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U.S. DEPARTMENT OF ENERGY AND GPU NUCLEAR CORPORATION RESEARCH AND DEVELOPMENT ACTIVITIES ON THREE MILE ISLAND UNIT TWO ANNUAL REPORT FOR 1985

Published April 1986

GPU Nuclear Corporation
Middletown, Pennsylvania 17057
and
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

AC01-84NE34110

Prepared for the Japanese Parties
In accordance with Agreement No. DE-GI-01-84NE-34109

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ABSTRACTS

Fuel and Waste Handling and Disposition Program

The major efforts for the Fuel and Waste Handling and Disposition Program were waste immobilization and core transportation. The Waste Immobilization Program oversaw the shipment of the last high-level submerged demineralizer system (SDS) liners to the Monitored Retrievable Burial Demonstration Program. One SDS vessel was buried in an instrumented concrete overpack and is being monitored. Research continued into the properties of EPICOR-II wastes.

Core transportation activities included drop testing of a quarter-scale shipping cask and a full-scale knockout canister. Procedures were developed for handling the shipping casks and the first cask was manufactured by Nuclear Packaging, Inc. The gas recombiner catalysts proposed for use in the defueling canisters were also tested.

The program to elute radioactive cesium from the makeup and purification demineralizers was completed.

Accident Evaluation Program

Analysis of core samples continued in an effort to complete a fission product inventory; however, samples of more and different regions of the core are required before the inventory can be completed. Analysis of the accident scenario is also continuing, based on the plant conditions, instrument histories, computer models, and severe fuel damage experiments.

Video inspections and debris sampling of several regions of the core were conducted and further data gathering, including use of a core drilling system, is planned.

Reactor Evaluation and Data Acquisiton Program

After a major engineering effort, the plenum assembly was lifted from the reactor vessel in May 1985. In October 1985, defueling operations began. The defueling system consists of a rotating work platform mounted above the reactor vessel, specially designed fuel canisters with a system to hold them in the reactor vessel, and various long-handled tools, including a vacuum system, for manipulating and loading core debris.

The Cables and Connections Program continued with the retrieval and testing of samples from the Reactor Building. Reports on the testing and evaluation of TMI-2 radiation, temperature, and pressure instruments were issued. Also, a calculational technique for determining hydrogen gas generation in sealed radioactive waste containers was developed.

Japanese Parties-Funded Activities

The Remote Reconnaissance Vehicle No. 1 (Rover) was used to collect sludge and concrete core samples in the Reactor Building basement. Further robotic applications are being engineered.

Ex-vessel fuel characterization was conducted and small quantities of fuel debris were identified in the makeup pump rooms and various primary system components.

Decontamination for dose reduction purposes was performed in support of ex-vessel fuel characterization and defueling activities in the Reactor Building. In the auxiliary and fuel handling building (AFHB), approximately 22,000 square feet of floor area was radiologically improved and the 282-ft 6-in. elevation was decontaminated to a level that allows entry without protective clothing.

SUMMARY

The year 1985 was significant in the cleanup of Three Mile Island Unit 2 (TMI-2). Major milestones in the project included lifting the plenum assembly from the reactor vessel and the start of operations to remove the damaged fuel from the reactor. This report summarizes these milestones and other TMI-2 related cleanup, research, and development activities. Other major topics include the following:

- waste immobilization and management
- fuel shipping cask delivery and testing
- sample acquisition and evaluation
- decontamination and dose reduction.

ACKNOWLEDGMENTS

The editor thanks Dave Scardena, EG&G Idaho, for coordinating the Idaho National Engineering Laboratory and EG&G Idaho input, and Carrie McArthur and Gordon Brown, EG&G Idaho, for coordinating production of this report.

Andrew P. Kelsey
Bechtel National, Inc.

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1. INTRODUCTION

The year 1985 was important in the cleanup of Three Mile Island Unit 2 (TMI-2). A major milestone of the program was attained in October with the start of defueling operations. This report summarizes the cleanup, research, and development activities related to TMI-2 during 1985. Other significant accomplishments include:

- Removing the plenum assembly from the reactor vessel in May
- Procuring and testing the fuel shipping casks
- Shipping the last submerged demineralizer system (SDS) liner from the processing of the original accident water in April
- Monitoring a buried SDS liner
- Examining the lower head region of the reactor vessel (beginning in February)
- Delivering core sampling equipment to the site in November
- Constructing the defueling system
- Delivering the first of the defueling canisters to the site in September

- Completing instrumentation and electrical projects on the resistance temperature detectors, dome area radiation monitor, pressure transmitters, and incore instrumentation

- Retrieving and testing 17 cable/connection samples from the Reactor Building

- Collecting sludge and concrete core samples in the Reactor Building basement, using a robot

- Locating and characterizing fuel that has been transported from the reactor vessel

- Continuing dose reduction and decontamination activities throughout the plant.

2. FUEL AND WASTE HANDLING AND DISPOSITION PROGRAM

2.1 Waste Immobilization

2.1.1 SDS Liner Monitoring

The last TMI-2 SDS liner for the Monitored Retrievable Burial Demonstration Program was safely shipped to Rockwell Hanford Operations at Richland, Washington on April 26, 1985. With the shipment of this liner, a total of 19 liners were accepted for use in the Department of Energy (DOE) zeolite disposition research and development program, which is now complete except for long-term monitoring of a buried liner. This SDS liner was buried in November 1984 in an instrumented concrete overpack to monitor actual burial conditions during the test period. Four parameters are being monitored: SDS liner internal pressure, shell temperature, overpack moisture, and fission product particulates. A summary of the conditions observed through September 30, 1985 follows:

- The two transducers used for monitoring SDS vessel pressure apparently lost their calibration, as a result of radiation damage, after being located on top of the SDS vessel for less than three months. The insulation material between the transducer bridge circuit and the stress beam is vapor-deposited silicon-monoxide. This insulation apparently lost a significant fraction of its dielectric strength during the first three-month period. However, pressure readings taken during March 1985 (and subsequently) show a remarkable change with improvement in stability after several months of widely varying data. Apparently, the resistance of the silicon-monoxide layer has increased to near its original value. While the accuracies of the absolute readings are still suspect, we now expect to gather valuable trend information over the coming months. Vessel pressure will continue to be closely monitored, and the stability of the apparent recovery of the transducers will be carefully evaluated.

- The fission product heating from inside the SDS liner has increased the temperature of the overpack and surrounding soil, as expected. Temperatures outside of the overpack (at 13.6 ft) and outside of the liner (at 13.3 ft) have steadily increased from 54.5°F and 107.2°F, respectively, in November 1984, to 83.8°F and 136.3°F in September 1985. The centerline temperature is expected to reach its peak in 1986. Temperatures outside the overpack are expected to continue to rise after the centerline peaks because of the thermal mass of the overpack and the large volume of soil being thermally monitored outside the overpack.

- Relative humidity in the overpack increased from 88.4% measured in February 1985 to 95% measured in March and May 1985. It decreased to 23% in July 1985 and then increased to 80% in August 1985. The unsaturated condition in the overpack indicates that moisture is not accumulating and, because of the fission product heating, moisture is not expected to accumulate.

- Gas samples were drawn from the bottom of the overpack and pulled through 0.45-micron filters. The amount of airborne fission products detected on the filter was essentially normal background.

2.1.2 EPICOR-II Research and Disposal of Waste

The Low-Level Waste Data Base Development--EPICOR-II Resin/Liner Investigation Program, funded by the U.S. Nuclear Regulatory Commission (NRC), is obtaining information on radioactive waste during NRC-prescribed tests. A paper by J. W. McConnell, R. M. Neilson, Jr., and R. D. Rogers entitled "Testing Waste Forms Containing High Radionuclide Loadings" was presented at Waste Management '86 in Tucson, Arizona, March 3-6, 1986. This paper describes the resin solidification task of that program, including the present status and results to date. An unusual aspect of this investigation is the use of commercial grade, ion exchange resins that have been loaded with over five times the radioactivity normally seen in a commercial application. That dramatically increases the total radiation dose to the resins. The objective of the resin solidification task is to determine the adequacy of test procedures specified by NRC for ion exchange resins having high radionuclide loadings.

A field study was designed to monitor the release (if any) of gamma-producing radionuclides from solidified EPICOR-II ion exchange resins. Both Portland Type I-II cement and Dow vinyl ester-styrene waste forms are being tested in lysimeter arrays located at Argonne National Laboratory in Illinois (ANL-E) and Oak Ridge National Laboratory in Tennessee (ORNL). The study is designed so that continuous data on nuclide release and movement, as well as environmental conditions, will be obtained over a 20-year period. Details on waste form formulation, lysimeter design, installation, instrumentation, and data acquisition and storage are provided in a report authored by R. D. Rogers, J. W. McConnell, E. C. Davis, and M. W. Findlay, entitled "Field Testing of Waste Forms Containing EPICOR-II Ion Exchange Resins Using Lysimeters," NUREG/CR-4498, March 1986 (Draft).

2.2 Core Transportation

Major accomplishments were made in preparation for shipping the damaged TMI-2 core from TMI to the Idaho National Engineering Laboratory (INEL), where it will be stored as part of the Core Examination Research and Development Program. Nuclear Packaging, Inc., (NuPac) completed the first NuPac 125B shipping cask with skid and rail car in December 1985. The second shipping cask with skid and railcar, along with the other miscellaneous cask handling equipment (i.e., vertical lift fixture and horizontal lift equipment), will be completed early in 1986. Equipment for dry loading canisters into the cask, also designed and fabricated by NuPac, will be completed in early 1986. This equipment includes the fuel transfer cask, mini hot cell, jib crane and its support tower, shipping cask loading collar, cask unloading station, and cask hydraulic lift assembly. Figures 1 and 2 show this equipment.

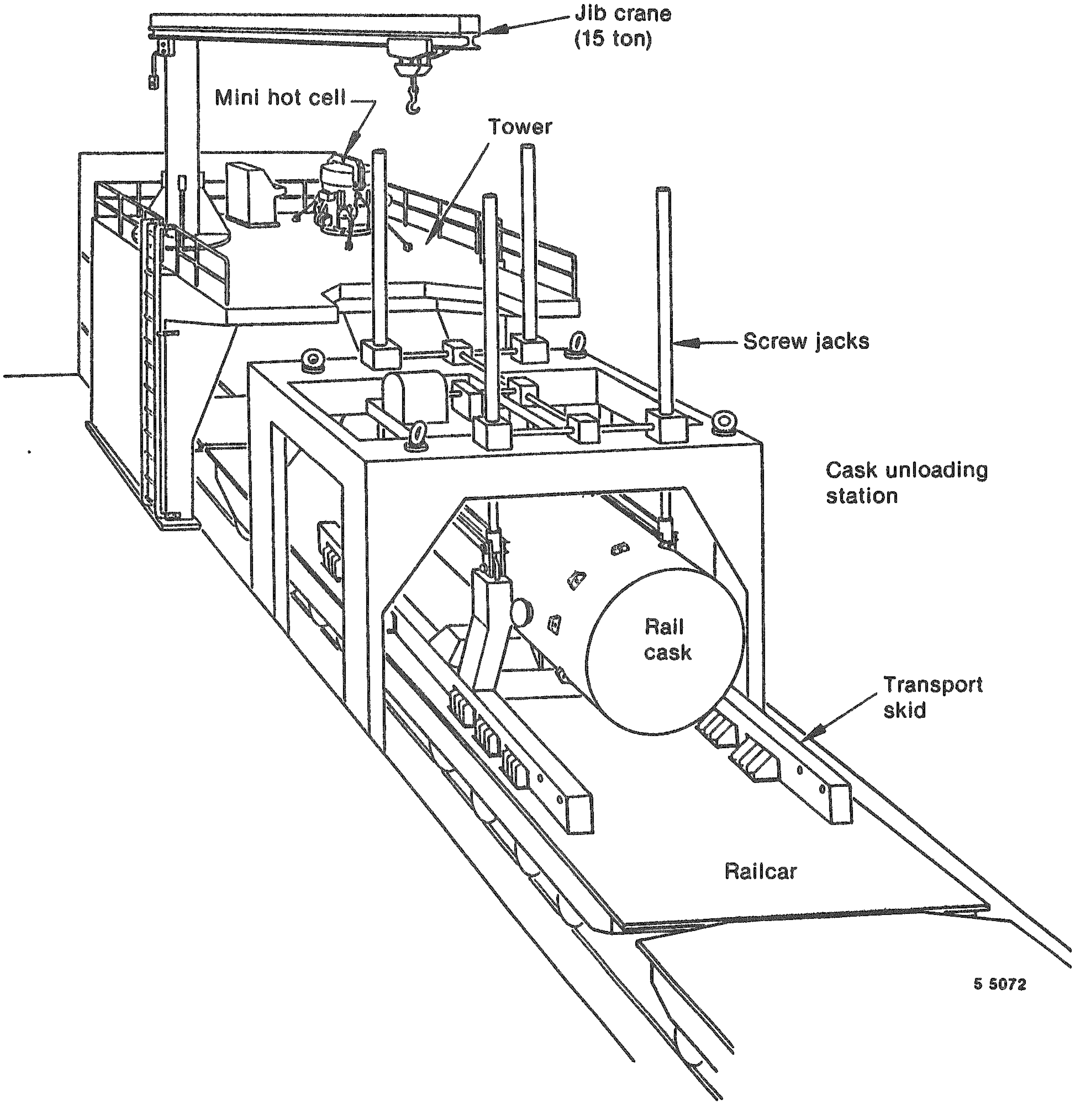

Figure 1. Rail cask on the railroad car.

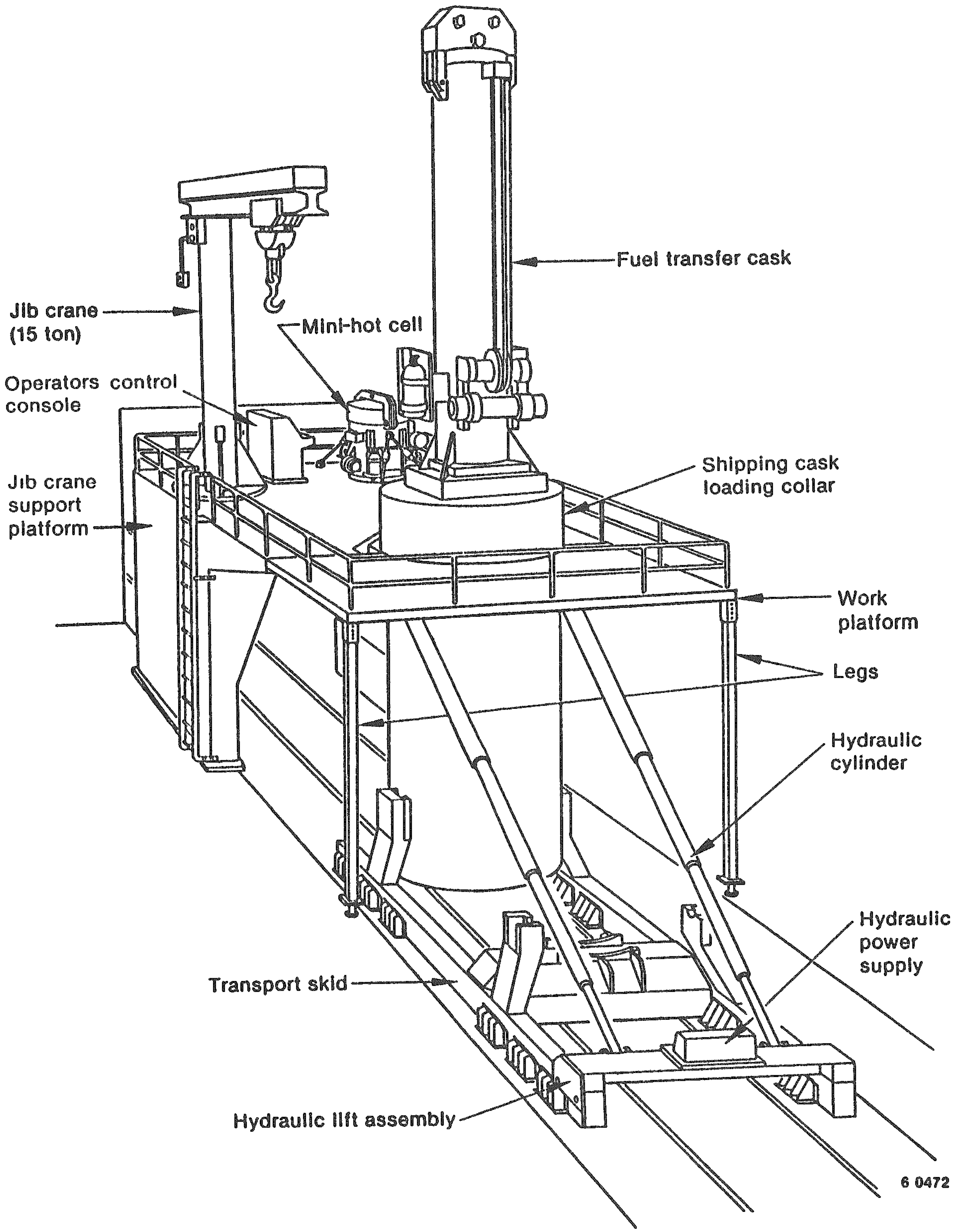

Figure 2. TMI fuel cask loading components.

2.2.1 Quarter-Scale Cask Drop Tests

Fabrication was undertaken in parallel with the licensing review process after NuPac prepared a safety analysis report (SAR) for the NuPac 125B shipping cask with detailed analyses of the cask. Before starting fabrication and submitting the SAR for Nuclear Regulatory Commission (NRC) review, a quarter-scale model of the cask was built and subjected to a series of five drop tests at the Transportation Technology Center of Sandia National Laboratories. The test series included three 30-ft tests and two 40-in. puncture tests. The 30-ft tests included bottom end, oblique, and side drops, and the puncture tests included side and lid end drops. The bottom end drop (Figure 3) was performed to determine the peak acceleration response of the lid and closure bolts and to qualify the internal canister energy absorbers in the inner vessel containment tubes. The oblique drop (Figure 4) was on the lid end at an angle to maximize cask body shell stresses, and the side drop (Figure 5) imparted maximum loads to the inner vessel. The side puncture (Figure 6) verified the integrity of the cask side wall and the end puncture drop (Figure 7) verified the integrity of the cask lid. The test requirements are listed in Table 1.

The quarter-scale drop tests were successful in demonstrating the ability of the NuPac 125B fuel shipping cask to survive hypothetical accident event loadings. Pre- and posttest dimensional checks demonstrated that significant permanent damage to the package was limited to the external overpacks and internal energy absorbers. The side puncture test did produce local deformation of the outer cask outer shell and lead shielding, but this damage was expected and did not cause loss of shielding or containment capability or damage to the inner vessel. Leak tests performed before and after the tests confirmed that the cask maintained its seal geometry and leaktight (10^{-7} atm cc/s) containment integrity of both the inner and outer cask vessels. X-rays taken before and after the tests showed no quantifiable amount of lead slump.

The quarter-scale model was instrumented with accelerometers, strain gauge rosettes, and thermocouples. Evaluation of test data from this

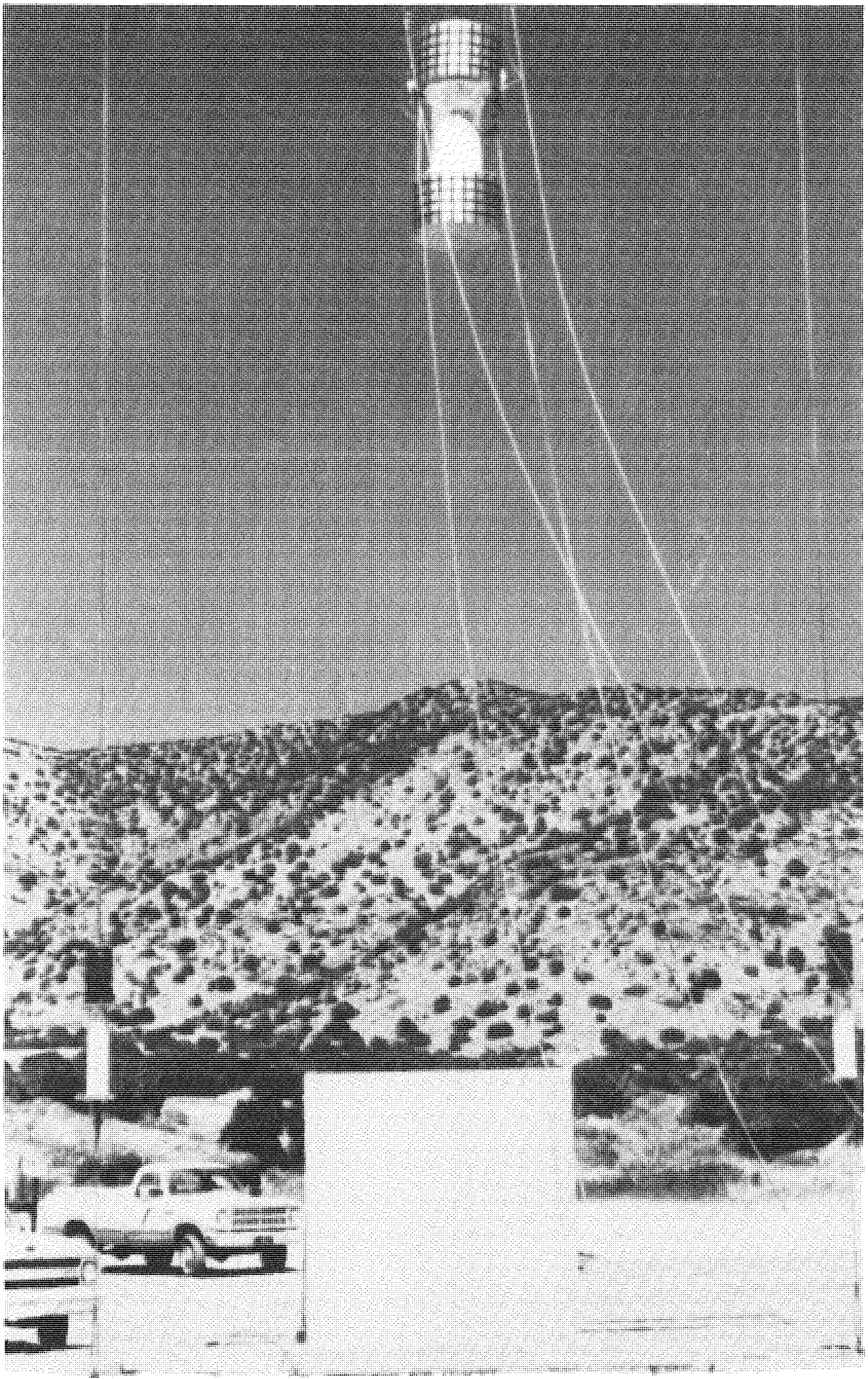

Figure 3. Bottom end drop height and orientation check.

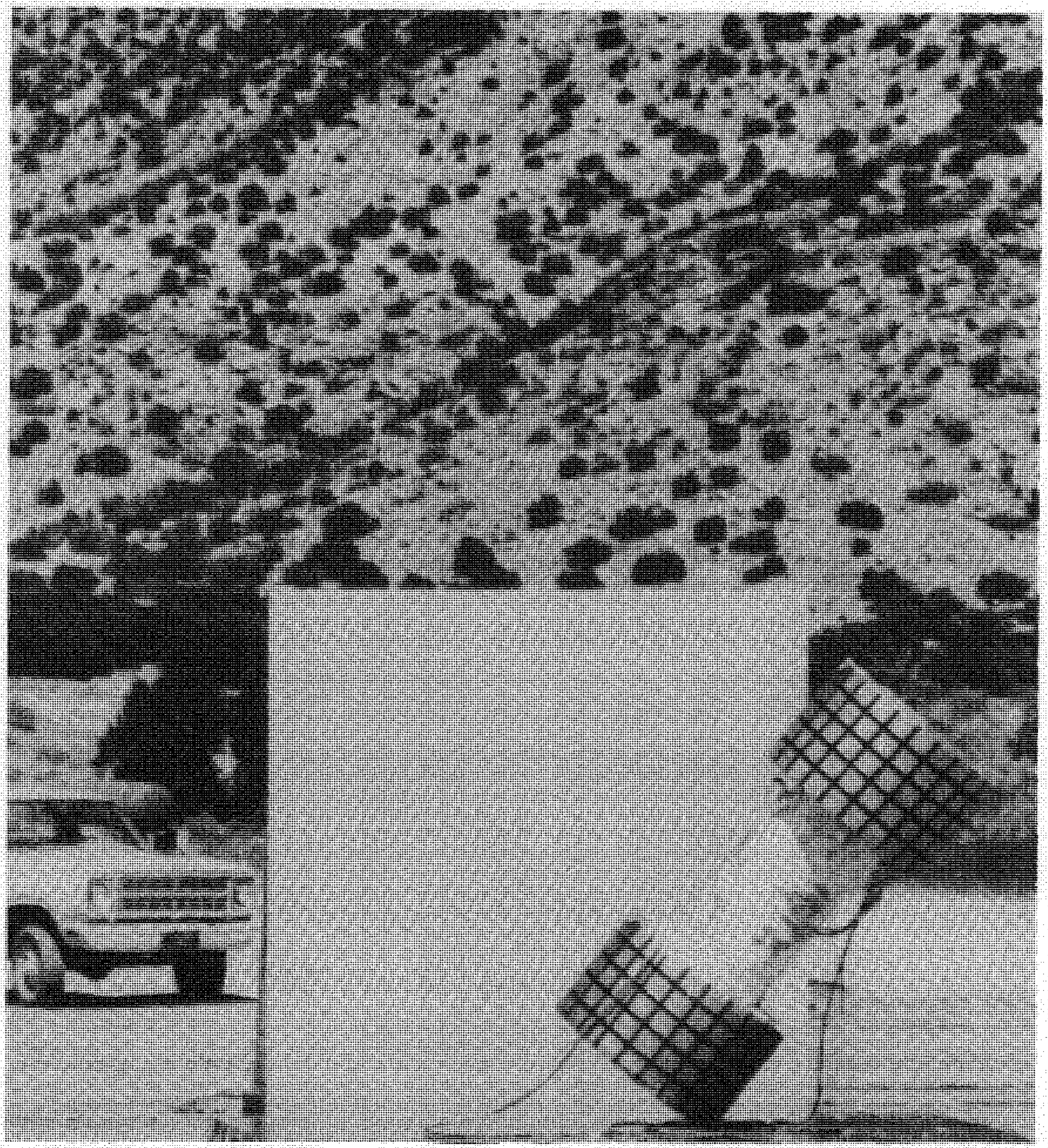

Figure 4. Oblique drop at the instant before impact.

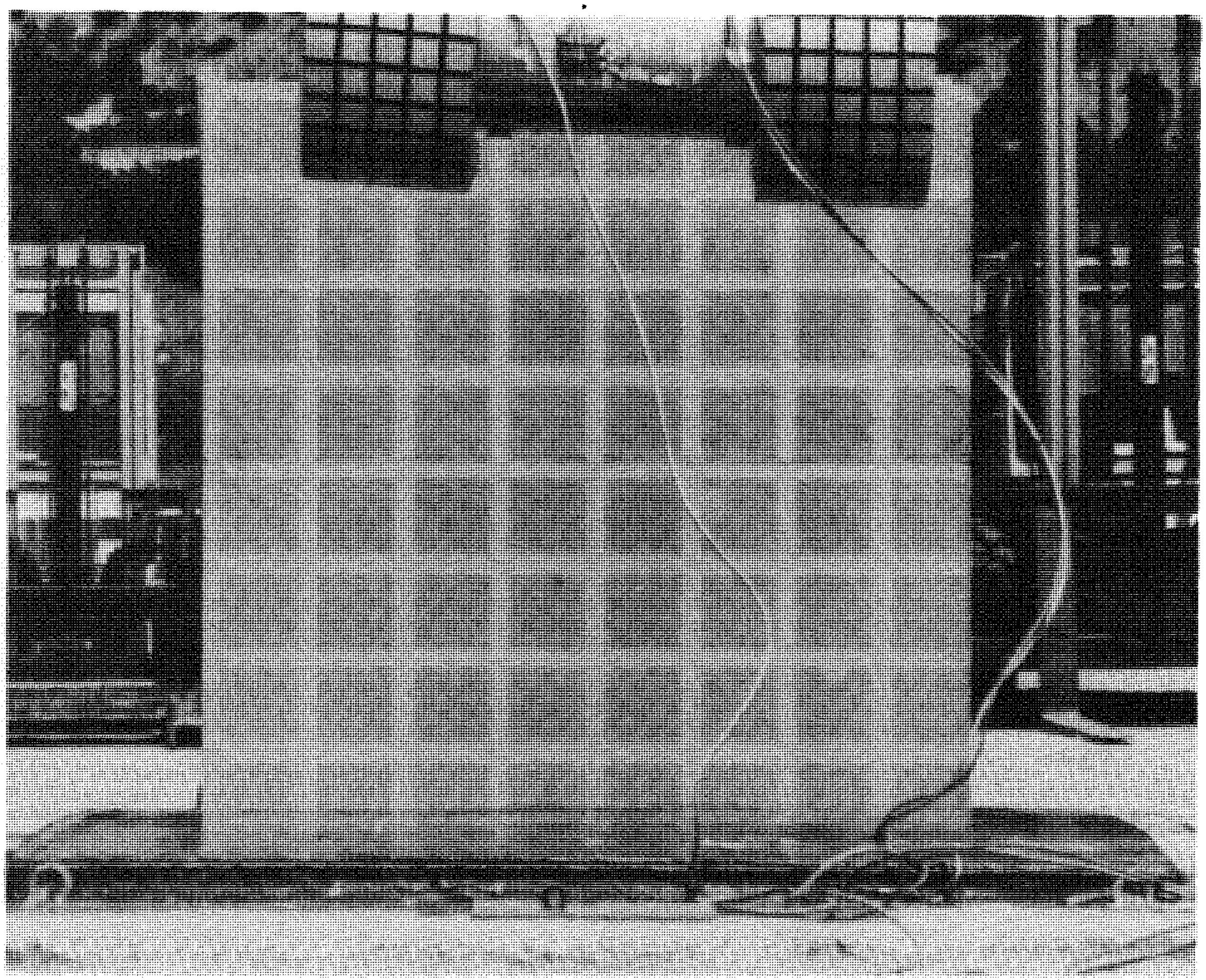

Figure 5. Side drop at the instant before impact.

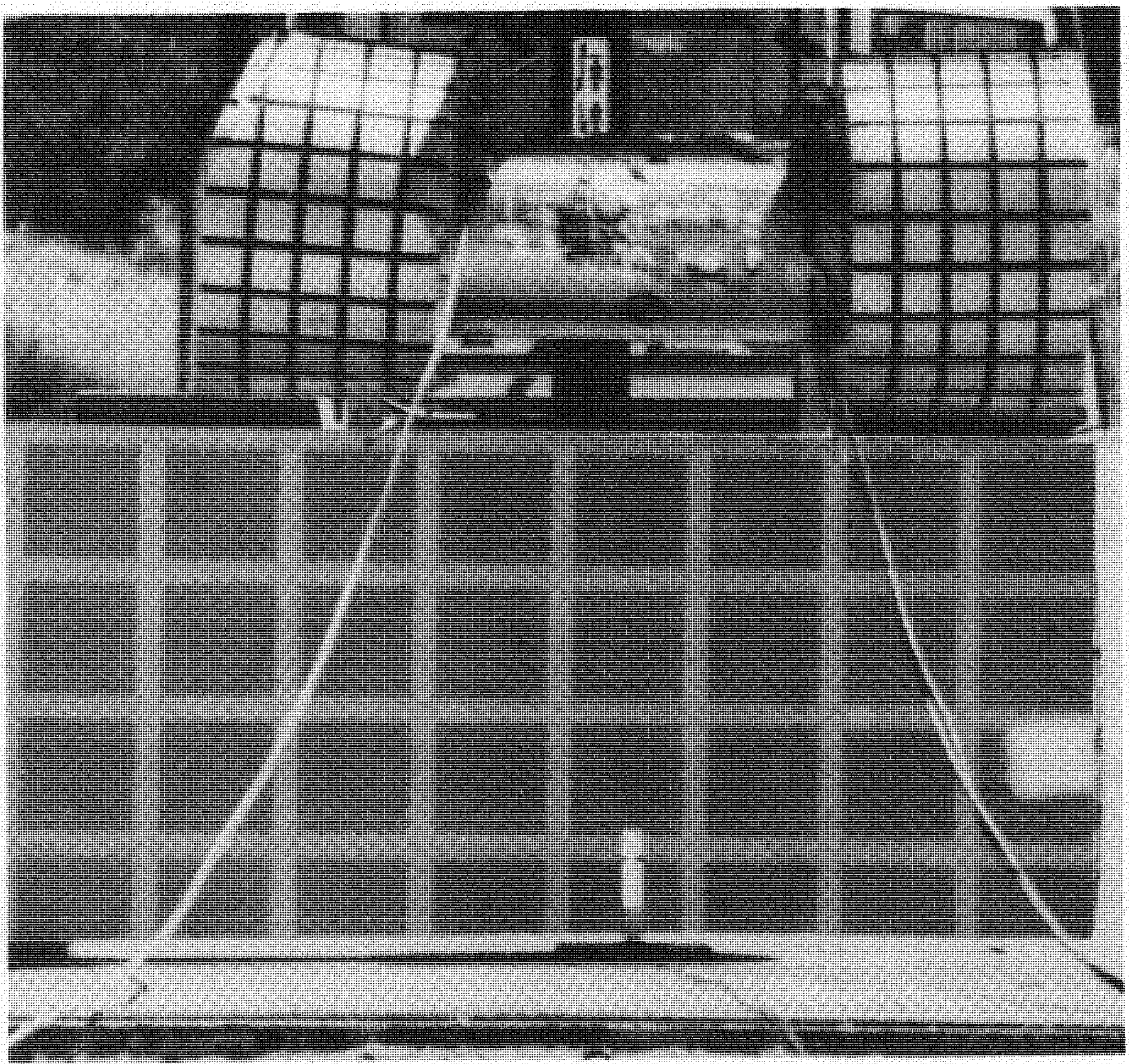

Figure 6. Puncture drop height and orientation check.

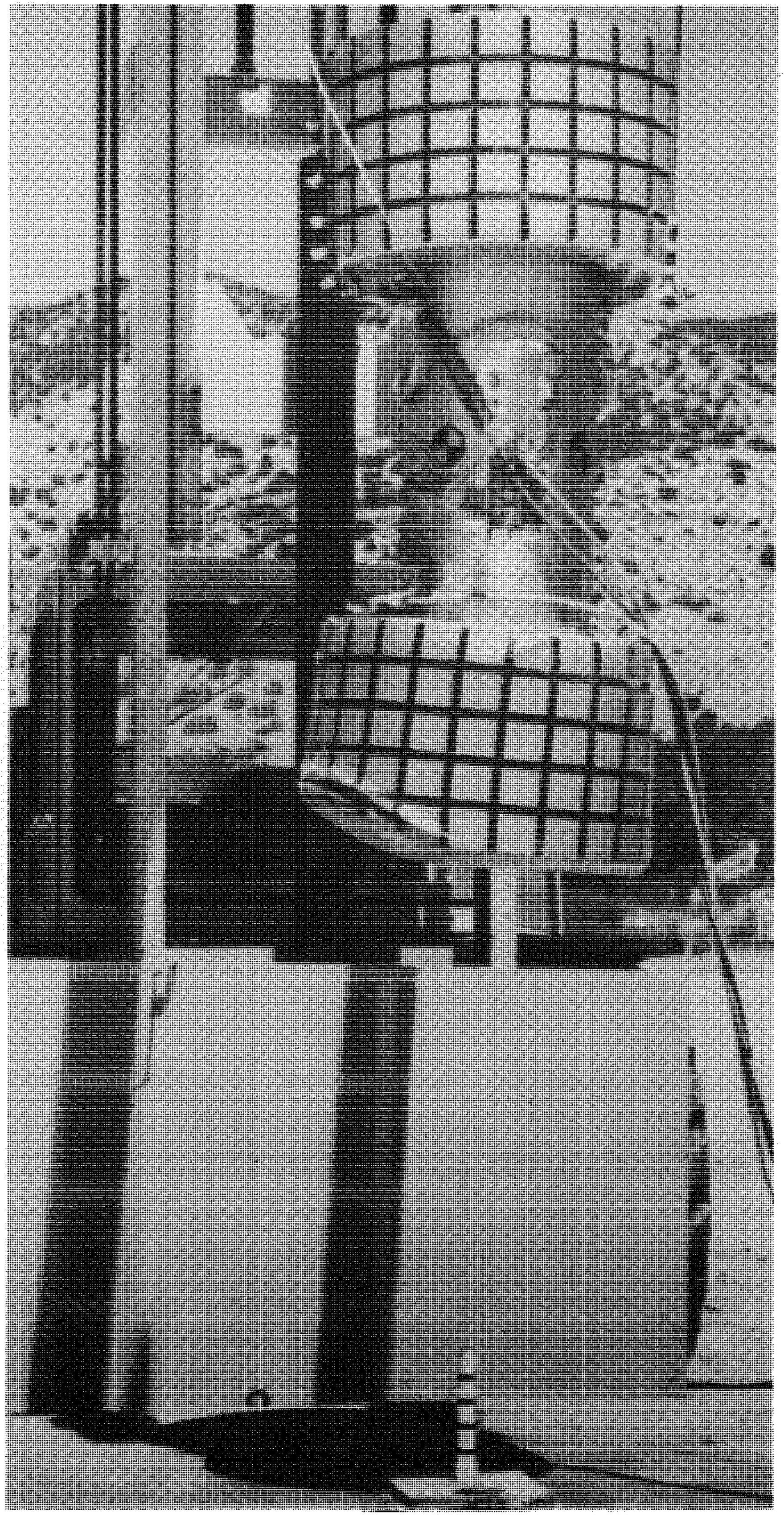

Figure 7. End puncture drop at the instant before impact.

TABLE 1. TEST REQUIREMENTS MATRIX

Test Configuration	Bottom End Drop	Oblique Drop	Side Drop	Side Puncture	Lid End Puncture
Impact end	Bottom	Top	Side	Side	Top
Orientation angle[a]	90°	62.5°	0°	0°	90°
Drop height	30 ft	30 ft	30 ft	40 in.	40 in.
Pretest Steps					
100% visual inspect	Yes	No	Yes	No	No
Dimensional survey	Yes	No	Yes	No	No
Torque lid bolts	Yes	No	Yes	No	No
Leak test	Yes	No	Yes	No	No
Install overpacks	Yes	No	Yes	No	No
Chill to below -20°F	Yes	Yes	No	No	No
Drop Steps					
Visual inspection	Yes	Yes	Yes	Yes	Yes
Check instrumentation	Yes	Yes	Yes	Yes	Yes
Check outer shell temp.	Yes	Yes	No	No	No
Document and photos	Yes	Yes	Yes	Yes	Yes
Posttest Steps					
Remove overpacks	No	Yes	No	No	Yes
Leak test	No	Yes	No	No	Yes
Inspect lid bolts	No	Yes	No	No	Yes
Disassemble and visually inspect	No	Yes	No	No	Yes
100% visual inspection	No	Yes	No	No	Yes
Dimensional survey	No	Yes	No	No	Yes

a. With respect to horizontal.

instrumentation and the structural analysis in the SAR demonstrate that the NuPac 125B cask will adequately survive hypothetical drop accident conditions. Table 2 shows the comparison between accelerations measured during the tests and those predicted by analyses. Close agreement is seen for the 30-ft drop tests; however, for the 40-in. drops, the test accelerations exceed the predicted values. The lower predicted values for 40-in. drops correspond to a puncture bar fabricated of A36 steel for the analysis while the tests were performed with a bar made of higher strength mild steel (SAE 1020 or 1018). Also, the stresses are well below the yield stresses of the material. Table 3 shows the test results and the analytical predictions for the stresses in the outer cask shell. Again, the stresses on the outer shell are well below the yield stress of the material.

2.2.2 Full-Scale Knockout Canister Drop Tests

In response to the first set of SAR review questions received from the Transportation Certification Branch of the NRC concerning criticality control for the knockout canister (Figure 8), GPU Nuclear and EG&G Idaho agreed to conduct drop tests using a full-scale canister to confirm its structural integrity. Oak Ridge National Laboratory (ORNL) performed the drop test at the Tower Shielding Facility (shown in Figure 9) on an extremely tight schedule (less than two months). To simulate the interface between the canister and cask, the full-size, production-run test canister was placed in a carbon steel pipe with a 14.62-in. inside diameter. Blocks of closed cell urethane foam were used as impact limiters to simulate deceleration loads that the canister could expect in the cask from overpacks and inner-vessel impact limiters. Figure 10 shows the simulated cask vessel with the impact limiters used for the vertical drops, and Figure 11 shows the simulated cask vessel with the impact limiters for the horizontal drops. To simulate the fuel debris, the canister was loaded with 1800 lb of lead shot, which was covered with water.

To test for damage expected from the 30-ft drop hypothetical accident conditions specified in 10 Code of Federal Regulations 71, the canister was

TABLE 2. COMPARISON OF ACCELERATIONS MEASURED BY TEST AND PREDICTED BY ANALYSIS[a]

Test	Measured Acceleration (g)	Predicted Acceleration (g)
End drop		
Cask	47.5 - 51	51.6
Canisters	37 - 40[b]	36.1 - 41.9
Oblique drop	25 - 28.5	31.6
Side drop	42 - 45	39.6
Side puncture	12.5 - 13[c]	7.3 - 11.9[d]
End puncture	16 - 18	7.3 - 11.9

a. All data have been normalized to full-scale equivalent values.

b. Canister accelerations were estimated from cask accelerations and crush of internal energy absorbers.

c. 12.5 g represents rigid body portion of response; 13 g represents apparent elastic body portion of response.

d. 7.3 g corresponds to puncture bar fabricated of A36 steel; 11.9 g corresponds to Sandia puncture bar fabricated of other, higher strength mild steel (SAE 1020 or 1018).

TABLE 3. COMPARISON OF CASK OUTER SHELL STRESSES MEASURED BY TEST AND PREDICTED BY ANALYSIS

Test	Measured Stress (psi)	Predicted Stress (psi)
End drop		
Axial	-8500 to -10000	-7761
Hoop	-300 to 1200	0
Oblique drop		
Axial hoop	-10600 to 4100	-13469 to 3558
Hoop	-500 to 100	0
Side drop		
Axial	14500 to 2600	-18488 to 16761
Hoop	-7000 to -2600	-2022 to 1549
Side puncture		
Axial	-17900	-17000
Hoop	-17300	-20453

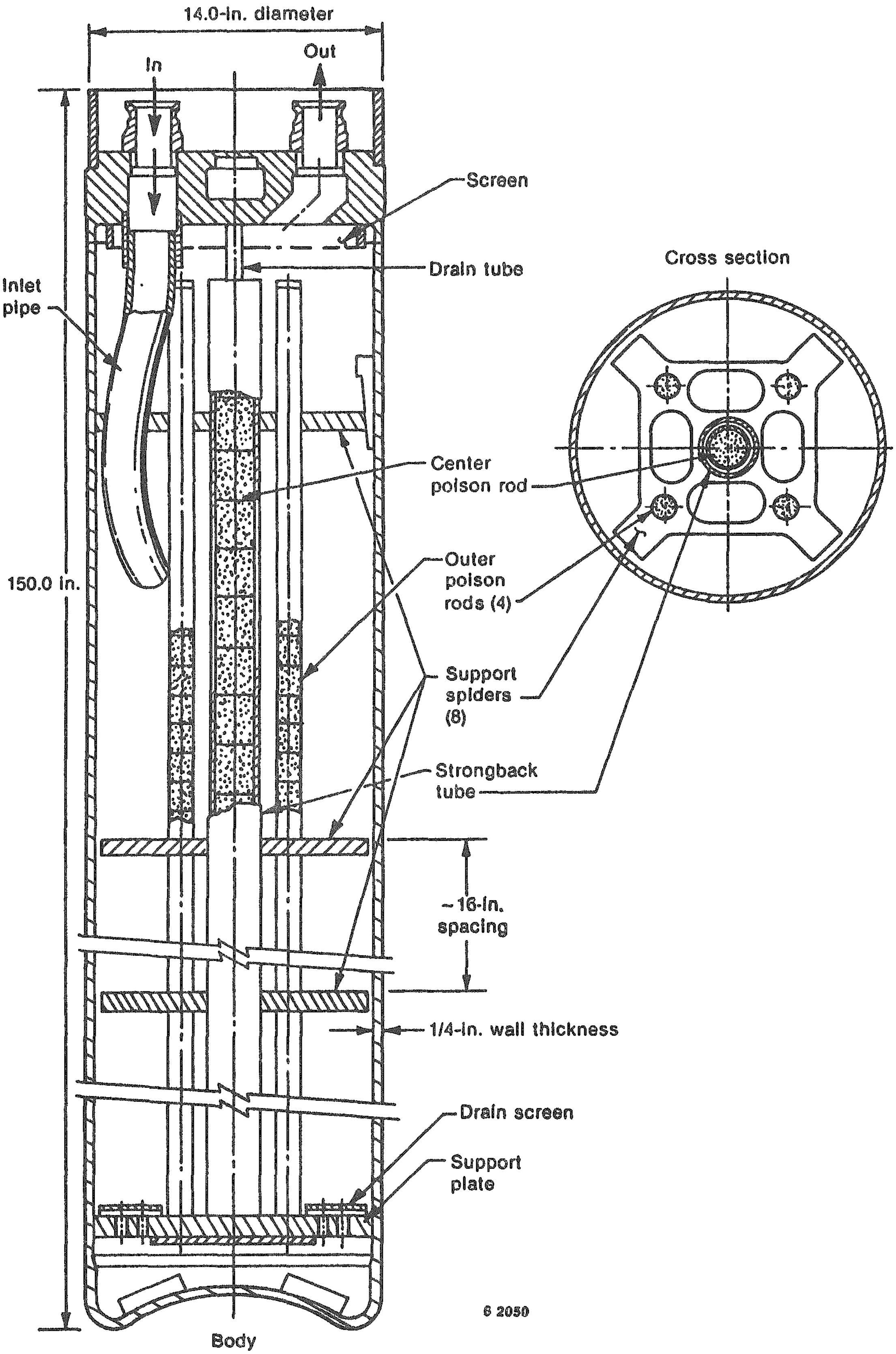

Figure 8. Knockout canister.

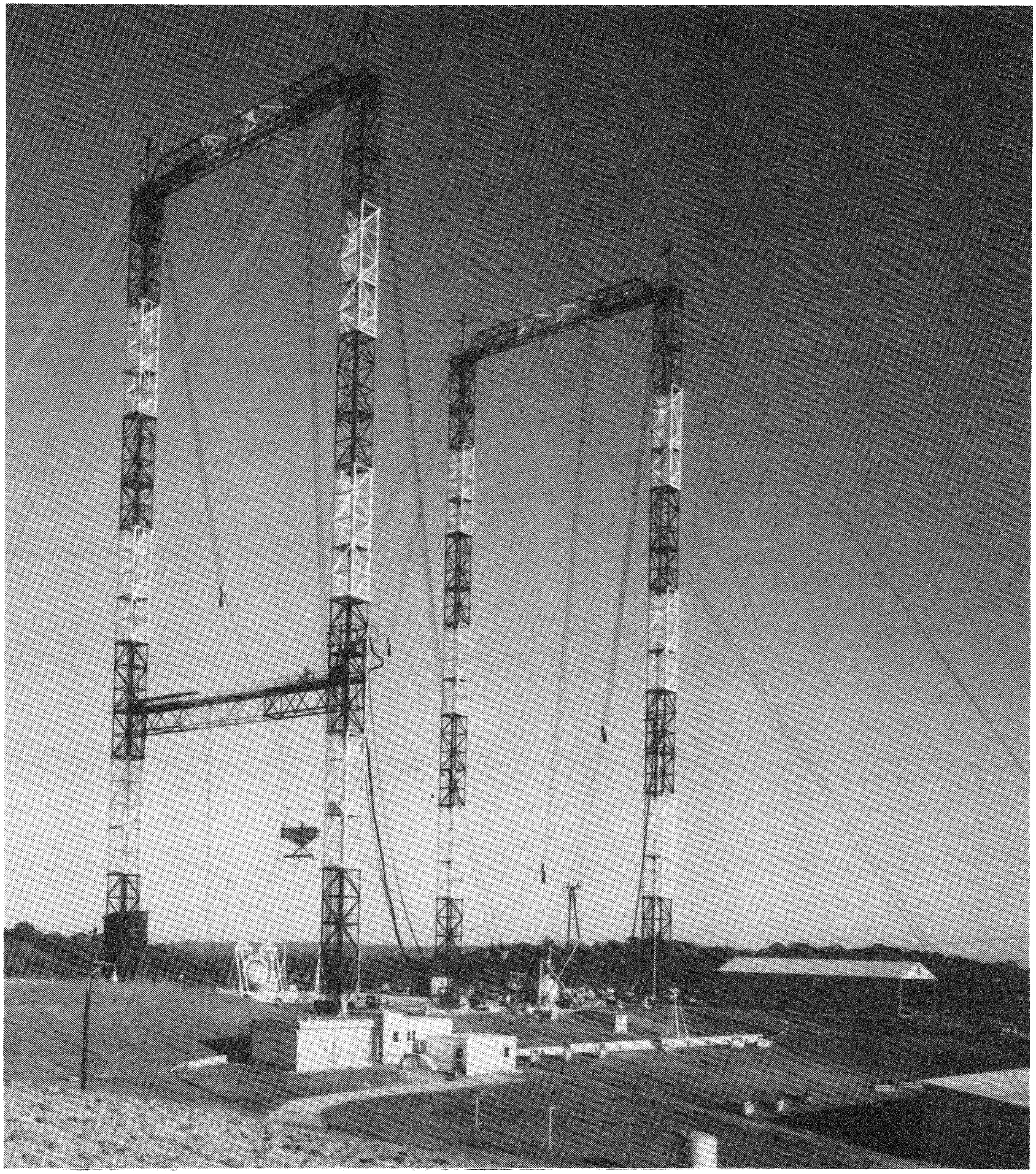

Figure 9. Drop test facility at Oak Ridge National Laboratory.

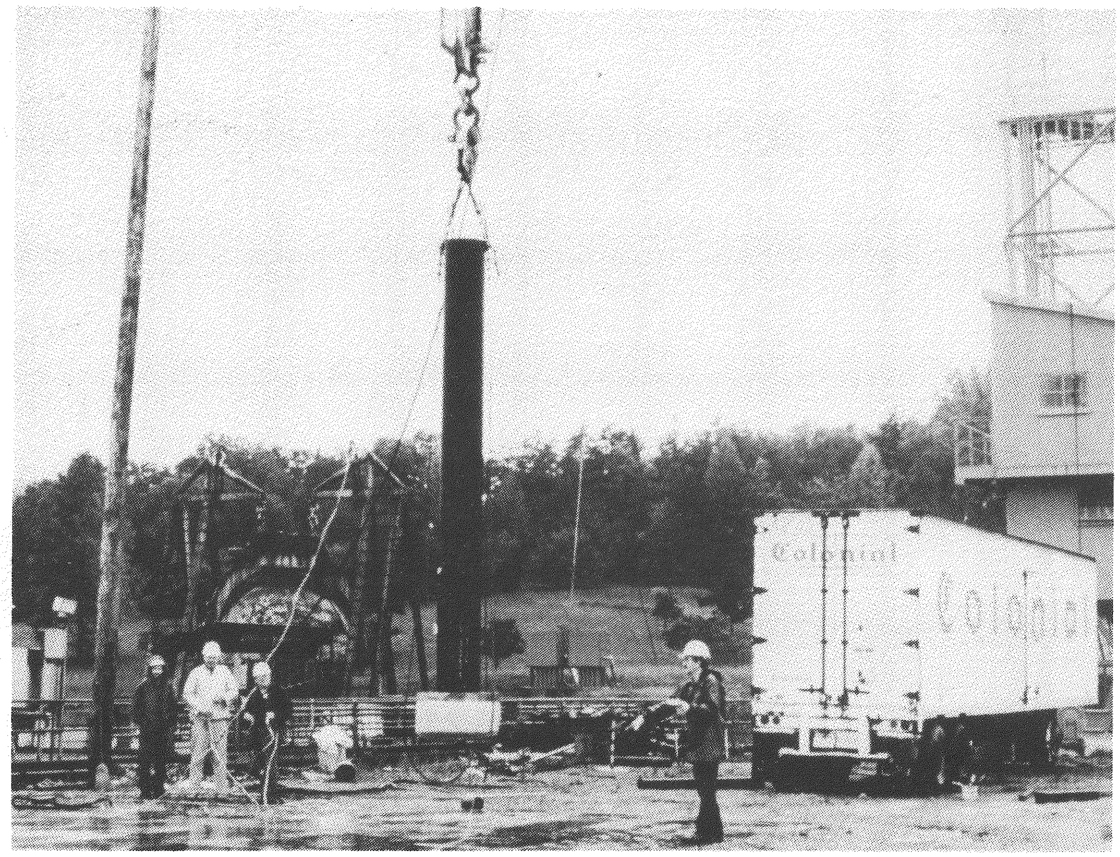

Figure 10. Cask simulation vessel and simulation impact limiter for vertical drops.

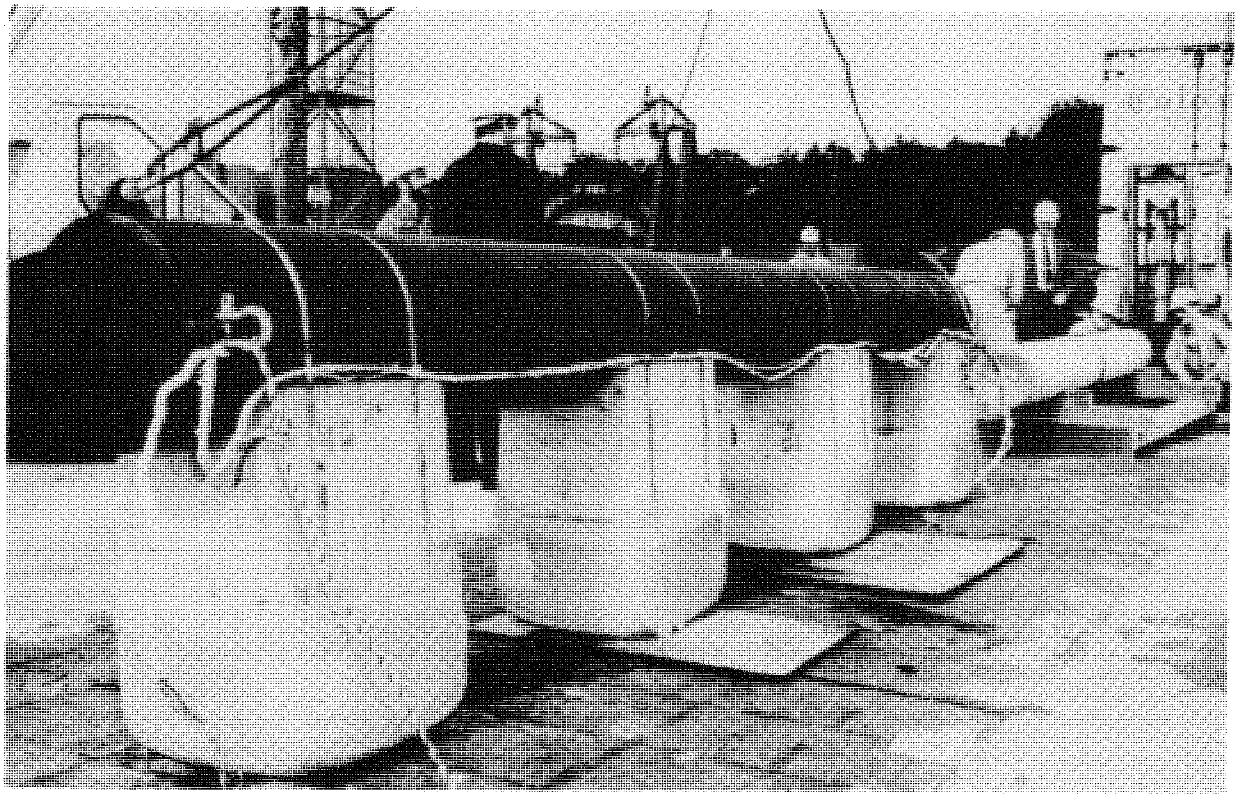

Figure 11. Cask simulation vessel with simulation impact limiters for horizontal drops.

dropped in four different configurations (see Figure 12). The first test was on the canister bottom with the simulated debris frozen to the side, which allowed maximum bending stresses to occur on the support spiders and support plate and the maximum crippling load on the poison rods and strongback tube. The second test was a side drop with the simulated debris frozen at the top and centered on one outer poison rod. This test allowed maximum bending stresses on the poison rod and maximum buckling loads on the support spider arm. The support spider arms that could have gotten maximum deformation from the bottom impact test were down to maximize rod displacement and offcenter arm loading. The third test was on the canister top with the debris loose (not frozen), loaded on the top support spider. This test allowed maximum shear forces between the strongback tube with the weld holding it to the support plate. The intermediate spiders also had a bending force due to the flow of the debris. The fourth test was a drop on the canister side with the debris frozen to one side. This test allowed maximum twisting or torsional moment on the internal assembly. The test parameters and canister test load results are shown in Table 4.

The canister was pressurized to 15 psig before each test. Pressure checks after the tests showed that the canister held pressure after each drop. X-rays were also taken after each test to verify the canister internals had not significantly deformed. After the last test, the canister was disassembled and the internals were measured. Analysis of loads from the test data, posttest measurements, and final visual examination showed that the poison structures in the canister had performed better than required for the cask criticality control, and the assumptions used in the criticality analysis were confirmed.

2.2.3 Gas Recombiner Catalyst Testing

Early in the planning of the defueling operations, it was determined that removing all of the water from the canisters would be very expensive and time consuming. Therefore, wet shipment and vented long-term storage of the canisters before disposal were planned. To ensure that flammable gas mixtures will not exist in these canisters because of hydrogen and

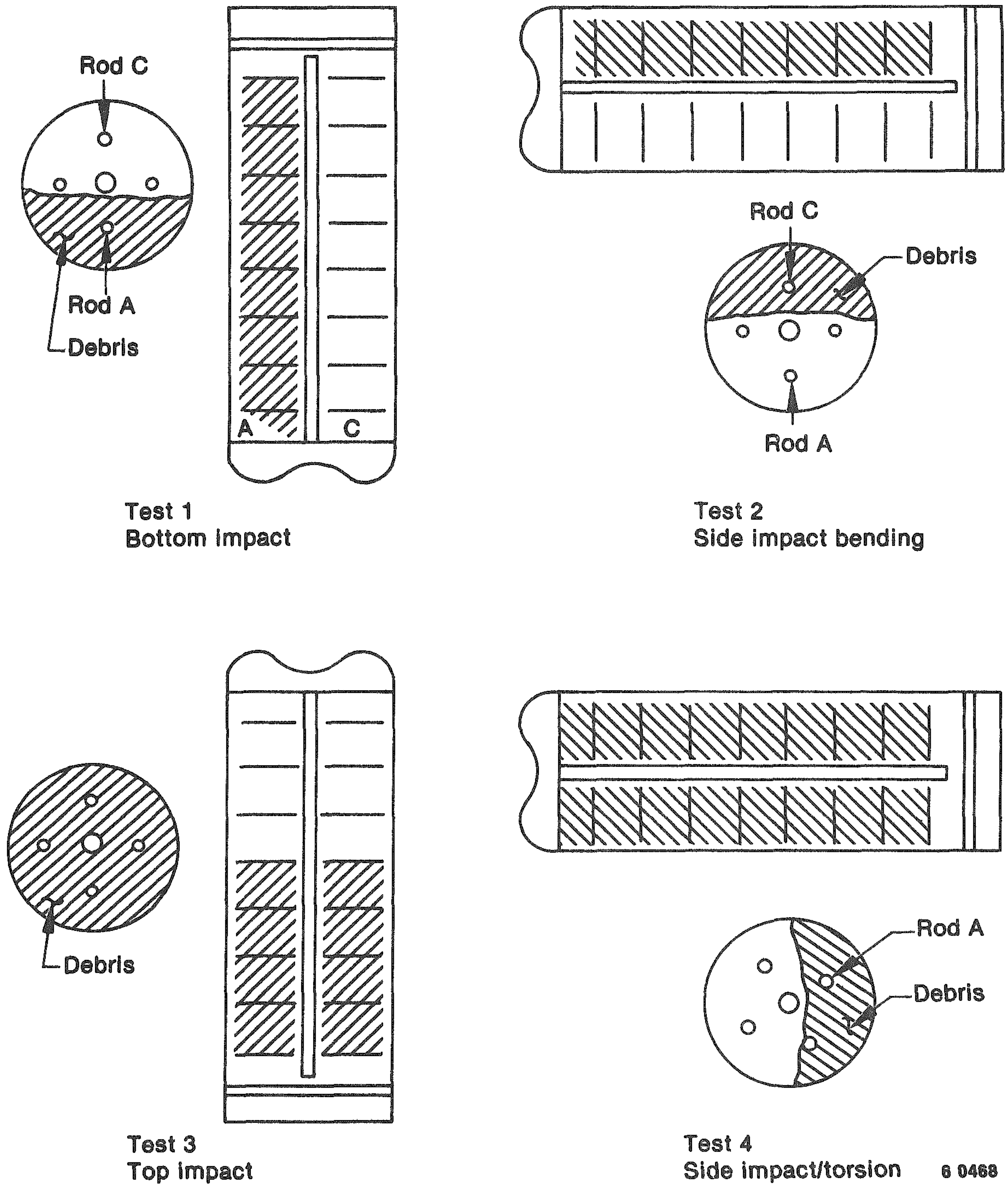

Figure 12. Drop test configurations.

TABLE 4. KNOCKOUT CANISTER TEST PARAMETERS AND TEST RESULTS

Test	Orientation	Frozen Debris	Drop Height	Target "g" Load	Reference[a] Cask "g" Measurement	Test Results "g" Loads
1	Bottom impact	Yes	30 ft	60-80	40	100
2	Side impact bending	Yes	30 ft	80-120	60	120 - 160
3	Top impact	No	30 ft	60-80	40	90
4	Side impact torsion	Yes	30 ft	80-120	60	79[b]

a. Reference cask "g" loads from design basis for canisters that were confirmed by measurements made during quarter-scale drop test of shipping cask.

b. An average value of 63 g at one end of the cask simulation vessel and 94 g at the other end. A secondary impact in Test 4 put a side load of at least 500 g on the canister, although the position of the debris put little of the load on the internals.

oxygen generation from radiolytic gas generation, Rockwell Hanford Operations was asked to determine which gas recombiner catalysts and bed configurations would be acceptable. The NRC requires that shipment of waste materials subject to hydrogen and oxygen generation must meet a safe-shipping time period, which is twice the expected shipping and handling period (from canister purging and closing to completion of shipment) to ensure safety during shipment. The concentration of hydrogen gas must not exceed 5% by volume or the amount of oxygen gas must be limited to 5% by volume.

Rockwell performed a series of catalytic recombiner tests. The tests were conducted using small (16-liter gas/vapor volume) pressure vessels that simulated the shipping canisters. Four specific catalysts were tested to determine the relative benefits of special wet-proof and proven industry-standard catalysts. The four catalysts tested were Engelhard Deoxo-D palladium on alumina, AECL silicon-coated platinum on alumina, AECL Teflon-coated platinum on alumina, and Houdry platinum on alumina.

The test series was designed to evaluate handling and shipping conditions that might affect catalyst performance. Such conditions included wetted catalyst beds; submerged beds; beds poisoned with waterborne chemicals, insoluble particulates, and carbon monoxide gas (generated radiolytically from organic substances); and highly irradiated catalysts. Tests to measure each of these effects on various sizes and shapes of catalyst beds were included in the series. The results of these efforts are summarized below:

- When catalysts were totally submerged in water, essentially no recombination occurred.

- Catalyst beds that were drained after having been submerged in water at two atmospheres for approximately 24 hours started recombining hydrogen and oxygen even in an atmosphere of 100% relative humidity. Recombination rates increased with bed drying as a result of the exothermic reaction. The AECL wet-proof catalysts began recovery earlier than the Engelhard catalyst, but were not as effective as the same volume of Engelhard catalyst in maintaining the gases at acceptable levels.

- o Irradiation of the AECL silicon-coated catalyst at 10^8 rad (an exposure level higher than expected in the canister's 30-year shipping and storage period) had definite effects on the catalyst. Microscopic examination of cross sections of the irradiated pellets indicated spreading of degradation products into the pellet. The surface of the silicon coating appeared to be more uniform and less porous except for fissures. This condition apparently occluded reaction sites within the pellet and decreased its effectiveness as a catalyst.

- o Rockwell's long-term experience with the Engelhard catalyst under high radiation exposures has shown no performance degradation and obviated further radiation testing of this material.

- o Thin beds with a larger surface area exposed to the canister interior performed distinctly better than compact beds.

- o Mixed-bed catalyst containing 80% Engelhard and 20% AECL silicon-coated catalysts by weight performed significantly better that either catalyst alone.

- o Catalysts subjected to freezing temperatures showed two results. If the catalyst was frozen before introducing a stoichiometric mixture of hydrogen and oxygen, the catalytic recombination was drastically reduced. However, if the catalyst was cooled after catalytic recombination has begun (which better represents actual conditions for transport), the catalysts performed well.

- o A series of tests that simulated canister poisoning from possible operational sources also showed minimal effects on the catalysts. These sources included hydraulic fluids, heat damage from welding, Licon concrete from the fuel canister, and cutting fluids.

2.2.4 Core Shipping

The sequence of events for a TMI-2 fuel shipment includes cask loading at TMI, cask transport, and cask unloading at the INEL. Cask loading operations at TMI involve removing the cask protective cover and overpacks from the cask and moving the cask and railcar into the TMI Fuel Handling Building truck bay. The cask unloading station will be attached to the cask and skid, the cask and skid will be raised off the railcar, the railcar removed from the truck bay, and the cask will be lowered onto the floor where the skid will be secured. The hydraulic lift assembly will be secured to the skid and cask lifting saddle, the cask uprighted to the vertical position and secured to the jib crane support tower, and the work platform installed on the cask overpack lugs. In this configuration, the cask is seismically restrained. The cask lids are removed, the shipping cask loading collar is installed, and the mini hot cell is used to remove a shield plug. The fuel transfer cask (FTC) transfers a canister from the storage pool to the shipping cask, where the FTC is aligned with the cask loading collar. The cask loading collar door is opened and the canister is lowered into the cask. The cask loading collar door is closed, the fuel transfer cask is removed, and the shield plug is reinserted from the mini hot cell. This sequence of events will be repeated until the cask is loaded. The shipping cask loading collar will be removed and the inner containment vessel will be installed and leak tested. The outer containment vessel lid will be installed and leak tested, and the cask will be lowered to the horizontal position. The hydraulic lift assembly will be removed, the cask unloading station installed, the cask lifted, and the railcar moved into the truck bay under the cask. The cask will be installed on the railcar, and the railcar with cask will be removed from the truck bay where it will be prepared for shipment. The overpacks will be reattached, the protective cover installed, and the final documentation will be prepared.

To ensure that all of the equipment used in the cask loading operation works properly, an integrated test will be performed at the Hanford Engineering Development Laboratory in Richland, Washington. In addition to checking out the equipment, the tests are also designed to check out detailed operating procedures and provide training for GPU Nuclear personnel.

Using regular train service, the railcar and cask will be moved 2500 miles across the country to the Central Facilities Area of the INEL. The move is expected to take about 10 days.

On arrival at the INEL, the cask will be unloaded onto a specially designed tractor/trailer. From there, the truck will transport the cask another 30 miles to a research complex called Test Area North, where the canisters will be stored for up to 30 years. The canisters will be remotely withdrawn from the cask, moved to a water pool, and lowered into storage racks. Following unloading, the cask will again be transferred to the railcar and sent back to TMI. The TMI-INEL round trip will take about a month.

2.3 Core Receipt and Storage at INEL

At the end of 1985, essentially all of the facility modifications and equipment necessary to begin core receipt at INEL were complete. The balance of the equipment is scheduled to arrive in January. The equipment and facility modifications include:

- Safety review of the gantry crane (reworked crane rail foundation)
- Overpack rigging
- Canister grapple system
- Underwater TV and light system

- Pool cart refurbishment and module adaptor

- Water filling and dewatering cart system

- Miscellaneous handling tools and vent tubes

- Fuel storage modules (sufficient to begin operations with the balance to be delivered by March 1986).

Other equipment and modifications scheduled for January 1986 completion include:

- Horizontal lift fixture (NuPac supplied)

- TWAMCO truck and Kalyn jeep modifications

- TAN hot shop cask stand modifications

- Vertical lift fixture (NuPac supplied)

Dry run testing of the equipment began in October. However, testing is not scheduled for completion until February 1985, awaiting availability of a NuPac 125B shipping cask.

2.4 Cesium Elution

As a consequence of the accident, the two demineralizers in the reactor coolant makeup and purification system were severely contaminated with fission product radionuclides. The resins in the demineralizers were significantly degraded both radiolytically and thermally because of the accumulation of radionuclides and fuel debris during the accident. GPU Nuclear has undertaken a program to remove the resins from these demineralizers. The first step in this process is to remove some of the contamination from them via a washing process known as elution. This is to lower the radiation levels from the resins so they can be handled and disposed of safely.

Radioactive cesium was removed from the makeup and purification demineralizer resins by gradient elution using sodium hydroxide solutions. The radiocesium was eluted from the resins and then resorbed onto the zeolites in the submerged demineralizer system (SDS). The process consisted of 23 batch elutions on the A-demineralizer and 24 batches on the B-demineralizer and was performed between September 30, 1984 and April 12, 1985.

Approximately 790 Ci of ^{137}Cs were eluted from the A-demineralizer (MU-K-1A) and 3,455 Ci of ^{137}Cs were eluted from the B-demineralizer (MU-K-1B). The majority of the radiocesium in the eluants was sorbed onto the zeolites in one SDS liner (U-00009) and shipped to Rockwell-Hanford operations in Washington for disposal. Approximately 68% and 89% of the original ^{137}Cs activity were removed from the A- and B-demineralizer resins, respectively, by the batch elutions. A total of 30,755 gallons of diluted eluant was processed by the SDS during these operations.

3. ACCIDENT EVALUATION PROGRAM

3.1 Examination Requirements and Systems Evaluation

3.1.1 Fission Product Inventory

During FY-85 (the fiscal year October 1984 through September 1985), the results of fission product analyses on samples collected from the TMI-2 plant in FY-84 were collated. Among the sample analyses completed in 1984 were the first samples obtained from the reactor core.

The core samples analyzed in 1984 were obtained from two locations in the upper debris bed that rests on the consolidated lower portion of the core. The upper debris bed is estimated to contain approximately 20% of the core mass. Samples of the remaining portions of the core have still not been fully analyzed and the characterization of core material below the upper debris bed remains unknown. No conclusions about the ultimate inventory closure on fission products should be made until this portion of the core has been explored.

Samples from the upper debris bed were found to contain approximately 5 to 6% of the core inventory of cesium and iodine, approximately 12% of the core inventory of strontium, and approximately 26% of the cerium. Qualitative extrapolation of these numbers to the entire core mass indicates that significant quantities of cesium and iodine may still be found in the core along with most of the strontium and essentially all of the cerium.

Large quantities of cesium and iodine have also been found in the sediment and water on the floor of the Reactor Building basement. These elements, highly water soluble, appear to have been transported to this location through the Pilot Operated Relief Valve and Reactor Coolant Drain Tank via RCS leakage.

In addition to investigations of the unknown portions of the reactor core, only a few areas of the plant are judged to have the potential for significantly changing the fission product inventory closure. They are the sediment and concrete in the Reactor Building basement and debris and plateout in the makeup and purification system components, and the reactor coolant drain tank. A summary of the fission product inventory found in the upper core debris bed and revised inventories for the reactor building basement is given in Table 5. An update of all fission product inventory data through FY-85 is in preparation.

3.1.2 Standard Problem

The TMI-2 accident will be used to benchmark severe accident analysis techniques that are being applied by industry and regulatory agencies to estimate the source term from low-probability severe accidents. The standard problem is a formal exercise in which several participants will apply their analytical methods to the TMI-2 accident using common initial and boundary conditions. The results of the analyses will be compared among the participants and with the measured or determined actual conditions during the accident. This constitutes the benchmarking process. Organizations expressing interest to participate include the NRC and foreign countries through the Organization for Economic Cooperation and Development Committee on the Safety of Nuclear Installations.

A standard problem package is being prepared for distribution at the end of FY-86. The package will contain the necessary information to perform an analysis: initial plant conditions; boundary conditions, such as operator actions; and plant configuration, i.e., a complete geometric description. Additionally, to assist an analyst, the package will contain a best-estimate accident scenario and selected results of a demonstration analysis performed with the state-of-the-art severe accident analysis code--RELAP5/SCDAP.

TABLE 5. FISSION PRODUCT INVENTORY Fractions--1985
(Decayed to December 31, 1979)

Component	Reactor Building Basement (Revised)	Upper Core Debris Bed
Tritium	0.57[b]	
Strontium	0.01[a]	0.12[c]
Iodine	0.18[a]	0.05[c]
Cesium	0.41[a]	0.06[c]
Cerium	0.001[b]	0.26[c]

a. From R. J. Davis et al., GEND-INF-047.

b. From C. V. McIssac, D. G. Keefer, GEND-042.

c. From analyses carried out this year.

3.1.3 Accident Scenario

The accident scenario developed for the initial four hours of the accident is based on the currently known end-state conditions of the core and reactor vessel, data from plant instrumentation recorded during the accident, the results from best-estimate analyses of the accident employing the SCDAP code, and results from severe fuel damage experiments in the Power Burst Facility at the INEL. The important features of the accident scenario are discussed here to identify the primary mechanisms controlling core damage progression and the primary questions remaining to be resolved.

Core uncovery started between 100 and 120 min. This is substantiated by the measurement of superheated steam detected in the hot legs at 113 min. Best-estimate core damage predictions indicate that core temperatures were high enough to balloon and rupture the fuel rod cladding at about 140 min, releasing the noble gases and volatile fission products such as iodine and cesium. Fission products were detected in the containment at about 143 min. These predictions also indicate that cladding temperatures rapidly increased at about 150 min because of cladding oxidation, and temperatures quickly exceeded cladding melting points. The molten zircaloy dissolved some fuel, and the liquified fuel flowed downward through the core, eventually solidifying in lower, cooler regions of the core. The minimum relocation level was probably coincident with the coolant liquid level, which is estimated to have been into the lower one-third of the core.

By 174 min (just before the primary pump transient), core temperatures probably had reached fuel melting in the central, highest-temperature regions of the core; and between one-quarter and one-half of the core probably attained cladding melting temperatures with subsequent dissolution of some of the fuel. The liquified and molten material flowed downward and froze. During the time period between 150 and 174 min, a relatively solid region of core materials composed of previously molten and intact fuel rods is postulated to have formed, as shown at the top of Figure 13. The top of the core probably consisted of highly oxidized and embrittled fuel rod

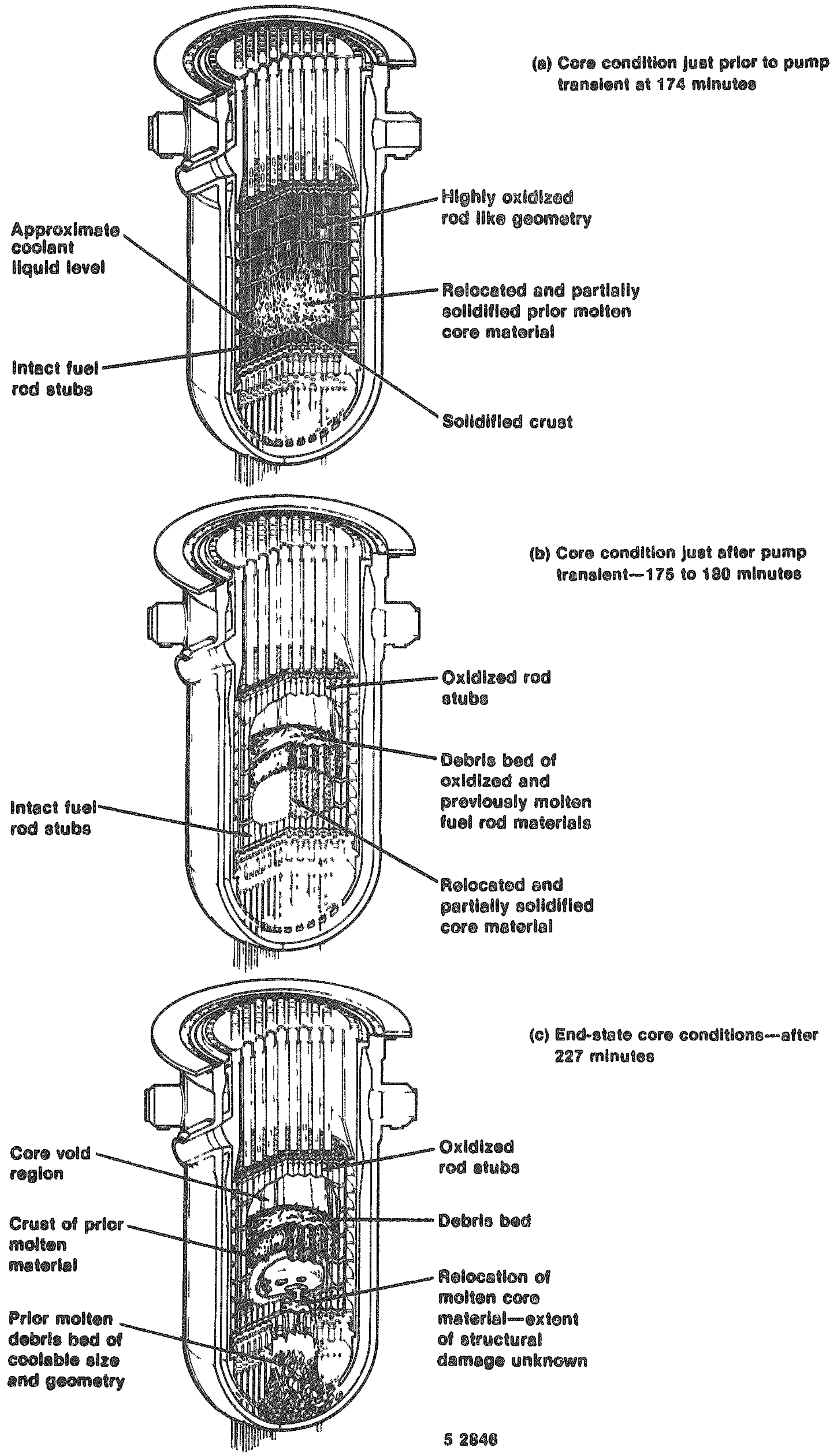

Figure 13. Hypothesized stages of the TMI-2 accident progression.

remnants. High-temperature molten material apparently had not yet penetrated below about 0.75 m, because the minimum water level is believed to have been at about that elevation. This was concluded from the fact that the self-powered neutron detectors (SPNDs) at Level 1 and about half of those at Level 2 (0.25 and 0.75 m above the core bottom, respectively) indicated no anomalous behavior during this time.

The primary system pump transient at 174 min injected some coolant into the core. However, the extent of core cooling is not known because of the potential flow blockage resulting from the relocated and frozen material in the lower regions of the core. Thermal and mechanical shock resulting from the injected coolant fragmented the embrittled fuel rod remnants in the upper regions of the core. These fuel rod fragments apparently collapsed onto the solidified surface of previously molten material, forming the rubble bed shown in the center of Figure 13. Thermal calculations suggest that the zone of the relocated core materials continued heating even after injection of coolant into the core. Those calculations are corroborated by the incore thermocouples, which indicated a second heatup between 180 and 227 min. The heatup occurred even though the level of coolant may have been near the midplane elevation in the core, indicating that a noncoolable geometry was present.

The primary relocation of molten core materials into the lower plenum probably occurred at approximately 227 min. This relocation of high-temperature material was verified by the lower head examinations and its timing was calculated based on anomalous output from the Level 1 and 2 SPNDs and a very rapid increase in the primary system pressure of approximately 2 MPa. The increased system pressure was apparently caused by the vigorous interaction between the downward-flowing hot core material and water, which generated a substantial quantity of steam. The expanding steam and the material flow through the core support assembly probably fragmented much of the molten material as it relocated into the lower plenum. This fragmentation may have had a significant effect on the eventual formation of a coolable configuration in the lower plenum. The progression of the accident was essentially halted at this time by the water in the lower plenum and the continued injection of water into the reactor coolant system (RCS) by the high-pressure injection system. The postulated final state and configuration of the reactor and support structures are illustrated at the bottom of Figure 13.

3.2 Analytical and Experimental Support

3.2.1 Data Evaluation and Data Base Development

The Data Evaluation and Data Base Development staff of EG&G Idaho has two principal areas of responsibility: (a) the evaluation of on-line measurements that pertain to the standard problem, and (b) the development of an integrated data base, which eventually will contain the information required to run a standard problem.

Most of the on-line data has been digitized and placed in a computer file. Some of this data has been evaluated; the pressurizer level measurement is one example. The result of this study was that the pressurizer level was indicating correctly to within 15 in.

Other measurements that have been evaluated include those from the SPNDs. The SPNDs are important during the core heatup (after the pumps were shut off) because, even though they have failed by this time, their response can be related to core temperatures at specific times. Two responses have been discovered that have identifiable temperature ranges associated with them. The first response is when an SPND starts to go negative--the temperature is 1000 to 1300°F. The second response occurs later when an SPND changes rapidly from negative to offscale positive--the temperature is then in the range of 1900 to 2500°F.

Source and intermediate-range neutron detectors are being evaluated now, and this evaluation will continue throughout 1986. It has been suggested, and preliminary studies indicate, that times of initial core movement can be determined using these detectors.

During the last half of 1985, the concept of an integrated data base for TMI-2 data became more firm. The data base will be built as several independent bases through 1986 and integrated the following year. The data base will have the data needed to perform the standard problem and the results of the fission product distribution inventory.

The independent data bases are: (a) time series data, (b) initial and boundary conditions, (c) plant physical configuration, (d) sequence of events and operator actions, (e) fission product data, and (f) reactor system examinations.

The preliminary, initial conditions-boundary conditions data base has been completed and a beginning has been made on the sequence of events data base. It is anticipated that the time series and plant configuration bases will be completed in 1986.

3.3 Information and Industry Coordination

Approximately 200 people attended the First International Information Meeting on the TMI-2 accident at DOE Headquarters in Germantown, Maryland, on October 21, 1985. The meeting agenda covered the accident scenario, the scope of the Accident Evaluation Program, and recent results from TMI-2 research. It is intended that these information meetings be annual events, held just before or after the water reactor safety research meetings that NRC sponsors. Full formal papers covering the topics discussed at the meeting will be assembled into the proceedings of the conference and will be published in 1986.

In conjunction with the publication of the draft of the Accident Evaluation Program Description document in September 1985, an ad hoc advisory group, the Accident Evaluation Advisory Group (AEAG), was formed to review the document and to advise the program on its objectives and planning. Comments have been received from the AEAG and are currently being addressed in a revised version of the description document that will be published early in 1986.

3.4 Sample Acquisition and Examination

3.4.1 Reactor Vessel Internals Characterization and Sampling

To document the condition of in-vessel components and identify possible samples, GPU Nuclear conducted various video inspections during 1985. The first lower head video inspection, which took place shortly after plenum jacking, was also documented using the video enhancement system described in the 1984 Annual Report. The video system was used again during plenum removal, upper core void, and lower head inspections. The videotapes were analyzed by GPU Nuclear engineers and sent to INEL for review and analysis. During the lower head inspections in July and December 1985, rock samples were obtained from below the flow distributor plate using a long-handled manipulator grab tool and 3-inch-diameter sample buckets. These samples underwent preliminary radiological analysis before shipment to an offsite laboratory. Preliminary analysis results of the debris are given in Table 6.

3.4.2 Core Sample Acquisition and Examination Project

Following the evaluation of commercially available core sampling equipment used by the geology and mining industry, work was initiated by EG&G Idaho in 1984 to design an extensively modified system for use above the TMI-2 reactor. During 1985, design of the system was completed, followed by component fabrication, assembly of the integrated system, checkout, and procedure development. The completed unit, along with its supporting equipment, was disassembled, packaged, and delivered to TMI-2 late in 1985. Use of the system to acquire samples is anticipated during mid-1986.

Application of the project's equipment at TMI-2 will be targeted at obtaining full-height samples from up to nine core positions. Figure 14 shows the candidate positions from which the core samples will be selected. The samples will have representative stratigraphy, including the crust material, standing fuel rods, and spacer grids, and the lower

TABLE 6. LOWER HEAD PHASE 1 SAMPLES PRELIMINARY ANALYSIS RESULTS

Size (in.)	Radiation Measurements[a] Beta/Gamma (R/h)	Radiation Measurements[a] Gamma (R/h)	Dry Weight W_d (g)	Saturated Weight W_s (g)	Immersed Weight W_i (g)	Envelope Density (g/cc)	Matrix Density (g/cc)	Open Porosity (%)	Pellet Volume (cc)
1.2 x 1.0 x 0.8	13	1.6	50.1	50.1	42.5	6.57	6.57	0.0	2.61
0.4 x 0.2 x 0.2	1.2	0.13	1.0	--b	--b	--b	--b	--b	--b
0.2 x 0.2 x 0.1	0.8	0.10	0.4	--b	--b	--b	--b	--b	--b
1.5 x 0.7 x 0.6	12	1.2	39.7	39.8	34.9	8.08	8.25	2.0	4.91
1.8 x 1.3 x 1.2	26	3.0	123.9	124.4	106.2	6.79	6.94	2.75	18.23
1.8 x 1.0 x 1.0	25	2.9	107.1	107.6	91.7	6.72	6.75	3.14	15.92
2.5 x 2.5 x 2.2	50 42[c]	7.5 5.5[2]	553.9	555.6	470.2	6.47	6.60	1.99	85.52
0.7 x 0.7 x 0.4	4	0.5	12.7	12.8	10.8	6.3	6.7	5.0	2.2
1.6 x 1.2 x 1.1	30	3.2	118.8	119.0	102.3	7.09	7.18	1.20	16.73
0.4 x 0.2 x 0.2	0.7	0.1	0.6	--b	--b	--b	--b	--b	--b
0.5 x 0.5 x 0.5	3.5	0.32	5.5	--b	--b	--b	--b	--b	--b

a. Radiation readings were taken at rear of the Auxiliary Reactor Area Hot Cell at the INEL. The background readings were: 80 mR/h gamma and 50 mR/h gamma at 8 in.
b. Because of the relatively small particle size and the sensitivity limits of the triple beam balance used to weigh the particles, saturated and immersed heights could not be made.
c. Reading taken at 10 inches. At 8 inches, the detector was offscale (50 R/h beta/gamma).

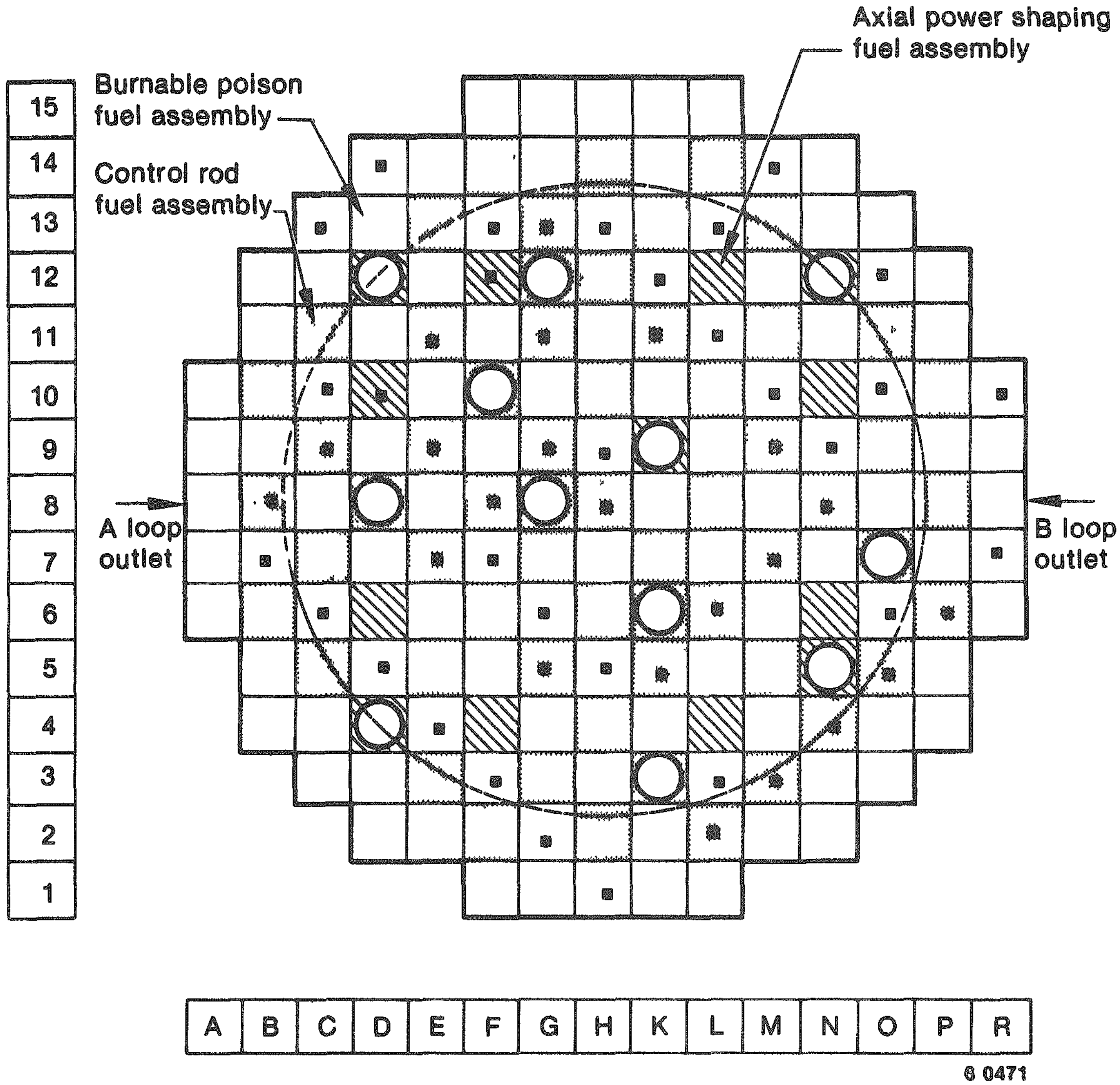

Figure 14. Core bore locations.

endfitting. The sampling activity will also include the acquisition of sample material from the space between the normal lower endfitting elevation and the upper surface of the elliptical diffuser plate. Immediately following the removal of each sample, the accessible space will be visually examined using a remotely controlled closed-circuit television camera.

Figure 15 is an illustration of the core boring mechanical equipment and support structures on the defueling platform above the reactor vessel. The figure depicts drill piping completely inserted in the core before sample withdrawal.

Following the delivery of the core samples to the INEL in the summer of 1986, an extensive examination task will begin. Supporting this effort will be a detailed evaluation of the data acquired during the actual drilling operations, which will help to identify the location and extent of void spaces and loose materials (if present). The detailed material examinations and the interpretation of the results are scheduled for completion in 1988.

3.4.3 Maximum Core Temperatures on the Upper Debris Bed

Analysis of a number of particles removed from the upper debris bed indicates the presence of regions of prior molten U-Zr-O in many of the particles. Of these, many are prior molten $(U, Zr) O_2$ indicating peak temperatures greater than 2800K. There are a few examples of prior molten material that are almost pure UO_2, indicating temperatures up to 3100K (the effect of small amounts of impurities on the melting point of UO_2 is not well known). However, the relatively unstructured appearance of much of the material suggests that much of the fuel probably remained at fairly low temperatures (<2000K) relative to the peak temperature or was exposed to high temperatures for only a short time.

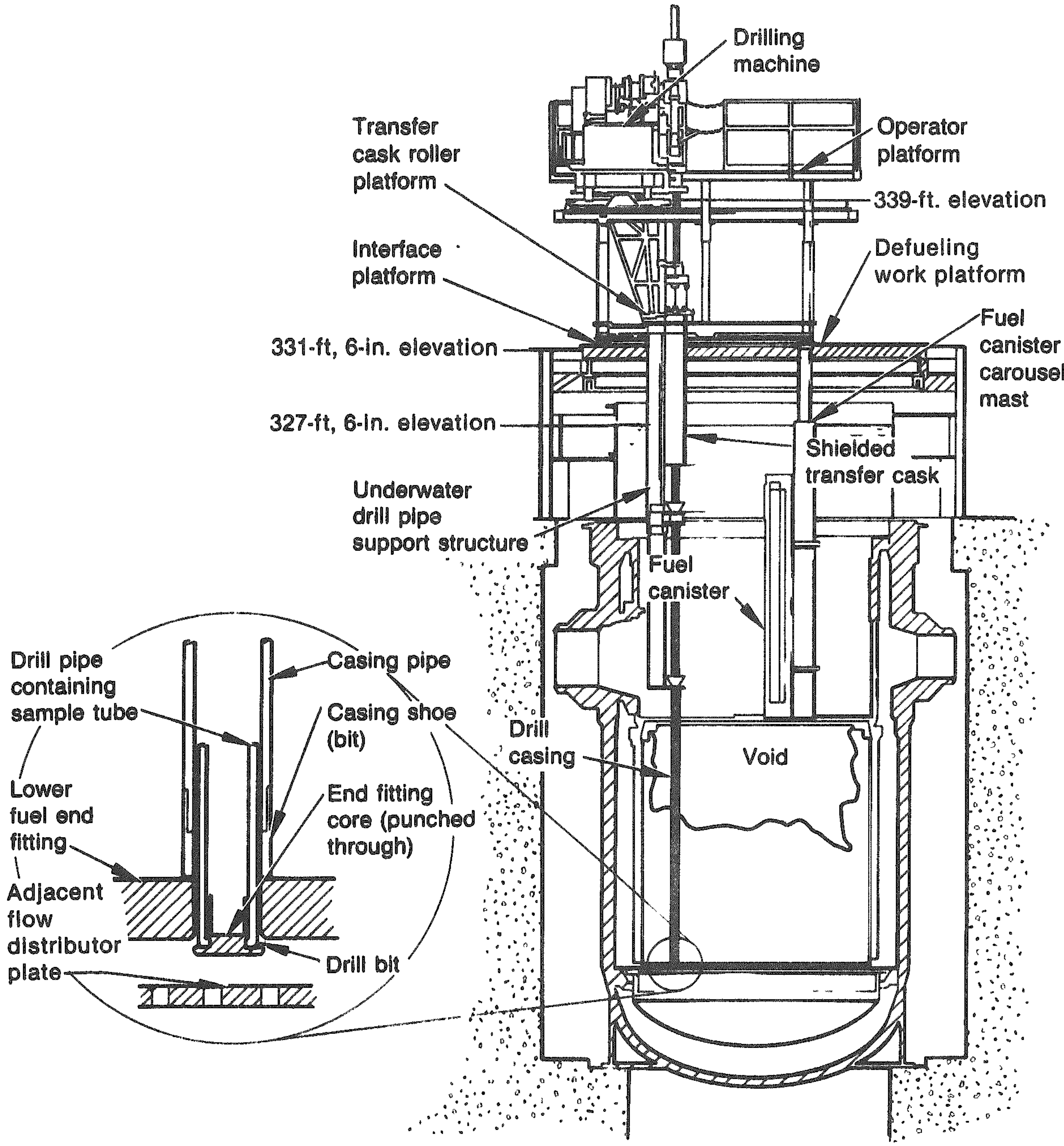

Figure 15. The core sample drill is shown completely inserted before sample withdrawal.

3.4.4 Gamma Scanning of In-Core Detectors

A series of measurements were made in an attempt to quantify the amount of debris present on the lower reactor head. This analysis was performed by measuring the gross gamma radiation profiles with a miniature ion chamber inserted into the calibration tubes of in-core instrument assemblies. At the L-11 location, the greatest penetration (approximately 19 inches above the bottom of the reactor vessel) was achieved. The gamma radiation profile of the L-11 instrument assembly indicates the presence of a low activity region extending about three inches above the level of the bottom of the reactor vessel head. It has been suggested that this low activity zone may be due to the presence of non-fuel material with a low melting point deposited on the lower head of the reactor during the accident.

4. REACTOR EVALUATION AND DATA ACQUISITION

4.1 Reactor Evaluation Program

4.1.1 Plenum Removal

Following the upper plenum assembly jacking, inspection, and debris removal activities in 1984, the plenum remained in its jacked position inside the reactor vessel during the early part of 1985. The upper plenum is a 55-ton cylindrical assembly approximately 12 ft high and 12 ft in diameter and contains the reactor control rod guide tubes.

There was a five-month interim period between jacking the plenum and the time the plenum was actually removed from the water-filled reactor vessel. During this period, workers concurrently prepared for the plenum's removal and installed defueling equipment in the Reactor Building's refueling canal. This defueling equipment installation had to precede plenum removal because the canal was partially flooded to shield the stored plenum.

In preparation for the final lift and transfer of the plenum assembly, a test assembly was fabricated and used to load test and check out the performance of the lifting and handling hardware. Further inspections were performed to make certain that no physical conditions were present that would preclude an interference-free removal. Additional plenum underside cleaning was done to minimize the possible spreading of contamination during the transfer of the plenum assembly. Separately, a video inspection of the lower reactor vessel head revealed new insights on the observed character, magnitude, and extent of core damage. As a precaution, the Plenum Removal Safety Evaluation Report was significantly expanded to address a plenum assembly load drop analysis.

Except in the deep end area, much of the Reactor Building refueling canal was left dry throughout the plenum removal operations. A specially fabricated dam was installed to isolate canal flooding in the canal's deep end. In this manner, a smaller volume of contaminated water will have to be processed during future cleanup activities. Moreover, cleanup-worker radiation exposure on the defueling platform is minimized by using the "dry" canal concept because of the shielding provided by the canal walls.

On May 15, 1985, the upper plenum assembly was successfully removed intact from the TMI-2 reactor vessel (see Figure 16). Using the Reactor Building's polar crane, the plenum was transferred to a storage stand immersed in water located in the dammed, deep end of the Reactor Building refueling canal. The canal water will shield cleanup personnel working inside the building from the plenum's radiation field. The plenum assembly, which was somewhat deformed by heat produced during the accident, will remain stored in the canal indefinitely. Cleanup planners are presently formulating plans that will include the eventual disposition of the plenum.

The actual plenum removal operation took less than three hours. The operation, in addition to a postplenum removal inspection, was videotaped to assist in future defueling and data acquisition efforts. The plenum removal gave cleanup workers the first direct, human-eye overview of the reactor vessel core region since the accident. Previously, cleanup planners relied on video camera probes to look inside the reactor and could only examine small areas of the plenum at any one time.

Actual radiation and worker exposure levels during the operation were less than those conservatively estimated by engineers before the start of operations. Whole body doses for the five-member plenum removal crew were held to a minimum because they worked from a shielded enclosure. Radiation levels inside the Reactor Building increased temporarily during the plenum's actual removal and transfer to the canal. After the plenum was lowered into the water-filled canal, radiation levels inside the Reactor Building returned to preremoval levels.

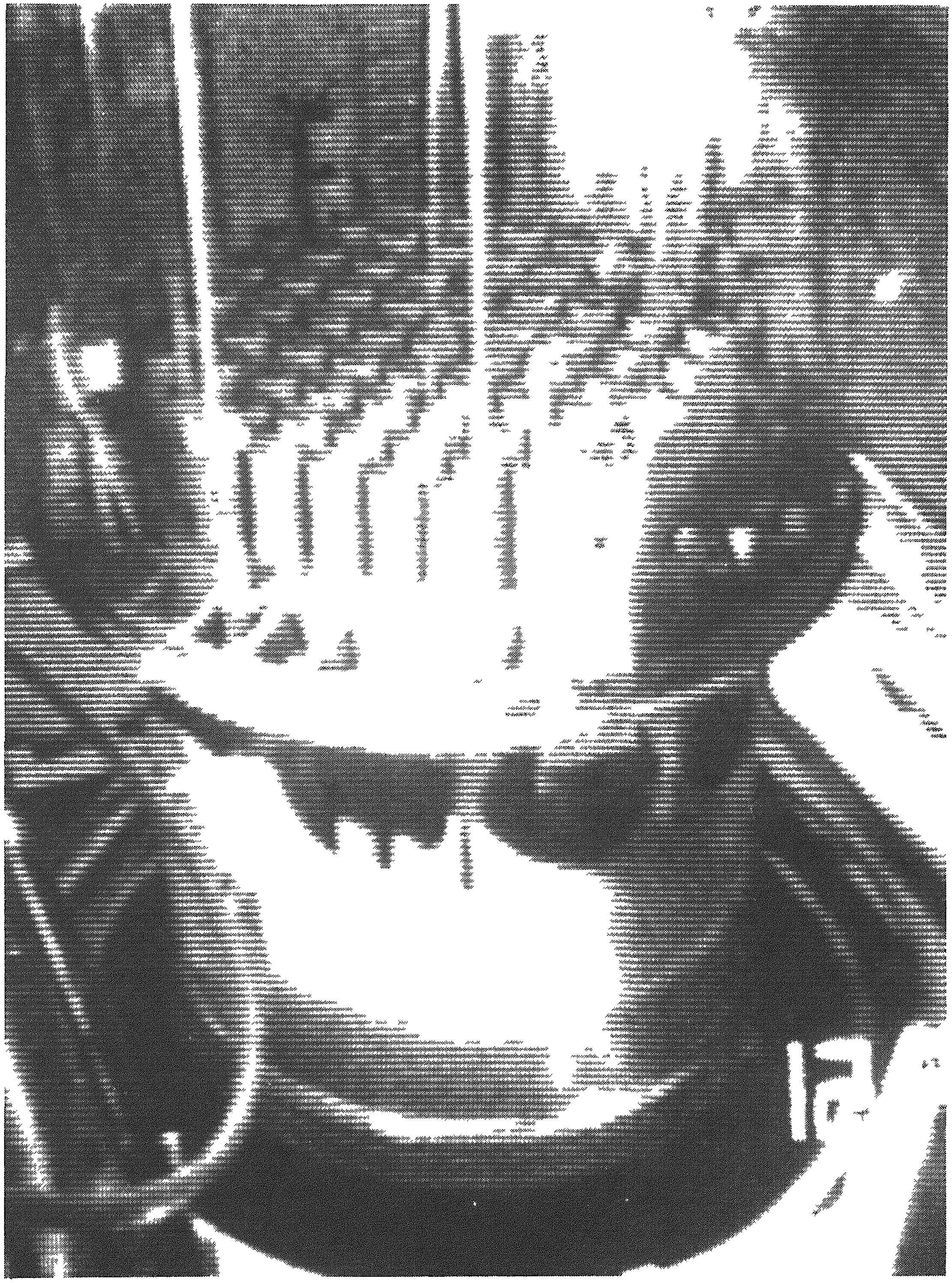

Figure 16. Removal of the plenum.

Finally, a plenum removal tooling report was prepared by the project's primary tooling subcontractor, Babcock & Wilcox Company, and published as GEND-INF-051, Equipment for Removal of the TMI-2 Plenum Assembly, April 1985.

4.1.2 Fuel and Core Debris Removal

During the year, the major defueling tooling system and support equipment components were installed and successfully checked out inside the TMI-2 plant (see Figure 17). Actual defueling operations, which involve packaging approximately 100 tons of uranium dioxide fuel and 50 tons of reactor vessel core components, began in October. Reactor vessel defueling is expected to take about 18 months. More specifically, defueling operations are defined as those activities involved with placing reactor vessel core debris into defueling canisters and the transfer of those canisters to temporary storage in the Fuel Handling Building.

The defueling system (see Figure 18), with tools designed to operate remotely underwater in both manual and power-assisted modes, continues to follow the basic technical approach that was established as a result of intensive planning in previous years. This approach initially calls for the simplest, least developmental tools. The tooling is only permitted to become more complex and developmental as proof-of-principle testing and operations experience are gained, and as known core conditions dictate.

The major defueling system components installed include a rotating shielded work platform and associated support structure; a rotating canister positioning system; and vacuuming, control, video, and lighting systems. The "toolbox" for initial defueling activities includes a flexible assortment of long-handled tools, debris buckets, and canisters. This initial defueling tooling is capable of picking and placing loose debris in the upper and central reactor vessel regions. More specifically, the long-handled tools consist of hooks, grippers, parting wedges, shears, and tampers; and camera, lighting, and debris container handling tools.

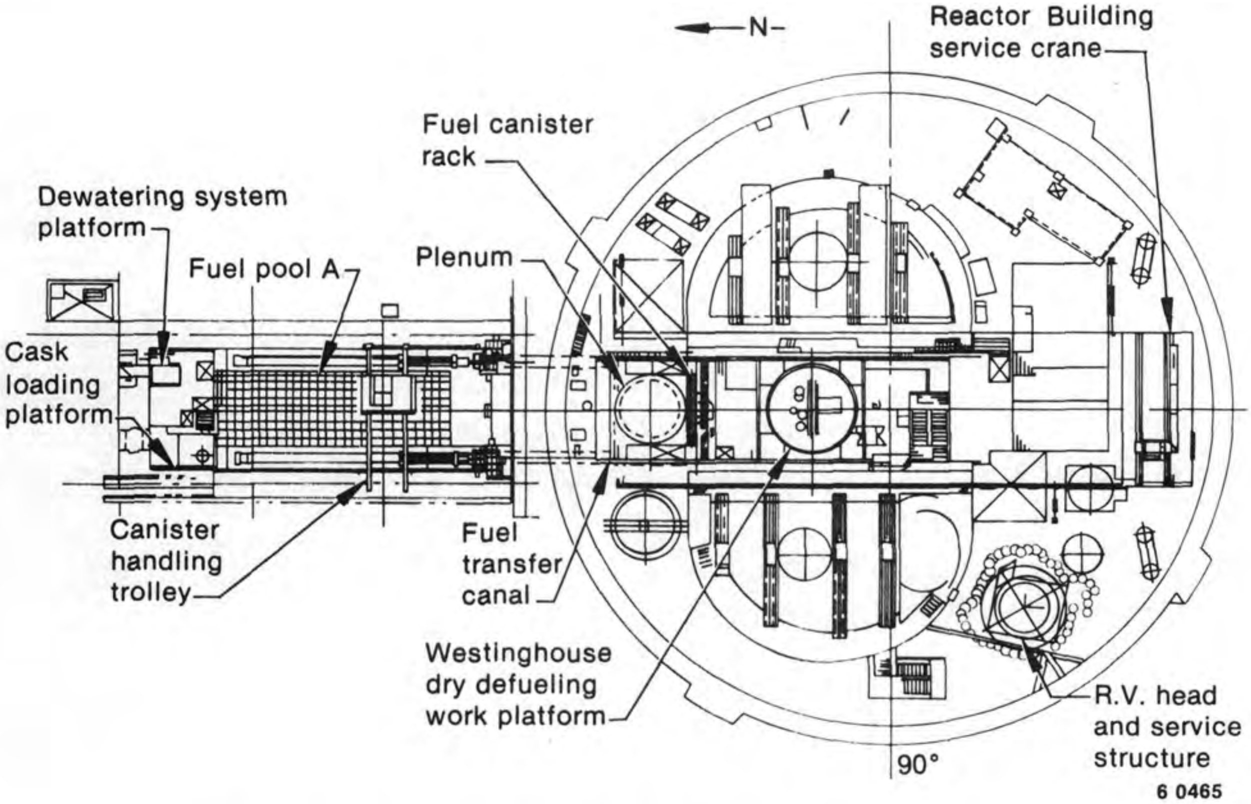

Figure 17. TMI-2 defueling general arrangement.

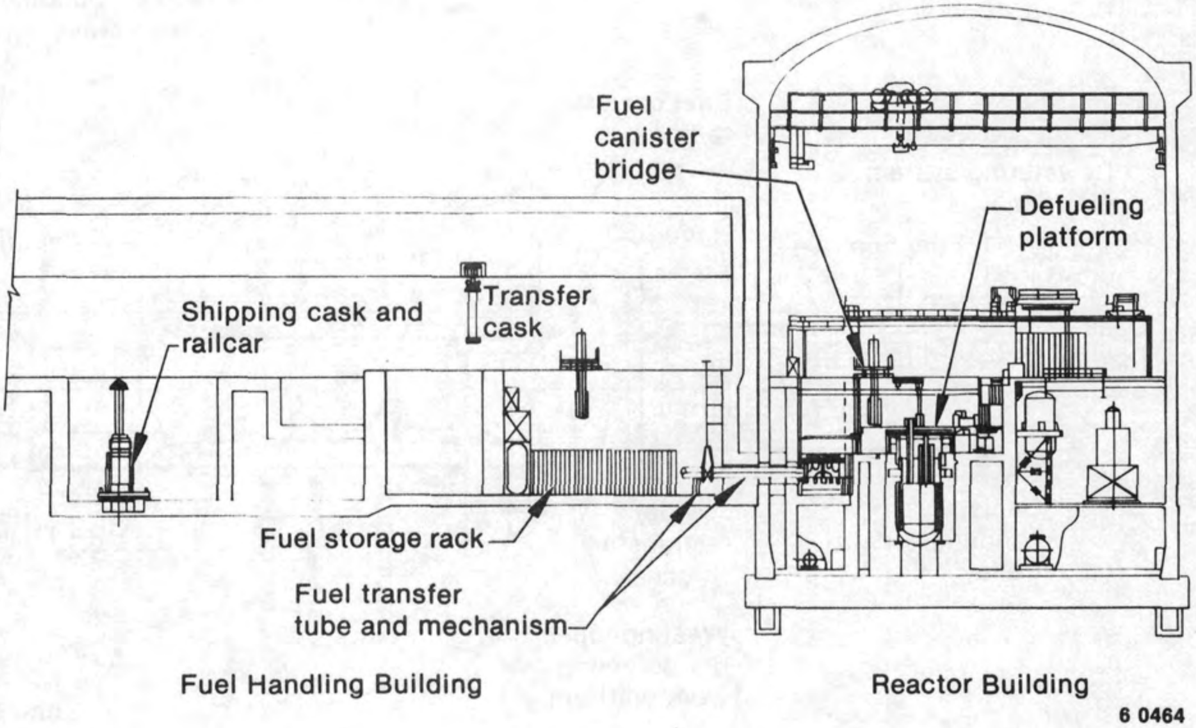

Figure 18. TMI-2 defueling system.

The central feature of the innovative defueling system is the shielded work platform built by Westinghouse. It is on this platform that defueling operators work 9 ft above the reactor vessel flange. The operators insert tools through removable hatches arranged in a T-slot configuration in the platform. To provide radial access to the reactor vessel, this platform has a rotational capability. The platform also provides support for major defueling subsystems including the independently rotating canister positioning system carousel and the fines/debris vacuum system. The carousel holds up to five defueling canisters.

While much of the defueling tooling support equipment was custom designed to satisfy the unique needs of the TMI-2 plant cleanup, the long-handled tools reflect commercially available hardware with minor modifications.

As the result of core condition data acquired earlier in the year, combined with theoretical assumptions, it is highly probable that a hard crust exists immediately below the reactor vessel's loose debris bed. The recently acquired data supporting this conclusion were collected as a result of a debris bed probe, three video inspections of the lower head, a hydraulic displacement test, and a variety of core debris samples. Core data further suggest that once-molten masses, composed of oxidized and unoxidized ceramic/metallic properties, could be fused to structural components in the core support assembly's lower grid and lower flow baffle regions. As a worst condition, it is possible that the hard crust may not be friable and it may be mixed with large stainless steel components. Unlike friable materials, ductile materials require more rigid and precise cutting techniques. In addition to accommodating these conditions, tooling will also be required to adapt to a wide range of in-vessel positions and orientations. Consequently, additional tooling for this "bulk" defueling is being developed, based on an integrated assessment of tooling requirements to defuel the hard crust transition zone and the lower core region, including the core support assembly and lower reactor vessel head. This tooling augments the initial, or pick-and-place long-handled defueling tools. It is recognized that this integral tooling system and its interfaces need to be reasonably flexible in the event that future data acquisition tasks result in modifications to enhance tooling effectiveness.

Some of the specialized tooling includes a clamping and cutting station, manual tool positioner masts, hydraulic impact chisels, a robotic manipulator arm, and incore instrument cutters. The proposed lower reactor vessel and core support defueling tooling, which will complement both the initial and bulk tooling, includes a water jet cutter, an abrasive saw, fuel assembly lift tools, and a large debris vacuum system (capable of vacuuming material larger than fuel pellet size). Aside from the initial defueling equipment that is currently at TMI-2, the balance of the required tooling is currently undergoing final design or is in fabrication.

A core boring machine, which is presently at TMI-2 for obtaining debris core stratification samples for research purposes, is being considered by TMI-2 defueling engineers as a contingency defueling production tool. This device is patterned after a conventional core drilling unit used in the petroleum industry. In addition to its data acquisition programmatic mission, GPU Nuclear is considering the core drilling unit for defueling those reactor vessel areas where previously molten, fused masses of core debris may be encountered. In this event, a solid-faced drill bit instead of a coring bit would be used to penetrate the core material.

In addition to the previously described defueling tooling system, specially designed defueling support equipment has also been staged and installed in the TMI-2 plant. A defueling water cleanup system (DWCS) has been installed to control reactor vessel water turbidity. This will ensure good visibility for the underwater camera. Another function of the DWCS is to control the presence of soluble radioisotopes present in the RCS water. A 5-ton-capacity service crane was installed and successfully tested inside the Reactor Building. This crane represents the defueling system's primary equipment handling device and will effectively eliminate direct reliance on the large-capacity Reactor Building polar crane. As a supplement to the service crane, two smaller-capacity jib cranes have been mounted on the defueling work platform above the reactor vessel flange.

To handle and transfer the loaded debris canisters, the auxiliary fuel handling bridge and trolley in the Reactor Building, and the storage fuel handling bridge and trolley in the Fuel Handling Building, were modified. The main fuel handling bridge in the Reactor Building was removed. Specially designed canister transfer shields were installed in conjunction with the Reactor Building and Fuel Handling Building canister handling bridges. The as-built fuel transfer systems connecting the Reactor Building refueling canal with the Fuel Handling Building spent fuel pool were modified to permit the underwater handling and transfer of loaded debris canisters.

The canister dewatering station to be installed in the Fuel Handling Building fuel pool was temporarily assembled and checked out in the Turbine Building. A limited amount of canister dewatering, using a recently developed, but much simpler system, will initially be performed inside the reactor vessel before the canisters are transferred outside the vessel. Newly fabricated canister storage racks were installed in the Fuel Handling Building fuel pool. These racks will hold up to 252 of the special defueling debris canisters.

Three special debris canister designs were developed for TMI-2 defueling needs. They have the same basic outside dimensions to provide handling device and storage system compatibility. Each design has engineered features to provide radiological, mechanical, and thermochemical safeguards for the type of fuel and core debris planned for placement in the canister. Defueling planners estimate that 280 canisters will be necessary to hold the TMI-2 reactor fuel and core debris. Of these, 45 canisters were delivered onsite by the end of the year. The canisters are being manufactured by Babcock & Wilcox, Joseph Oat Company, and Nuclear Energy Services.

Once loaded, the canisters, which have a design life of at least 30 years, will be lifted from the vessel into the dry transfer shield and lowered into the deep end of the Reactor Building's refueling canal. At that point, they will either be placed in a storage rack or passed directly through a fuel transfer tube into the recently flooded spent fuel pool in the Fuel Handling Building by one of two fuel transfer mechanisms. The fuel pool can store up to 252 canisters until GPU Nuclear is ready to transfer them to the Fuel Handling Building truck bay. There, the canisters will be transferred to a railroad cask for shipment to INEL for research.

Initial defueling operations began on October 30, 1985, after an intensive training and tooling/procedure checkout program for defueling personnel. There was also a review by GPU Nuclear's Readiness Review Committee, NRC certification of GPU Nuclear's fuel handling senior reactor operators, and NRC approval of defueling safety analysis and procedural software before operations began. The first step was for workers to use long-handled tools to rearrange core debris that interfered with completing the installation of the canister positioning system. Following the successful completion of this work, debris was picked-and-placed inside fuel canisters that were mounted on the canister positioning system carousel. By the end of the year, two full and two partially filled fuel canisters were on the canister carousel inside the reactor vessel. Additional core debris was sized for future canister loading and final preparations to operate the fines/debris vacuum system were completed.

A defueling tooling development report was prepared by the project's primary tooling subcontractor, Westinghouse Electric Corporation, and published as GEND-INF-065, <u>TMI-2 Defueling System Design Description</u>, March 1985.

4.2 Onsite Data Acquisition

4.2.1 Instrumentation and Electrical

The Instrumentation and Electrical (I&E) staff of EG&G Idaho completed evaluation of the resistance temperature detectors, dome area radiation monitor, pressure transmitters, and incore instrumentation. In support of the final I&E program element, Cables and Connections, 17 cable/connection samples were retrieved from the Reactor Building. In addition, a major TMI-2 I&E program objective was completed with the transfer of the cable/connection program technology to the nuclear power plant industry.

During the past year, the I&E staff collected the third set of data on approximately 75 circuits that have been repeatedly tested over a period of 2 1/2 years. This data has provided valuable trending information that shows some circuits continuing to degrade, but overall demonstrates the strength of the basic instrument and control designs at TMI-2. From the cable connection program, it has generally been concluded that most circuits, cable/connections, and end instruments that failed at TMI-2 did so not as a direct result of the accident but failed because of cumulative effects that were accelerated by the accident. It is highly probable that these components would have eventually caused operational problems under normal operation unless they had received preventive maintenance.

The technology to test electrical systems in situ, detect weak areas, and use this information for maintenance planning appears to be one of the major benefits coming out of the TMI-2 research programs. The I&E staff has been working in earnest with industry to transfer this technology. The electrical circuit characterization and diagnostic (ECCAD) system (Figure 19), which was designed to acquire the in situ data, was demonstrated at several workshops and conferences in the past year with considerable response from industry. During the next year, the I&E program will be working with the Electric Power Research Institute (EPRI) to develop an actual pilot demonstration program with a nuclear power plant.

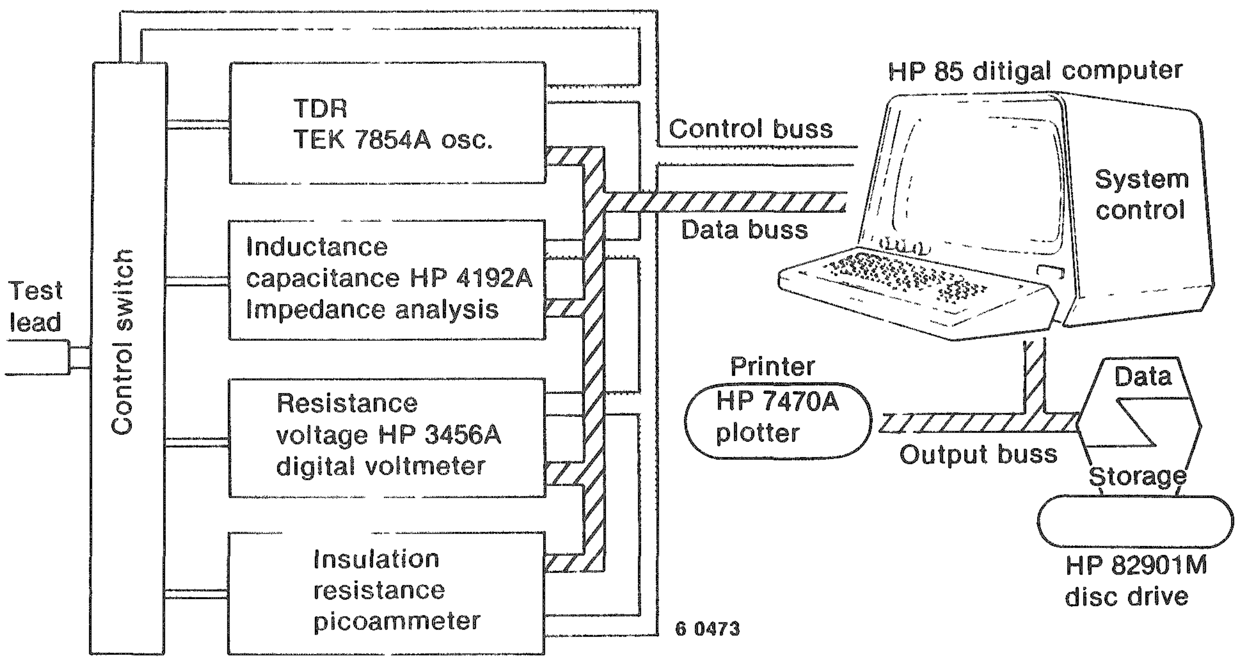

Figure 19. ECCAD system configuration.

The ECCAD system uses standard test equipment under computer control to quickly acquire meaningful data to analyze a circuit. The computer removes human errors that can be introduced in data acquisition and provides a repeatable and trendable data base. This data base can indicate circuit condition and is useful in maintenance and plant life planning.

4.2.1.1 Cables and Connections. The TMI-2 Cables and Connections Program was established to investigate the consequences of the loss-of-coolant accident (LOCA) on cable and connector components in the Reactor Building. The capability to receive readout signals from, and supply energizing voltages to, Class 1E instruments is essential to reactor control during periods of environmental stress. Therefore, it is important to characterize the functional properties of cable channels during accident and postaccident conditions.

The cable channels were characterized with the ECCAD system through a series of static, in situ cable tests designed to determine the effects of the accident on the operation of all cable channels. The cables and connections program includes all components in a given electrical channel or circuit from the Reactor Building electrical penetration assembly up to, but excluding, the end instrument. This definition encompasses penetration assemblies, terminal blocks, splices, bulk cable, and connections. The data gathered are already beginning to assist the nuclear industry in the maintenance area as utilities begin to adopt the ECCAD system technology and as standards groups begin to emphasize maintenance and good practice through publication of guidelines. The I&E staff is directly involved in preparation of these guidelines. As the final data are obtained and analyzed, the information is expected to help the nuclear industry improve the reliability of these components, as well as reevaluate stringent qualification testing procedures and regulatory requirements in light of a LOCA event.

To accomplish the cables and connections investigation, the program was divided into two distinct phases. Phase I, in situ testing of the cables from the outer electrical penetration junction box, was completed in

1984. Based on the data obtained, cables and channels exhibiting anomalous behavior were identified. Phase II involves the removal of selected cables from the Reactor Building for a thorough offsite examination. Seventeen samples were removed from accessible areas in the Reactor Building during 1985.

4.2.1.2 Results of In Situ Test Data Analysis. The results obtained from analyzing the in situ data demonstrate that the electrical circuits can be characterized in terms of the electrical operating parameters and that these electrical parameters define the condition or health of the electrical circuit.

The data obtained to date from TMI-2 indicate that the failure rate of electrical circuits is increasing. Analysis of types of degradation detected suggests that these are a result of moisture intrusion, probably initiated by the LOCA of 1979. The significant trend shown in Figure 20 is that the percentage of total circuits tested that exhibit new indications of degradation has increased from 7.5% in 1983 to 23.5% in 1985.

4.2.1.3 ECCAD System. The ECCAD system is a computer-controlled measurement system designed to characterize electrical circuits in nuclear power plants. The I&E Program developed the system to assess the damage to electrical circuits caused by the accident at TMI-2. The system has been demonstrated to enhance maintenance activities by diagnosing problems in electrical circuits and can provide a data base for trending electrical circuit conditions.

The ECCAD system characterizes the electrical parameters that might impact the ability of a circuit to perform its electrical function. For example, if the circuit is a motor for a motor-operated valve, the ECCAD system will determine if all connections or contacts are good, if proper voltage can be applied to operate the motor, and if the motor is electrically functional. The ECCAD system has a built-in capability to check internal calibration before each measurement and to confirm that the circuit is returned to its original condition after testing is completed.

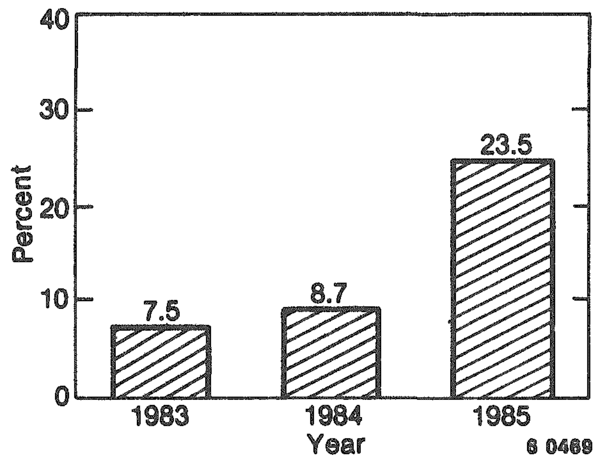

Figure 20. Newly detected anomalies by percentage of circuits not previously known to have anomalies.

In an application at Shippingport (Pennsylvania) for the NRC Nuclear Plant Aging Research program, approximately 500 circuits were tested with the ECCAD system in a two-week period.

The ECCAD system is composed of electronic test equipment that is readily available on the commercial market. The system is controlled by a Hewlett-Packard 85 personal computer. The computer controls the individual instruments, setting critical factors. It performs a self test on the systems, and it sequences the instruments. It also formats the data, ensuring a standard data set of high quality and eliminating much of the operator uncertainty that often plagues long-term (trending) data acquisition.

4.2.1.4 Analytical Approach. The basic analytical approach is to make a comparison between the actual test data and the predicted data or earlier test data. When this comparison is made, any changes are noted and analyzed to determine their cause.

At TMI-2, the most effective approach to determining the cause of change has been to define the various expected operational problems and to search the data for clues to those expected problems. This is accomplished by modifying the electrical model of the circuit to simulate the operational problem. The expected change in the electrical data can then be predicted. Using this approach, a relational matrix was developed for the TMI-2 data and has been highly effective in flagging suspected problem areas. The two most common anomalies appear to be wet cables/connections and wet end devices. Recent data also are showing an increasing indication of degraded conduction paths.

4.2.1.5 Example of Data. Long-term degradation was detected in a circuit connecting to normally open contacts of a relay on reactor coolant pump RCP-2B. The connecting cable, three twisted pairs of 16-gauge conductors, was typical for TMI-2. The circuit included a Reactor Building penetration that was known to have been partially submerged during the accident. The circuit was tested in August 1983, May 1984, and again in July 1985. A comparison of the time domain reflectometry (TDR) data is shown in Figure 21.

In August 1983, the TDR plot indicated a slight offset at the front end, a sign of early degradation. This circuit was considered functional.

When this circuit was tested again in May 1984, the TDR plot indicated an impedance increase at the penetration. This indicated that the cable was severely degrading at the penetration and was opening.

The latest testing in 1985 indicated on the TDR plot that the circuit was completely open at the penetration.

Laboratory experiments of similar type circuits produced the same effects that were observed above. By wetting the inner-liner penetration box on a circuit that is routed to a normally closed switch, the circuit characteristics changed in a manner similar to the TDR plot above.

All of the evidence for this example strongly indicates that the problem was initiated by the water during the LOCA, but that degradation was slow and due to corrosion of the circuit connection at the Reactor Building inner-liner penetration box. This evidence of damage is consistent with known environmental effects at this location.

4.2.1.6 Significance of TMI-2 Data. The data base established at TMI-2 could likewise be established at any nuclear power plant or similar facility as part of a normal maintenance and surveillance program. This would provide the capability for rapid, accurate diagnostics as well as trending data to ensure that there is no electrical deterioration of the circuits. The data are easily acquired and can be interpreted without complex analysis. The cost benefits are obvious because maintenance could be planned. Proper maintenance action would often involve very simple housekeeping after detecting early signs of degradation. Such actions might include:

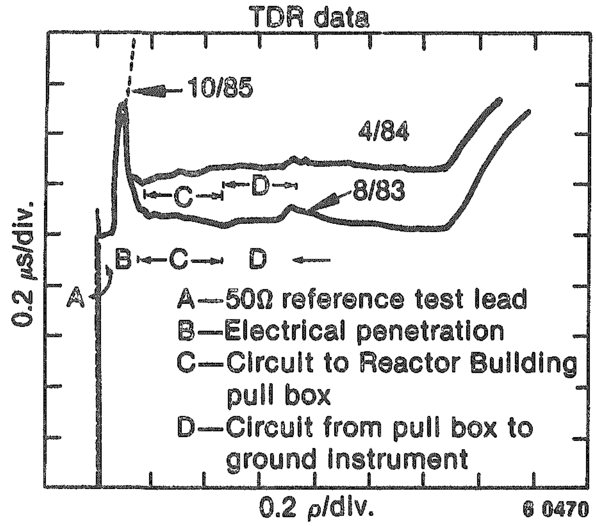

Figure 21. Time-domain reflectrometry plot of circuit connecting to relay contacts at reactor coolant pump, showing continuing degradation.

- Cleaning the penetration boxes

- Eliminating moisture intrusion paths

- Cleaning all termination points

- Replacing terminal blocks with environmental splices when possible

- Replacing seals

- Replacing defective components.

Further, these actions would not be necessary until an anomaly was detected. Most circuits might never need additional attention; however, those in stressful environments might require surveillance in order to ensure functional capability.

4.2.1.7 Material Analysis. The 17 samples removed from the Reactor Building were shipped to the Hanford Engineering Development Laboratory, where they are being examined for any changes in material or electrical properties caused by the accident. The first samples to be examined were the connection/cable assemblies from the seal table for the incore instruments (Figure 22). These assemblies were given priority because their safety classification has been upgraded at nuclear plants since the TMI-2 accident and because data from these assemblies, obtained by the I&E program in the previous year, has played a key role in understanding the progression of damage to the reactor core during the accident. Data being obtained include the following:

- Both full-length and detailed x-rays of each assembly

- Both full-length and detailed photographs of each assembly

- Detailed visual inspections for mechanical damage, areas of severe corrosion, or unusual residue deposits on each assembly

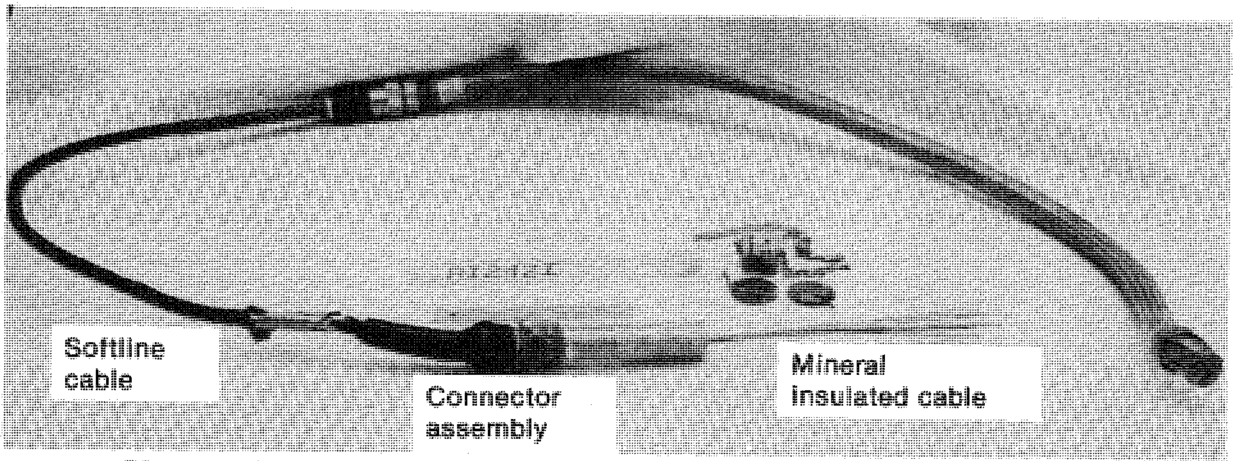

Figure 22. TMI-2 incore instrument connector cable assembly.

- Measurements of conductor electrical conductance

- Measurements of insulation resistance

- Each test assembly has been parted at the electrical connector after the above tests and electrical measurements repeated on each subassembly

- Tensile tests, using ASTM Standards D-3032 and D-638, on conductor insulation and cable sheath materials.

Test results on these connection/cable assemblies from the incore instruments include the following:

- All male connector/soft cable subassemblies possessed conductances of less than 1 ohm and values of insulation resistance greater than 10^{10} ohms, thereby demonstrating good functionality

- Corrosion was evident on some connector interior surfaces, but all surfaces of electrical pins and plugs, as well as the pin/plug insulator materials, were bright, clean, and free from visible contamination

- Several cable subassemblies showed failure due to low resistance shorting of the electrical conductor to ground

- All test assemblies exhibited discontinuities or comparatively high resistances in ground circuits. X-rays suggest the cause of failure was associated with the manner of connecting sheath grounds in the splice adjacent to the connector

- The majority of failures occurred in the female connector/hard cable assemblies

- Several conductance measurements exhibited erratic behavior that did not repeat after the connector was mated and demated

- Two test assemblies exhibited abnormally high resistance values in active circuits--15 ohms and 9 ohms (less than 1 ohm is required)

- The electrical connector exterior components exhibited a green and white corrosion area or deposit, which is relatively radioactive compared to other areas of the assembly

- White, powdery deposits were observed on the braided conduit. These deposits are "radiation hot spots"

- The tensile strength of cable conductor insulation material appeared reduced between 13 and 17%

- The average elongation at the tensile break point is reduced from 12 to 70% below control cable material

- The TMI-2 accident environment severely deteriorated the outer sheath of the test cables in a manner that would allow the environment constituents to enter the cable interior.

Analysis of data to assess the functional condition of the assemblies and establish causes of impairment where they have occurred will be completed in early FY-86.

4.2.2 Status Report

An interim status report, TMI-2 Cable/Connections Program FY-85 Status Report (GEND-INF-068), was published. This report discusses anomalies in the electrical circuits located in the Reactor Building as detected with the ECCAD system. Most of these circuits have not been physically inspected to date due to radiation environments.

This report also presents the results of laboratory tests on cables and terminal blocks. The tests measured the variation in the cable parameters for various test conditions, including a dry and wet cable, a cable looping in a cable tray, cables inserted into a dry and water-filled conduit, and a cable terminated with a terminal block submerged in water.

An evaluation of the available TMI-2 in situ test data indicates that of the circuits inside the Reactor Building that were tested, 3.5% are presently nonfunctional.

4.2.3 Instruments

The I&E program completed all evaluations of radiation, temperature, and pressure instruments in FY-84. During FY-85 all reports were completed and a summary report (GEND-050) was prepared that brought together all of the I&E program results. This will be followed by an industry assessment of the results which will be documented as a GEND report.

4.3 Information and Industry Coordination

The EG&G Idaho Information and Industry Coordination (I&IC) and the GPU Nuclear Industry and Government Coordination staffs continued the refinement and development of technology transfer topics that will assist the commercial nuclear power industry. Work continued on prediction of hydrogen gas generation in sealed radioactive waste containers. Other areas of technology transfer and assistance were in plant maintenance of electrical equipment, beta dosimetry, waste management, outage maintenance, and American Nuclear Society (ANS) standards.

Aside from publications and conferences, a major medium for distribution of this information is the workshops sponsored by the EPRI-funded Maintenance Equipment Applications Center (MEAC). The objective of MEAC is to commercialize products useful in nuclear power plant maintenance that have been developed by EPRI. In 1985, GPU Nuclear and EG&G Idaho participated in three MEAC workshops on the following topics: outage maintenance/personnel factors, the DOE's ECCAD system, and the use of video/CADD systems at operating power plants.

4.3.1 Calculation of Safe Storage Time for Radioactive Waste

A major portion of the I&IC effort during 1985 was in the area of developing a safe and economical (person-rem and jobhours) method of calculating hydrogen gas generation in sealed radioactive waste containers. This work is being done in response to an NRC inspection and enforcement notice regarding hydrogen gas generation. The NRC is requiring nuclear plant operators to ship wet radioactive waste containers within 10 days of preparation and sealing or, if they cannot meet the 10-day deadline, to vent the containers before shipment.

This NRC requirement is in response to concerns that even low-level radioactive waste may generate hydrogen gas. On January 1, 1986, the Low-Level Radioactive Waste Policy Act went into effect, and many utilities still do not have a low-level waste disposal site available to them. Without a method of determining if and when a particular waste container has attained a combustible gas condition and being unable to ship the container within 10 days, these utilities will have to assume that all of their wet low-level radioactive waste poses a gas generation problem. The utilities will have to store and handle the waste accordingly, which could be expensive in the manhours and manrem exposure.

Based on experience with gas generation in the EPICOR II and submerged demineralizer system containers and the experiments and data collected to determine gas generation rates, the nuclear industry may have another, more reasonable alternative: it is possible to calculate, based on known parameters concerning the waste and container type, the safe storage time before hydrogen gas concentrations reach a combustible mixture (4 to 5% by volume). I&IC has prepared such a calculation. With this information, a utility can produce a plant-specific procedure to determine safe storage time. Accepted by the NRC, the calculation will, in most cases, allow utilities much longer than 10 days for storage before shipment. Because the method does not require special tooling or equipment and the calculation is made with data the utilities already have, the method is cost-effective.

A meeting was held with the NRC Waste Transportation Certification Branch in early April. Representatives of the Edison Electric Institute and I&IC presented the EG&G Idaho/DOE method for calculating combustible gas generation in sealed waste containers. The NRC staff noted that the proposed method is a valid technique and agreed to amend certificates of compliance for waste shipments to provide for calculational analysis.

GEND-041, A Calculational Technique to Predict Combustible Gas Generation in Sealed Radioactive Waste Containers, has been prepared as a result of the work initiated by the I&IC group. It details a step-by-step method for predicting safe storage times for sealed radioactive waste containers. Numerous utilities have requested information and assistance on this issue. The fuel cycle and waste management division of the ANS requested that the I&IC staff prepare a paper on the hydrogen gas generation problem. In response, I&IC organized a session at the ANS Winter Meeting.

4.3.2 Technology Transfer, 1985

Members of the I&IC and GPU Nuclear staff have made lectures and presentations on TMI related topics at a number of professional meetings. The information disseminated at these meetings has been well received and valuable to the commercial nuclear audience, which includes nuclear utilities, architect/engineers, equipment manufacturers, and other interested parties. The TMI-2 Organization staff continues to be invited to participate in upcoming workshops, professional meetings, and standards committees.

4.3.3 Enhanced Surveillance Techniques of Electrical Equipment

Plant maintenance and surveillance of electrical equipment has been given new possibilities as a result of testing methods developed at TMI-2. I&IC has had two technical articles on this topic published in national trade journals. Also, EPRI has published a technical brief prepared by I&IC that tells how the ECCAD system was developed and its value in preventive maintenance programs.

These publications tell how in situ testing of electrical circuits provides information for engineers to evaluate the health of the electrical cable, junction points, and the end piece of equipment. The short-term goals of this I&IC program are to provide the utility audience with information that will enable them to:

- Enhance surveillance procedures to allow for a determination of the quality of operation and the likelihood of continued reliable operation of equipment being monitored.

- Receive technical specification relief through quick verification of the accuracy of abnormal instrument indications and provide a verifiable and reproducible basis for operator action and continue operations while a specific appropriate action is selected.

- Extend qualified life of class 1E components. In many cases qualified life estimates prove to be overly conservative. The ECCAD system provides evidence to support seeking an extension of qualified life when no circuit degradation is detectable.

The long-term objective of this program is to provide the utility with a means to acquire a comprehensive, auditable data base for quality assurance that is evidence for extending the "qualified life" of the electrical portion of a nuclear power plant. The data base will include information that can be computer sorted for directing maintenance planning, scheduling upcoming outages, and purchasing new equipment.

4.3.4 Summary Reports

The I&IC, EPRI, and GPU Nuclear staffs prepared a number of summary reports that present the results of various TMI programs, address issues of general interest to the commercial nuclear power industry, and provide a concise overview to help users see how TMI program activities could affect their areas of responsibility.

These summary reports have been made available to the commercial nuclear industry through a number of networks, including the Update, an international newsletter published by the DOE Technical Information & Examination Program; the Nuclear Operations and Maintenance Information Service; and the Nuclear Network of the Institute of Nuclear Power Operations. EPRI Technical Briefs on the following subjects were prepared in 1985:

- In situ, nondestructive methods of detecting debris in reactor coolant system piping
- Loose parts monitoring charge converters
- Calculational technique for predicting hydrogen gas concentration and safe storage and shipment periods for sealed radioactive waste containers
- Multi-element beta dosimeters for measuring dose rates
- Electrical circuit characterization system
- Video cameras for TMI-2 in-containment tasks
- Remote reconnaissance vehicle surveys exclusion area
- TMI-2 strippable coating test

- New decontamination technique using a self-stripping copolymer.

I&IC and Government and Industry Coordination continue to communicate new technologies and cost-efficient methods of plant operations learned from the cleanup of Unit 2. The emphasis of the work for 1986 will be transferring new technologies on waste management and defueling operations.

5. ACTIVITIES FUNDED BY THE JAPANESE PARTIES

Work that is at least partially funded by the Japanese Parties is described in this section.

5.1 TMI-2 Robotics Program

5.1.1 Reactor Building Sediment and Concrete Samples

The Remote Reconnaissance Vehicle 1 (RRV-1 or Rover), which had been used to perform video surveys of the Reactor Building basement in 1984, was also used to take samples there in 1985. GPU Nuclear developed a sludge collection system that employed Rover to scoop up sludge samples from the basement floor. This system consisted of a scoop mounted on an arm on Rover (Figure 23). An eductor-type pump on 305-ft elevation sucked the samples from the scoop into a shielded collection container, also on the 305-ft elevation, via a hose connected to the collection arm. On September 24, 1985, Rover successfully sampled three composite locations that were selected as representative of the sediment deposited in the impingement area of the Reactor Building basement. Table 7 shows the preliminary onsite analysis of these sediment samples.

On November 25, 1985, Rover was used to collect two concrete cores from the D-ring and impingement walls in the basement. The first sample was removed from the D-ring wall approximately 2 ft 6 in. above the floor and the second was from the impingement wall at 8 ft 4 in. above the floor. These samples were taken by a core boring device developed at Carnegie-Mellon University (CMU) and mounted on Rover (Figure 24). Sample dimensions and radiation measurements are shown in Figure 25. Onsite analysis of these samples, including a leach test, is continuing.

5.1.2 Makeup Demineralizer Cubicle Investigations

Radiation surveys were performed in demineralizer cubicles A and B using the robot Louie with an R07 gamma survey instrument attached at the

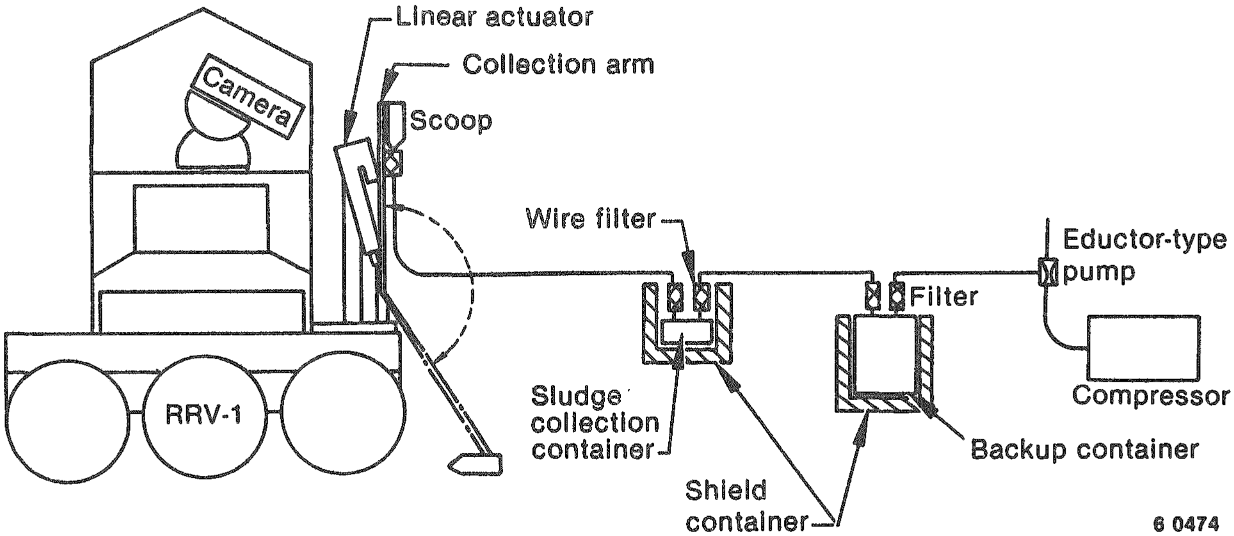

Figure 23. Sludge collection system.

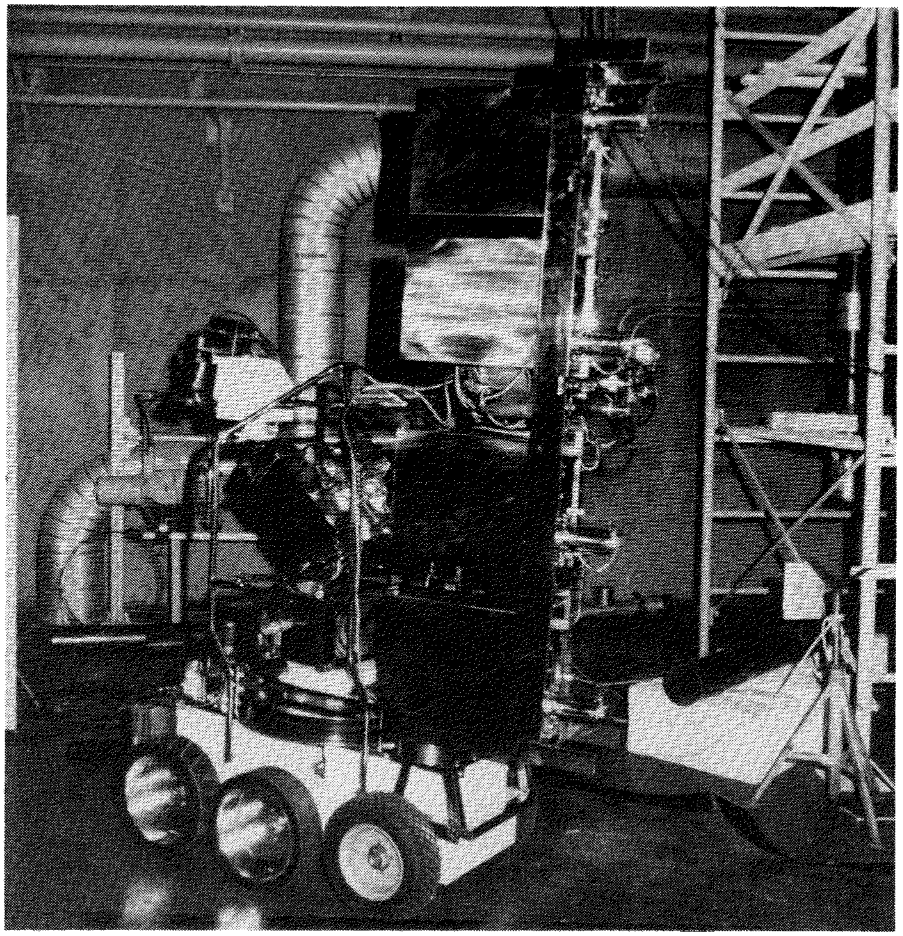

Figure 24. RRV-2 with core drill.

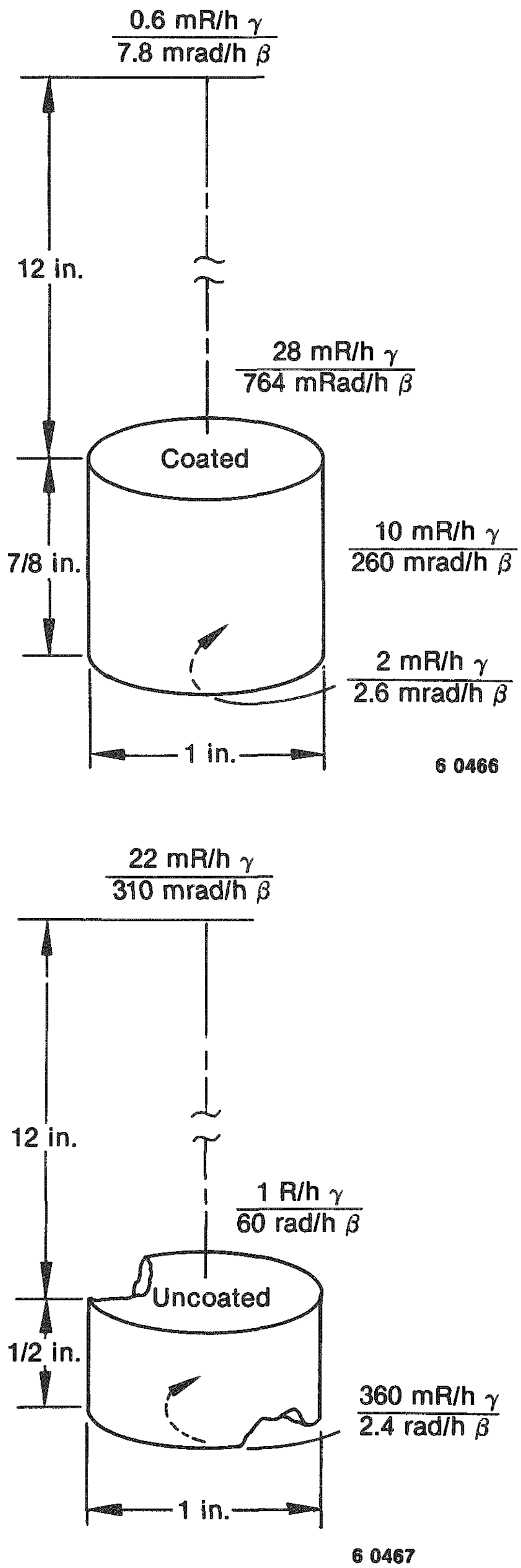

Figure 25. Core samples taken from D-ring wall (top) and from impingment wall (bottom).

TABLE 7. PRELIMINARY ANALYSIS RESULTS

	NaI Scan Results		
	Sample 1	Sample 2	Sample 3
Cs-137 activity (μCi)	674±70	826±107	69±14
μCi/g dry solids	86	23	30
Volume (mL)	75	163	10
Dry solids (g)	7.84	35.89	2.28
Wet solids (g)	9.68	44.31	2.81
Volume % dry solids	9.6	22.5	24.6

top and thermoluminescent dosimeters (TLDs) attached to the front of the robot (Figure 26). The R07 was used to obtain cubicle gamma exposure rates and the TLDs profiled the gamma exposure from the demineralizer vessels. The TLD data were modeled to estimate the postelution Cs-137 activity in the demineralizers. The quantities of Cs-137 estimated to be retained in the demineralizers are 380±75 Ci in A and 440±90 Ci in B. This information will be used in planning for removal of the resins from the demineralizers.

5.1.3 Development

Development of robot-based technologies continued in 1985 with cooperation between GPU Nuclear and CMU. A second Rover-type vehicle, RRV-2, was delivered from CMU and is being used to train operators in the TMI-2 Turbine Building. Development is also continuing on a Remote Working Vehicle, known as Workhorse. This robot will be fitted with various heavy duty tools to be used for decontamination tasks in the basement of the Reactor Building. Engineering work also began on a remotely controlled arm, which is planned for use in decontamination tasks in the D-rings.

As of January 1986, all of the $200,000 allocated to TMI-2 robotics by the Japanese Parties was billed.

5.2 Core Support Assembly Defueling/Disposition

GPU Nuclear completed two studies related to the core support assembly (CSA) in 1985. The first study, based on two studies performed by Babcock & Wilcox, presented a plan for CSA defueling (Core Support Assembly Defueling, TPO/TMI-145, September 1985). It provides a recommended sequence for defueling the CSA in-place within the reactor vessel and recommends tool types for each task. The plan also describes prerequisites to and limitations on various types of defueling tools that need to be developed. Since the publication of this report, design and fabrication of some of the tooling recommended therein has begun.

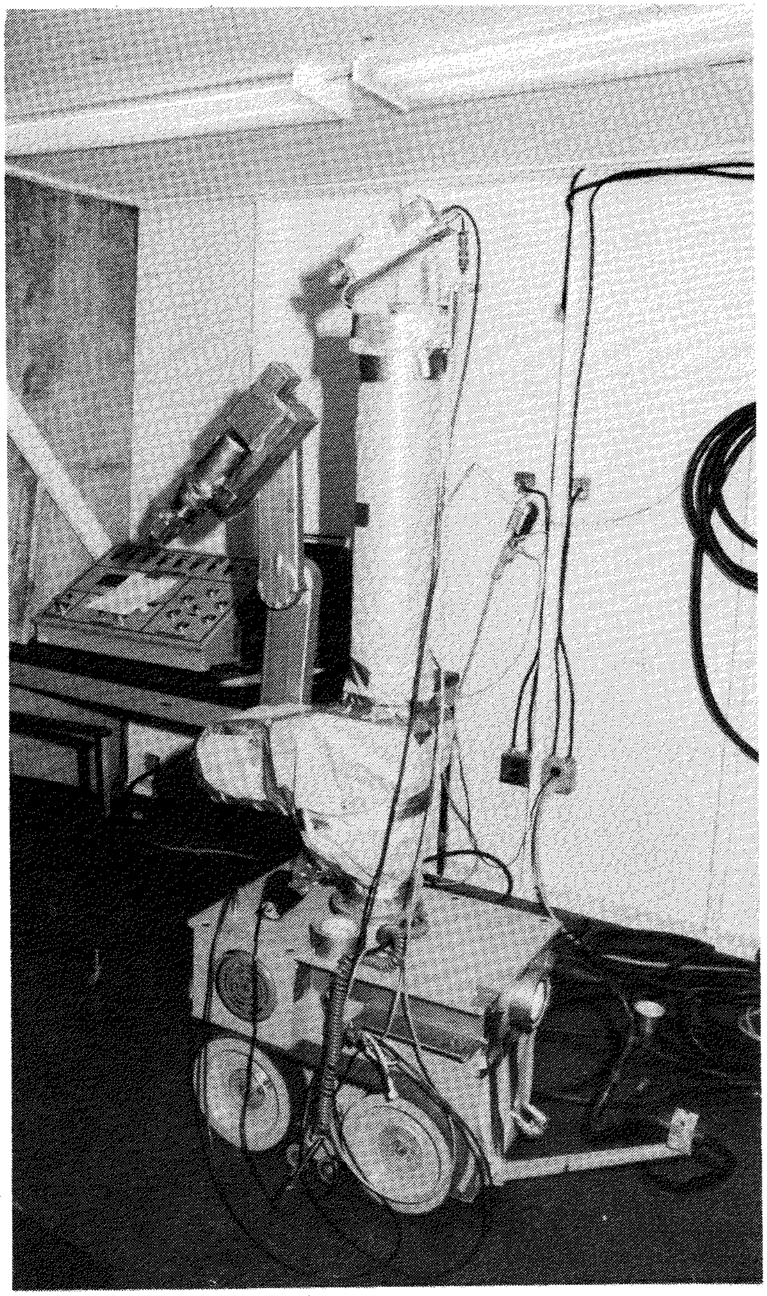

Figure 26. Louie, a robot used in demineralizer cubicle radiation surveys.

A second study was done on Large Reactor Component Disposition (TPO/TMI-144, November 1985) including the CSA. This study concluded that the reactor vessel head, plenum, CSA, internals indexing fixture, service structure, and defueling platform should remain in their current locations through the end of Phase III of the recovery program.

As of January 1986, $190,400 of the $200,000 allocated by the Japanese Parties to this task for Contract Year 1985 (May 1985 through April 1986) was billed.

5.3 Ex-Vessel Fuel Characterization/Removal

In addition to defueling the reactor vessel, engineers are currently studying possible approaches to defueling locations outside the vessel and inside the plant where fuel debris was transported as a result of the accident. In preparation, technicians are conducting surveys to locate fuel and fission products. This work, in addition to the accomplishments of the previous six years, will provide a sound technological basis for formulating decisions that will lead to the ultimate disposition of the TMI-2 plant.

The ex-vessel fuel removal task in 1985 consisted solely of characterization activities. These activities included gross gamma directional surveys, TLD mapping, gamma spectroscopy measurements, and some sampling outside of the reactor vessel.

Using a portable sodium iodide system to detect the characteristic 2.18 MeV gamma of $^{144}Ce/^{144}Pr$, various primary system pipes and components were examined for location of fuel debris. Table 8 contains results reported by GPU Nuclear of total fuel/fuel debris material located in the components examined.

In order to further characterize the amount of fuel debris and condition of the pressurizer, GPU Nuclear has performed a video scan and

TABLE 8. FUEL AND FUEL DEBRIS LOCATIONS

Primary System Component	Estimated Average Fuel Quantity (grams)
Makeup pump room 1A	1.7±0.4 to 8.4±1.7
1B	9.7±1.0 to 17±1.3
1C	0.97±.021 to 0.26±0.1
Pressurizer lower head	1100 to 25000
"B" core flood tank system - drain line	30 to 120
- check valve	2 to 10
OTSG "A" External - upper tube sheet	0 to 600
- manway	11

obtained a debris sample from the pressurizer internals. This sample will be sent to an offsite facility for detailed radiological and chemical evaluation.

Gamma scan measurements in the Reactor Building basement show evidence of fuel debris in an amount that would extrapolate to approximately 18 kg of fuel on the total basement floor area, assuming a uniform distribution.

Characterization of the primary system for fuel debris will continue in 1986. Components such as the steam generator A handhole cover backing plate, pressurizer manway backing plate, and an RTD thermowell will be acquired and sent to INEL for surface deposition studies. There will also be a task to investigate the once through steam generator (OTSG) internals. This will include video inspections, sampling, and gamma spectroscopy of the upper tube sheets, lower heads, and J legs.

As of January 1986, $497,200 of the $600,000 allocated by the Japanese Parties for Contract Year 1985 has been billed.

5.4 Reactor Building Dose Reduction/Stabilization

Reactor Building decontamination and dose reduction in support of defueling continued through 1985. The A and B D-ring walkways and the pressurizer missile shield were scabbled using the equipment described in the 1984 Annual Report. Also, contaminated cable trays in the refueling canal were shielded and shielding for the polar crane operator was installed on top of the A D-ring.

Other general dose reduction/decontamination activities involved scabbling floors on the 305- and 347-ft elevations, which had begun in 1984. Dose rates were further reduced on the 305-ft elevation by shielding of the lower part of the air coolers. This reduced dose rates in the immediate area by 50%.

Dose reduction/decontamination was also done in support of ex-vessel fuel characterization activities in the D-rings. Gross flushing of the upper elevations of the D-rings resulted in small reductions in dose rates and contamination levels. Radiological surveys then showed that the majority of the dose was coming from under the mirror insulation on the primary RCS components. Some of these insulation panels were removed from the pressurizer head, the A steam generator head, and the B steam generator. The exposed component surfaces were then flushed, but this caused no significant dose reduction. Studies are underway to determine the effectiveness of coating and corrosion removal as a decontamination method for these surfaces.

Tests were performed on one of the removed mirror insulation panels. The panel was flushed, which removed almost all of the surface contamination, and cut open. The inside of the panel was found to contain high levels of contamination and, as a result, the decision was made that they must be disposed of as radioactive waste. A program to remove and dispose of the majority of the mirror insulation panels in the upper D-rings is being planned for 1986. Engineering activities in 1985 included plans for shielding the elevation 347 floor hatches and elevation 305 floor. Lastly, engineering to reach the Phase III decontamination end points continued.

As of January 1986 $662,000 of the $700,000 authorized by the Japanese Parties for Reactor Building dose reduction/stabilization had been billed.

5.5 AFHB Dose Reduction/Stabilization

Decontamination activities in the Auxiliary and Fuel Handling Building (AFHB) were primarily aimed at improving the general radiological condition of the building. Approximately 22,000 sq ft of floor area in the building were radiologically improved and the 282-ft 6-in. elevation was decontaminated to a level that allows entry without protective clothing. The majority of this work was done using two methods: hands-on wiping and steam cleaning. The hands-on decontamination is normally done with towels and spray-on cleaning fluids. The steam cleaner, a Kelly Corporation Vacu-Mac, is a combined unit that sprays steam on a contaminated surface, thereby loosening contamination, and then immediately vacuums it up. Other new equipment that was procured includes a second high-pressure pump (Elliot) and two ultra-high-pressure pumps (Wakatsuki and Admac).

Another project completed in 1985 was cork seal removal. After the accident, contaminated water in the AFHB and Service Building basements seeped into the cork seal joint between those buildings and the Reactor Building. This cork became highly contaminated and seepage out of the plant was feared. Using specially modified digging tools, the cork was removed and disposed of. The resulting gap was filled in April by a dam composed of asphalt and polyurethane foam.

Developmental activities included the delivery and some testing of three drain cleaning devices: a mole nozzle, a cavijet nozzle, and a flex-hone. Also, a spool piece was removed from makeup pump 1 A and packaged for testing. Engineering for velocity flushing of piping systems and for ex-RCS criticality safety completed the major 1985 activities.

As of January 1986, all $350,000 that was authorized by the Japanese Parties was billed.

5.6 Onsite Waste Management

In 1985, 760 drums and 34 boxes of low specific activity radioactive waste were shipped from TMI-2. The drums had a total activity of 4.181 Ci and a volume of 386.9 m^3 while the boxes had an activity of 372.86 Ci and a volume of 150.9 m^3.

Engineering was completed and procurement began on the waste handling and packaging facility (WHPF). This facility will have equipment for handling, sectioning, decontaminating, and packaging solid radwaste.

As of January 1986, all $600,000 and $350,000 authorized by the Japanese Parties for solid radwaste processing and the WHPF, respectively, had been billed.