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# U.S. DEPARTMENT OF ENERGY THREE MILE ISLAND RESEARCH AND DEVELOPMENT PROGRAM 1985 ANNUAL REPORT

Edited by Gordon R. Brown

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EG&G Idaho, Inc.
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## PROGRAM PURPOSE AND HISTORY

The U.S. Department of Energy (DOE) Three Mile Island Research and Development Program in 1985 continued its research and development work at Three Mile Island Unit 2 (TMI-2). In 1980, plant operator GPU Nuclear Corporation, the Electric Power Research Institute (EPRI), the U.S. Nuclear Regulatory Commission (NRC), and the U.S. Department of Energy (DOE) established the Technical Integration Office at TMI to carry out DOE's research and development objectives. These objectives included obtaining and analyzing data on the March 1979 accident and its aftermath, developing new techniques for responding to the unique challenges at TMI, and transferring these findings and technologies to the commercial nuclear power industry.

The Technical Integration Office is DOE's primary data gathering and distribution arm, with primary interests in accident and postaccident performance of instrumentation and electrical equipment; physical, chemical, and metallurgical behavior of fuel cladding and core components; fission product behavior, transport, and deposition; general physical damage to surfaces, structural components, and equipment in the building; and techniques for decontaminating the surfaces and equipment. Through its evaluations, the DOE program expects to (a) identify possible design changes in equipment standards and regulations; (b) improve the understanding of fission product release pathways and retention mechanisms; (c) provide the industry with new techniques for decontamination and recovery; and (d) have an understanding of the accident scenario and sequence. In conjunction with the program, an Information and Industry Coordination staff was established to communicate program findings directly to the industry through various information networks.

DOE established other programs to respond to new research and development challenges. The Fuel and Waste Handling and Disposition Program is adapting and developing advanced waste processing technologies in answer to the special wastes generated as a result of the accident. The Accident Evaluation Program is acquiring data and determining the appropriate methods for disassembling and defueling the damaged reactor.

Since the accident, DOE, GPU Nuclear, and their contractors have reached a number of milestones in the recovery operation. In 1979, the EPICOR II system began cleaning the contaminated water in the basement of the Auxiliary Building. Television cameras and radiation instruments were used in the first inspection of the Reactor Building. In 1980, 43,000 Ci of radioactive krypton gas were safely vented from the Reactor Building, allowing workers to begin entries on a routine basis. In 1981, the submerged demineralizer system (SDS) began to decontaminate the radioactive water from the basement of the Reactor Building. Shipments of the SDS vessels containing the waste were started in May 1982. Also in 1982, workers lowered cameras into the damaged reactor and conducted the first inspection of the core.

In the summer of 1983, the last solid waste from the processing of original accident-related water was shipped from TMI-2. Later that year, further explorations inside the reactor vessel produced the first samples of the damaged core, as well as a topographical map of the core void and the clearest videotapes of the damaged core to that date.

A number of major steps toward reactor disassembly dominated activities during 1984. In February, the TMI-2 polar crane was load tested for qualification to lift the reactor vessel head. Five months later, the head of the reactor was successfully moved to its storage stand in the Reactor Building, and shielding was installed over the vessel, giving workers safe access to the reactor's internal components to prepare for defueling. Finally in

December, the plenum assembly was inspected, cleaned of hanging debris, and jacked 18.4 cm above its seated position.

In October 1985, actual reactor defueling operations began with core debris being loaded into specially designed fuel canisters. As a prelude to defueling, the plenum assembly was lifted from the reactor vessel after a major engineering effort.

The accomplishments to this point have been significant not only in moving the entire recovery effort closer to completion, but in demonstrating that every new challenge this unique situation presents can be met; and the DOE program has been instrumental in keeping the industry well informed about the progress at TMI-2.

# SUMMARY OF SIGNIFICANT ACCOMPLISHMENTS OF 1985

The year 1985 was significant in the cleanup of TMI-2. Major milestones reached 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.

## 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 ended with the shipment of the last SDS vessels to the Monitored Retrievable Burial Demonstration Program for burial and monitoring in an instrumented concrete overpack.

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.

## Accident Evaluation Program

Analysis of core samples continued in an effort to complete a fission product inventory; however, additional samples of different core regions are required before the final 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 Program

After a major engineering effort, the plenum assembly was lifted from the reactor vessel in May 1985. In October 1985, actual 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 17 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.

# FUEL AND WASTE HANDLING AND DISPOSITION PROGRAM

## Waste Immobilization

The last TMI-2 SDS vessel for the Monitored Retrievable Burial Demonstration Program was safely shipped to Rockwell Hanford Operations at Richland, Washington. With the shipment of this vessel, a total of 19 vessels were accepted for use in the DOE zeolite disposition research and development program. This program is now complete except for long-term monitoring of a buried vessel. This SDS vessel is inside an instrumented concrete burial overpack to monitor actual burial conditions during the test period. Four parameters are being monitored: SDS vessel internal pressure, shell temperature, overpack moisture, and fission product particulates. A summary of the data collected is as follows:

- The two transducers used to monitor the vessel internal pressure started malfunctioning after three months of service. Readings taken in March, May, and September of 1985 indicated that the malfunction seems to be correcting itself. The absolute pressure readings were 10.5 psia in March, 12.2 psia in May, and 14.0 psia in September 1985. If the indicated pressure rise continues at this rate, it will indicate an accumulation of hydrogen gas in the vessel. If the pressure levels off at atmospheric pressure, it will indicate that air has been slowly leaking into the vessel.

- The fission product heating from inside the SDS vessel has increased the temperature of the overpack and surrounding soil and seems to be leveling off. The temperatures of the outside of the overpack (at 13.6 ft) and the outside of the lower vessel (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. The temperature outside the overpack is 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.

- 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 indicates that moisture is not accumulating, and is not expected to, due to the fission product heating the inside of the concrete overpack.

- 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 less than 100 counts/min, which is essentially normal background.

## Core Transportation

Major accomplishments were made in preparation for shipping the TMI-2 damaged core from TMI to the Idaho National Engineering Laboratory (INEL) where it will be stored and used for the Core Examination Research and Development Program. Nuclear Packaging, Inc. (NuPac) completed the first NuPac 125B shipping cask with skid and railcar 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 in early 1986. Equipment for dry loading canisters into the cask, also designed and fabricated by NuPac, will be completed in early 1986. That 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.

Fabrication of the cask 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. The SAR includes a description of the package (cask and contents), and structural, thermal, containment, shielding, and criticality evaluations for normal and hypothetical accident conditions. It also provides operating procedures, acceptance tests, and maintenance programs, and finally a quality assurance plan. Before starting fabrication and submittal

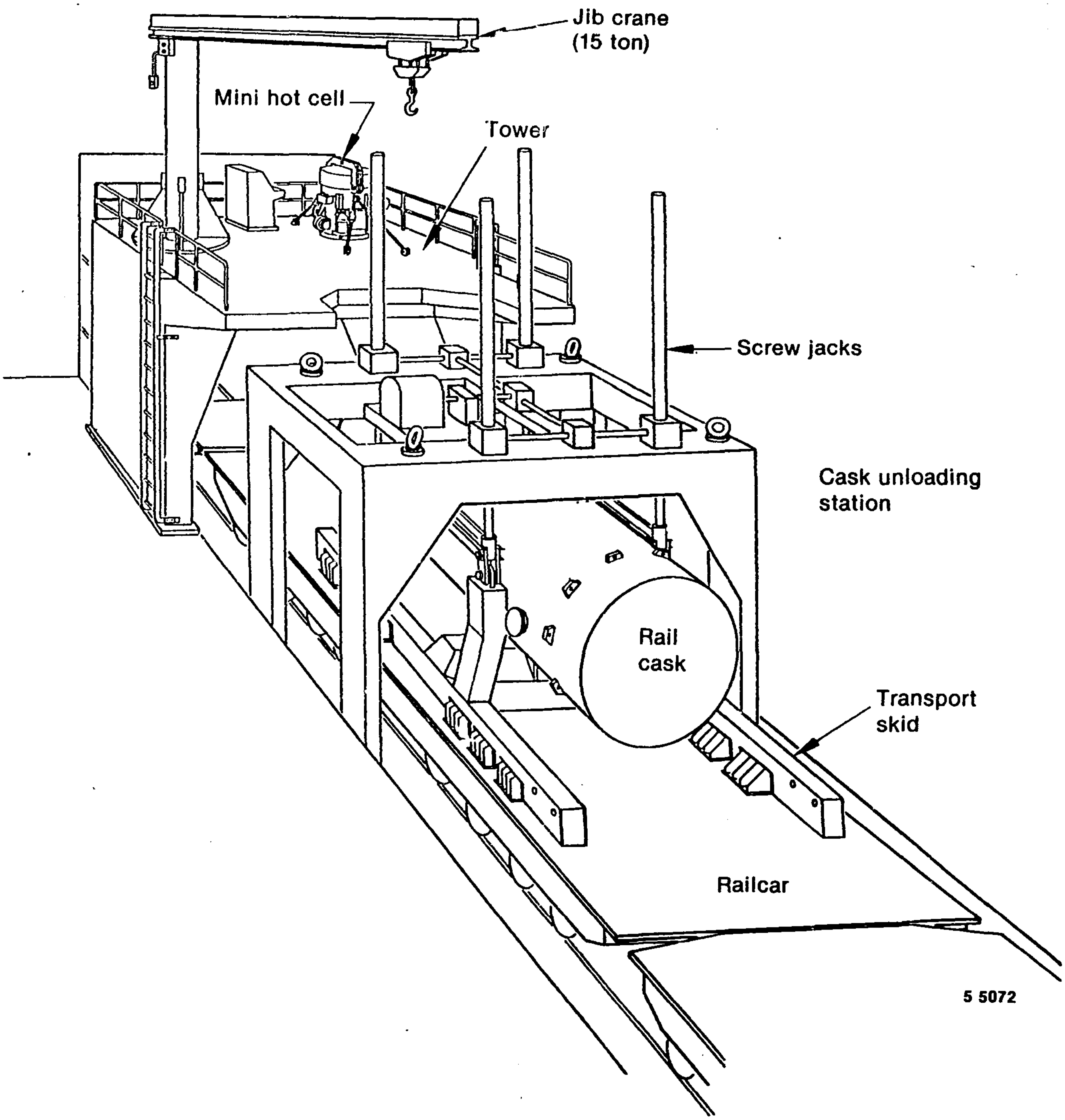


Figure 1. Rail cask on the railroad car.

of the SAR for 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. The test also qualified 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. The side drop (Figure 5) imparted maximum loads to the inner vessel. The side puncture (Figure 6) verified the

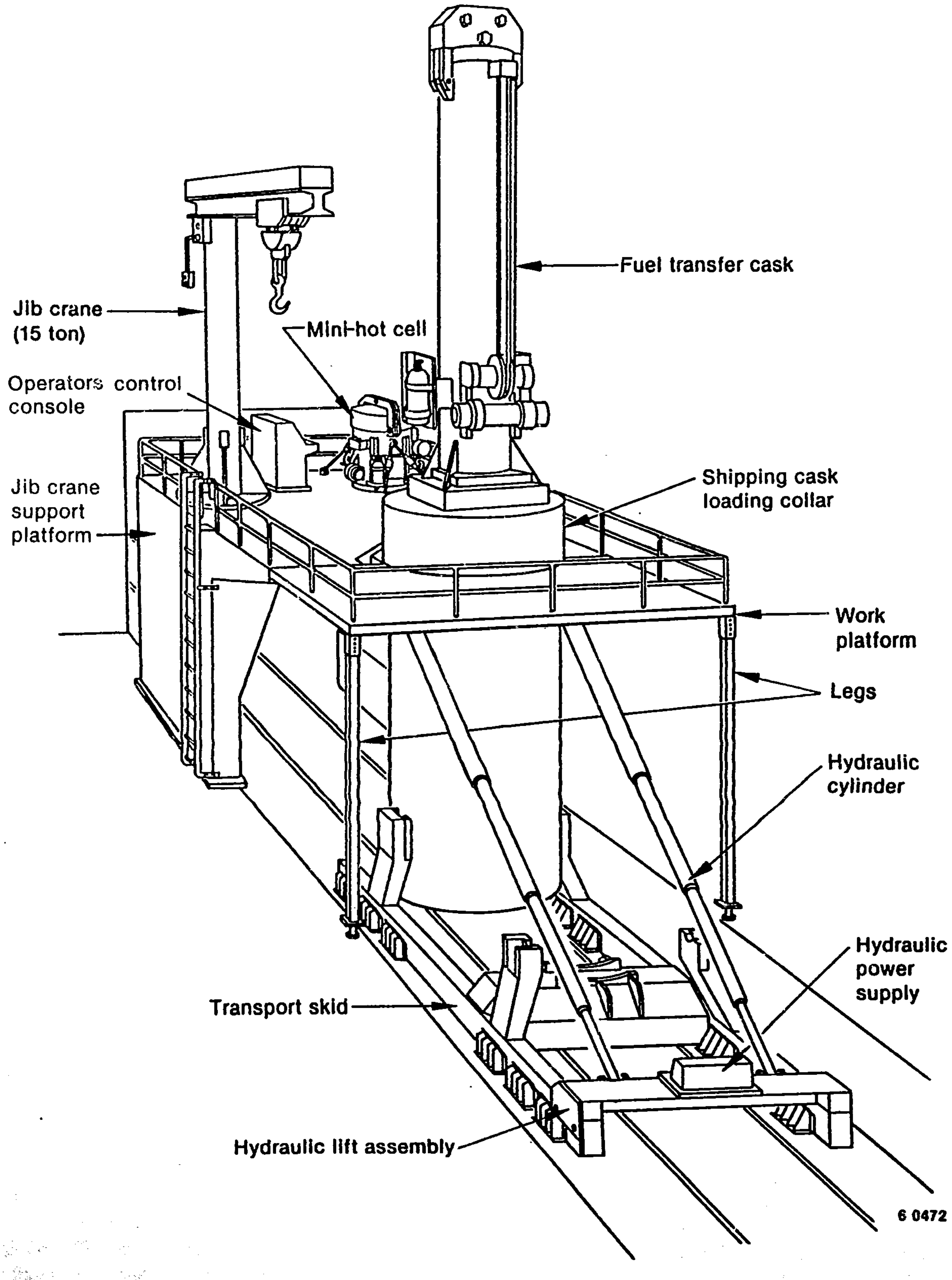


Figure 2. TMI fuel cask loading components.

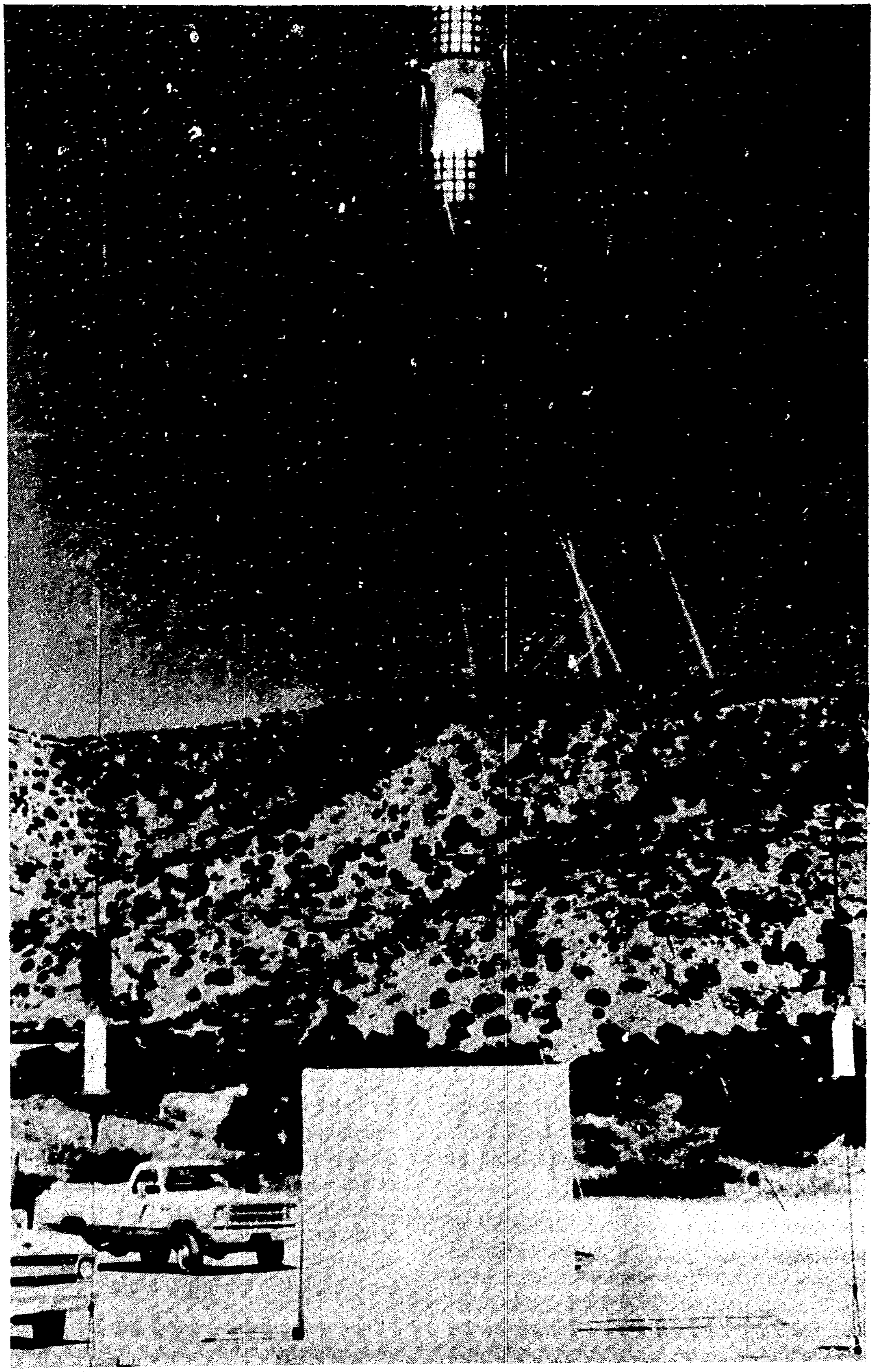

Figure 3. Bottom end drop height and orientation check (85-346-3-7).

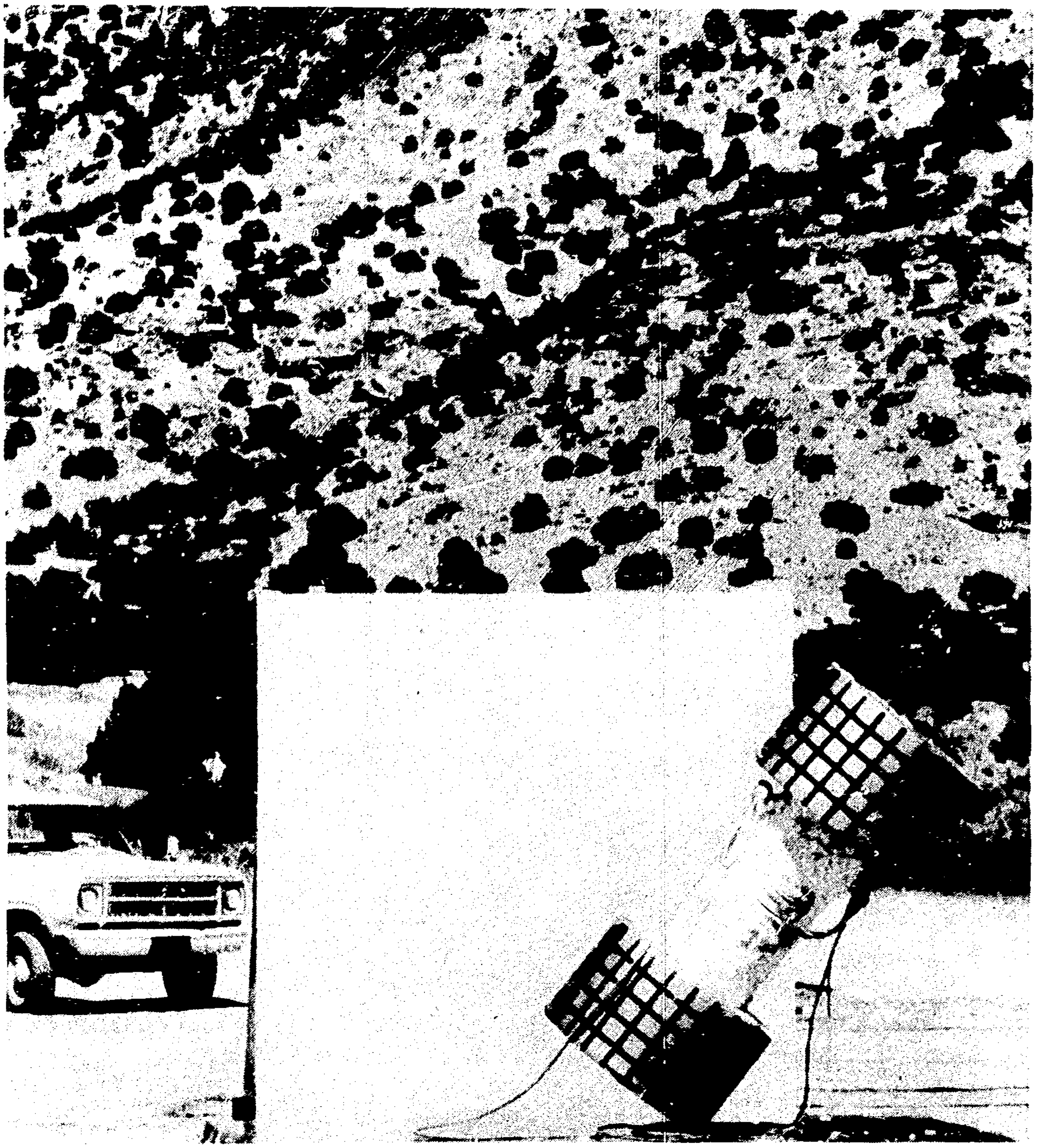

Figure 4. Oblique drop at the instant before impact (85-346-3-12).

integrity of the cask side wall. And the end puncture drop (Figure 7) verified the integrity of the cask lid. The test requirements for the tests 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. This damage was expected and did not cause loss of 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 instrumentation and the structural analysis in the SAR

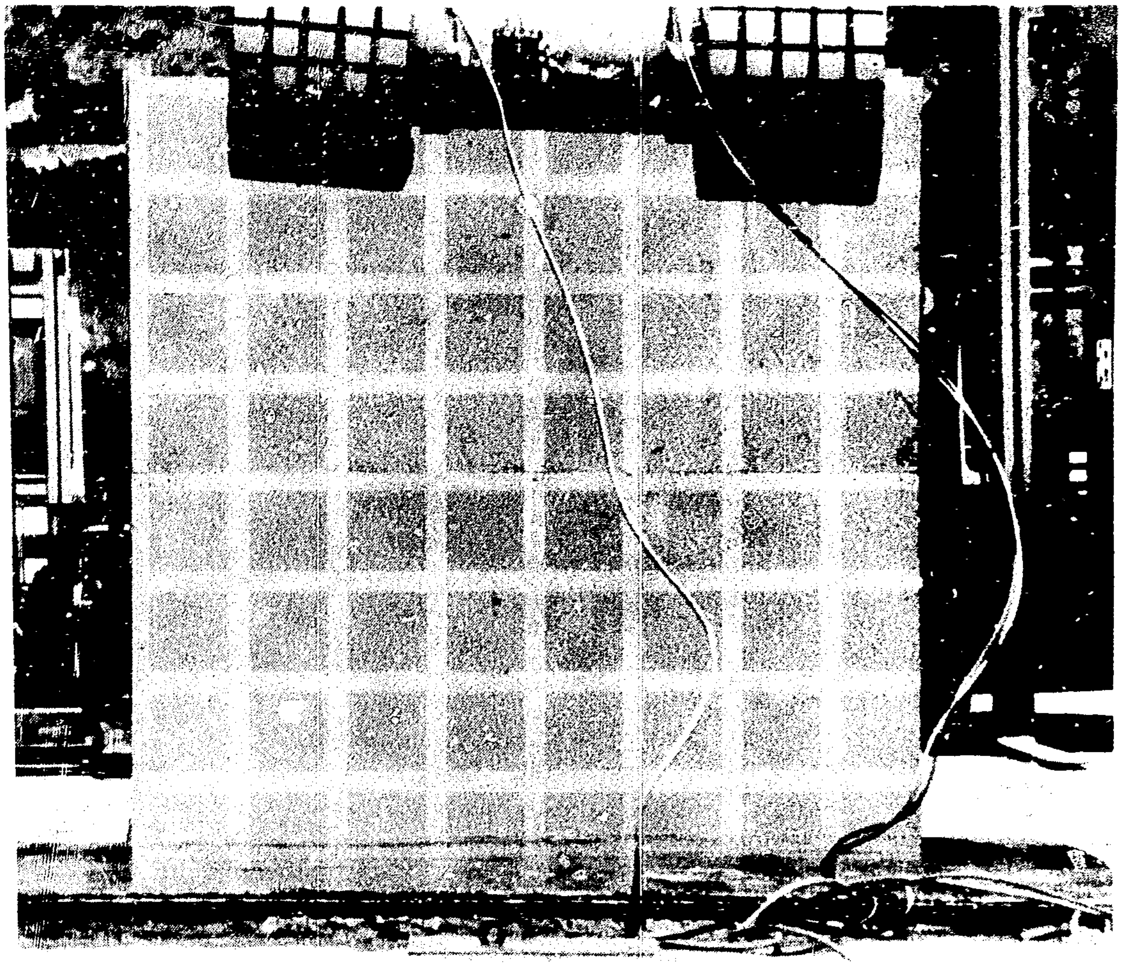

Figure 5. Side drop at the instant before impact (85-346-8-5).

demonstrates 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.

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 performed the drop test at the Tower Shielding Facility, (seen on 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 deacceleration 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. Figure 11 shows the cask simulation vessel with the impact limiters for the horizontal drops. To simulate the fuel debris, the canister was loaded with a total of 1800 pounds of lead shot that was covered with water.

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

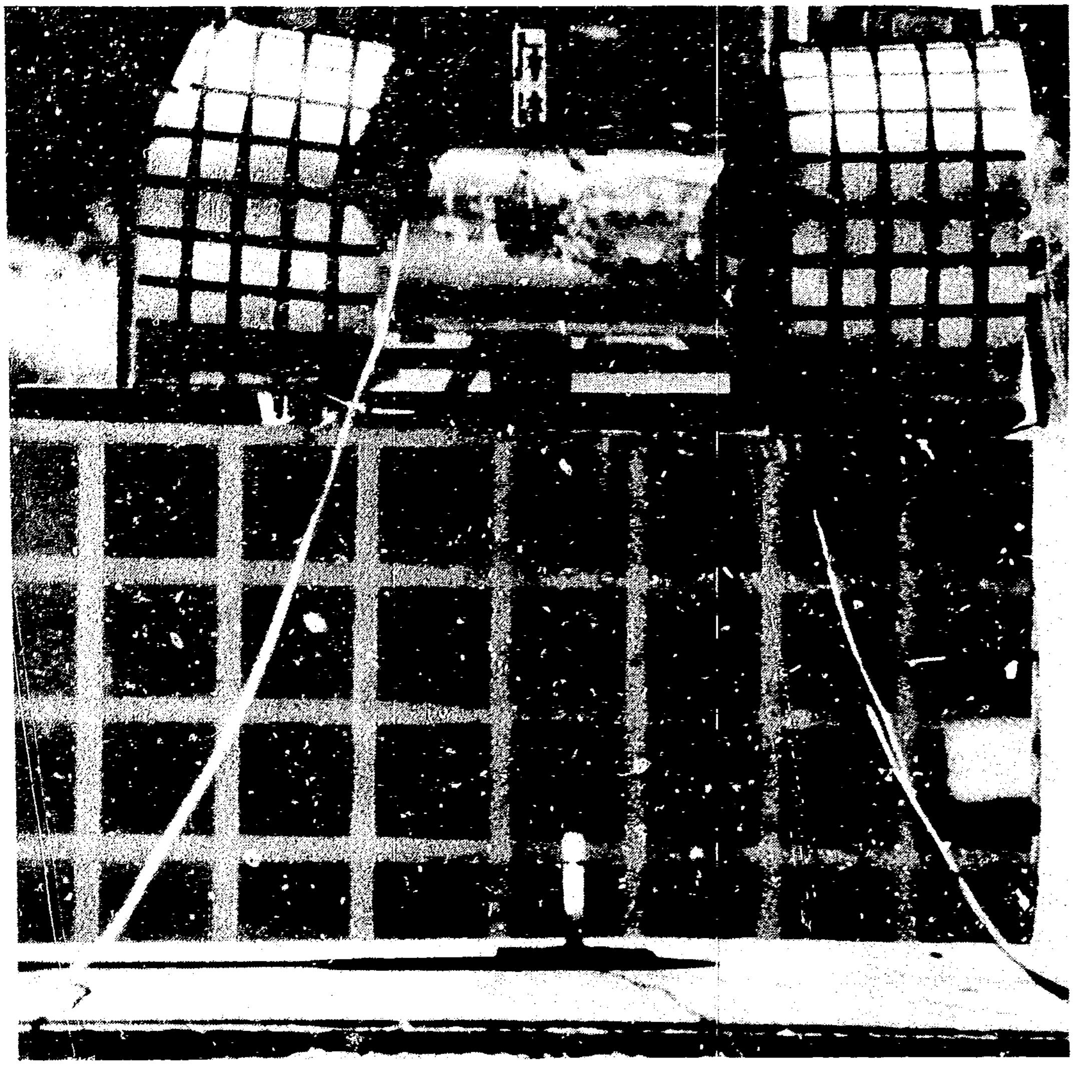

Figure 6. Puncture drop height and orientation check (85-346-7-4).

was dropped in four different configurations, as shown on 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 off-center 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 on 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

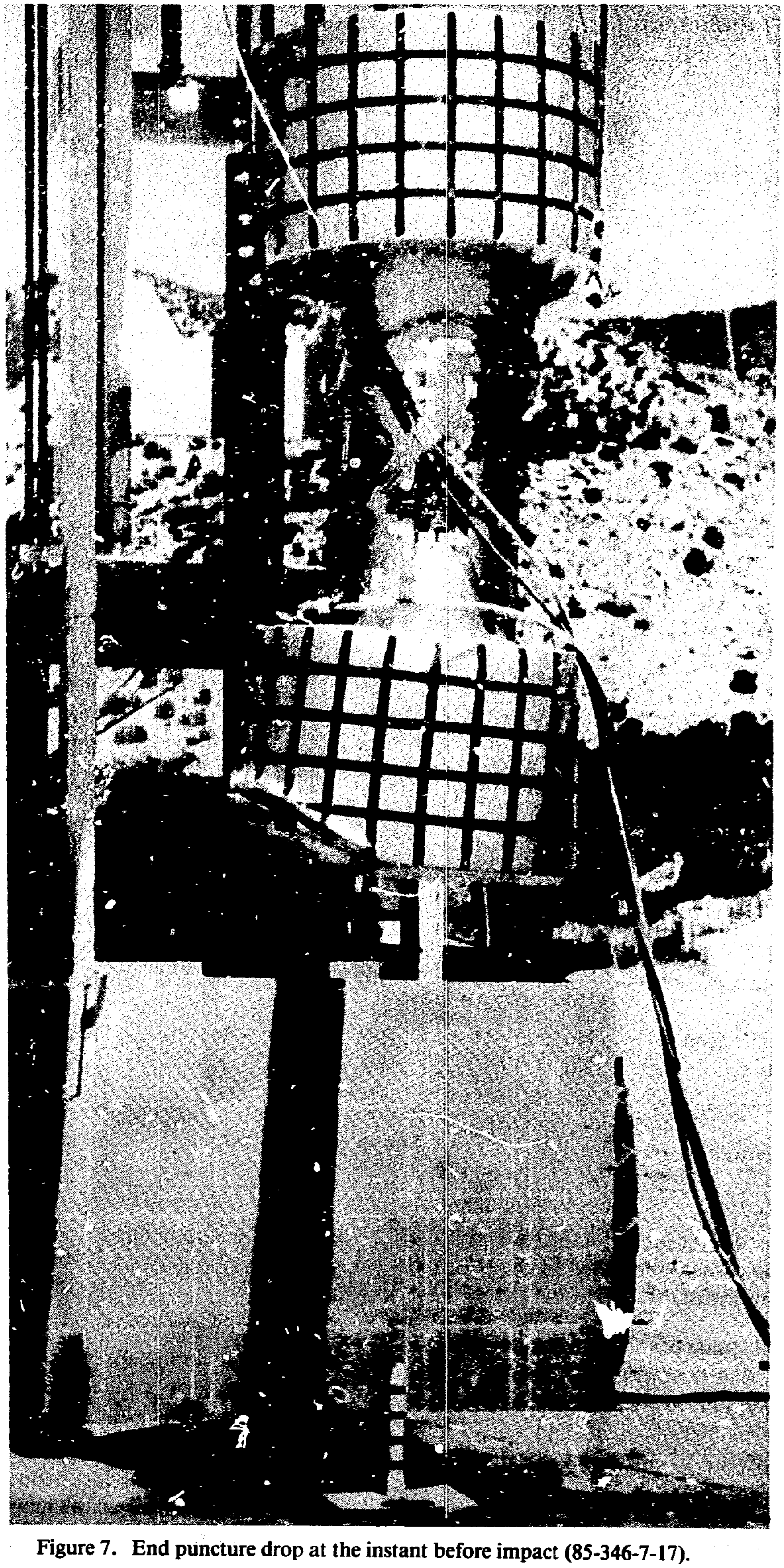

Figure 7. End puncture drop at the instant before impact (85-346-7-17).

**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.

performed better than required for the cask criticality control, and the assumptions used in the criticality analysis were confirmed.

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 is planned. To ensure that flammable gas mixtures will not exist in these canisters due to hydrogen and oxygen generation from radiolytic gas generation, Rockwell Hanford Operations was asked to determine which gas recombiner catalysts and bed configuration 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.

**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).

Rockwell performed a series of catalytic recombiner tests. The tests were conducted using small (16-L 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.

**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 | -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 |

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 h 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.

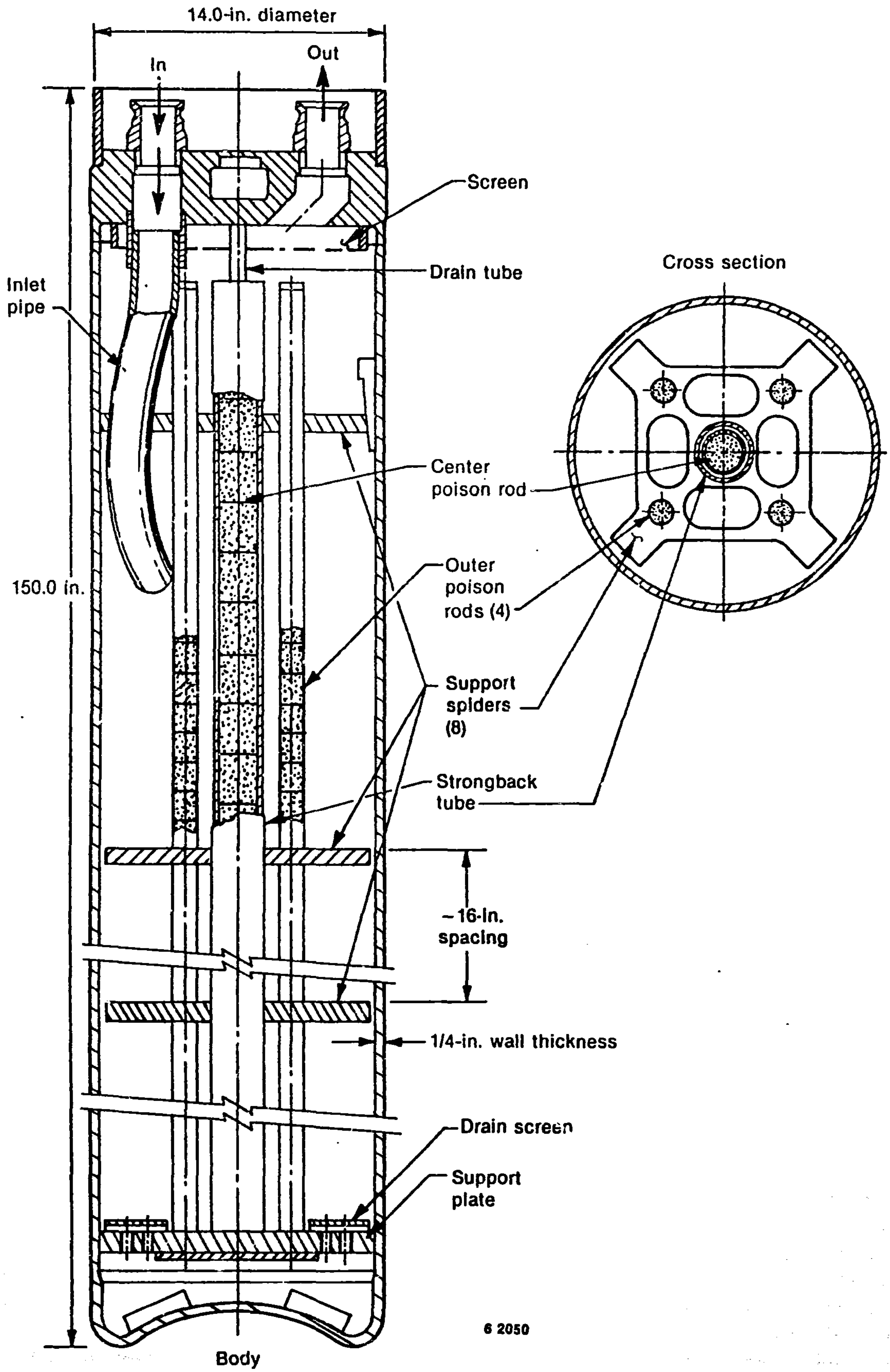


Figure 8. Knockout canister.

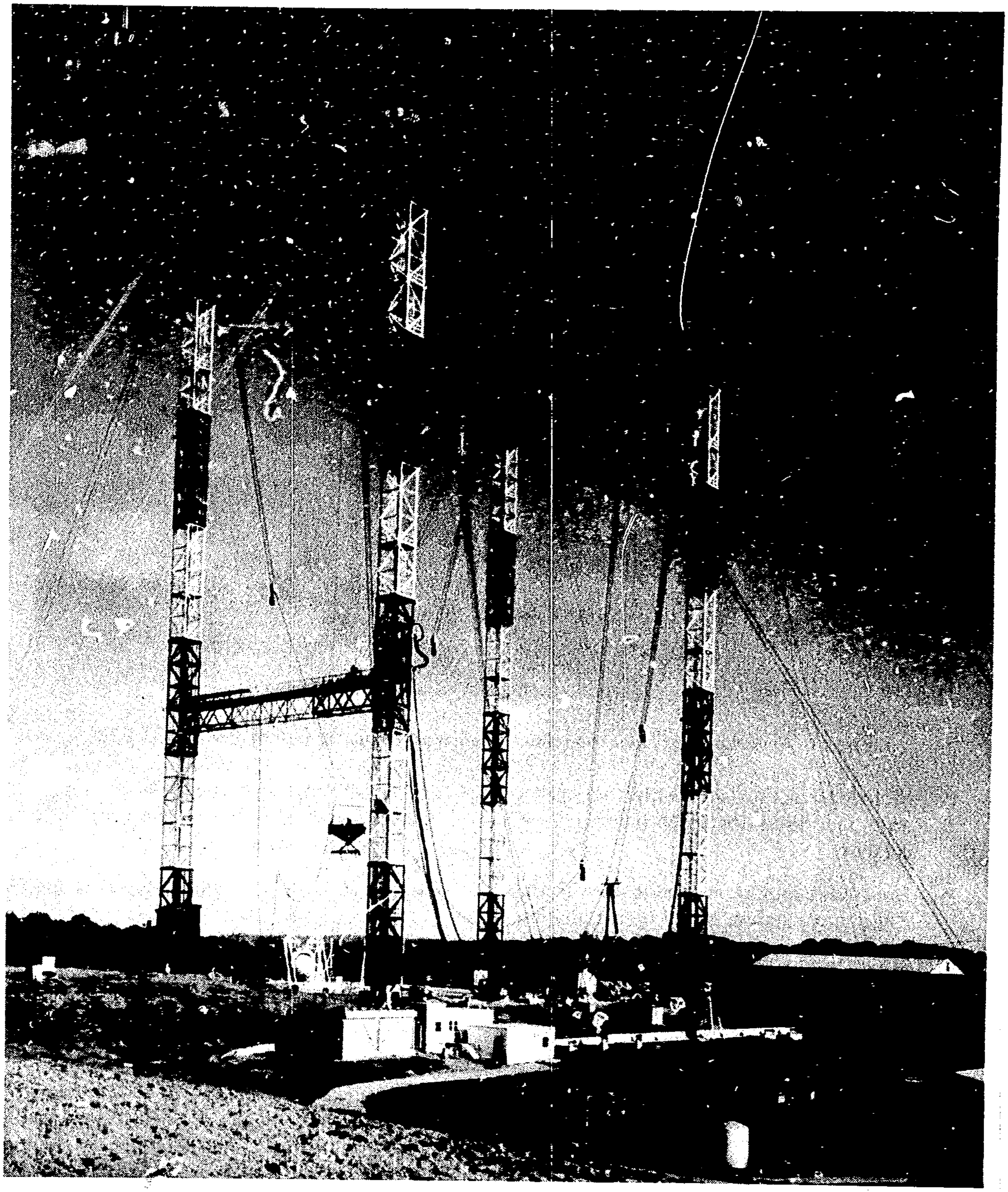

Figure 9. Drop test facility at Oak Ridge National Laboratory.

- 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

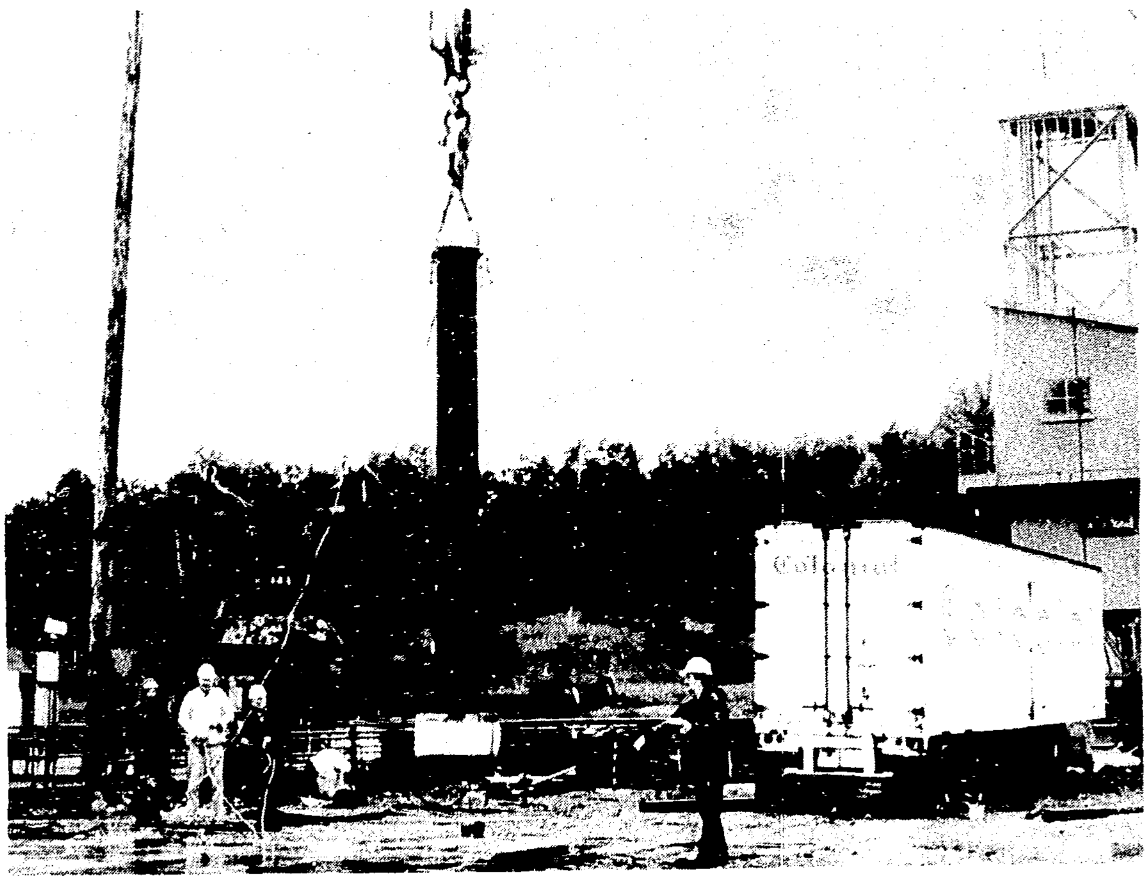

Figure 10. Cask simulation vessel and simulation impact limiter for vertical drops (86-45-2-3).

occluded reaction sites within the pellet and decreased its effectiveness as a catalyst.

- 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.

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

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

- 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 had begun (which better represents actual conditions for transport), the catalysts performed well.

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

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 onto the TMI Unit 1 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

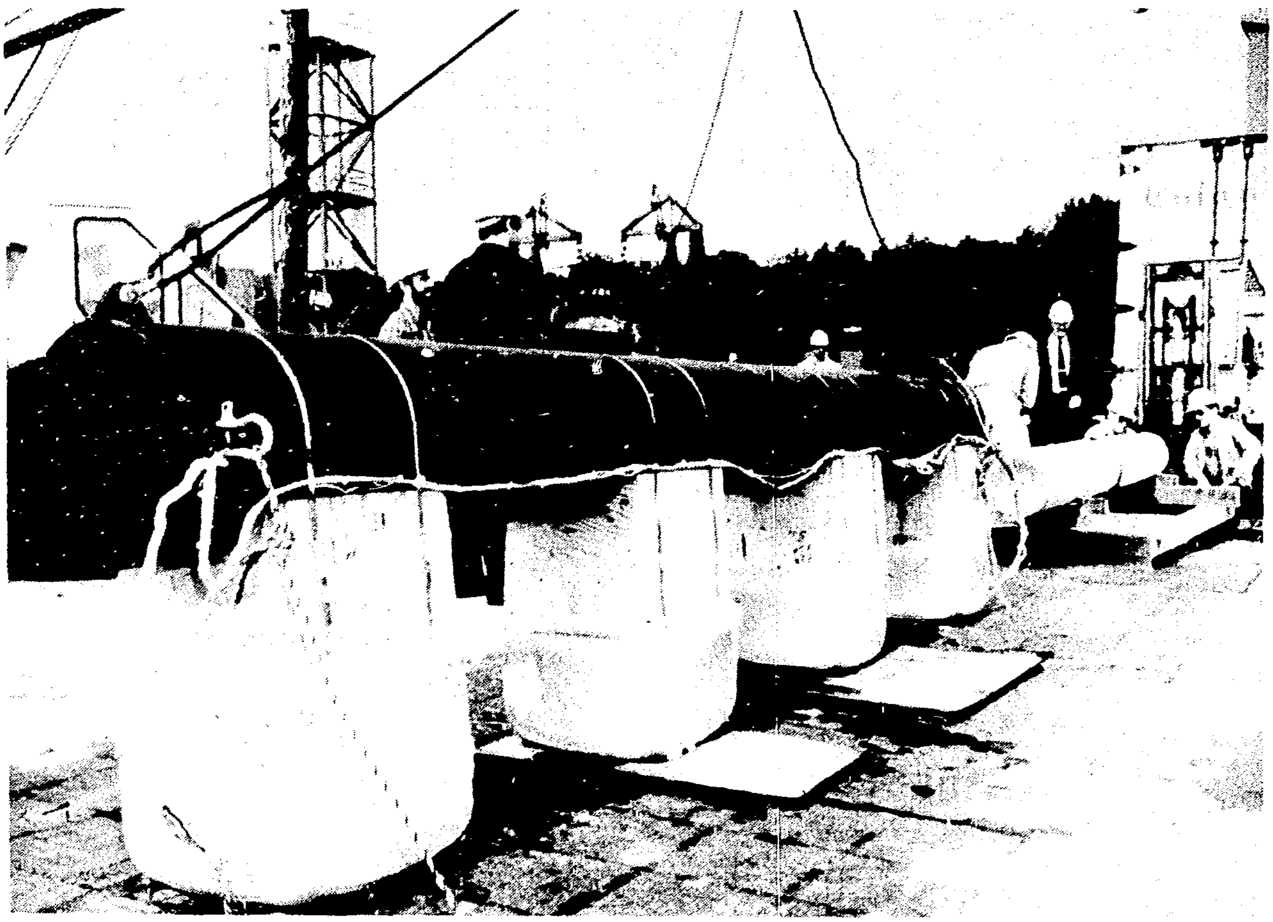

Figure 11. Cask simulation vessel with simulation impact limiters for horizontal drops (86-45-3-10).

position and secured to the jib crane support tower, and the work platform is 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 transfers a canister from the storage pool to the 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. 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 to GPU Nuclear personnel.

Using regular train services, the railcar with cask will be transported across country and arrive at the Central Facilities Area of the INEL.

On arrival at the INEL, the cask will be unloaded onto a specially designed tractor/trailer. The time required to transport the loaded cask 2500 miles across the country is expected to take about ten days. 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

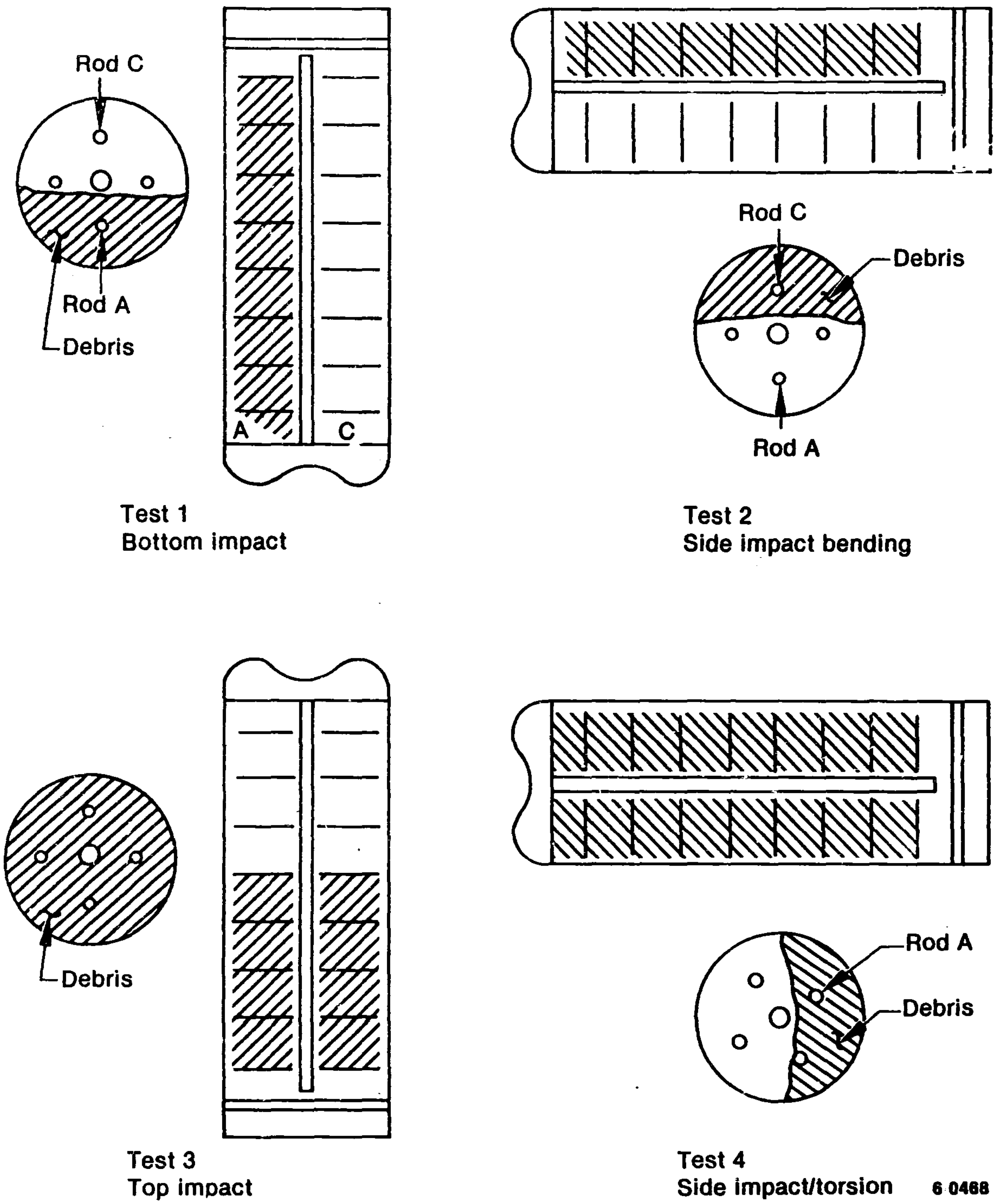


Figure 12. Drop test configurations.

up to 30 years. The canisters will be remotely withdrawn from the cask, conveyed 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. This roundtrip will take about a month.

**Table 4. Knockout canister test parameters and test results**

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

a. Reference Cask "g" loads from design basis for canisters, which were confirmed by mesurements made during the 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.

## Examination Requirements and Systems Evaluation

**Fission Product Inventory.** During FY-85, the results of fission product analyses on samples collected from the TMI-2 plant in FY-84 were collated. Among the samples analyzed 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 resting 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 obtained 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 about 5 to 6% of the core inventory of cesium and iodine, about 12% of the core inventory of strontium and about 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 appear to have been transported to this location because of their high solubility in water.

In addition to investigations of the unknown portions of the reactor core, only a few additional 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, 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.

**Table 5. Fission product inventory fractions—1985[a] (decayed to December 31, 1979)**

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

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

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

c. From analyses carried out this year.

**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.

## Accident Scenario

The accident scenario developed for the initial 4 h of the accident is based on the known 25% of the 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 due to cladding oxidation, and temperatures quickly exceeded cladding melting. 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. It is believed that during the time period between 150 and 174 min, a relatively solid region of core materials composed of previously molten and intact fuel rods formed, as shown at the top of Figure 13. The top of the core probably consisted of highly oxidized and embrittled fuel rod remnants. It is judged that high-temperature molten material had not yet penetrated below about 0.75 m, because the minimum water level was probably at about that elevation. This was estimated 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) indicate no anomalous behavior during this time period.

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 predicted flow blockage resulting from the relocated and (partially) frozen previously molten 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. It is postulated that these fuel rod fragments collapsed onto the projected 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 lower head visual examinations and the in-core 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 first estimated after visual examinations of the lower head and then indicated by a review of the SPNDs that showed an anomalous output from the Levels 1 and 2 SPNDs and by 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 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 is estimated to have 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.

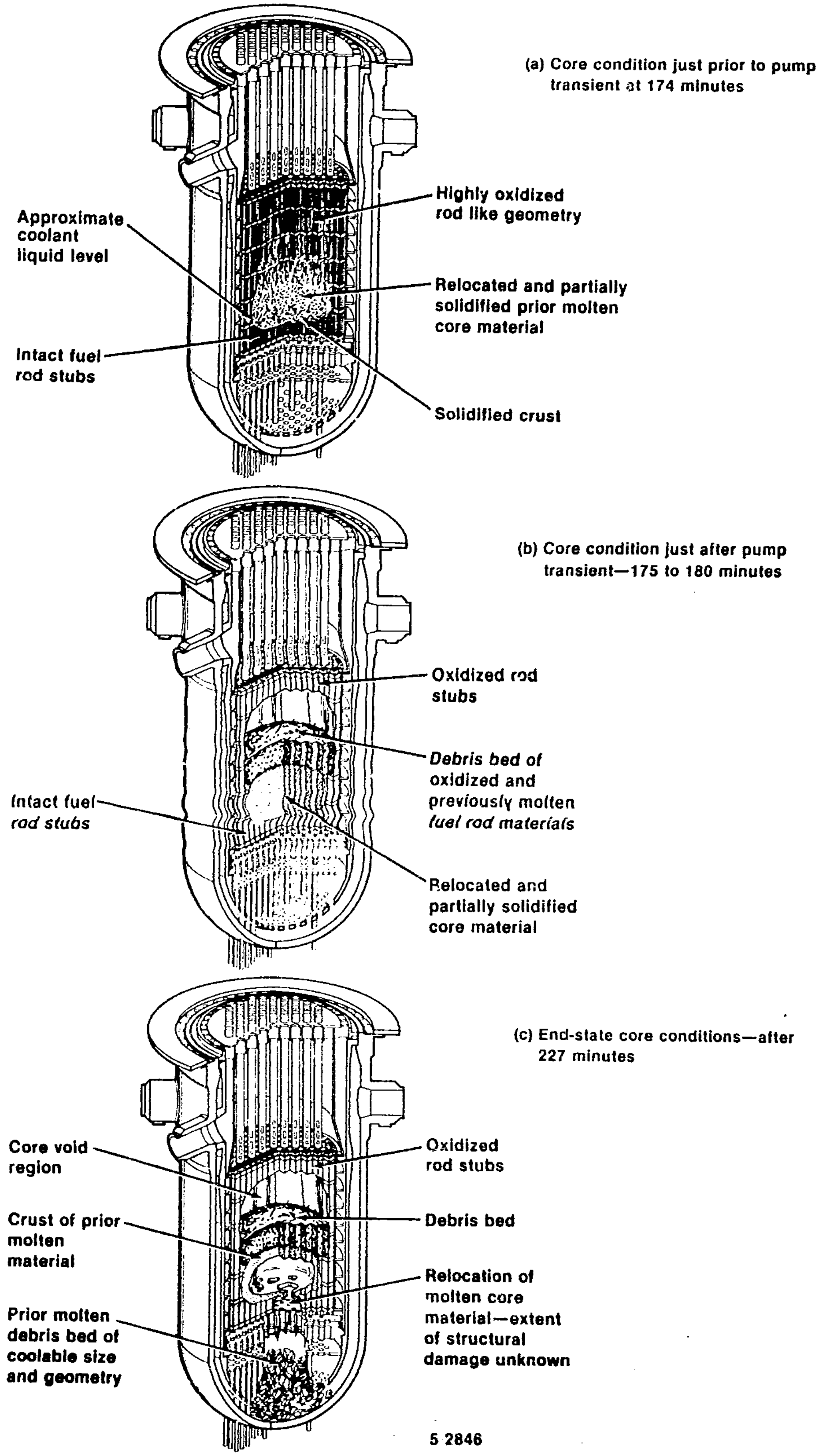


Figure 13. Hypothesized stages of the TMI-2 accident progression.

## Analytical and Experimental Support

**Data Evaluation and Data Base Development.** Data Evaluation and Data Base Development focused on the evaluation of on-line measurements that pertain to the standard problem, and the development of an integrated data base that eventually will contain the information required to run a standard problem.

Most of the on-line data has been digitized and placed on 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 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 the SPND starts to go negative, the temperature is 1000 to 1300°F. The second response occurs later and is when the SPND changes rapidly from negative to off-scale positive, then the temperature is 2000°F, 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. Also, the amount of fuel material in the bottom of the reactor vessel possibly can be estimated.

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 will be completed in 1986.

## Sample Acquisition and Examination

**Ex-Vessel Characterization.** In 1985, GPU Nuclear continued the gamma spectra and radiological measurement program in an effort to locate and characterize the deposition sites for the ex-vessel fuel debris. Using gross gamma directional surveys, thermoluminescent dosimeter mapping, and gamma spectroscopy measurements, various primary system components were studied.

In support of GPU Nuclear's characterization efforts, EG&G Idaho has been providing technical support and hardware to take in situ gamma measurements of primary system components. This gamma spectral data is being acquired using a portable sodium iodide system and the EG&G Idaho mobile gamma spectrometer system described in the 1984 Annual Report.

**Primary System Piping and Components.** Using the 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 6 contains results reported by GPU Nuclear for 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 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 of 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, worst-case 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 the INEL for surface deposition studies.

**Reactor Building Concrete and Sediment Samples.** Basement sediment and concrete samples were obtained in 1985 to provide information leading to the development of decontamination methods. The robotic vehicle (Rover) collected sediment samples in several Reactor Building basement locations. Using the EG&G Idaho mobile gamma spectrometer, onsite analysis of the samples indicates that the samples contain gross gamma and Cs-137 activity on the

**Table 6. 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 ± 0.021 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 |

order of 10 to 90 $\mu$Ci/g. Rover also removed two concrete cores from the D-ring and impingement area walls in the Reactor Building basement. The first sample was removed from the D-ring wall approximately 2 ft 6 in. above the basement floor. The second sample was removed from the impingement wall at an elevation 8 ft 4 in. above the floor. After onsite characterization, the samples will be sent to an offsite laboratory for extensive evaluation.

**Reactor Vessel Internals Characterization and Sampling.** In order to document the condition of in-vessel components and identify possible samples, GPU Nuclear conducted various video inspections during 1985. Using the video enhancement system described in the 1984 Annual Report, EG&G Idaho engineers documented the first lower head video inspection that took place shortly after plenum jacking. Eight samples of debris in the lower vessel head were obtained during the inspection. Selected analysis results of the debris are given in Table 7. The video system was used again during plenum removal, upper core void, and lower head inspections. Data collected were sent to the INEL for review and analysis. During the December lower head inspection, another small rock sample was obtained from below the flow distributor plate using a long-handled manipulator grab tool. This sample will undergo preliminary radiological analysis before shipment to an offsite laboratory.

**Core Sample Acquisition and Examination Project.** Following the evaluation of commercially available core drilling equipment used in the petroleum industry, work was started in 1984 to design an extensively modified system for use over 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 shipped to TMI-2 late in 1985. Use of the system to acquire samples is anticipated during 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 sampling 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 end fitting. The sampling activity will also incorporate the acquisition of sample material from the space between the normal lower end fitting elevation and the upper surface of the elliptical diffuser plate. Immediately after 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

## Table 7. Lower head phase 1 samples selected 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<br>42[c] | 7.5<br>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).

installed over the reactor vessel. The figure depicts drill piping completely inserted into the core before sample withdrawal.

Following the delivery of the core samples to the INEL in 1986, an extensive examination task will commence. 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.

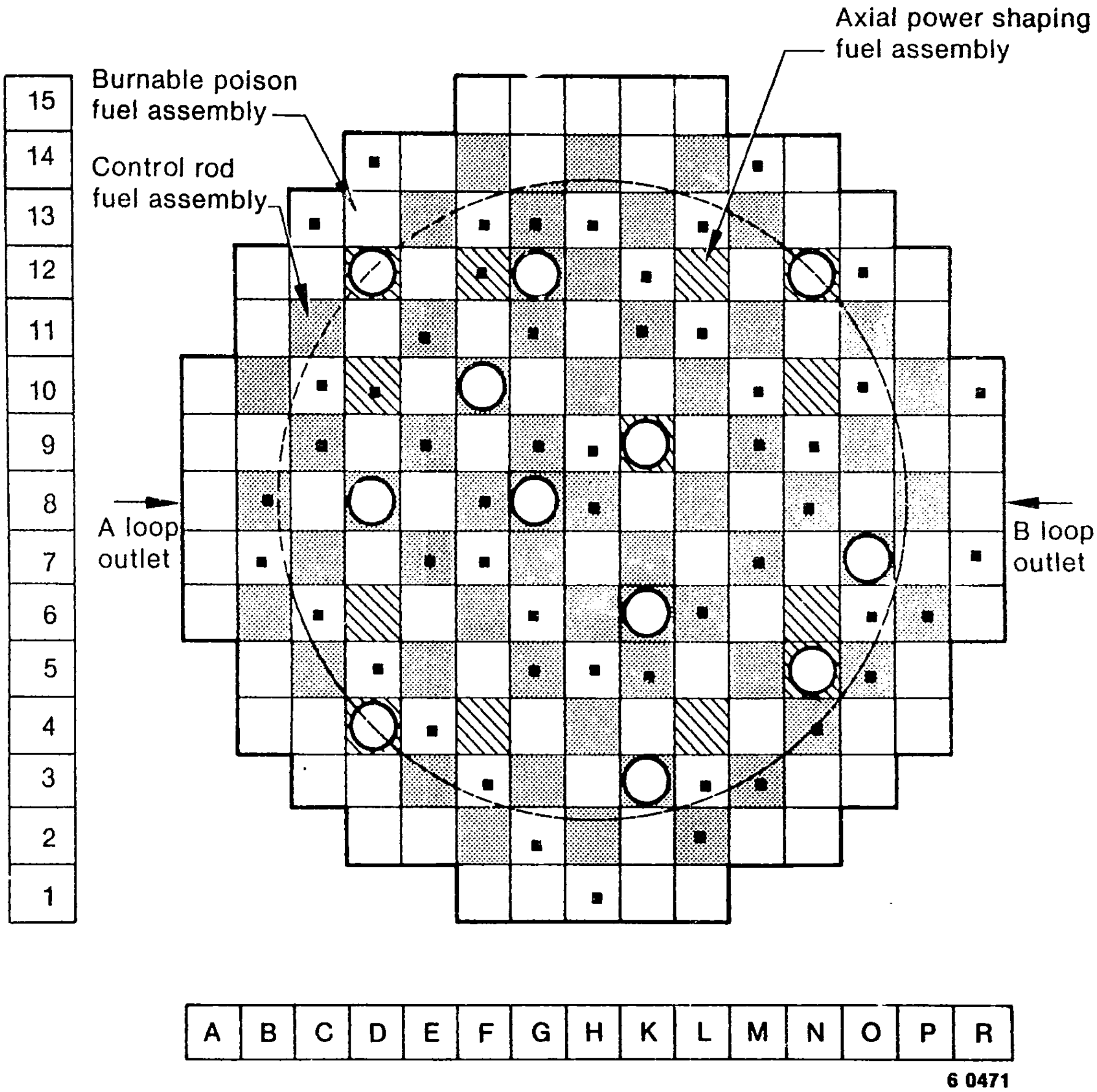


Figure 14. Core bore locations.

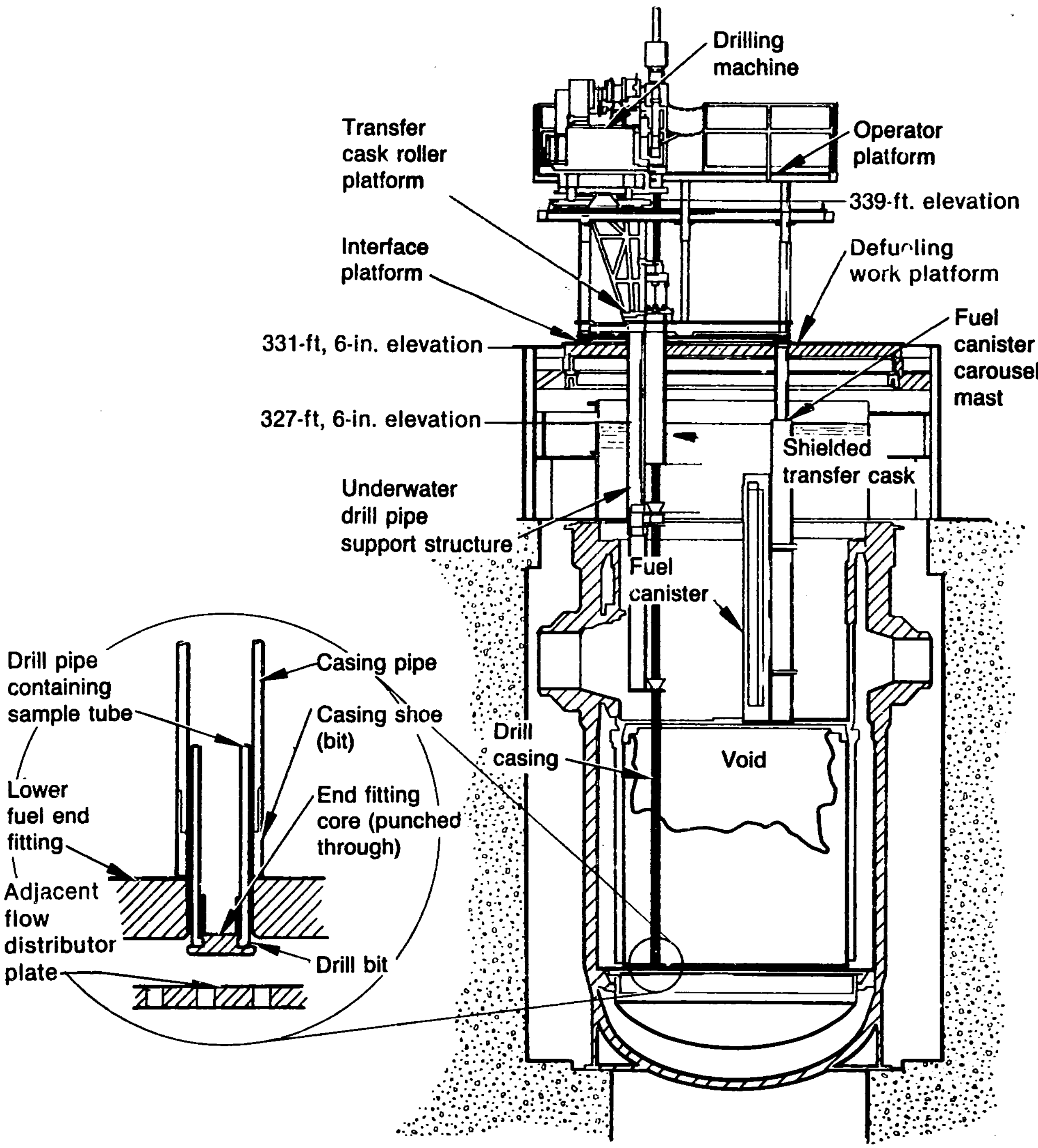


Figure 15. The core sample drill is illustrated here as completely inserted before sample withdrawal.

## 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 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's control rod guide tubes.

The plenum was actually removed from the water-filled reactor vessel five months after the initial jacking of the plenum. During this period, workers concurrently made preparations for the plenum's removal and installed defueling equipment in the Reactor Building's refueling canal. This defueling equipment installation had to precede the plenum's removal since the canal was partially flooded shortly before the plenum was removed.

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 conducted 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.

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 and stored underwater on a stand in the dammed, deep end of the Reactor Building refueling canal. The canal water shields 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. Plans are being formulated to include the eventual disposition of the plenum.

The actual plenum removal operation took less than three hours. The operation, in addition to a post-plenum-removal inspection, was videotaped to assist in future defueling and data acquisition efforts. The plenum's removal gave cleanup workers the first direct human-eye view of the reactor vessel's core region since before the time of the accident. Previously, cleanup planners relied on video camera probes to look inside the reactor.

Actual radiation and worker exposure levels during the operation were less than those conservatively estimated by engineers before the start of operations. The five-member plenum removal crew's whole body doses were held to a minimum as 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.

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.

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

## 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 late in the year. Reactor vessel defueling is expected to take from 18 to 24 months. More specifically, defueling operations are defined as those activities involved with placing reactor vessel core debris

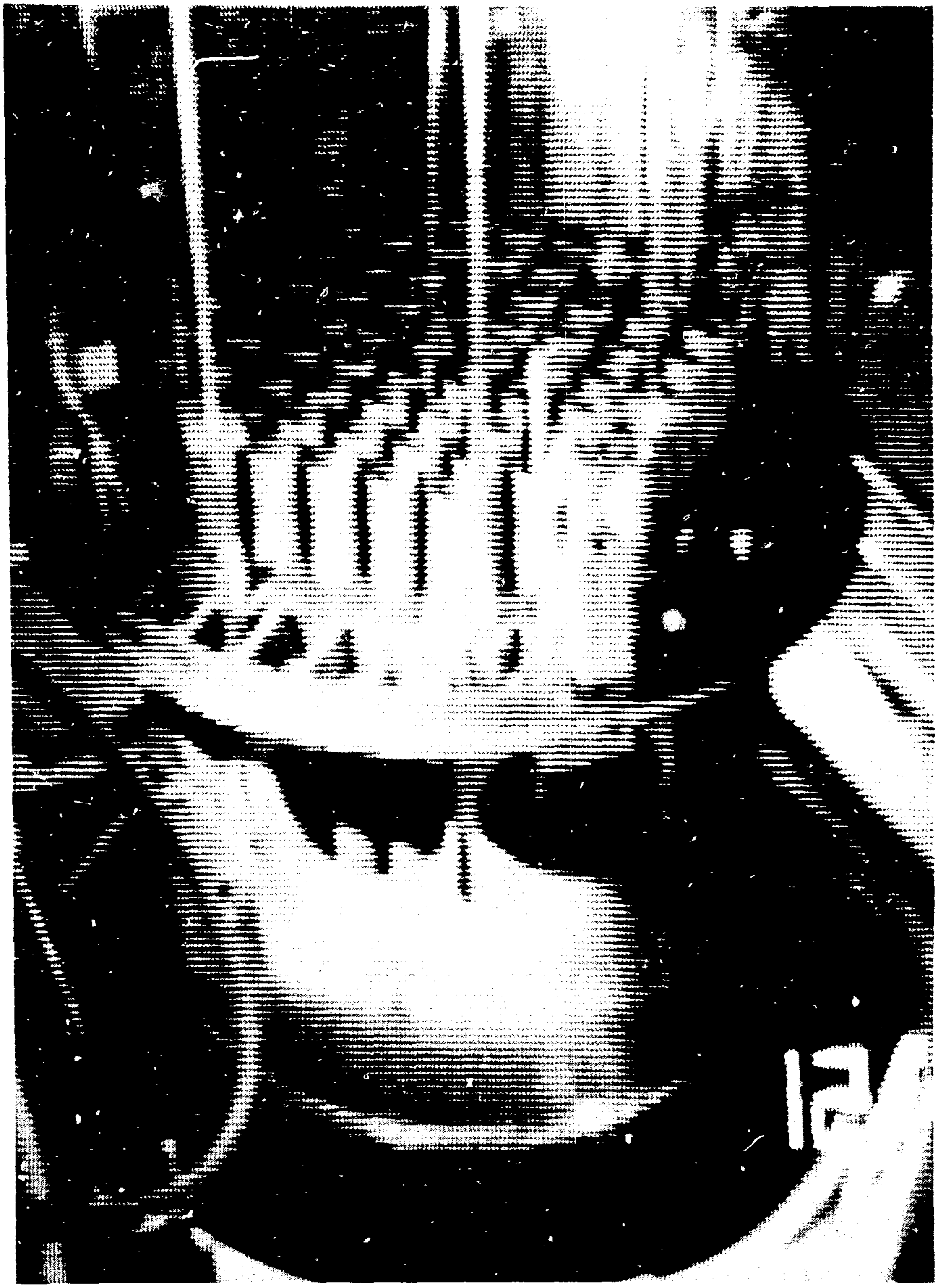

Figure 16. Removal of the plenum.

into defueling canisters and the transfer of those canisters to temporary storage in the Fuel Handling Building.

The defueling tooling system (see Figure 18), designed to operate remotely underwater in both manual and power-assisted modes, continues to follow the basic technical approach that was

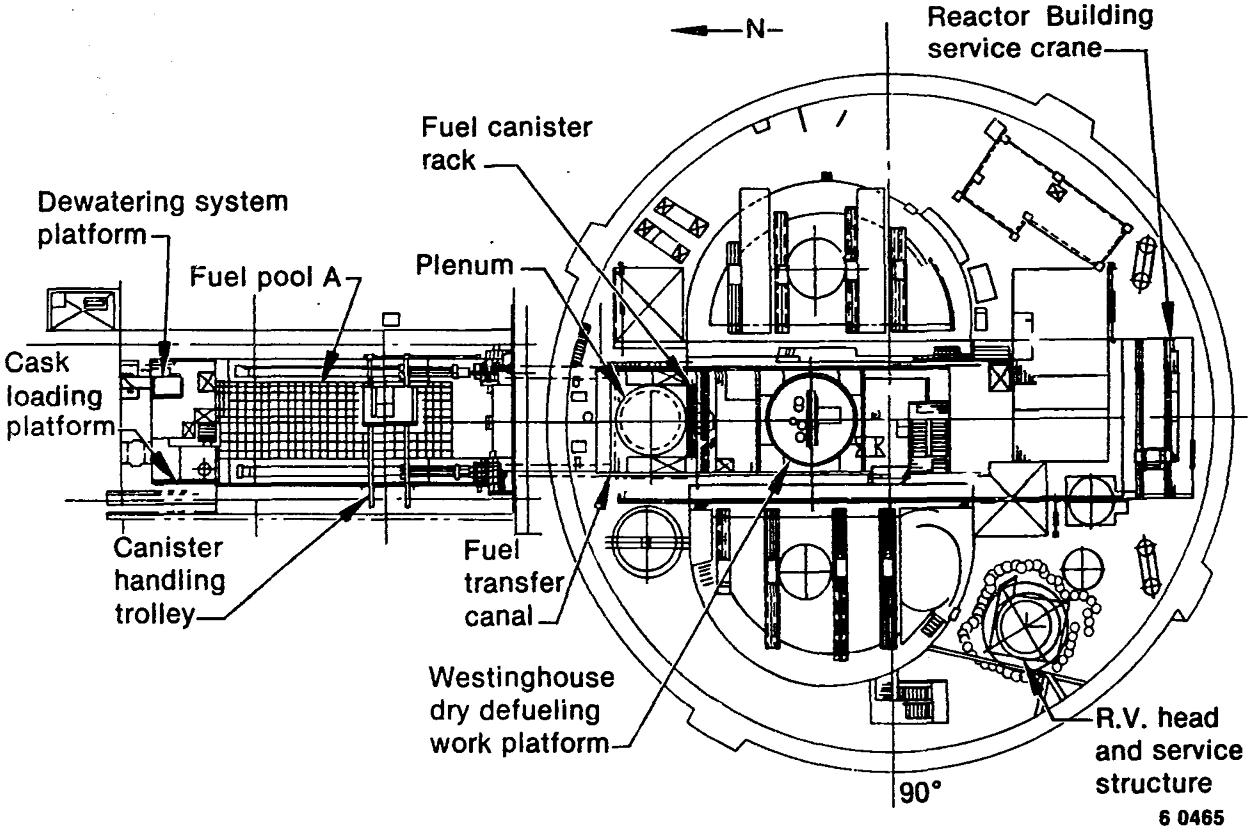


Figure 17. TMI-2 defueling general arrangement.

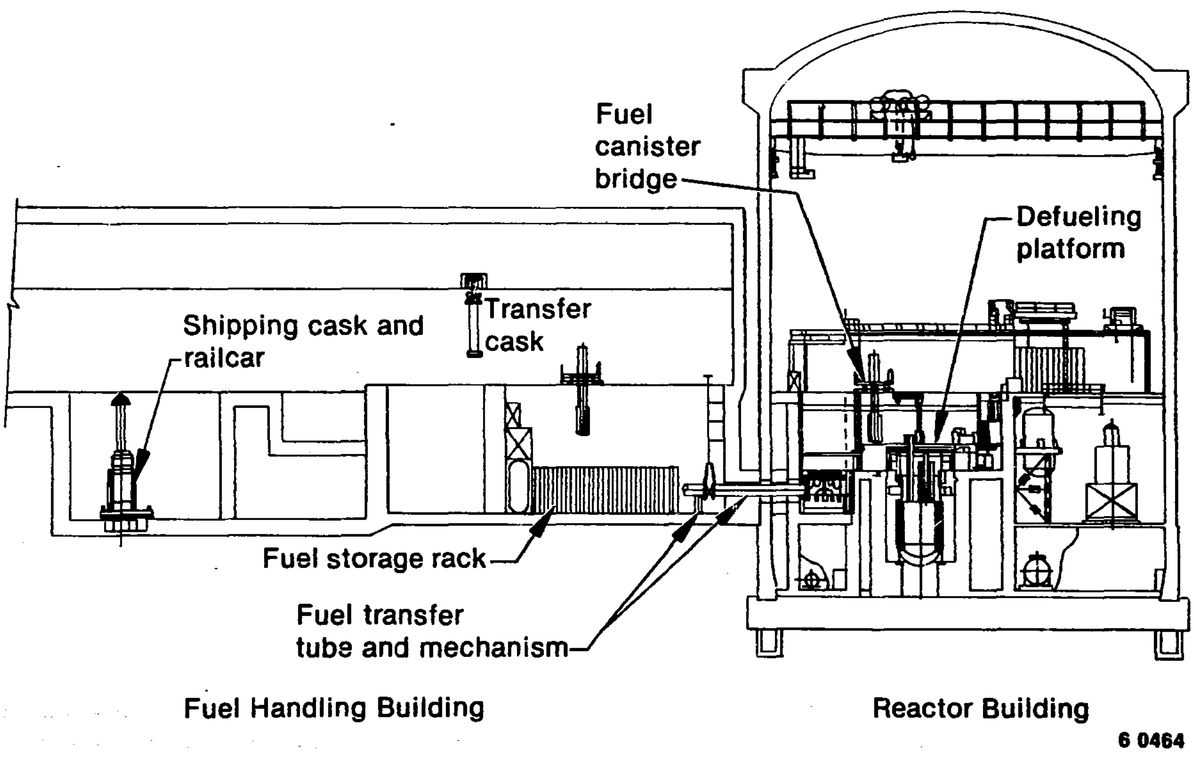


Figure 18. TMI-2 defueling plan.

established as a result of previous years' intensive planning. This approach initially calls for the simplest, least developmental methods. 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, tappers and camera, lighting, and debris container handling tools.

The central feature of the innovative defueling tooling system is the shielded work platform. It is on this platform, that defueling operators will work nine feet over the reactor vessel flange. These operators will insert tools through removable hatches arranged in a T-slot configuration. To provide radial access to the reactor vessel, this platform has a rotational capability. The platform also provides support for major defueling tooling subsystems including the independently rotating canister positioning system carousel, and the fines/debris vacuum system. The carousel will hold 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 postulated 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, two lower head video inspections, a hydraulic displacement test, and a variety of core debris samples. Core data further suggest that once-molten masses, comprised of oxidized and unoxidized ceramic/metallic properties, might 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 is being developed based on an integrated assessment of tooling requirements to defuel the hard crust transition zone and the lower core region to include the core support assembly and lower reactor vessel head. This tooling augments the initial pick and place defueling tooling system. 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 lower reactor vessel and core support defueling tooling, which will complement both the initial and bulk tooling, will include 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, which is presently at TMI, the balance of the required tooling is currently undergoing final design or is in fabrication. These tools are proposed and not all of them are final, especially the lower-head tools.

A core boring machine, which is presently at TMI for obtaining debris core stratification samples for research purposes (see the section Core Sample Acquisition and Examination Project earlier in this report), is being considered by TMI-2 defueling planners as a contingency defueling production tool. This device is patterned after a conventional petroleum industry core drilling unit. In addition to its data acquisition programmatic mission, GPU Nuclear is considering its deployment 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 underwater camera visibility. Another function of the DWCS is to control the presence of soluble radioisotopes present in the reactor coolant system water. A five-ton capacity service crane was installed and successfully tested inside the Reactor Building. This crane represents the defueling tooling workhorse 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 linking 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' transfer outside the vessel. Newly fabricated canister storage racks were installed in the Fuel Handling Building fuel pool. These racks will accommodate up to 252 of the special defueling debris canisters.

Three special debris canister designs were developed to support the TMI-2 defueling needs. They have the same basic outside dimensions to provide handling device and storage system compatibility. Each design is equipped with safety features engineered to provide radiological, mechanical, and thermochemical safeguards with respect to the type of fuel and core debris they will contain. Defueling planners estimate that 280 canisters will be necessary to accommodate the TMI-2 reactor fuel and core debris. Of these, 45 canisters were built and delivered to TMI by the end of the year.

Once loaded, the canisters, which have a design life of at least 30 years, will be lifted from the vessel in a 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 tube into the flooded Fuel Handling Building spent fuel pool by one of two fuel transfer mechanisms. The 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 railcar for shipment to the INEL for research.

After an intensive defueling personnel training and tooling/procedure checkout program, a review by GPU Nuclear's Readiness Review Committee, the NRC certification of GPU Nuclear's Fuel Handling Senior Reactor Operators, and the NRC approval of defueling safety analysis and procedural software, initial defueling operations began on October 30, 1985. The first step was for workers to use long-handled tools to rearrange core debris which 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 which 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.

In addition to defueling the reactor vessel, engineers are currently studying possible approaches to defueling locations outside the vessel and inside the reactor coolant system where fuel debris was transported as a result of the accident. In preparation, technicians are conducting radiological 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.

A significant amount of decontamination and shielding work has been conducted in an effort to reduce worker exposure at various points of interest associated with the primary reactor coolant system. From the survey and characterization work conducted thus far, relatively little fuel material has been located outside the reactor vessel. Specific areas where work in support of this effort has been conducted include the A and B D-rings, the two steam generators, the pressurizer, the A D-ring core flood line, the letdown coolers, and the primary water treatment system piping.

GPU Nuclear's robot-like, remotely controlled vehicle, Rover-1, was used to obtain the first samples of concrete from the Reactor Building basement. Rover was fitted with a core boring device that secured the samples from two different internal walls. Preliminary indications suggest the presence of only surface contamination on the walls. Rover was used earlier in the year to obtain samples of sediment from the Reactor Building basement floor.

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

## Onsite Data Acquisition

**Instrumentation and Electrical.** The Instrumentation and Electrical (I&E) staff completed evaluations of the resistance temperature detectors, dome area radiation monitor, pressure transmitters, and incore instrumentation. To support the formal I&E program, cables and connections, 17 cable/connection samples were obtained 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 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.

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 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 develop an actual pilot demonstration program for maintenance and surveillance with a nuclear power plant.

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.

***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

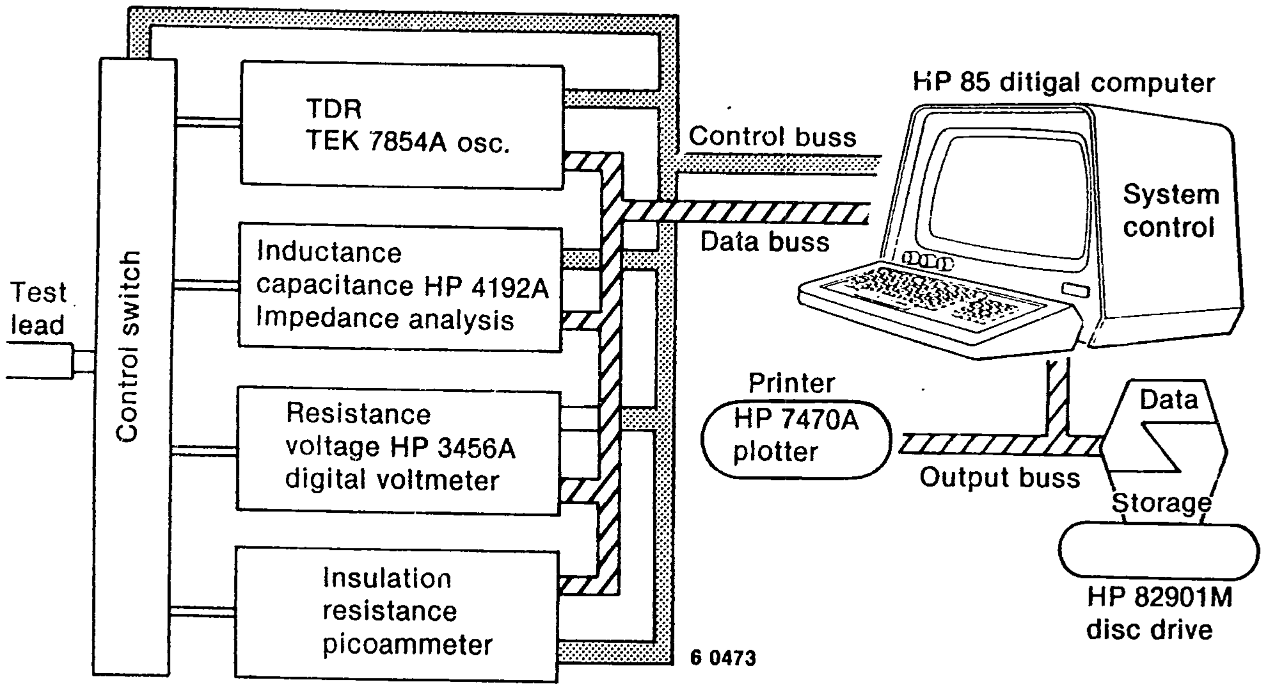


Figure 19. ECCAD system configuration.

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, involving the removal of selected cables exhibiting anomalous behavior from the Reactor Building for a thorough offsite examination. Seventeen samples were removed from accessible areas during 1985. Findings from this work are discussed below in the section Material Analysis.

***Results of In Situ Test Data Analysis.*** The results obtained from analyzing the in situ data report 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.

***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

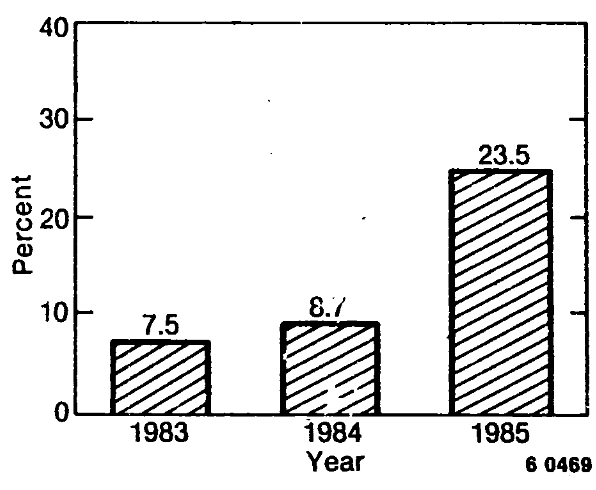


Figure 20. Newly detected anomalies by percentage of circuits not previously known to have anomalies.

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.

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.

***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 identifying 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.

***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 21). These assemblies were given priority because their safety classification has been upgraded at nuclear plants since the 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

- 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.

***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:

- 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.

**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.

**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 that will be documented as a GEND report.

## 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 predicting 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.

**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 at that time many utilities may 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 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.