TMI Unit 2 Technical Information & Examination Program

# UPDATE

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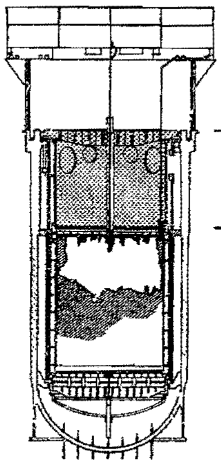
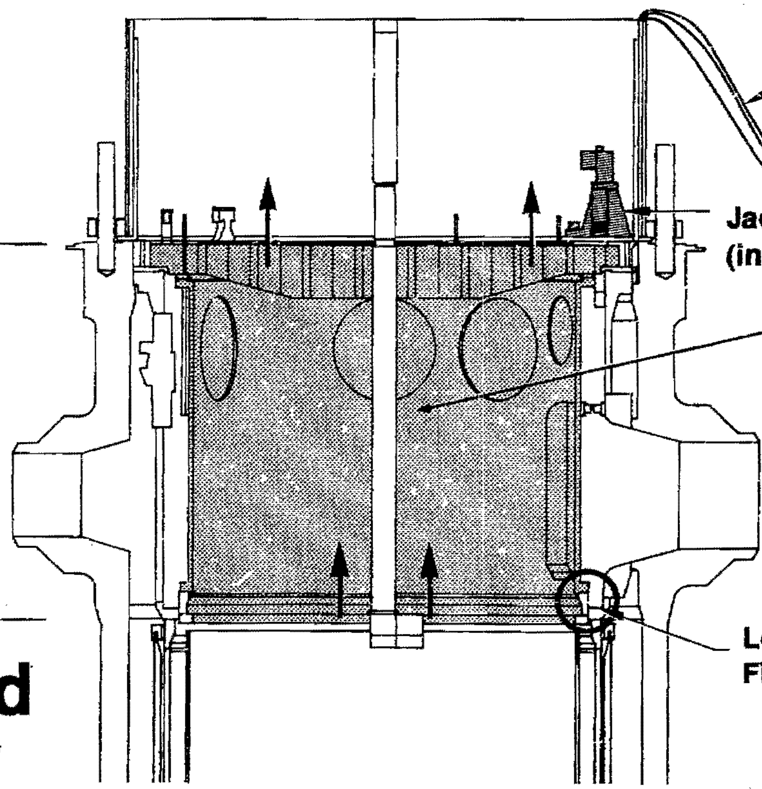


# Plenum Jacking Performed Smoothly




The plenum assembly, the 55-ton cylinder on top of the core containing the guide tubes for the control rods, was jacked in December 1984 in preparation for its removal in May (see Figure 1). Removal of the plenum will provide access for defueling the damaged reactor.

*Figure 1.* **In December 1984, the plenum assembly was jacked to 7-1/4 in. The highlighted area indicates the location of the photographs in Figures 2 and 3.**

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Plenum jacking operations began at Three Mile Island Unit 2 (TMI-2) last October with video inspections under the plenum and between it and the core support assembly. The inspections, which continued through November, were to assess the condition of the plenum, specifically to determine its available clearances and freedom from interference from other reactor components.

The inspections revealed that GPU Nuclear can expect little difficulty in plenum removal. Babcock & Wilcox (B&W) had predicted before jacking that potential thermal deformation in the way of binding could occur. GPU Nuclear did find distortion at the bottom of the plenum, but any binding resistance would have been well within the lifting capacity of the hydraulic jacks.

In concert with the inspections, which also revealed debris on the lower regions of the plenum assembly, workers dislodged unsupported fuel assembly end fittings. Many of the end fittings were already missing when workers inspected the region under the plenum. Plans for selectively knocking off the remaining end fittings reflected GPU Nuclear's expectations that many of them would drop off as the plenum was jacked. All eight of the axial power shaping rods also were removed during the end fitting separation activity.

After receiving Nuclear Regulatory Commission (NRC) approval of the required Safety Evaluation Report and permission to jack the plenum, GPU Nuclear moved the jacks into the Reactor Building from their mockup positions in the Turbine Building. During staging in the Reactor Building, two of the jacks required alterations to fit the plenum, but no other significant delays occurred.

The original plan was to jack the plenum 2-1/2 in., remove any remaining fuel assembly material, and then jack it another 2-1/2 in. Another inspection and clean-off procedure was then to follow before the plenum was to be jacked 2-1/4 in. more, to an overall 7-1/4 in.

However, jacking to 2-1/2 in. was performed with no apparent binding, so following inspection and peripheral end fitting knock-down, the plenum was jacked directly to 7-1/4 in. Jacking the plenum resulted in no measurable increase in either Kr-85 or radiation levels.

The video inspections, jacking, and end fitting knock-down were observed on remote closed-circuit television monitors and recorded on high-resolution, broadcast-quality video recording equipment. From these high-quality videotapes, enhanced photographs were obtained as further illustration of the plenum activities. In enhanced video photography, several frames of videotape (an average of 17 frames was necessary for each of the photographs seen here) are compiled into one image, resulting in a much clearer view than could be obtained from any individual frame.

Figure 2 is a closeup of "ears" to a fuel assembly's upper end fitting; the "ears" guide the upper end fitting into the plenum assembly.

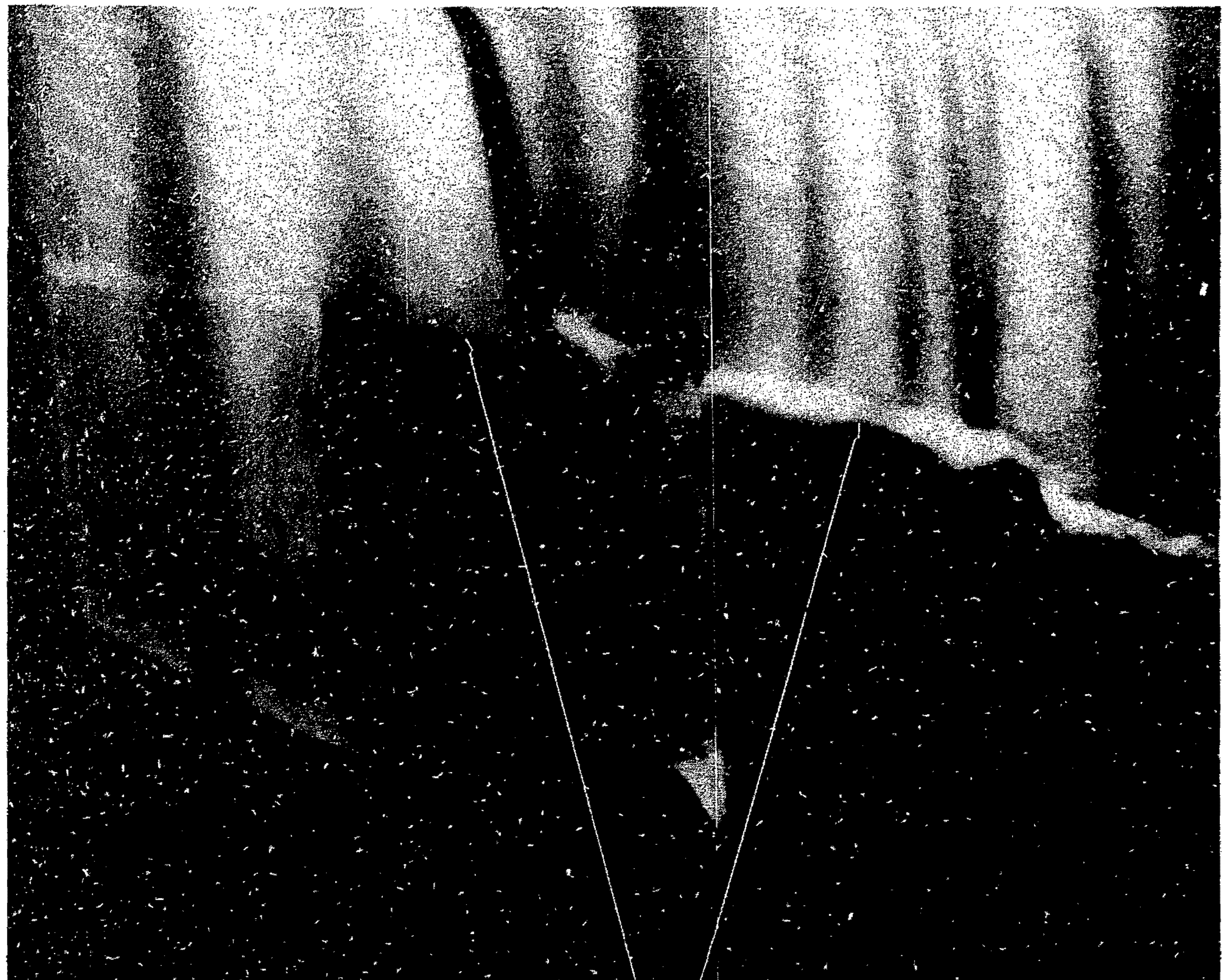



*Figure 2.* **This closeup focuses on "ears" to a fuel assembly's upper end fitting. The "ears" are beveled to guide the upper end fitting into the plenum assembly.**

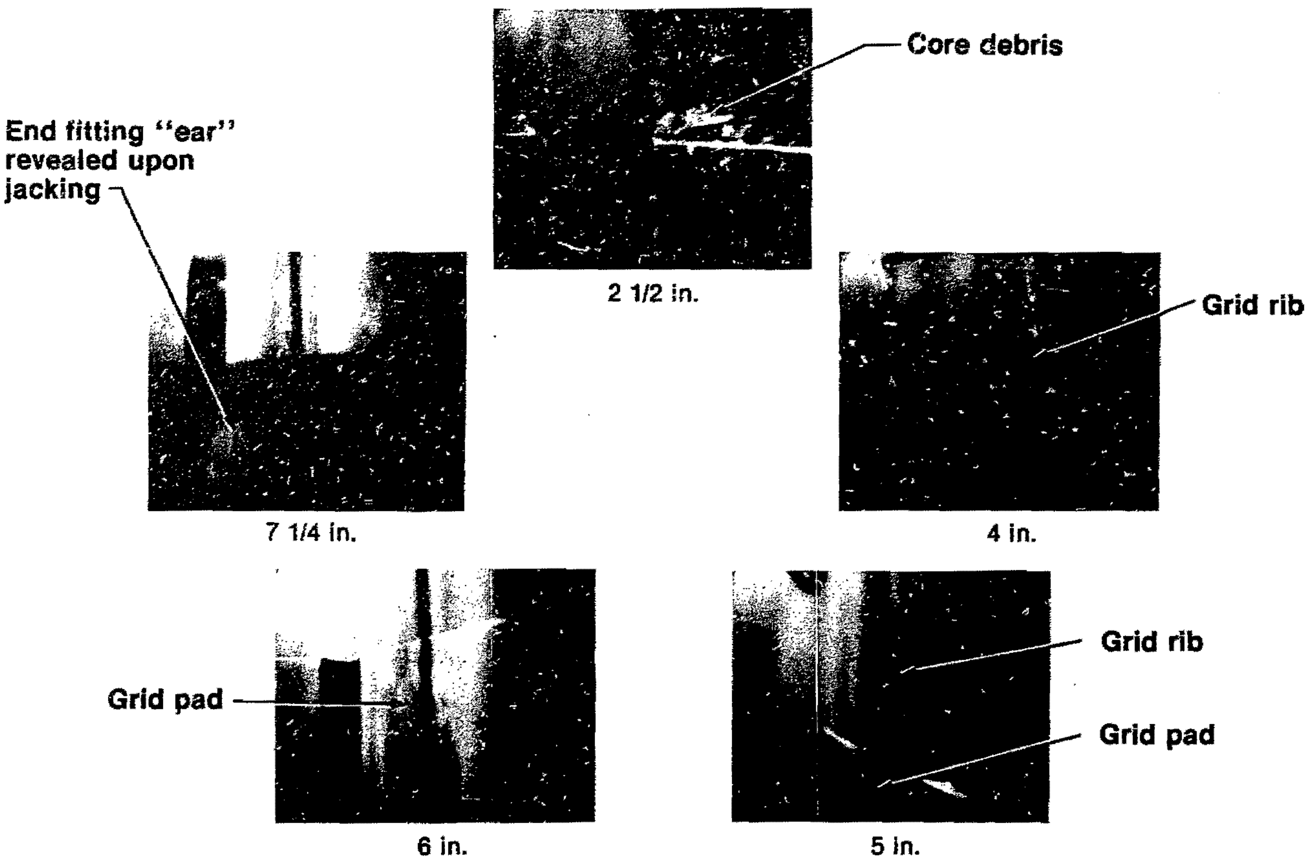



Figure 3 illustrates the jacking sequence from 2-1/2 in. to the final jacking position of 7-1/4 in., with the camera closely focused on an inside section of the plenum. Clockwise from top center, the figure shows the plenum at 2-1/2 in., 4 in., 5 in., 6 in., and 7-1/4 in. In the final picture, an end fitting "ear" has come into view, after previously being obscured by a grid pad. In enhanced photographs of stages between 6 and 7-1/4 in., the position of this "ear" is seen as unchanged, indicating the end fitting did not rise with the plenum.

Following the second jacking, workers separated the rest of the end fittings from the plenum, inspected the rubble bed and the underside of the plenum, and probed the rubble bed in an effort to determine its depth and the condition of the core below.

Preparations for removing the plenum this spring include installing a dam to hold water in the deep end of the refueling canal, putting the plenum storage stand in place, laying on the stand a large cover in which the plenum will be wrapped, and flooding the deep end of the canal. After the plenum is lifted with the polar crane, it will be placed on its storage stand and wrapped. ☐

*Figure 3.* **Clockwise from top center, these enhanced photographs show the plenum at five jacking stages. The photographs, magnified two to three times the actual size, were taken from a small area inside the plenum, using an 11-mm lens. (See Figure 1 for location of photographs.)**

# Reactor Vessel Defueling Scheduled to Start in July

Within just a few months, workers will begin defueling the damaged Unit 2 reactor vessel, whose head was removed in 1984 and plenum is scheduled for removal in May. GPU Nuclear will defuel the vessel by loading the debris into canisters that will then go through several stages of transfer and storage before being shipped off the Island to the Idaho National Engineering Laboratory (INEL). A number of contamination controls have been incorporated into the scheme to keep radiation levels as low as reasonably achievable. Figure 4 is a schematic of the defueling system for TMI-2.

Defueling will occur in two major phases: early defueling—removal of loose core debris by vacuuming, to begin in July 1985—and bulk defueling—removal of the remaining larger core debris using manually operated tools and robotic devices, to begin in November 1985.

Both activities will be carried out with much of the refueling canal dry; only the deep end of the canal will be flooded to provide shielding from the relocated plenum and the canisters loaded with core debris. Shielding over the open reactor vessel will continue to be provided by the Reactor Coolant System water in the internals indexing fixture (IIF) that sits atop the reactor vessel. Among the advantages of keeping the canal dry: a smaller volume of water will be contaminated an$^{d}$ have to be processed.

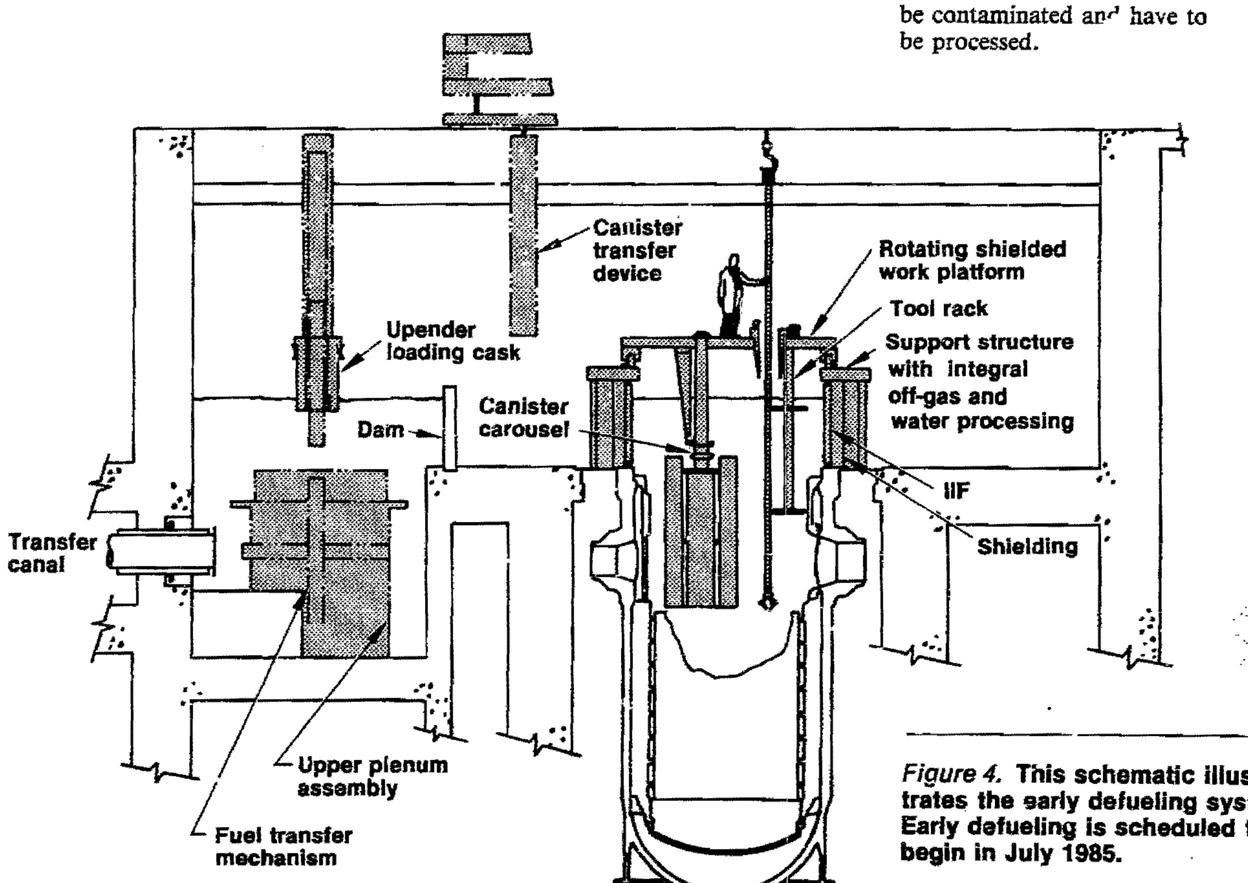



*Figure 4.* **This schematic illustrates the early defueling system. Early defueling is scheduled to begin in July 1985.**

In preparation for early defueling, workers will use long-handled tools to remove fuel assembly end fittings and some of the other large pieces of debris to clear the rubble bed for vacuuming. The workers will stand on a newly designed, steel shielded work platform and manipulate the tools through an 18-in.-wide slot. This work platform and the water in the IIF together should provide enough shielding to keep dose rates 18 in. above the platform at an average 7 mR/h. Dose rates over the 18-in. slot in the platform will be maintained at 17 mR/h. The workers will continue their "pick-and-place" work later as they vacuum out the debris.

The vacuum system will be located under the shielded work platform and will comprise a pump, piping, valves, and a filtration system. Its control console, to be located on top of the platform, will give workers the necessary valve actuation readouts, pump monitoring, manifold control, backflush control, and other fail-safe information for the entire system.

Debris and fine particles, all in Reactor Coolant System water, will be drawn through a nozzle that will be manually operated with a long-handled tool. The debris will flow into knockout canisters that will retain particulates ranging in size from about 130 μm to the size of whole fuel pellets. The knockout canister removes the medium-sized debris from the water by reducing the flow velocities, thereby allowing the particles to settle.

The smaller debris that are not retained in the knockout canisters will be drawn through the vacuum pump and discharged through filter canisters that will retain particles as small as 0.2 μm. The processed Reactor Coolant System water will then be channeled back into the reactor vessel. In the event of excessive wear or clogging, system components can be backflushed or replaced. Later, when bulk defueling begins larger pieces of debris, such as partial fuel assemblies, will be loaded into fuel canisters.

All three canister types—fuel, filter, and knockout—have an outside diameter of 14 in. and length of 150 in. The filter canisters will be positioned, two at a time, in a bracket in the vessel, below the work platform. The knockout and fuel canisters, meanwhile, will be positioned in a carousel, also inside the reactor vessel. The carousel permits one canister at a time to rotate into the loading position and will be able to hold as many as five loaded canisters in-vessel.

The three canister types have a design life of at least 30 years and can be vented, dewatered, and leak-tested. The fuel canister has an internal shroud that controls the size of the internal cavity and provides a means for encapsulating the neutron absorbing material that will provide criticality control during shipment. Also, catalyst recombiners will be incorporated at the top and bottom of each of the three types of canisters to recombine hydrogen and oxygen gases formed by radiolytic decomposition of the water in the debris.

The central feature of the defueling system is the earlier mentioned, newly designed shielded work platform. This new platform will be placed over the IIF, 9 ft above the reactor vessel flange, replacing the temporary platform that was installed after head lift. The platform rotates to provide workers with full core access. The platform also will serve as a support for in-vessel equipment, including the vacuum system and canister carousel, and provide shielding to workers standing on top.

Between the work platform and its own support structure will run various lines for water treatment and air ventilation to control off-gassing. So that the platform's ability to rotate is not impaired, careful management of cables and service lines went into the platform's design.

Because the defueling operators will not have full, direct view of their work, a system of lights and cameras will be incorporated, with techniques to improve viewing through turbid water. Monitors will be stationed on top of the work platform, as well as in the Unit 2 Command Center. Technicians will also consult reliable control and console readouts to be sure all operations are running smoothly.

Much of the defueling work will be done manually using tools mounted on the ends of 30- to 37-ft-long handles. High-pressure lines will run through the handles to activate the tools.

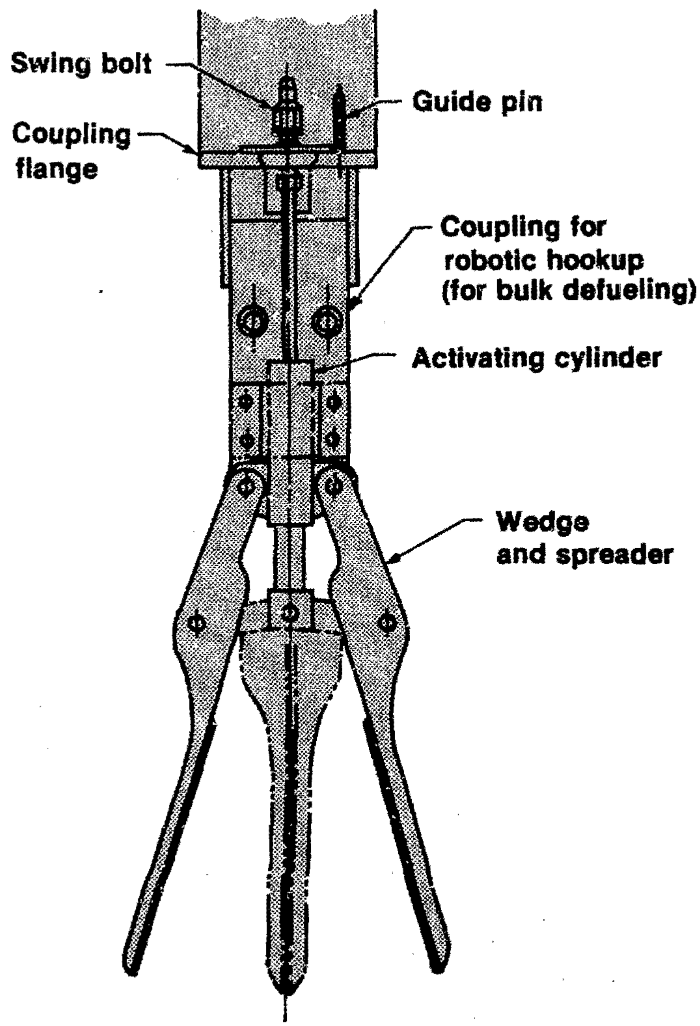
Among the tools are locking pliers to grip large pieces of debris or adjust hoses and cables; three- and four-point grippers to pick up objects from the debris pile; a grapple to lift irregular pieces, such as end fittings and spider assemblies; single rod shears, similar to scissors and capable of cutting one or two fuel rods at a time; a hydraulic parting wedge to separate and fracture material for easier handling and vacuuming (see Figure 5); bolt cutters for light-duty vertical and horizontal cutting; and hooks to lift and move debris.

*Figure 5.* **Among the defueling tools being designed is this hydraulic parting wedge to separate and fracture material for easier handling and vacuuming.**

A number of the tools have undergone proof-of-principle tests. A hydrolaser, for example, was capable of cutting through 1/2-in.-thick stainless steel by means of a high-pressure water stream and an abrasive. Some of the other tools, meanwhile, have been part of reactor servicing operations for years.

The defueling activities may release substantial amounts of soluble and suspended solids in the reactor water. Meanwhile, the Makeup and Purification Demineralizer System that normally cleans the Reactor Coolant System water has been inoperable since the accident. A defueling water cleanup system (DWCS) was therefore designed as the means to maintain a low, stable level of radioactivity in the water. The DWCS is capable of processing water from the reactor vessel, as well as from the fuel transfer canal and spent fuel pool where the loaded canisters will be temporarily stored.

The eight filter canisters that are part of the water cleanup system each will be capable of filtering 3 L/s of water. The filters are made of sintered stainless steel metal and will remove fuel fines and debris particles as small as 0.2 μm. As designed, the DWCS will be able to process out suspended solids from reactor coolant water at a rate of up to 400 gpm and soluble radioisotopes at up to 60 gpm.




Once the canisters—still in the reactor vessel—are loaded with debris, they will be hoisted through the opening in the platform into a shielded transfer cask attached overhead to the fuel handling bridge. A collar around the cask will be lowered to the platform to contain radiation fields as the canister is withdrawn from the vessel. The canister, inside the cask, will then be transported over the refueling canal to the canal's dammed and flooded deep end. Then the canister will be lowered into the water, where it will either be placed in a storage rack or immediately placed in the fuel transfer mechanism that will move the canister through the fuel transfer canal and into spent fuel pool A.

The loaded canisters will sit in storage racks in the water-filled spent fuel pool until GPU Nuclear is ready to transfer them to the Fuel Handling Building truck bay, where they will be prepared for shipment. The storage racks will have room to accommodate at least 250 canisters, during the interim, in the spent fuel pool.

In the following article, the Technical Information and Examination Program (TI&EP) presents details on the shipping cask that will be used to safely transport the loaded canisters to the INEL. □

# Selected Shipping Cask Design Stresses Safety

Loaded with debris from the damaged TMI-2 core, 238 canisters will be transported by train to the INEL, where they will be temporarily stored and later used for research. Two rail casks—each capable of carrying seven debris-filled canisters at a time—will be required for the operation; after unloading their freight in Idaho, the casks will be returned to the Island for their next shipment.

Designed by Nuclear Packaging, Inc. (NuPac), the casks will ensure that the TMI-2 core debris will be safely carried off the Island and transported to the INEL. At the Test Area North facility in Idaho, the canisters will be unloaded remotely and placed in storage racks in a water pit.

Using a conservative approach to meeting U.S. shipping regulations, the cask design provides for two levels of containment. Federal regulation 10 CFR 71.63 requires two separate containers for shipping plutonium-bearing material. In addition, the cask and its inner containment vessel will have seals that meet "leak-tight" leak rate criteria. At this low leak rate, specified in ANSI N14.5, the cask can be used to transport the core debris canisters without precise isotopic information that would be needed to calculate allowable release rates for higher leak rate seals.

As designed, the cask could be loaded wet—in a fuel pool. But to optimize the operations, these casks will be loaded dry. Additionally, the fuel pool that would be used for wet loading now holds equipment that precludes placing the rail cask in the underwater cask loading pit. Therefore, equipment is being designed to load the casks standing upright on a rail car in the truck bay, and a special loading system will be used to transport the canisters to the cask from the fuel pool.

This loading system is being designed such that operations personnel will always be shielded from the canisters and thus protected from direct radiation exposure.

Canisters containing core debris will be transferred inside a protective transfer bell from the spent fuel pool to the rail cask waiting in the truck bay (see Figure 6). The transfer bell, whose base has a sliding gate, will come to rest on the floor valve that is part of the temporary loading head that sits atop the cask. The transfer bell's sliding gate and the valve together will open, allowing the canister to be lowered into one of the cask's seven cavities. After the transfer bell is removed and sent to bring the next canister for loading, a shield plug will be placed in the cask cavity above the canister that was just loaded. (A "mini hot cell" will provide the necessary shielding over the open cavity during the plug's installation.)

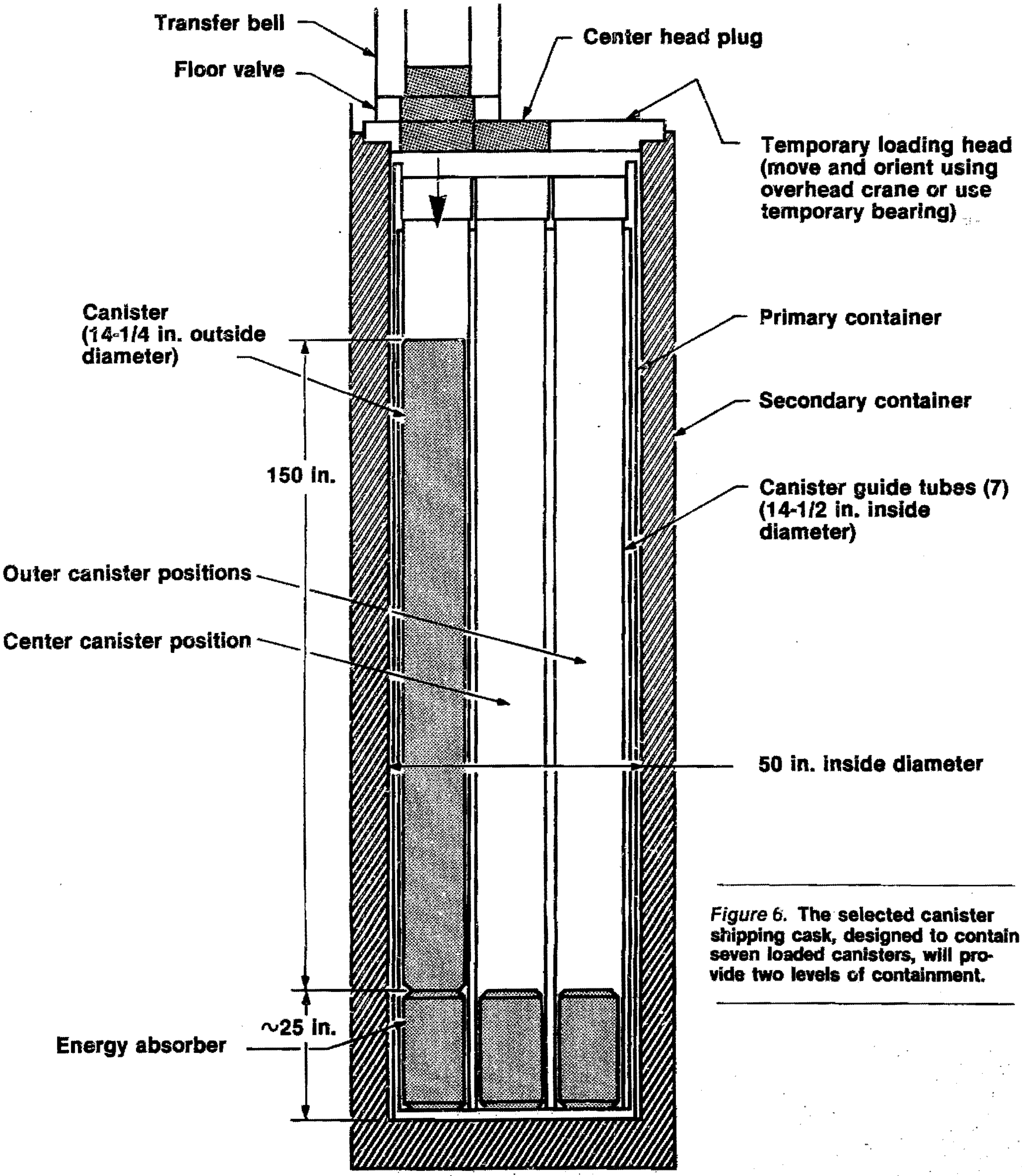



*Figure 6.* **The selected canister shipping cask, designed to contain seven loaded canisters, will provide two levels of containment.**

The temporary loading head allows the cask to be filled with canisters without leaving any of the already loaded canisters exposed (see Figure 7). This loading head has an outer head plug port that rotates over the six outer canister positions, leaving one position open at a time for loading, and a center head plug port to fill the center cavity. Once loaded, the cask will be sealed, leak-tested, and rotated from a standing to a horizontal position on the rail car using a crane.

Each cask can be loaded in four days, after which the two casks will be gone for 32 days. While the casks are away, other operations in the truck bay can be performed. Shipping 238 canisters, seven canisters per cask, two casks per train, will take about 23 months.

In addition to meeting federal regulations, the NuPac cask meets NRC licensing requirements regarding brittle fracture, containment vessel stresses (allowable stress criteria, fabrication stresses, transportation vibratory stresses, and hypothetical accident-condition impact stresses), and containment requirements (double containment provisions and containment "leak-tight" leak rates). The cask will provide criticality control for the array of seven canisters placed inside. This measure for criticality control is in conjunction with that being provided within the individual canisters for the most severe accident postulated.

In parallel with the preparation of the cask Safety Analysis Report, a drop test sequence of a one-quarter-scale cask model is being planned to verify the structural performance of the casks during impact events and thus provide experimental verification of assumptions used in analytical models. In addition, the drop test will provide the public with a readily understandable demonstration of the safety of the cask, complementing the analytical approaches to demonstrating cask safety. □

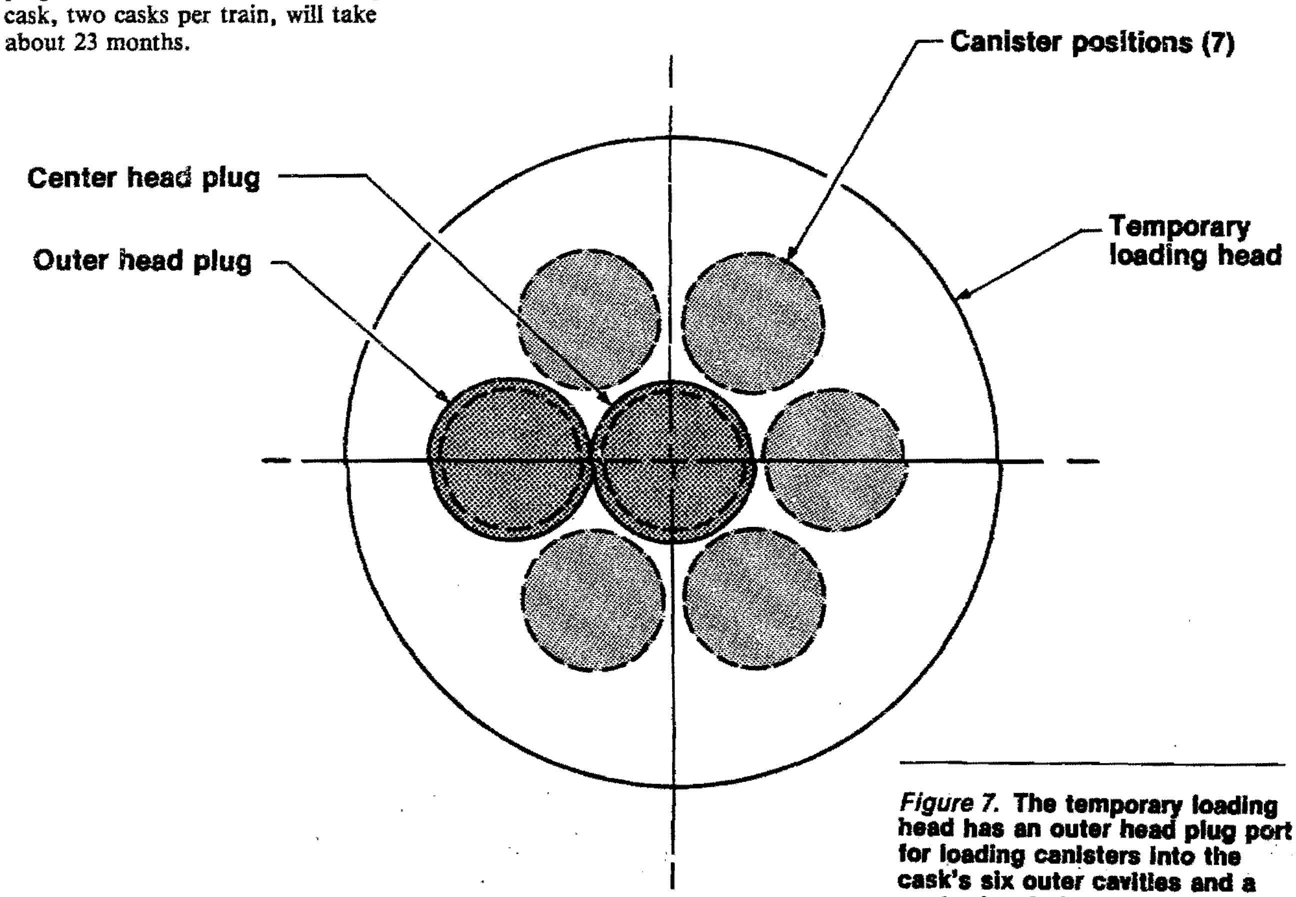



*Figure 7.* **The temporary loading head has an outer head plug port for loading canisters into the cask's six outer cavities and a center head plug port to load the seventh canister into the center cavity.**

# Cesium Elution of Demineralizers Begins

In the autumn of 1984, processing began for the two makeup and purification demineralizers that were contaminated as a result of the accident at TMI. The demineralizers contained the highest concentration of radioactive isotopes outside the Reactor Building and, as a result, left them inaccessible to workers.

During normal plant operation, the demineralizer tanks remove impurities from Reactor Coolant System water. But during the 1979 accident, highly contaminated coolant water passed through the tanks, whose resins captured about 11,000 Ci of radioactive cesium. The tanks also contained as much as 9 lb of reactor fuel particles. GPU Nuclear, with the U.S. Department of Energy (DOE), EG&G Idaho, Inc., and contractors, developed a program to remotely characterize the demineralizers, elute the high activity radionuclides from the resins, and process the resulting waste stream. (In 1983, the demineralizer resins were characterized in preparation for cesium elution. For more about this characterization, see *Update* Volume 3, Number 2, August 15, 1983.)

To remove the radioactive fission products, a mixture of water and sodium hydroxide is first pumped into each demineralizer vessel, where ions of cesium are exchanged for sodium ions from the sodium hydroxide. Consequently, the cesium is no longer bound to the resin, but dissolved in the water. Boric acid is added to this mixture to reduce its pH.

The elution equipment was designed using the data gathered from characterization and sample testing. The equipment removes the high specific activity liquid from the demineralizers through existing access points on the resin fill lines, filters it, and delivers batches to the plant neutralizer tanks. The water is then processed through the Submerged Demineralizer System (SDS).

Each batch of eluant is pumped out of the demineralizer and delivered to a filtration system located about 20 ft away. This filtration system uses a 20-μm stainless steel filter that prevents suspended particles and resin debris from being transported to downstream equipment. The filtered eluant, which contains cesium, strontium, organic contaminants, and other radionuclides, is then stored in tanks and sampled.

These samples tell engineers how effective the process is in releasing the cesium and decontaminating the demineralizer resins, whether more or less sodium hydroxide is needed to release cesium from the resins in the next batch, and what the cesium concentration is in the water in the neutralizer tanks before the mixture moves on to the SDS.

The inorganic material in the liners of the SDS captures the radioactivity that was released from the demineralizer resins, and packages it in a state that is safe for shipment to an approved waste disposal site.

The two SDS liners generated from cesium elution are expected to contain 90% of the cesium originally in the demineralizers, and will be shipped to a DOE laboratory for disposition. ☐

# Researchers Analyze Samples to Define Core Condition

Over the past year, scientists at the INEL and the B&W laboratory in Lynchburg, VA, have been closely examining material acquired from the TMI-2 core. Not only does this study help them to determine the state and nature of the core's damage and its postaccident condition, but it is providing information especially helpful for developing tools and procedures for defueling the reactor.

Eleven samples of loose debris were obtained—six at the center of the core, from the surface to the depth of 30-1/2 in., and five at the mid-radius point, again from the surface but to the depth of 37 in. (see Figure 8).

In the course of their work, analysts have been gathering data on the

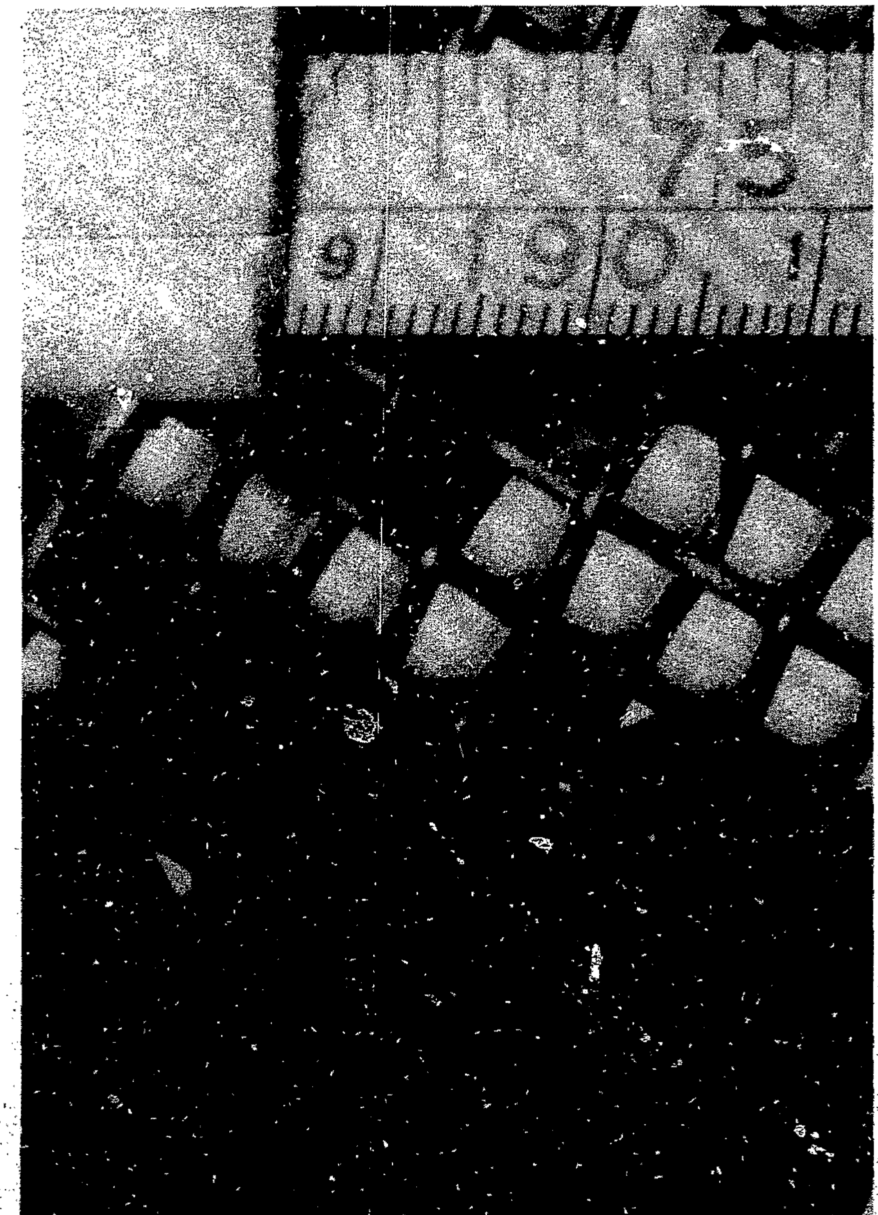



*Figure 8.* **These particles of greater than 0.157 in. are among the debris in the sample obtained 37 in. into the rubble bed at the core's mid-radius.**

samples' physical makeup—specifically size, shape, structure, and origin. They have been studying the chemical and microstructural makeup of some of the particles by conducting metallographic examinations, scanning electron microscopy, X-ray diffraction, and Auger analyses. Researchers also have been quantifying the particles' fission products. In pyrophoricity tests of the samples, they found that the particle content was not combustible in the presence of oxygen, reducing this concern during fuel shipment.

Key among their findings are the temperatures the core apparently experienced during the accident. Researchers found particles of a ceramic material of uranium and zirconium that could only have formed at temperatures above 4800°F—which is 280°F below the melting point of $UO_2$ fuel (see Figure 9). The ceramic forms when $UO_2$ fuel pellets, in contact with zircaloy cladding at that high temperature, are dissolved by the zirconium, forming a liquid phase of Zr-U-O, termed "liquefied fuel."

While this ceramic material is the first concrete evidence of such temperatures, the finding generally agrees with predictions by computer codes for severe core damage. In fact, one of the computer analyses predicted peak temperatures in the range of 5000°F.

Fission products have been retained in the core to different extents, according to their chemical characteristics. The data obtained from the grab samples provide information on the fractions of core inventory retained. The Cs-137 concentrations showed a much lower retention level than Sr-90 and I-129. Researchers hypothesize that the majority of Cs-137 was released into the Reactor Coolant System due to its high solubility in water. That I-129 and Sr-90 had a higher degree of retention in the core is significant because these radionuclides carry with them considerable consequences to personnel, as well as the public, if released to the environment.

In lesser quantities, other gamma-emitting radionuclides present in the core debris were Co-60, Ru-106, Ag-110m, Sb-125, Eu-154, Eu-155, and Am-241. A comparison with computer code data indicated that some of these fission products, such as Ce-144, Eu-154, and Eu-155, remained primarily with the fuel and were not transported out of the reactor core. An analysis of one sample, for instance, indicated that while 13% of Cs-137 was retained, 27 to 40% of Ru-106 and 70 to 100% of Ce-144 and Eu-154 were retained.

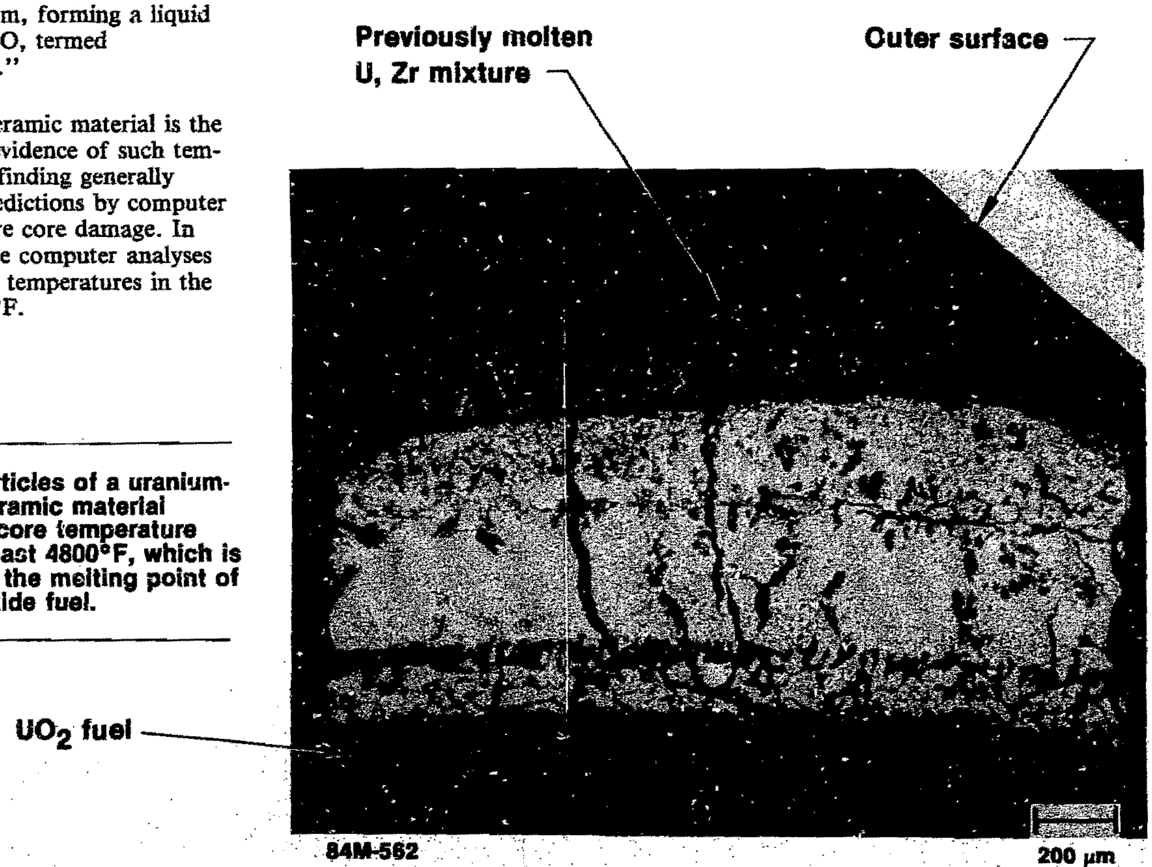



*Figure 9.* **Particles of a uranium-zirconium ceramic material indicate the core temperature reached at least 4800°F, which is 280°F below the melting point of uranium dioxide fuel.**

Through this effort and the ongoing mass balance project, research engineers will be able to establish the behavior of these radionuclides in situations like the one at TMI. As these studies continue, the INEL plans to determine the core materials to which the radionuclides tend to attach themselves to thus allow retention. In the end, this information will help engineers gain a better understanding of fission product transport and may help to change the approach for siting nuclear plants; the current 10-mile evacuation plan may be unnecessarily restrictive.

In work geared specifically toward the defueling effort, a series of turbidity, cesium release, and airborne activity potential tests were performed in two stages: undisturbed, without fracturing the debris particles, and disturbed, after crushing the debris particles to expose freshly fractured surfaces.

Researchers found that crushing the debris had minimal impact on turbidity. This work directly influences plans for defueling the damaged reactor because maximum water clarity is essential.

Disturbing the debris by crushing did, however, increase the soluble Cs-137 concentrations by a factor of 4 or 5. The soluble Cs-137 went into solution in 5 min, with little subsequent leaching. In evaporation tests, airborne activity increased 2 to 3 orders of magnitude near the end of the process, just before the liquid dried out. As soon as the liquid evaporated, airborne activities decreased almost to zero, indicating the activity possibly was transported with the water vapor. These studies regarding cesium release and airborne activity potential are essential to establishing radiation exposure controls for personnel who will participate in the defueling operations.

Among some of the general physical observations of the 11 samples: they contain fuel pellet fragments and shards of cladding or guide tubes, as well as other core structural material.

All of this research will have an impact on defueling in a number of ways. First, the physical form of the debris is significant because small particles, for example, may be suspended in the Reactor Coolant System water during defueling and cause cloudiness. Knowledge of the particle size distribution is necessary to determine the type, number, and effectiveness of filters to be used to clean the water.

The content of retained fission products also is important because it represents a potential radiological source that must be controlled. Researchers must define core source term and the levels of leachable radionuclides, such as Cs-137, that could potentially dissolve in the water during the defueling operation.

Meanwhile, the type of material in the rubble bed will influence tool designs and the defueling method.

Overall, this research is important to defining the behavior of a commercial light water reactor core under the accident conditions found in Three Mile Island's Unit 2. □

# Robot Inspects Basement Where People Are Still Prohibited

Late in 1984, GPU Nuclear initiated the most extensive examination of the Unit 2 Reactor Building basement since the 1979 accident. But *workers* were not assigned to conduct the inspection in this most highly contaminated part of the building. Instead, a six-wheeled, remote-controlled vehicle, nicknamed Rover, was called in to do the job.

Rover, a 6-ft-tall "remote reconnaissance vehicle " equipped with radiation instruments and three television cameras (see Figure 10),was twice lowered into the basement.

Two workers stationed in the basement of the adjacent Turbine Building remotely operated the robot.

Technicians used an electrically operated hoist to lower the 1000-lb robot through a hatch in the building's 305-ft elevation floor. As Rover was eased 24 ft downward, its six lights shone on the walls, still marked with a "bathtub ring," a reminder of where accident water once stood. The robot's cameras also sent back views of digital readouts from the radiation instruments.

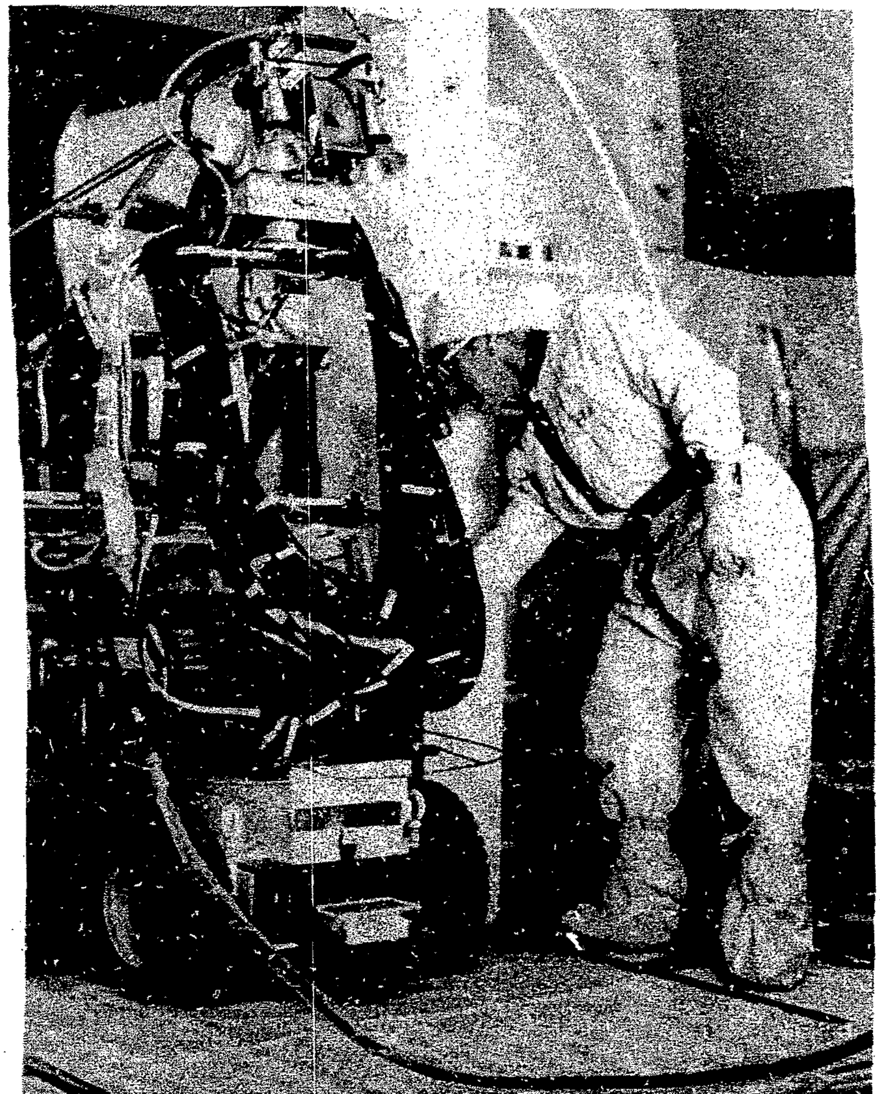


*Figure 10.* **This 6-ft-tall, remote-controlled vehicle used radiation instruments and three television cameras to inspect the TMI-2 basement.**

Once it reached the floor, Rover set out on its mission of simultaneously conducting radiological and camera surveys over preplanned paths. Before the close of the year, Rover surveyed almost half of the circumference of the basement that at one time was flooded with radioactive water from the accident. The 2 to 4 in. of water Rover waded through is water that collects in the basement with decontamination work; the water is pumped out periodically and processed.

As a result of Rover's work, GPU Nuclear has learned that an apparently thin layer of sediment lies in patches on the basement floor. General area gamma radiation readings, taken 4 to 5 ft above the floor, were between 5 and 35 R/h; and localized readings, from 4 to 7 ft up the walls, were anywhere from 5 to 1100 R/h. The highest localized radiation readings were recorded at the concrete block wall of the enclosed stairwell and elevator shaft. In comparison, radiation levels where people now work in the building are generally 0.035 to 0.1 R/h.

Data from these and future entries will help planners prepare for decontamination of the basement. After Rover and other remote-controlled vehicles complete their inspections, they will be modified to collect samples and actually carry out the decontamination activities. Technicians demonstrated on Rover's first entry that these vehicles can in fact be recovered for modifications and reused: the surface contamination that accumulated on the remote reconnaissance vehicle easily washed off with hot water before it was lifted out of the basement. □

# Videotape Reviews TMI Activities of 1984

The TI&EP recently completed a videotape program titled "1984 in Review: A DOE TMI-2 Programs Brief." Now available for loan without charge, this program reviews accomplishments in the recovery and research and development activities of 1984. Specifically discussed are head removal; plenum jacking; preparations for defueling, including tool design and characterization of the reactor core through sample acquisition and analysis; plans for shipment of core contents; preparations for core receipt; studies of various electrical components; and the continued immobilization of highly radioactive waste.

The program is available in a 20-min version and a condensed 9-min version. To obtain either of these versions of the program, contact Kim Haddock, EG&G Idaho, Inc., TMI Site Office, P.O. Box 88, Middletown, PA 17057, telephone FTS 590-1019 or (717) 948-1019. □

# TMI-2 Topics

In the TMI-2 Topics, you will read news items of interest to the nuclear power industry which may not cover work conducted under the auspices of the DOE TI&EP. The TI&EP Information and Industry Coordination group transmits such news items at technical meetings and through the Electric Power Research Institute (EPRI), the Institute of Nuclear Power Operations (INPO), and the Nuclear Operations and Maintenance Information Services (NOMIS). For more information, contact John Saunders or Jim Flaherty, TI&EP Information and Industry Coordination, FTS 590-1063 or (717) 948-1063.

## GPU NUCLEAR PROBES REACTOR CORE

In December 1984, GPU Nuclear conducted a series of probes of the damaged reactor core and found that the depth of the rubble bed averages 14 to 46 in. Using a 39-ft-long, 130-lb, stainless steel rod, and with the help of closed-circuit television cameras, workers probed the core at 18 locations.

The depth of the void, they confirmed, averaged 56 to 80 in. from the 312-ft, 1/2-in. elevation—the underside of the plenum had it been in its seated position.

Workers first carefully lowered the tapered, 7/8-in.-diameter probe until they saw on camera monitors or felt, when no visual aid was available, that it was touching the surface of the rubble bed. Then they let it drop freely to sink into the bed by its own weight. After recording the penetration, workers manually pushed the rod into the bed as deeply as they could, recording the penetration each step of the way, and then hammered it in until it would go no farther.

In most cases, the workers noted, the rod sank relatively easily into the rubble, requiring only 3 to 10 hits of the 30-lb hammer until the rod reached a hard surface. But in one probe, the workers had to drive the rod 18 times with the hammer; each time the rod sank by just fractions of an inch. The average depth at which penetration ceased was 90 to 106 in. from the 312-ft, 1/2-in. elevation.

These data most consistently indicate the bottom of the rubble bed—the level at which the probe hit impenetrable material. Analysts, however, were unable to determine the state of the material below the rubble, except that the workers had no trouble withdrawing the tool; it did not get lodged or stuck in any substance.

In conjunction with this work, GPU Nuclear took a number of radiation readings from thermoluminescent dosimeters positioned in the jacked plenum. Readings ranged from 3 to 350 R/h, which was considerably lower than the expected 800 to 1000 R/h.

Monitoring of the operation was possible using carefully positioned underwater cameras and drop lights, and the entire operation was recorded on videotape. Personnel meanwhile took advantage of the probing project to test out a new visual enhancement technique that improves low-light or photon-starved images, such as those taken from a video monitor.

## SIMPLE MEASURES CAN PREVENT INSTRUMENT FAILURE

Research engineers at TMI have learned that water damage was the most prevalent cause of failure in instrumentation and electrical equipment in the Unit 2 Reactor Building. In most cases, however, simple measures could have been taken to prevent or significantly delay moisture-related problems.

Researchers arrived at this conclusion after examining two representative pressure transmitters and three representative level transmitters from the core flood tanks. Laboratory tests confirmed the pressure transmitters operated flawlessly, while the level transmitters failed from water damage.

The difference? The pressure transmitters were equipped with moisture barriers; the level transmitters were not. Simple, inexpensive protective devices, such as conduit seals and drip shields, installed during plant construction or when the plant is down for refueling, can prevent or delay failures like those found in Unit 2.

Research engineers also have found that operator confusion could result when level transmitters and related signal conditioners and control room readouts are recalibrated for dc output signals of -10 to +10 V. In this case, a transmitter that fails will have a 0-V output signal that results in a midscale control room meter reading, giving control room operators an erroneous indication. By using a level transmitter with a dc output signal of 4 to 20 mA or 0 to 10 V and recalibrating the readout circuitry, operators have a clear indication of device operability; a control room meter reading of zero means a possible system problem or instrumentation failure.

The performance, or failure, of the TMI-2 transmitters illustrates the value of implementing such preventive strategies for all instrumentation installed in a reactor building. If you are interested in further documentation of this work, contact the Information and Industry Coordination office at FTS 590-1063 or (717) 948-1063.

## SCABBLING FURTHER REDUCES DOSE RATES IN UNIT 2

Technicians at TMI have significantly reduced radiation dose rates in the Unit 2 Reactor Building by removing contaminated paint from concrete floors. In this latest dose-reduction activity, GPU Nuclear decreased general area gamma dose rates by an average of 38% on the 347-ft elevation floor, from a dose rate of approximately 80 mR/h to about 50 mR/h.

The decontamination process, known as scabbling, called for the loosening of paint and about 1/16 in. concrete from 3700 ft$^2$ of concrete floor. Once scabbled, the floor was repainted with a nuclear-grade paint.

The scabblers use the up-and-down motion of pistons to loosen the material and are used routinely by the construction industry. TMI-2 technicians have adapted the machines for decontamination work, the major modification being the installation of a vacuum system for collecting and packaging the loosened material and minimizing the amount of airborne contamination generated during the operation.

Scabbling is part of an overall dose-reduction program, begun in early 1983, that has reduced dose rates at the 347-ft elevation of the Reactor Building to approximately 50 mR/h from 117 mR/h. TMI-2 engineers estimated that total radiation exposure to workers from early 1983 through August 31, 1984 was reduced 43%—from a potential total exposure of 893 man-rem to an actual total exposure of 510 man-rem.

TMI-2 workers have since begun scabbling the 305-ft elevation, whose floors were contaminated by radioactive material in the water that spilled during the 1979 accident.

## TMI RECEIVES INDUSTRY FUNDING SUPPORT

On January 1, the electric utility industry began aiding the Unit 2 cleanup by making volunteer payments to a program set up by the Edison Electric Institute (EEI). The EEI board of directors in 1983 adopted a resolution to create a program to voluntarily provide $150 million over six years, $25 million per year, as part of its cost-sharing effort for the TMI-2 cleanup. The board subsequently modified its plan in 1984 in order to maintain the cleanup schedule at TMI and approved a two-part program: an industry voluntary program and a program of supplemental research and development grants from six Pennsylvania and New Jersey utility companies.

The $25 million per year is the sum of approximately $11 million from the EEI industry voluntary program and $14 million from the supplemental program. Thirteen utilities have pledged to support the industry voluntary program for a total of about $66 million. Monies for the supplemental program come from funds that the six utilities otherwise would have paid as dues to the research and development organization, EPRI.

The companies participating in the supplemental program are GPU Corporation, Pennsylvania Power & Light Company, Duquesne Light Company, Rockland Electric Company, Philadelphia Electric Company, and Public Service Electric & Gas Company; Atlantic City Electric Company, while not a member of EPRI, is also a participant in the supplemental program. □

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# UPDATE

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TMI Unit 2 Technical Information & Examination Program

# UPDATE

Volume 5, Number 3 August 1985

# New Electrical Diagnostic System Supports Maintenance Activities

A primary goal of the Three Mile Island (TMI) research and development program is to assess how the TMI-2 accident affected the general condition of all systems in the Reactor Building. During the accident, some of the instrument and control signals began deteriorating to the point of ambiguity, severely handicapping control room personnel. Because the hostile Reactor Building environment prevented direct access, it was necessary to assess system performance by the electrical characteristics gathered from remote locations. To perform the task, the Electrical Circuit Characterization and Diagnostic (ECCAD) system was designed as a means to acquire basic electrical data on electrical channels and to store and format the data for easy handling and analysis (see Figure 1).

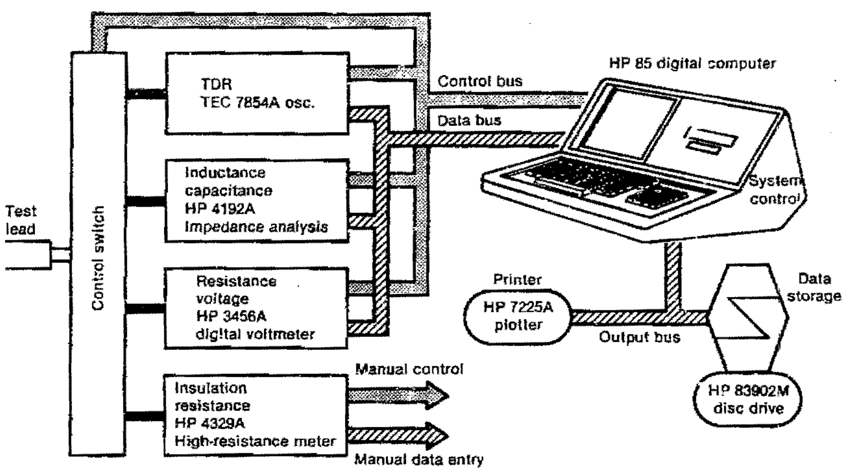


*Figure 1.* **Electrical circuit characterization and diagnostic system.**

## *Data Easily Accessed*

The ECCAD system uses commercially available electronic test equipment with computerized control to provide a means to obtain a high-quality standard data set. The data set encompasses the electrical measurements typically performed in a plant maintenance program. The most significant feature of this system is that the data, which is stored and formatted, can be easily accessed for the quality assessment and diagnosis of electrical circuit and equipment conditions. Although the system is still under development, it could be used in its present configuration after some operator training in data analysis.

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Teams of personnel started working with basic data obtained from points outside the Reactor Building to establish the validity of signals coming from the Reactor Building and assigned a degree of confidence to these signals. Since then, electrical characteristics have been monitored, recorded, analyzed, and compared with laboratory test results to gain quantitative data from which analysts could assess the ability of instruments to function.

### *Forecasting More Reliable*

Electrical characteristics monitored include indicated output signal, insulation resistance, circuit resistance, capacitance, time domain reflectometry, bias voltage, starting current, equipment actuation time response, operating current, and spectral content of output signals (feedback blanking or "noise"). Data acquisition consisted of passive and active surveillance of the above characteristics for the cable, junction points, and the end device. For the purpose of analysis, each circuit or channel was treated as a transmission line, with the end device being the load. Because the circuit's ability to function generally was not defined in terms of its electrical characteristics, analysts had to start with assumptions and arrive at qualitative results rather than make absolute, quantitative assessments. The results appear to be well suited to forecasting channel reliability.

Use of the ECCAD System at TMI-2 saved considerable test time and also resulted in high quality, repeatable data with minimum operator error. Direct storage of data on magnetic disks in the computer also eliminated paper work.

### *System Enhances Surveillance*

The ECCAD System is still in its infancy, but at least one nuclear service vendor is planning to provide an identical system to support plant maintenance. In support of maintenance, ECCAD will enhance surveillance procedures to allow for a determination of the quality of operation and the likelihood of continued reliable operation of equipment being monitored. The System can quickly verify the accuracy of abnormal instrument indications, provide a verifiable and reproducible basis for operator action, and detect degraded circuits that, with maintenance, would be returned to proper operating condition. □

# A Calculational Approach to Determining Combustible Gas Concentrations in Sealed Radioactive Waste

The Technical Integration & Examination Program (TI&EP) has developed a calculational method for determining the rates at which gas is generated in radioactive waste containers. The work is significant to facilities generating radioactive waste, because the method will decrease costs and reduce personnel radiation exposures during various venting and storage operations.

### *Gas Production A Safety Concern*

The production of combustible gases in sealed radioactive waste containers has been identified as a significant safety concern relative to handling, shipping, and storage of radioactive waste. A Nuclear Regulatory Commission (NRC) evaluation of the hydrogen gas generation problem resulted in issuing new requirements for certain certificates of compliance related to radioactive waste shipment packages.

These new requirements address hydrogen gas generation and applicable safe storage and shipment periods. The requirements state that for waste containers that have the potential to radiolytically generate combustible gases, a determination must be made by tests and measurements of canisters for hydrogen and oxygen content.

Basically, hydrogen gas concentrations must be limited to no more than 5% by volume, or the container must be inerted to ensure that oxygen is limited to 5% by volume. Compliance with this requirement is unnecessary if the containers are shipped within 10 days of sealing or venting.

## *New Requirements Too Conservative*

These new requirements affect most radioactive waste shipments from operating nuclear power plants. The TI&EP considered the new NRC requirements conservative and costly relative to financial expenditures and increased personnel radiation exposures and sought to improve predictive techniques.

In addition to the NRC requirements, utilities must consider that the determination of safe storage periods for radioactive waste containers is more significant with the enactment of the "Low-Level Radioactive Waste Policy Act" of 1980. The Act provides for the formation of interstate regional disposal facilities to relieve the present burden on the three states with low-level waste (LLW) disposal sites. After January 1, 1986, states with regional waste compacts will not accept LLW from nonmember states, thus requiring on-site storage for the affected utilities.

## *Task Force Organized*

The Utility Nuclear Waste Management Group of the Edison Electric Institute formed a "Hydrogen Generation Task Force" to study and evaluate the new NRC requirements. The task force acquired direct technical and operational experience assistance from the TI&EP. This resulted in the development of a calculational method to quantify hydrogen gas generation in sealed containers.

The calculation model was developed by applying the results of the NRC and Department of Energy (DOE) funded research projects to the gas generation problem. A modified computer shielding code was used to reduce uncertainties associated with previous predictive models. Actual TMI EPICOR II measurements were compared to predicted values with excellent agreement.

## *Calculational Method Verified*

Based on the work by the TI&EP and the Electric Power Research Institute (EPRI), the NRC acknowledged the validity of the calculational method. The Commission has modified certificates of compliance to allow calculation of hydrogen concentration as well as tests and measurements as an acceptable method of compliance to regulation.

Calculating combustible gas concentrations is now an acceptable means of determining quantities of gas in sealed radioactive containers. Waste generators will realize cost savings and reduce manrem exposures by eliminating special handling requirements for the majority of their radioactive wastes. Waste management safety will be enhanced by the ability to quickly identify those containers that present a potential hazard. □

# Drop Tests Verify Design of Shipping Cask for Safety

*Figure 2.* **Height and orientation check before end drop.**

In spring 1985, engineers at Sandia National Laboratory, in Albuquerque, NM, dropped a quarter-scale model of the NuPac 125B transport cask onto a 526,000-lb mass of concrete faced with about 4 in. of battle ship armor plate. This model of a double-containment rail cask, designed by Nuclear Packaging, Inc. (NuPac), underwent a series of drop tests as a demonstration of the cask's structural integrity and capability to survive hypothetical accidents without rupture, leakage, gross deformation, or compromise to its payload.

EG&G Idaho, Inc., the U.S. Department of Energy's contracting manager of the TMI-2 cleanup project, selected the 125B rail cask to transport the damaged fuel to the Idaho National Engineering Laboratory (INEL). The results of the drop tests are a major chapter in the Safety Analysis Report that the Nuclear Regulatory Commission is now reviewing for cask licensing. The drop tests in fact confirmed the positive results of earlier computer analyses: that the cask can safely contain the TMI-2 core debris under the extreme conditions of hypothetical accidents.

## *Materials Effectively Protect Payload*

Constructed principally of stainless steel and lead, the 125B rail cask has four basic components: foam-filled overpacks to absorb energy and protect the ends of the outer cask; the outer cask containment vessel, with lead shielding; the inner containment vessel, with borated concrete for neutron moderation and criticality control; and aluminum honeycomb energy absorbers at the ends of each canister tube in the inner vessel, to limit the axial "g" loads that could develop on the core debris canisters.

Additionally supporting the canister tubes are steel plates that make up a hub, spoke, and wheel arrangement in the inner vessel. Both vessel lids have rupture discs that contain pressure buildup in normal and hypothetical accidents. The discs are designed to rupture if the cask experiences a fire of longer duration and with temperatures significantly higher than considered even for hypothetical accidents.

While the full-scale NuPac rail cask will weigh approximately 183,000 lb, its quarter-scale model, a full representation of the actual cask, was 1/64th that weight, or 2,830 lb.

## *Regulation Establishes Accident Conditions*

Federal regulation 10 CFR 71, subpart F, requires the evaluation of this package dropped onto a flat, essentially unyielding surface, given certain hypothetical accident conditions. The regulation specifies that the package strike the surface in a position for which maximum damage is expected.

*Figure 3.* **Instant prior to impact from oblique drop.**

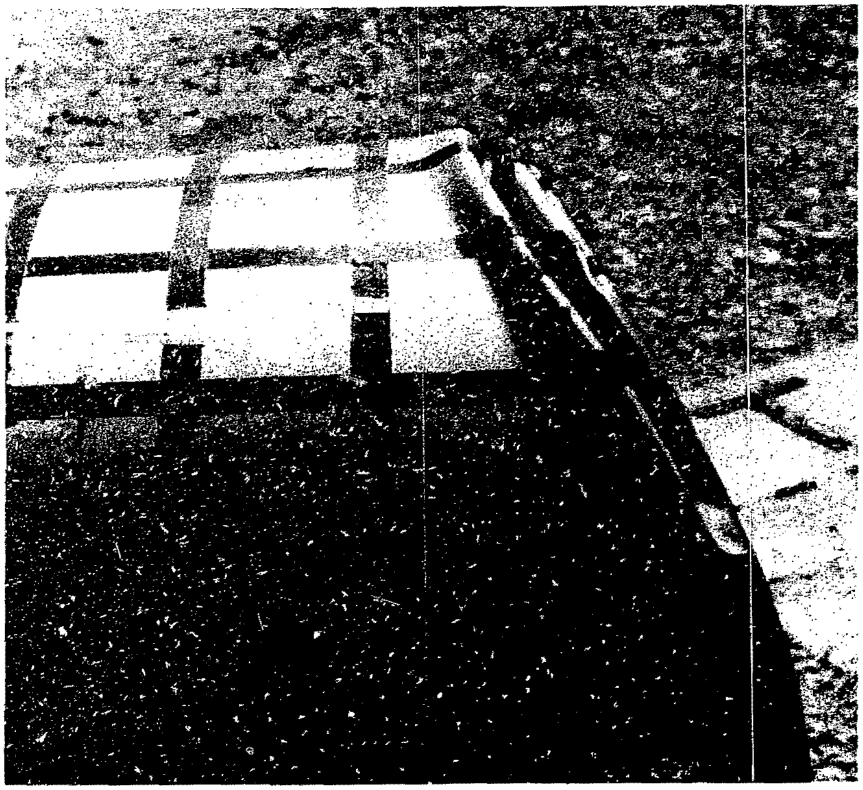
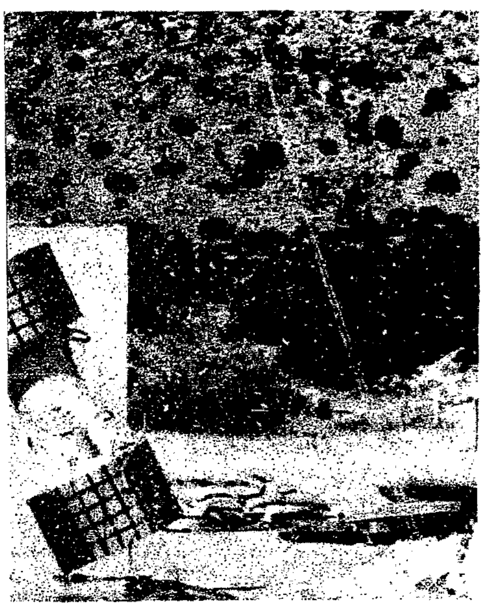


*Figure 4.* **Overpack damage from oblique drop.**



The model was dropped three times from 30 ft: flat on its bottom end, at a 62-1/2-degree (oblique) angle onto its lid, and on its side. It was then dropped twice from 40 in. onto a 1-1/2-in.-diameter pin. In these two puncture tests, the cask came down on its side and then on its lid onto the pin, demonstrating the integrity of the side wall and closure of the cask. Figures 2 through 4 show preparations, an actual drop, and a closeup of damage.

Nearly all permanent damage to the package was limited to the external overpacks and internal energy absorbers, as expected and desired. The only significant exception was damage to the outer cask outer shell and lead shielding on the side of the cask where it came in contact with the pin. The inner vessel experienced no damage.

Before the series of tests began, the model was instrumented with accelerometers, strain gauges, and thermocouples to obtain a detailed record of responses. Preceding each drop, workers swept, wet down, and reswept the target surface to eliminate dust that could cloud up and obscure the model upon impact. Portable, gridded stadia boards erected behind and to the sides of the landing surface provided a contrast for filming the experiment and for measuring velocity, elasticity, and deformation.

For the first two drops, the cask model was refrigerated to test its response at low temperatures, specifically to confirm that the unit is not subject to brittle fracture—a "worst-case" condition. By the time the model reached the drop site and was prepared for the test, its temperature was the desired -25°F. The test confirmed that the kind of stainless steel being used to construct the cask does not lose its ductility at temperatures this low, eliminating concerns about brittle fracture. The model was at ambient temperature for the side drop and puncture tests.

## *Damage Expected by Design*

After the end and oblique drops, inspectors returned the model to the laboratory to examine the overpacks, leak-test containment seals for both vessels, and torque-check the lid bolts. After the last drop test, the cask model again was leak-tested and then X-rayed and measured to be sure there was no hidden damage. Also, the overpacks were sectioned and examined to see how well they performed.

Removing the cask lid, inspectors found only minor deformation to the seven top energy absorbers—damage that was intended by design. Just as expected, the lower seven energy absorbers clearly protected the payload; the seven quarter-scale canisters were undamaged. Leak tests performed before and after the drops confirmed that the seals maintained their integrity.

X-rays showed that even after repeated impacts, no quantifiable amount of lead slumped; only the side puncture drop reduced lead shielding, but to an extent consistent with federal regulations.

The puncture tests verified the equations used to determine the thicknesses of cask materials. In the side puncture test, most of the deformation occurred at the point of contact; the outer shell was indented by less than its thickness and maintained its integrity against puncture. Slight residual elastic stresses were induced in the package shells due to a modest inelastic deformation of the lead shield.

Consequently, the results of the actual drop tests verified the positive findings of earlier computer analyses conducted to determine cask safety. Most important, the stresses on the cask model were well below the yield stresses for cask materials. Also, the damage assumptions for input to the computer thermal analyses were found to be quite conservative compared to the actual damage from the drops. □

# NuPac Rail Cask Featured in Videotape

"A Shipping Cask Developed for Safety" is the title of an 18-minute videotape produced by DOE contractor EG&G Idaho, Inc., now available for loan without charge. The program reviews the criteria behind the selection of the double containment rail cask designed by Nuclear Packaging, Inc., explains the cask design, and features various drop tests of a quarter-scale model, conducted to demonstrate cask safety.

To obtain a copy of the program, contact Kim Haddock, Administrator, EG&G Idaho, Inc., TMI Site Office, P.O. Box 88, Middletown, PA 17057, telephone FTS 590-1019 or (717) 948-1019. □

# Program Highlights

**CESIUM ELUTION COMPLETE**

In mid-March 1985, GPU Nuclear, with technical support by DOE contractor EG&G Idaho, Inc., completed cesium elution of the two makeup and purification demineralizers in Unit 2. Cesium radioactivity was reduced by approximately 70% in demineralizer vessel A and by approximately 90% in vessel B.

The Submerged Demineralizer System liner that was used to decontaminate the eluant containing the cesium from the vessel was then shipped in April from TMI to Rockwell Hanford Operations in Washington State. It was buried there the following month, the last liner DOE accepted for its research and development monitored retrievable burial demonstration program.

**DEFUELING BEGINS THIS FALL**

GPU Nuclear will begin defueling the damaged reactor vessel at TMI this Fall. Debris will be loaded into canisters that will then go through several stages of transfer and storage before being shipped to the Idaho National Engineering Laboratory. For details of the defueling operation, see Update Volume 5, Number 2, dated February 1985.

**PLENUM SUCCESSFULLY LIFTED**

The TMI-2 reactor plenum was lifted successfully in May and now rests on its storage stand in the deep end of the refueling canal. The plenum was made accessable last summer by the removal of the reactor vessel head and is the last remaining component to be removed in preparation for the start of defueling this fall.

**CORE TEMPERATURES ALLOWED FUEL MELTING**

Metallurgical studies conducted by EG&G Idaho, Inc., indicate that portions of the core reached 5100°F during the TMI-2 accident, which is the melting point of uranium dioxide fuel. Although previous results showed the temperatures may have exceeded 4700°F during the accident, the latest study provides the first firm evidence that some fuel melting occurred. Studies are continuing to determine how much of the core reached temperatures that would allow melting of fuel.

**FINAL LOFT TEST SIMULATES TMI ACCIDENT**

The final test at the Loss-of-Fluid Test Facility (LOFT) at the Idaho National Engineering Laboratory simulated the partial meltdown conditions experienced during the TMI-2 accident. The partial LOFT meltdown took 4.5 minutes and produced temperatures of more than 4,400°F at the center of the core. Eight of the fuel rods at the center of the 100-rod assembly were deliberately deprived of cooling water for the simulation. During the 4.5-minute period, temperatures of the eight rods rose from 120°F to more than 4,400°F. ☐

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