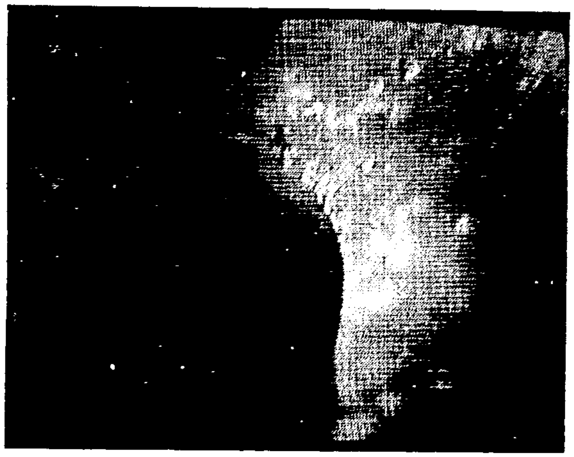

Figure 65. Portion of core-center location H-8 control rod spider assembly on top of the rubble bed.

Figure 66. Individual control element nut on H-8 control rod spider assembly on top of the rubble bed.

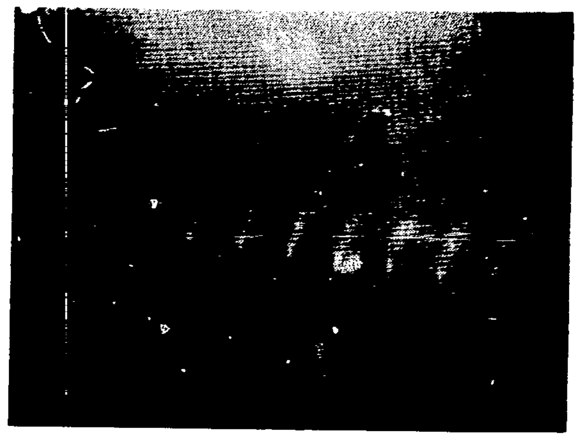

Figure 67. Pellet hold-down spring on top of rubble bed at location E-9.

Figure 68. Believed to be the end of a burnable poison pellet at location E-9.

Figure 69. Unidentified rod on top of rubble bed at location E-9.

one shown in Figure 70, were also protruding upward from the rubble at position E-9 from position E-7. A "forest" of rods could be seen looking radially outward toward the west edge of the core about 0.5 to 0.6 m from the camera (see Figure 61). These stubs were suspended from the remains of the upper end fitting that was still in place. All these observations led researchers to conclude that the core void area did not extend much further outward than the core mid-radius point.

During the H-8 and E-9 inspections, the rubble bed was probed by inserting a 1.3-cm diameter steel rod into the reactor vessel through the CRDM guide tube until the probe came in contact with the rubble. The rod was then rotated and allowed to penetrate the debris to a depth of 35 cm, where it was stopped by an unyielding obstruction. The rod penetrated the rubble bed to the same depth at both locations.

Preliminary Conclusions—Following the Quick Looks, a Videotape Review Group convened to review the Quick Look tapes and summarize their observations. Experts from General Public Utilities Nuclear Corporation (GPUNC), MPR Associates, Babcock & Wilcox, Westinghouse Hanford Corporation, Bechtel Northern, and

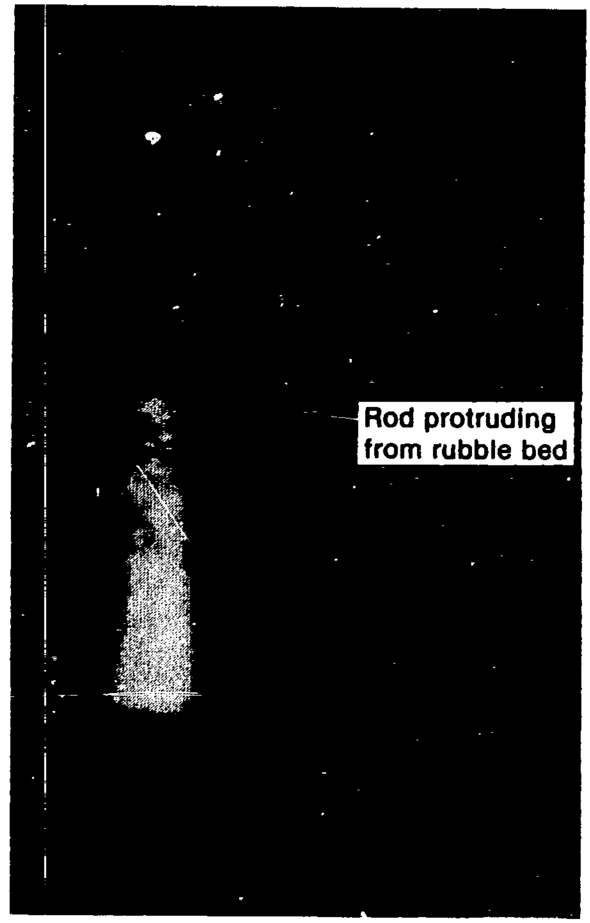

Figure 70. Rod protruding from rubble bed.

EG&G Idaho participated. Their preliminary conclusions pertain only to those areas explored during the Quick Look camera inspections. These preliminary conclusions are as follows:

- A significant number of the Unit 2 fuel assemblies sustained considerable damage during the 1979 accident, causing some of them to form a rubble bed.

- A void approximately 1.5 m deep exists in the upper portion of the core and extends from the core center line to about midway to the outer edge.

- At two points, one at the center H-8 position and the other midway to the outer edge at the E-9 location, the rubble bed is composed of loose material to a depth of at least 35 cm, indicating that in those two areas and to a depth of 35 cm the rubble is not a fused mass.

- No evidence of melted fuel pellets was found; however, no general conclusions have been made as to whether or not any of the fuel pellets within the core have melted.

- As expected, there was some evidence of partial melting of nonfuel material in components that have melting points much lower than uranium oxide fuel.

- The plenum assembly appeared to be substantially undamaged. At one point, located between the center and the periphery of the core, portions of fuel assemblies were seen hanging from the lower plate of the plenum.

These preliminary conclusions, when taken together with other core damage estimates, provided engineers with a more accurate description of core damage than they had before the Quick Look. For example, from the hydrogen burn studies conducted under the Radiation and Environment Program, engineers calculate that approximately 45% of the core zirconium inventory reacted to produce hydrogen. This and other information continue to contribute to the overall efforts of pre-head removal core damage assessment.

Reactor Vessel Head Removal. During 1982, several Reactor Vessel Head Removal task activities were completed. Included in these activities are CRDM uncoupling and Axial Power Shaping Rod (APSR) uncoupling and parking. In addition, a Head Removal Task Group was formed to conduct technical evaluations and to recommend the best method for accomplishing head removal.

CRDM and APSR Uncoupling—As a follow-up to the Quick Look inspections, leadscrew uncoupling was attempted on all the leadscrews not uncoupled for the Quick Look camera inspections. The uncoupling consisted of a series of operations that attempted to disconnect drive mechanism leadscrews from core control elements. Information gained from this activity was to be used to determine what special tooling might be required to support reactor vessel head removal. CRDM uncoupling and core mapping were planned to obtain the earliest possible data on the condition of the CRDMs.

These operations aimed to uncouple CRDM leadscrews from control rod assemblies to permit leadscrew parking and CRDM removal (if required). In the uncoupling process, engineers planned to obtain data that can be used to assess the condition of the core, and obtain data on the condition of individual CRDMs in order to develop the most efficient plan for contingency uncoupling and parking.

Specific data that engineers gathered during uncoupling helped in the overall evaluation of CRDM and core conditions. These data consisted of:

- Four pressure readings from the hydraulic pressure gauge on the alternate uncoupling tool

- The extent of leadscrew rotation and any indication of cessation of rotation felt by the operator

- Other information, such as the operator's feel for leadscrew rotation, the inability to rotate the leadscrew back to the coupled position, and movement or drop of the control element spider.

In order to allow access for the Quick Look camera, three CRDM leadscrews were uncoupled and removed using normal uncoupling methods. An alternate uncoupling method called the "torque tube" method was used on the remaining CRDMs and the APSRs to reduce the time and man-rem expenditures associated with normal CRDM uncoupling and parking. Torque tube uncoupling allows the leadscrew to be disengaged from the control rod spider without disturbing the leadscrew-to-torque-taker connection. This is accomplished by uncoupling the CRDM torque tube from the motor tube and rotating the internal assembly until the leadscrew disengages from the control rod spider. Figure 71 shows technicians in the Reactor Building uncoupling a CRDM using the alternate uncoupling torque tube method.

Key data from which core damage can be inferred are the locations where control rod spiders dropped greater than 5 cm during uncoupling. A spider drop of this distance might indicate that the fuel assembly end fitting is not at its normal elevation, implying that a core void exists below that location. However, during the

Figure 71. Technicians performing alternate torque tube uncoupling method on a control rod drive mechanism.

Quick Look inspections, end fittings were observed stuck in the upper grid above the core void; consequently, failure of the spider to drop more than 5 cm does not exclude the possibility of a core void below that location. Refer to Figure 50, which shows those locations where the spider dropped more than 5 cm.

All CRDMs and APSRs were uncoupled during 1982. Now that they are uncoupled, they must also be parked before reactor vessel head removal can occur. While the eight APSRs have already been raised to their parked positions using their drive mechanisms, the CRDM leadscrews will be parked using a hoist after the reactor missile shields are removed. Testing to date has confirmed that all CRDM leadscrews are free and movable, and that contingency techniques, such as leadscrew cutting, will not be required to complete the parking operations.

In November 1982, the leadscrew removed from the H-8 position for the Quick Look inspections was cut into three short sections and removed from the Reactor Building. One section remained on-site for examination in the Mobile Response Laboratory described in the Radiation and Environment section of this report. The other sections were shipped to laboratories for examination. The specimens had contact radiation readings ranging from 30 to 60 R/h gamma, primarily due to ^{137}Cs. Analyses of these specimens will be completed early in 1983 and will provide additional data for reactor disassembly planning. All remaining sections of the leadscrews removed for the Quick Look will be shipped off-site for analyses during 1983.

Reactor Vessel Head Removal Task Group Activities—In June 1982, a task group was formed to conduct technical planning and to recommend a method for removal of the TMI-2 reactor vessel head. Members of the group include representatives from Babcock & Wilcox, Bechtel National, Bechtel Northern Corporation, EG&G Idaho, Inc., and GPUNC. The planning evaluation included development of functional requirements, and evaluation of alternate head lift methods. The recommendations of this task group are summarized in the following paragraphs.

The task group recommended that the preferred reactor vessel head removal technique would be the "Dry Method." This method is essentially the normal refueling method. The alternative to a dry head lift is the "Wet Method," which includes provisions for flooding the refueling canal around the vessel head during head lifting to provide radiation control. If the wet method is employed and the canal is flooded, the task group further recommended that the Submerged Demineralizer System (SDS) be modified to increase processing capacity to further control radiation levels and reduce the cleanup time of the canal water. The task group also recommended that technicians should:

- Install a refueling canal temporary fill and drain system to overcome inaccessibility and maintenance problems associated with the current canal fill and drain valves

- Modify the canal seal plate to ensure that long-term sealing could be provided over the space between the canal and the reactor vessel

- Place the internals indexing fixture, covered with a temporary cover, over the reactor vessel to serve as a shield tank which will allow water to fill up the area above the vessel after head removal.

Based on the task group recommendations, design and procurement actions were initiated to modify the SDS system and canal seal plate, and to provide fill and drain capability for the refueling canal. Procurement of the required special tools and equipment was also initiated. These specialized tools and equipment include the tools for removing the reactor head hold down nuts and studs, specialized handling and rigging items, a 5-ton gantry crane, and electrical components for a temporary power distribution system in the

Reactor Building. These procurement actions were necessary to achieve the target schedule for head removal in the summer of 1983.

Underhead Radiation Surveys—On December 16, 1982, a "Quick Scan" or underhead radiation survey was performed to determine radiation levels inside the reactor. The Quick Scan was accomplished by lowering an ionization chamber through the removed leadscrew openings at CRDM positions B-8 and E-9. The detector, calibrated in excess of 100,000 R/h with a ^{60}Co source, obtained data at elevations from just under the head at the 327-ft elevation down to the top of the plenum at the 322-ft elevation. The data collected by the Quick Scan are shown in Table 22. GPUNC's preliminary analysis of the data indicates that the radiation levels during head removal may be greater than previously estimated. Further underhead data gathering and analysis will be performed as expeditiously as possible in early 1983 with the reactor vessel full of water, as well as partially drained. In addition, plans also call for removing one or more of the CRDMs to improve access to the top of the plenum area. This improved access will facilitate debris sampling and video camera inspection as well as underhead decontamination flushing and other dose reduction measures that may be required.

Plenum and Fuel Removal. Efforts toward removal of the reactor plenum assembly and the damaged core itself consisted of two preliminary engineering tasks in 1982. Two task groups were formed. The first of these completed a general technical plan in June and the second task group issued a more detailed plenum removal plan in November.

Table 22. Quick scan underhead radiation levels, December 16, 1982

	Core Location	
Elevation	E-9 (R/h)	B-8 (R/h)
327 ft 7 7/32 in.	40	—
326 ft 6 in.	120	—
326 ft 5 3/4 in.	—	50
325 ft 6 in.	170	100
324 ft 6 in.	200	200
324 ft 4 in.	240	220
324 ft 0 in.	320	340
323 ft 6 in.	550	600
323 ft 0 in.	540	540
322 ft 6 in.	530	540
322 ft 0 in.	520	580

The first task group, the TMI-2 Reactor Disassembly and Defueling Task Group, developed the technical data base and the functional requirements and design criteria for the tooling system necessary to remove the TMI-2 reactor plenum assembly and core. Task group responsibilities covered the following areas:

- Design bases, techniques, and tooling concepts for nondestructive and destructive plenum removal

- Design bases, techniques, and tooling for fuel removal, including a fines and debris vacuum system, viewing systems, and fuel storage and shipping canisters.

The results of the task group's work include a detailed evaluation of the potential plenum conditions to be considered during tooling design and planning for nondestructive removal. The group identified special plenum removal equipment and tools as well as an assessment of the probable need for these items. The group also recommended data acquisition tasks to add a degree of confidence to this probability assessment. Because the need for destructive removal has not yet been determined, the task group did not provide a detailed destructive removal plan. However, several destructive removal assessments and possible alternatives were reported by the group.

The task group work also included an evaluation of the potential core conditions to be assumed on the basis of the Quick Look which will impact tooling design, and a general plan for treating each of these conditions. Fuel removal tooling system alternatives were discussed and evaluated in five key categories. These categories are radiation exposure, success potential, operating efficiency, capital cost, and schedule impacts. The first task group also provided recommendations for further data acquisition and development.

A second task group, called the Plenum Removal Task Group, was formed in late August 1982 at Babcock & Wilcox. The purpose of this task group is to review the previous plenum removal planning in light of current information and to update that planning where appropriate.

The task group evaluation included activities associated with plenum assembly removal, starting from the conclusion of head removal and continuing through preparations, lifting, storage, and off-site disposal of the plenum. These evaluations provided predictions of the expected physical and radiological condition of the plenum and identified the base plan sequences for intact removal. Included in the base plan sequence are descriptions of the necessary steps and associated special tooling, as well as a comparison with the sequence presented in the preliminary design study. The task group also identified the base plan sequence for plenum disassembly in place, including descriptions of the necessary steps and the associated special tooling requirements.

The task group concluded that work on the intact-plenum removal plan should continue to develop and define the necessary activities. The major elements of the recommendations in the intact-plenum removal plan are:

- The internals indexing fixture should be installed and filled with water to approximately 1 m above the reactor vessel flange to provide shielding, and should be fitted with a cover that will serve as a work platform.

- Key inspections should be performed as soon as possible after the head is removed to determine actual debris deposition, cleaning requirements, and any possible damage or distortion to the plenum.

- Dipsticks equipped with removable slide hammers should be installed through the guide tubes and around the cover plate to monitor plenum-core separation during the initial lift and to separate fuel assembly end fittings from the plenum where required.

- The initial lift of the plenum should be performed with hydraulic jacks.

- The canal should remain dry to permit work on and around the indexing fixture for initial plenum lift by jacking. The canal should then be flooded prior to removal of the plenum assembly from the reactor vessel.

- The polar crane should be rigged to the plenum after the initial lift to complete lifting the plenum from the vessel into a flexible confinement bag for transport.

- The plenum should then be moved to the shallow end of the canal, placed in a storage container, and shielded as necessary.

- The use of video support should be limited to confirmation and troubleshooting.

Based on the two task groups' conclusions and additional planning discussions with GPUNC, preliminary engineering was initiated in the major task areas. The engineering tasks include refurbishment of the 'A' spent fuel pool, modification of the fuel transfer system and canal cleanup system, design of plenum and fuel removal tooling, and design of fuel and debris canisters and racks. Completion of the preliminary engineering on these tasks in early 1983 will allow programmatic decisions to be made that support critical items associated with removing the plenum assembly and defeuling the reactor.

Full Scale Defueling Test Assembly. During 1981, Technical Information and Examination Program contractors developed a conceptual design of a full scale Reactor Defueling Test Assembly which was reviewed and approved by DOE in 1982. The design will be used to complete functional requirements and to develop the actual assembly. This unit, or a variation, will be constructed in the turbine building at TMI-2 during 1983.

The concept includes a tank simulating dimensional constraints in the defueling canal and reactor vessel. This tank can be flooded and will include the necessary water cleanup systems and special lighting. It will also contain simulated reactor components required for operational checkout during reactor disassembly.

The TMI-2 Defueling Test Assembly will support worker training and equipment and procedure verification, and will aid in resolution of problems that may arise during reactor disassembly and defueling. Previous experience at TMI-2 has demonstrated the value of such training and testing in keeping worker exposures as low as reasonably achievable, and in ensuring smooth, efficient recovery operations.

CONCLUSION

Since its establishment the Technical Information and Examination Program has contributed to significant advances in the cleanup of the damaged plant at Three Mile Island Unit 2. Just as significant have been the Program's advances in developing technology of lasting benefit to the entire nuclear industry. Careful examination of electrical safety systems has produced and will continue to yield information leading to improved testing and fabrication standards for nuclear industry instrumentation and electrical components. Continuing development of mass balance and source term information will increase understanding of fission product transport and deposition during an accident. Analysis of the hydrogen burn event at TMI-2 will contribute to a better understanding of such occurrences and how to mitigate them in the future. Data gathered and analyzed in the mass balance, source term, and hydrogen burn studies will provide the industry with a point of reference for future computer code validation.

Regular shipments of wastes from water processing systems will have removed all 50 EPICOR II prefilter liners and 12 highly loaded Submerged Demineralizer System waste liners from the Island by the end of 1982. The ensured safety of both sets of shipments depend on technology developed with TI&EP research. The catalyst recombiner which suppresses radiolytic gas generation in SDS waste shipments will ensure safe shipment of other high activity radioactive wastes in the future.

Such reactor evaluation techniques as camera inspection of the inside of the damaged reactor core have allowed extensive planning for future cleanup activities which focus on the disassembly and defueling of the core and reactor internals. Reactor evaluation and removal technology will advance the state of the art in damaged core handling procedures while also affecting a major milestone in the TMI-2 recovery.

The accomplishments of the Technical Information and Examination Program have played an important role in the progress of the cleanup while pursuing the Program's founding goals of obtaining, developing, and distributing vital information to the nuclear power industry. The Program's focus on taking advantage of the lessons to be learned from the TMI-2 accident will continue to ensure that as the cleanup progresses, the overall safety and effective application of nuclear power will progress with it.