TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 4, Number 1 December 15, 1983

New Tool Maps Shape of Damaged Core Internals

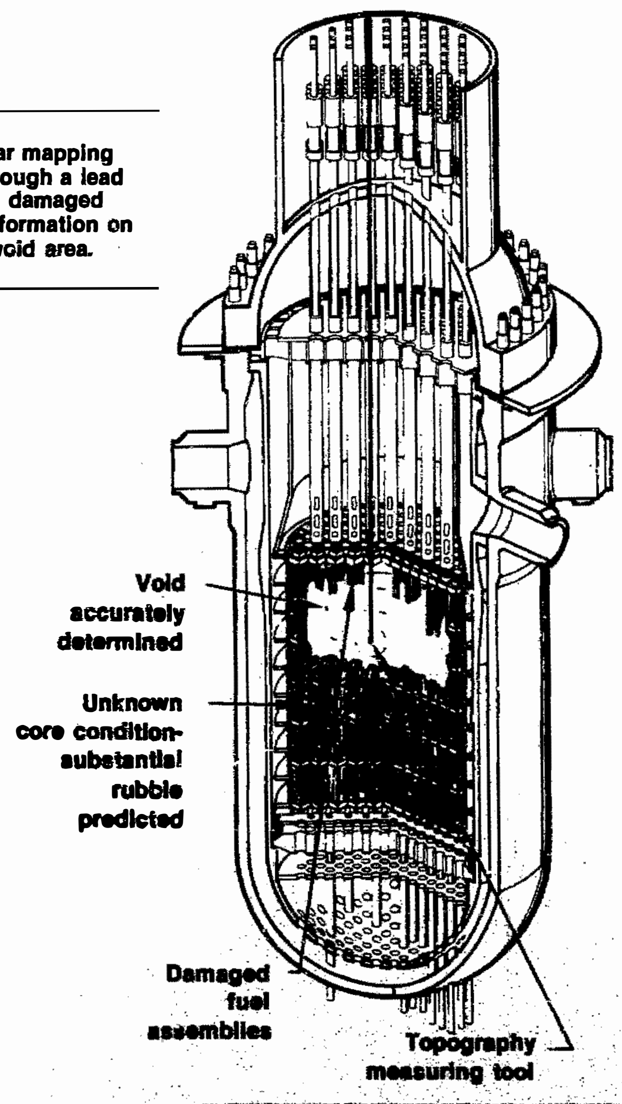

Figure 1. The CTS sonar mapping device was lowered through a lead screw opening into the damaged TMI-2 core to obtain information on the shape of the core void area.

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A preliminary review of the CTS data indicates that the void is roughly symmetrical and in some locations extends nearly to the core former wall (the structure marking the core boundary). That review, based on only about 5% of the total data, clearly shows that very few fuel assemblies in the core appear to remain intact, and those that may still be intact are located primarily in the peripheral row of fuel assemblies. The CTS data also show many objects, presumably the uppermost portions of damaged assemblies, hanging down into the core region from the underside of the plenum. After several months of computer-assisted data compilation and laboratory analysis are completed, engineers will be able to report definitive results from the CTS work.

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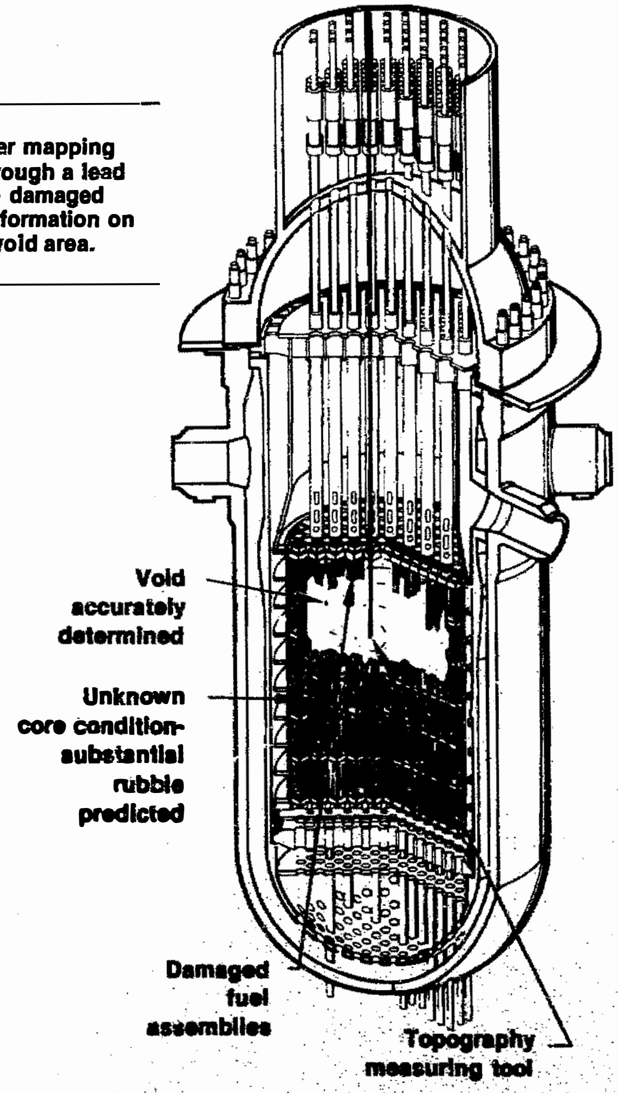

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are lowered into the core, and a positioning head, which supports the boom and provides up and down as well as rotational movement for the system. The 1-3/8-in.-diameter boom is 40 ft long and has acoustic transducers at the bottom end. The transducers, located in the "sensing head," are lowered into the reactor through a manipulator tube, which replaces the control rod drive mechanism motor tube normally occupied by a lead screw. Once inside the core void, as shown in Figure 1, the sensing head sends out an ultrasonic signal that reflects off the first barrier it encounters and returns to the transducer which sent it. The time required for the signal to return is directly related to the distance the reflecting surface is from the sensing head.

The sensing head contains six pairs of transducers that point in six predetermined directions. When the CTS was installed in the core, it was oriented relative to fixed points on the core service structure. One pair of transducers points straight down and records the sensing head's altitude above the bottom of the void. Another transducer set, pointing horizontally outward, measures the diameter of the void area. The four remaining transducer sets point at angles both above and below the horizontal at +30, +45, -35, and -60 degrees. The sensing head transducers can provide data on the location of an object within 1-1/2 in.

During operation of the CTS, the sensing head was driven to within 6 in. of the bottom of the core void. The sensing head then rotated a full 360 degrees in an automated, continuous motion. As the head rotated, a selected transducer at each location transmitted a signal every 0.9 degree of rotation, resulting in 400 data points per transducer per 360-degree revolution. The sensing head then automatically raised 1 in., and the entire procedure was repeated until the tool reached the top of the core cavity. A total of roughly 500,000 data points were obtained during CTS operation.

During the weeks and months which followed the incore work, all data collected were processed using corresponding computer software to develop a complete series of horizontal and vertical "cross-sections" of the core, called slices. The data are being used to develop a topographic map showing overall shape of the core cavity, location and shape of damaged fuel assemblies, and other such features.

Using the CTS before head and plenum removal will allow researchers to analyze the configuration of the core before disassembly and defueling work alters that configuration. The data gathered will be used to refine core relocation predictions in severe core damage accident assessment models.

The data will also be used in analyses of reactor system components which may be built-in mitigators of accident effects. For example, early review of the data indicates that the core void is nearly, but not entirely, symmetrical. The CTS data will provide information on where both the greatest and least amount of damage occurred. Engineers can then examine these areas to identify possible reasons for variances in the amount of damage.

General Public Utilities Nuclear Corporation (GPU Nuclear) will use CTS data to assist planners working toward removal of the plenum in 1984, for the data show material hanging from the central portion of the plenum which must be removed before that component can be lifted. The CTS also provides detailed information on the radial extent of the core void, its shape and prominent features such as partial and intact fuel assemblies. All this information will influence how and where fuel assembly removal—the last major stage in TMI-2 cleanup—will begin. □

First Samples of Damaged Core Obtained for Analysis

Engineers participating in the DOE's TMI Reactor Evaluation Program entered the Unit 2 Reactor building in September and October 1983 to obtain the first actual samples of damaged core materials. The March 1979 accident caused part of the fuel to fragment into gravel-sized and smaller pieces which now constitute a rubble bed in the core. Prior to the sample gathering, scientists could only speculate on the particle size and makeup of the granular debris.

EG&G Idaho, Inc., engineers, supported by DOE and in cooperation with GPU Nuclear, designed and built the specialized sampling tools used in the Reactor Building entries. To obtain six debris samples, team members used two types of rubble bed samplers—one a surface sampler and another for sampling below the surface. The surface sampler, shown in Figure 2, is called the clamshell sampler because of its open-shut mode of operation. It was designed to obtain core debris from the top of the rubble, especially any large rubble chunks. The stainless steel device is 6 in. long and 1.4 in. in diameter. Operators remotely open its hinges to a width of several inches to obtain samples.

The rotating tube or subsurface sampler is the same size as the clamshell sampler. In Figure 3, an engineer practices attaching the subsurface sampler to the sampling boom in a mockup facility. The subsurface sampler readily enters the rubble bed surface with its pointed bottom tip. Once the device is below the surface, operators remotely slide open the sampling chamber's revolving door, admitting up to 2 in.^{3} of subsurface rubble.

Each sampling device was lowered into the core at two locations: core center location H-8, and location E-9, at one-half the core radius. The first samples were taken at three different depths at each location: the surface, 2 to 3 in. down, and 22 in. below the surface. The samplers were lowered into the core one at a time on the end of a 46-ft-long boom. This boom, lowered in four sections to the rubble bed, had demarcations along its length to provide operators with sampler depth positions throughout the operation. Once operators completed sampling, they raised the boom and sampling tools up through a sample container situated over the control rod drive mechanism opening. The 12-in.-high, steel-shielded container had a trap door bottom which sealed shut after the sampling tool containing the core debris was secured inside. Based on radiation readings taken after sample acquisition, six good-sized samples were obtained. The readings ranged from 220 mR/h gamma to 1100 mR/h gamma at the outer surface of the steel sample container.

The six samples obtained during the grab sample work are currently being analyzed at Babcock & Wilcox research facilities and at the INEL. The results of these thorough chemical and microstructural analyses are expected in the spring of 1984. They will reveal for the first time the actual makeup of the rubble bed contents. These results will be studied to characterize the nature and history of damage to the core and will yield information needed for fuel handling and removal operations. □

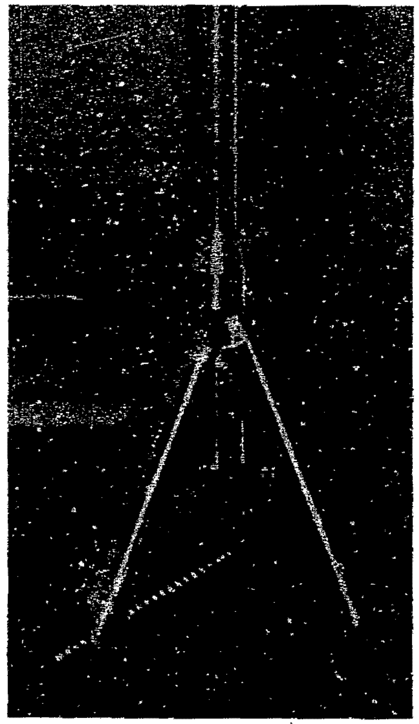


Figure 2. **Seen in its open position, this clamshell sampler was used to obtain debris from the surface of the rubble bed inside the TMI-2 core.**

Figure 3. **An engineer works with the subsurface sampler used to obtain core debris from beneath the surface of the core rubble bed.**

Control Rod Drive Mechanism Lead Screw Samples Evaluated

One of three control rod drive mechanism lead screws removed from the reactor vessel to provide access for closed-circuit television inspection of the reactor internals and for core damage assessment has undergone extensive laboratory examination. The H-8 lead screw selected for examination came from an area of core damage at the center of the vessel.

In November 1982, a 30-in. threaded section of the H-8 lead screw was cut into three pieces and removed from the Reactor Building. Figure 4 shows the location of the lead screw in the reactor and the sections cut and sent to laboratories for evaluation. The first section was sent to Battelle Pacific Northwest Laboratory, the second to Babcock & Wilcox's Lynchburg Research Center, and the last remained on site for solution chemistry studies.

Contact radiation readings ranging from 30 to 60 R/h gamma were detected on the three sections, primarily due to the presence of ^{137}Cs. One section was used in investigations of techniques for removing cesium and other fission products from stainless steel reactor internal components. The sample was soaked in increasingly strong chemical solutions, ranging from borated water to nitric-hydrofluoric acid. A noticeable decrease of ^{137}Cs and ^{125}Sb activity occurred only after the aggressive nitric-hydrofluoric acid bath, indicating that the cesium deposits were very tightly bound to the lead screw. Additional experiments showed that there were no detectable amounts of metallic zirconium, zirconium hydride, or zirconium-silver alloys present on the lead screw. (There had been speculation that the presence of these materials could constitute a pyrophoricity hazard during head lift.)

In detailed analyses of the second section, the lead screw was examined visually and samples of surface debris were collected. Nine metallographic specimens were cut from selected locations for microstructural evaluation and reconstruction of the peak temperature profile of the lead screw. This section was found to have loose particulate debris on the surface and a multilayer film on the stainless steel. The lead screw deposits indicated extensive core materials reaction. Lead-screw-deposit particles bearing uranium and zirconium (indicating fuel cladding interaction) accounted for approximately 10% of the particles analyzed. The presence of strontium-silver-bearing particles, which constituted approximately 6% of the particle population, indicated that the silver-indium-cadmium control material reacted with the zircaloy fuel rod cladding or guide tubes after control rod failure.

Metallography and microscopy revealed three distinct layers on the lead screw. An inner layer, approximately 3 μm thick, was identified as being a typical reactor water corrosion film. A second chromium-rich layer, 10 to 90 μm thick, was also identified. About 90% of the cesium on the lead screw sample was associated with this second layer. As noted in examination of the first lead screw section, the cesium could not be effectively removed from the lead screw by any decontamination solution except the nitric-hydrofluoric acid. This implies that a large concentration of cesium may remain on vertical underhead surfaces even after proposed flushing efforts. The third and outermost layer, which ranged in thickness from 25 to 75 μm, was readily removed with a wire brush. This layer contained approximately 85% of the ^{90}Sr and over 90% of the uranium on the lead screw section.

In March 1983, the rest of the lead screw was cut into sections and shipped to the INEL for detailed examination. Extensive testing is now underway at the INEL to more completely identify the type and quantity of fission product and uranium deposits on the lead screw. Examinations will determine the maximum temperatures to which the lead screw was subjected and the amount of tellurium deposited in the lead screw debris and on its surface. The testing should also reveal if there is a change in composition and quantity of radioactive material depending upon lead screw location in the core and should further characterize the nature of the loose and tightly adhered layers discovered in the early analyses. □

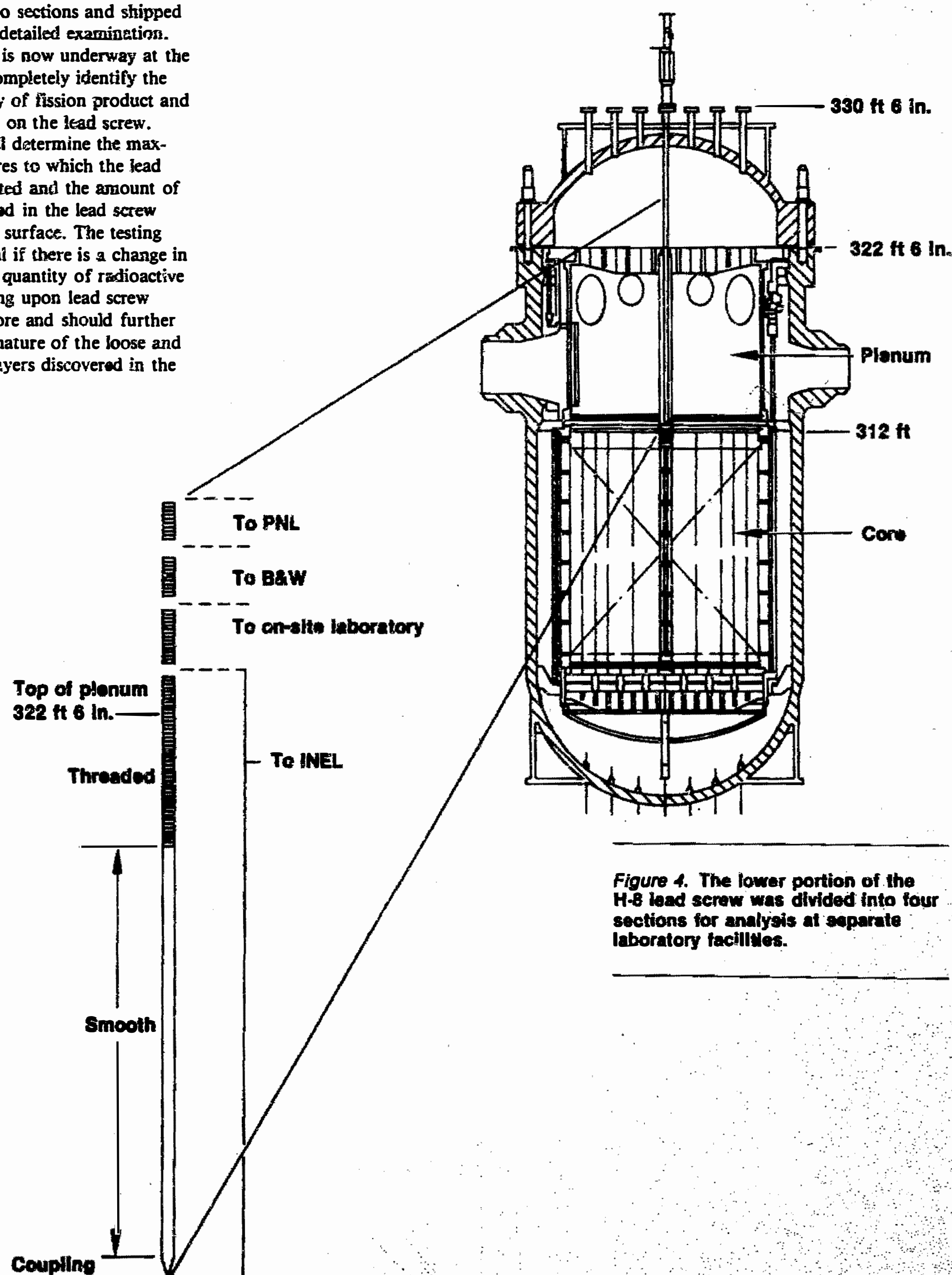


Figure 4. The lower portion of the H-8 lead screw was divided into four sections for analysis at separate laboratory facilities.

Accident Waste Shipment Goals Reached

The DOE Technical Information and Examination Program (TI&EP) reached a milestone in the TMI-2 recovery program during the summer of 1983, when the last of 50 EPICOR II canisters left the TMI site for the INEL, and the last Submerged Demineralizer System (SDS) liner used to process Unit 2 Reactor Building basement water left for Rockwell Hanford Operations, a DOE contractor in Richland, Washington.

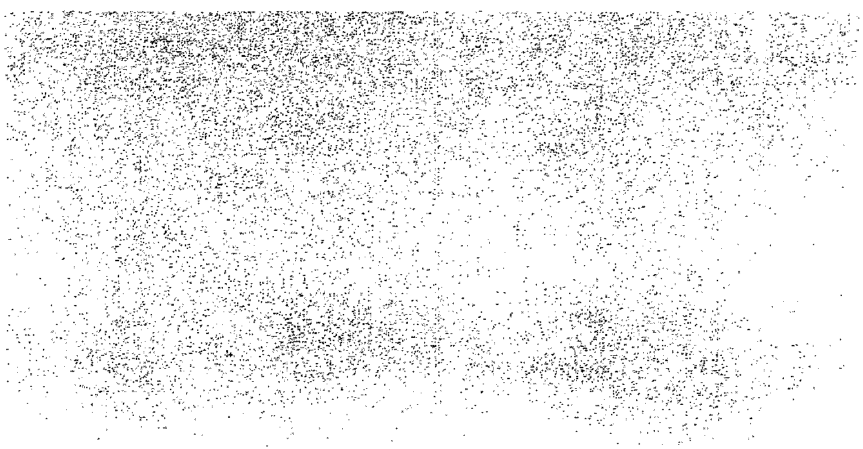
In a letter to the TI&EP staff marking completion of the EPICOR task, Shelby Brewer, DOE Assistant Secretary for Nuclear Energy, noted that the final shipment of an EPICOR liner in July 1983 "not only opened the way for a valuable research program," but also alleviated "concerns of the people around the TMI site" that the plant would become the permanent disposal location for the wastes. In August, Dr. Brewer attended formal ceremonies as the last SDS liner used to process basement water was shipped off the island. See Figure 5.

The EPICOR II canisters are prefilters from the EPICOR II water processing system at TMI-2, which decontaminated 565,000 gal of accident water from the Auxiliary and Fuel Handling buildings. The curie loadings on the canisters after processing accident water ranged from 160 to 2200 Ci. The SDS is a water decontamination process that uses inorganic material called zeolite, rather than the predominantly organic resins used in EPICOR II, to adsorb the fission products from the water, concentrating them in a form suitable for safe shipment and disposition. The SDS processed a total of 600,000 gal of highly contaminated water from the Unit 2 basement. Zeolite, a substance resistant to radiation damage, has been found to accommodate radioactivity loadings in excess of 20,000 Ci/ft^3, while the resins in the EPICOR II system normally accommodate loadings of less than 40 Ci/ft^3.

Figure 5. **Dr. Shelby Brewer, DOE Assistant Secretary for Nuclear Energy, accepts the last SDS liner used to process accident waste water while GPU Nuclear President Robert Arnold looks on.**

Of the 19 SDS liners that DOE is accepting for research and disposition projects, only six were not used in basement water processing. These six liners, used to process Reactor Building decontamination water and reactor coolant system water, remained on the island following the August ceremonies there, with one liner scheduled for shipment to the Richland site in October, two in November, one in December, and the final two in February 1984. Of the 19 SDS liners, 16 will be buried in special concrete overpacks for a monitored burial demonstration program. The other three liners were used in 1983 in a vitrification demonstration, whereby the zeolite-ion-exchange media were mixed with glass formers and vitrified into a glass log to trap the radioactive contaminants.

Of the 50 EPICOR II canisters sent to the INEL, 47 will be buried at a commercial site in high integrity containers, capable of immobilizing the wastes for 300 years. A related article on these special containers appears in this issue of the *Update*. The remaining three EPICOR canisters are being used for research and disposition projects, such as resin solidification and resin degradation studies. □



New Container Handles TMI-2 Wastes

Technicians at the INEL near Idaho Falls, Idaho, have begun loading EPICOR liners into high integrity containers or HICs in preparation for shipment to a permanent storage area. The loading marks the end of a pioneering effort to design the first disposal containers for items with high radiation levels.

Nuclear Packaging Incorporated of Tacoma, Washington, built two prototype containers and will build 45 additional containers following a DOE decision to demonstrate that wastes generated as a result of the TMI-2 accident could be handled as commercial reactor wastes. EG&G Idaho, Inc., worked with Nuclear Packaging during the 18-month design and development process. The containers are 7 ft tall by more than 5 ft in diameter. In Figure 6, a HIC sits on a trailer with an EPICOR liner nearby.

"The container is designed to hold up to 2500 Ci of beta-gamma emitting wastes," according to one of the design team members, Ray Chapman of EG&G Idaho. Immobilizing wastes for a minimum of 300 years and meeting design criteria in 10 CFR 61 were other goals for the project. The Transportation Technology Center at Sandia National Laboratories developed the final design criteria. A peer group made up of representatives from throughout the nuclear industry evaluated and approved the design.

The containers are made of steel reinforced concrete with a steel inner liner. The inner shell is epoxy coated for corrosion protection. The concrete walls range in thickness from 6 to 11 in. A special feature of the lid is a venting mechanism that will allow dispersion of hydrogen and oxygen gases generated as a result of water breakdown within the container.

Videotape on Waste Management Available for Loan

A videotape program, "The Submerged Demineralizer System: Meeting the Waste Management Challenge," is available for loan from the TI&EP without charge. The program documents an entire TMI-2 waste management sequence from waste generation, through processing and shipment, to waste disposition. Presented in a narrative style suitable for a general technical audience, the program discusses the difficulties and successes encountered during development and use of the Submerged Demineralizer System. To borrow a copy of the videotape, contact Kim Haddock, EG&G Idaho, Inc., P.O. Box 88, Middletown, PA 17057. Phone FTS 590-1019 or (717) 948-1019. □

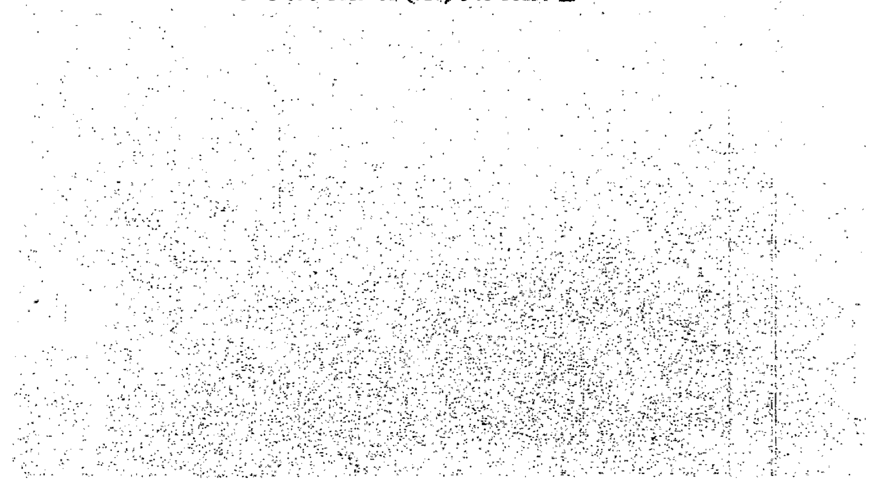


Figure 6. **A new HIC, still wrapped in shipping material, sits on a low-boy trailer next to a nonradioactive EPICOR liner used to practice loading at the INEL.**

Instead of using bolts or mechanical fasteners for the container lid, the lid is sealed with epoxy around its perimeter. The seal successfully passed drop tests conducted by the manufacturer and by INEL engineers: its integrity was unaffected by the impact of the drop. The epoxy seal, after a 48-h curing period, withstands radiation of more than 10^9 R, based on exposure tests. Using the epoxy also minimizes radiation exposure to workers. Surface exposures on the container with an EPICOR liner inside are about 100 R/h on the sides and about 25 R/h on the top.

The workers at the INEL will use special handling fixtures and procedures in the Test Area North Hot Shop to place each EPICOR liner into a container. In Figure 7, a nonradioactive EPICOR liner is lowered into a container during a dry run. A permanent storage site for the liners and containers is still under study, but the containers will be transported in a Chem-Nuclear Systems Incorporated CNSI-14-190 cask that is being built specifically for the project. □

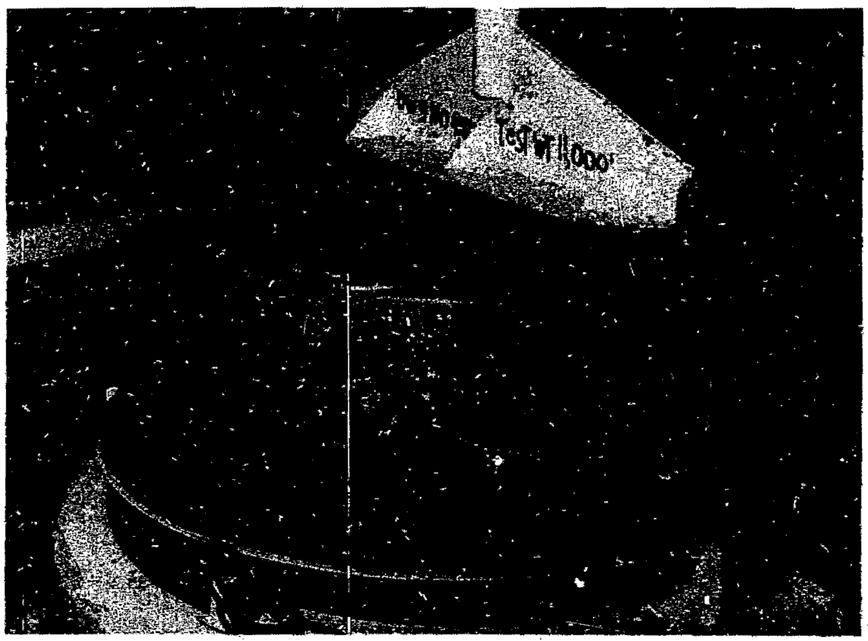


Figure 7. **A nonradioactive EPICOR liner is lowered slowly into the HIC.**

First Demineralizer Resin Sample Results Assist Waste Management Work

The DOE TMI Waste Immobilization Program is working with government laboratories and GPU Nuclear to develop methods for safely removing contaminated resins from the Unit 2 makeup and purification system demineralizers located in the Auxiliary Building of the damaged plant. As reported in the *August 15, 1983,* issue of the *Update,* sampling activities to assess demineralizer conditions began in early 1983. Preliminary results have given waste management engineers new information to assist in planning for resin removal.

In February 1983, gas samples were successfully obtained from both the A and B demineralizer vessels. Analysis of the samples confirmed GPU Nuclear's supposition that both the A and B resins were wet when they were exposed to fission product contamination, although fiberoptic borescope examination of the A demineralizer showed that the A vessel resins are now dry. The borescope examination, conducted in April 1983, showed what may be a crust of boron crystals on the surface of the dry A-vessel resins, but confirmed that the resins in the B vessel are still under about 1 ft of water.

Solution and solid samples were obtained from the B vessel in March and April, and a solid sample was obtained from the A vessel in April. Analysis of resin samples from both the A and B vessels is continuing at the Oak Ridge National Laboratory (ORNL), but selected preliminary results are listed in Table 1. All samples confirm that the fuel content of the demineralizer resins is well below criticality levels, but the presence of plutonium in the resins means they will have to be handled as transuranic wastes. The samples also indicate cesium activity levels that far exceed known values for any other accident-generated waste in the plant. The ^{137}Cs activity ranged from 220 μCi/g in the A vessel solid sample to 16.9 x 10^3 μCi/g in the B vessel solid sample obtained in April. Cesium will have to be removed from the resins before existing plant systems can be used to transfer the resins out of the A and B vessels.

In normal plant operations, demineralizer resins are removed by sluicing them in slurry form through existing sluice piping to spent-resin storage tanks. Tests performed on irradiated resin as part of the TMI research efforts confirm that the resins here are sluiceable. However, the high activity would make normal sluicing a high radiation-exposure task for plant workers. The DOE Waste Immobilization Program, GPU Nuclear, Westinghouse Hanford, and ORNL have developed a two-phase plan to first remove the cesium from the resins, and then sluice the resins from the vessels for packaging.

During Phase 1, ^{137}Cs will be removed from the resins and processed through the plant's Submerged Demineralizer System, a water decontamination process. To accomplish cesium removal, engineers will add water to the vessels to rinse and elute their contents. The resins will be rinsed with borated water and "fluffed" with nitrogen gas, and then the water will

Table 1. **Sample Analysis of Resins in the A and B Demineralizer Vessels**

	March 1983 Sample		April 1983 Sample		
Element	**B Solution (ppm)**	**B Solid (ppm)**	**B Solution (ppm)**	**B Solid (ppm)**	**A Solid (ppm)**
Cs	30	—[a]	30	100	100
Sr	1	—[a]	<1	1	4
U	0.064	1620	0.109	283	1250
Pu	0.72 E-3	3.550	0.64 E-3	0.787	3.520
Isotope	**(μCi/g)**	**(μCi/g)**	**(μCi/g)**	**(μCi/g)**	**(μCi/g)**
Cs-134	0.181 E+3	0.778 E+3	0.101 E+3	1.13 E+3	15
Cs-137	2.64 E+3	11.2 E+3	1.48 E+3	16.9 E+3	220
Sr-90	0.014 E+3	0.49 E+3	9.46	0.85 E+3	200

a. No analysis conducted.

be decanted. Essentially the same operation will take place during elution; however, chemicals such as sodium borate will be added to the flush water to remove additional radioactive cesium from the resin. During both the rinse and elution steps, the flow rate of water through the vessels will be restricted to below 5 gpm, a rate slow enough to ensure that very little of the resin will be carried out with the rinse water. Because even this slow velocity is capable of carrying some resin and fuel particles out with the water, a filter will be installed in the flow path to guard against particle carryover to the Submerged Demineralizer System.

Engineers estimate that about 2000 gal of water will have to flow through each demineralizer vessel before the cesium activity is significantly reduced. The water will be added to the vessels in 300-gal batches, and each vessel will be rinsed three times and then eluted three times. Engineers will feed a batch into a vessel, soak the resins, fluff them, let them settle, and then will decant the water. Because the cesium concentrations are so high, the discharge stream from each vessel will have to be diluted with additional process water immediately after the rinse water leaves the demineralizer cubicle. The entire procedure can be repeated more than three times if it appears that still more cesium could be removed from the resins. Cesium removal will reduce the dose rates both in the demineralizer cubicles and along the sluice path to the spent-resin storage tanks. Removal of the cesium will also minimize the handling problems associated with the packaging of the sluiced resins for shipment.

Once the resins have undergone rinsing and elution, Phase 2 of the removal plan will begin. In this phase, the actual sluicing, packaging, and disposition of the demineralizer resins will occur. While analysis of the ORNL samples continues, the information contained in the preliminary results have allowed waste management planners to develop this two-phase resin removal process. Additional samples will have to be obtained from the dry resin bed in the A vessel before engineers can confirm that the two-phase process developed on the basis of the wet resins in the B vessel will work equally well for the dry A resins. Barring any complications, the resin removal process should begin early in 1984. The removal of the resins from the plant will accomplish another significant milestone in the TMI-2 cleanup. □

TMI-2 Cables and Connectors Under Evaluation

DOE is supporting an effort to determine the effect of the TMI-2 accident on the cables and connectors inside the Reactor Building. Accident effects on both the electrical and material properties of the cables and connectors are being assessed. The components under evaluation include penetration assemblies, terminal boxes, terminal blocks, splices, cables, and connectors. The aim of the cable and connector evaluation program is to determine what impact cable and connector degradation had on the functional capability of instruments in the Reactor Building. DOE's TMI Instrumentation and Electrical Program is being assisted in its characterization efforts by three national laboratories: the INEL, Hanford Engineering Development Laboratory, and Sandia National Laboratories.

TMI-2 contains approximately 1800 instrumentation and electrical channels. In the cables and connectors evaluation program, engineers are reviewing post-accident data to identify channels and penetrations most likely to contain impaired cables or connectors. From these data, candidates for further evaluation are selected.

The instrument channels are first studied remotely using in situ data scanning techniques. Approximately 300 cable channels are undergoing these data scans. Cable channels and penetrations known to have been subjected to such environmental stresses as high temperature, radiation, and moisture are selected for the data scan. Data taken include insulation resistance as a function of voltage and time applied, capacitance and dissipation factors as a function of frequency, loop resistance, inductance, and changes in characteristic impedance as measured by time-domain reflectometry methods.

Several factors are considered when the scan data of instrument channels are being analyzed. Using empirical data on the channels obtained from the cable or connector vendor, empirical data measured on prototype components, and theoretical calculations—including calculations based on computer modeling—engineers learn what they can about the way the channels should operate and what might be causing channel impairment. Analytical characterization also takes into account the effect of the end instrument on the channel; this effect cannot be separated from the cable or connector in the scan data. Also considered in scan data analysis are drawings that locate cable channels according to the environmental stress they are projected to have received. This information is then correlated with the data taken in the scan tests.

By September 1983, 60 channels had been scanned. Of these 60, six showed signs of impairment. Preliminary analysis suggests the causes of impairment may include corroded penetration contacts, water penetration through the bulk cable sheath, and corrosion in spliced regions. In further testing planned on the impaired channels, the exact nature of each cable malfunction will be pinpointed.

Based on information gathered in the in situ data scans, sections of cables, connectors, and other components are identified for laboratory evaluation. If possible, these samples are removed from the Reactor Building, and their electrical and material properties are characterized in investigations at the participating national laboratories. Noticeable changes in material properties of the samples are correlated with impairment in dielectric properties of the sample. The dielectric properties of cable samples are characterized over an appropriate temperature range, and the data are compared with that taken on an experimental control section of cable.

The examination of a particular TMI-2 cable sample, the polar crane pendant cable, offers one example of the kinds of data being gathered as a result of in situ data scanning and correlating laboratory examination of TMI-2 cables and connectors. The polar crane pendant cable was normally used to operate the 500-ton Reactor Building polar crane at TMI-2 from a remote location on the building's 347-ft elevation. During the TMI-2 accident, the cable hung suspended through approximately 50 ft of free space at nearly the center of the Reactor Building; thus, the cable was considered a prime source for studying both radiation levels and hydrogen burn patterns.

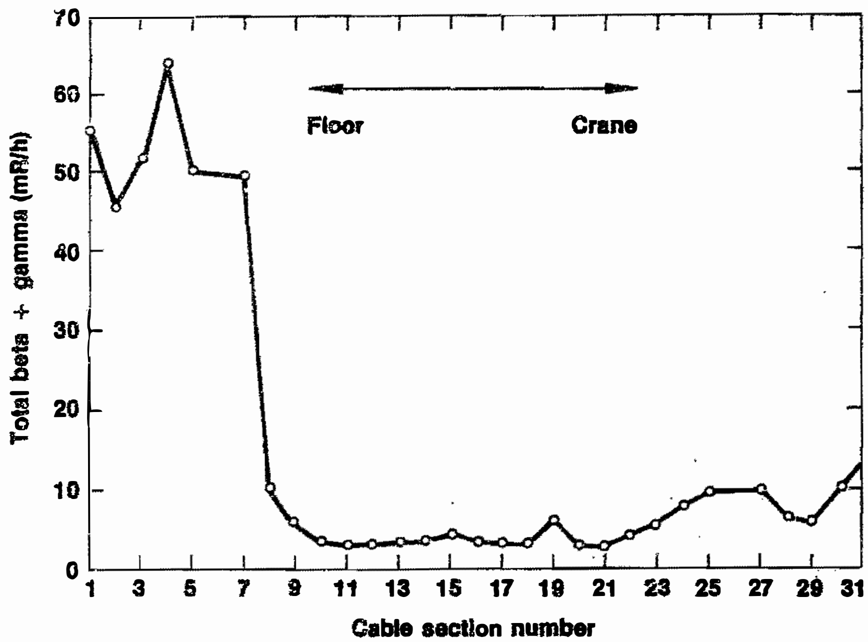


Figure 8. **Radiation levels on the polar crane cable were a function of location along the length of the cable from the crane to the floor. Radiation levels decreased as the distance of the cable from the floor increased.**

Visual inspection of the cable in place in the building provided the only in situ information obtained on the sample. In this visual examination, engineers observed the effects of the hydrogen burn on the cable sheath: greater burn damage occurred at higher elevations. The cable was removed as part of the polar crane refurbishment program, as reported in the related article in this issue of the *Update*. With the cable's building orientations carefully marked, the sample was cut into 30-in. sections. The measured radiation levels from the cable are shown in Figure 8. In this figure, an increasing section number corresponds to increasing elevation in the building. The relatively high radiation levels for lower elevations are attributed to the fact that these cable sections were lying flat in the building, thus exposing more surface area to settling contamination. In all studies of the cable sections, no significant difference in electrical properties was observed between the different cable sections.

The examination of this particular cable sample confirmed studies of both hydrogen burn patterns and radiation levels in the building being conducted in other research programs. It confirmed that the cable's electrical properties were uniform along the entire length of the cable, regardless of extremes of environmental conditions.

Information such as this will continue to be gathered on cable and connector specimens from the TMI-2 Reactor Building. By October 1984, engineers participating in the characterization program hope to be able to define the extent to which cable and connector degradation impaired the functional capability of the instrumentation in TMI-2. By late 1985, results from laboratory tests characterizing the nature of impairment to the cables and connectors will be available for inspection by utilities and by manufacturers of cable and connector equipment, as well as by the Nuclear Regulatory Commission and other standards-setting organizations. □

Polar Crane Refurbishment Complete, First Load Testing Planned

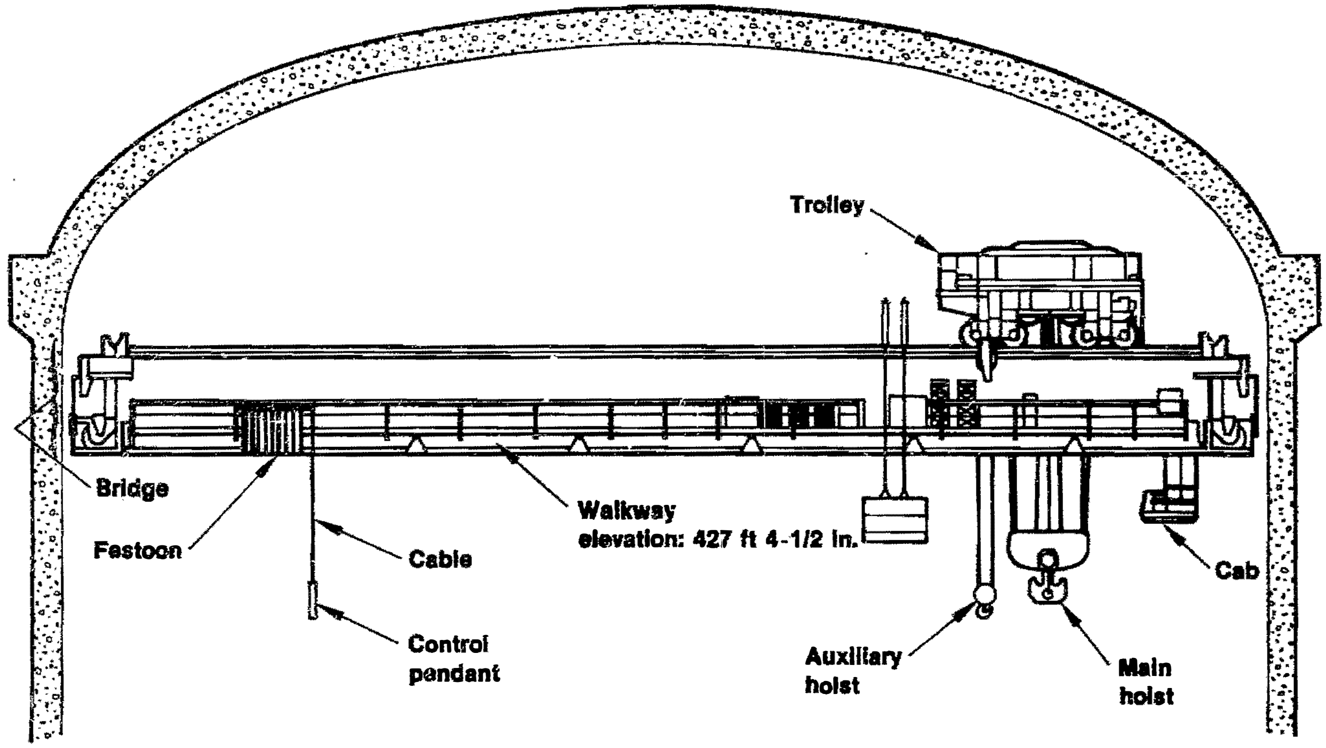


Figure 9. **Major components of the TMI-2 polar crane were refurbished during recent work preparing for reactor vessel head removal.**

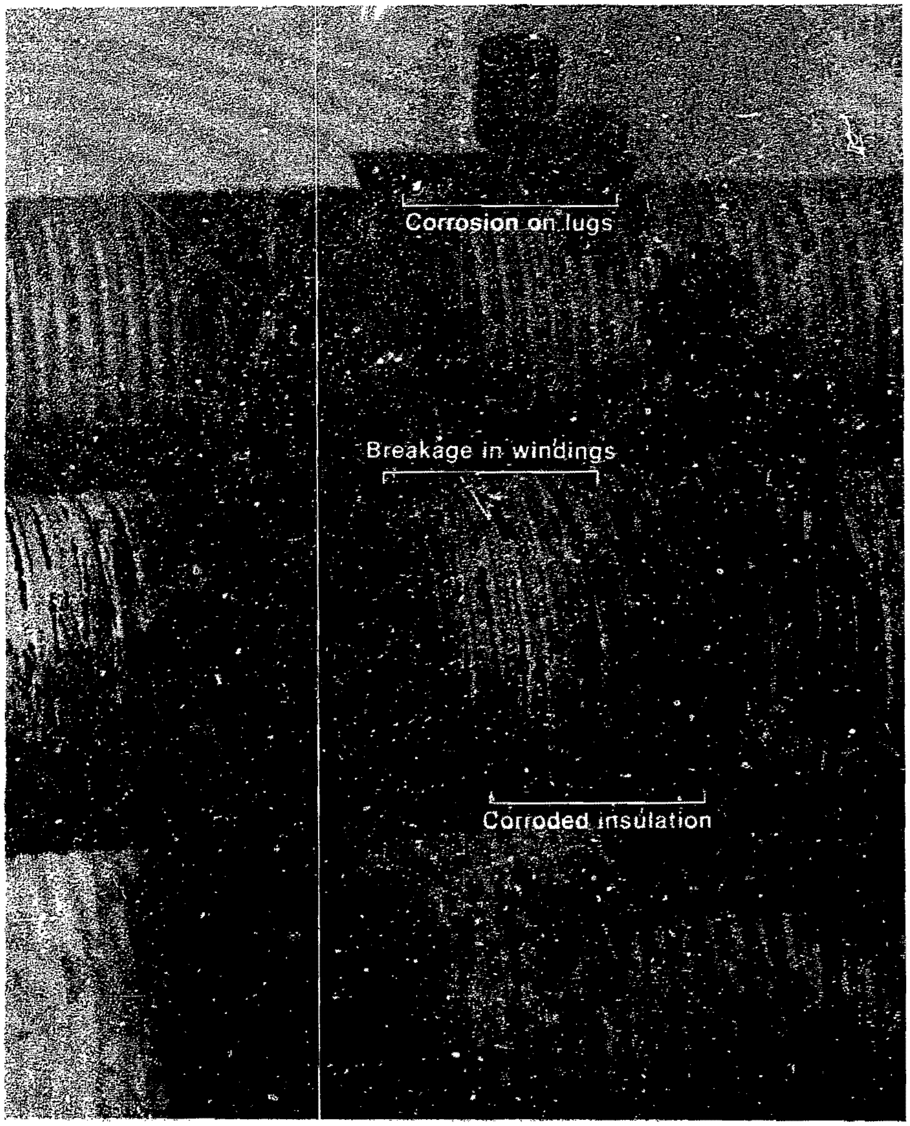
Following nearly a year of inspections, analyses, repairs, replacements, and no-load operational testing, the polar crane at TMI-2 is structurally, mechanically, and electrically ready to undergo load testing—the final stage in preparation for reactor vessel head lift operations. The 500-ton Whiting polar crane installed in the TMI-2 Reactor Building became inoperable as a result of the March 1979 accident. Figure 9 shows the main components.

Because of the polar crane's strategic importance in removing the reactor vessel head, GPU Nuclear and the Polar Crane Task Group decided inspections, refurbishments, and tests should be aimed at restoring functional capabilities of strictly the bridge, trolley, and main hoist mechanisms using pendant control. The industry experts making up the task group agreed that concentrating recovery efforts on these operating functions would help to control costs, save time, and minimize man-rem exposure, while achieving the main objective of the polar crane recovery project: reestablishing those crane motions necessary to move the missile shields and reactor vessel head.

The Polar Crane Task Group included personnel from Bechtel North American Power Corporation, responsible for overall management and implementation of refurbishment activities; consultants from United States Crane, Inc., sponsored by the Electric Power Research Institute and responsible for mechanical component repair work; and DOE-sponsored consultants from United Engineers and Constructors Inc., responsible for electrical component repair work.

Much of the damage to electrical systems and components appeared to be the result of the hydrogen burn. In fact, the power and control conductor-collector system from the crane bridge to the trolley showed extensive damage. In some areas, the supporting insulators fractured, and in other cases, the insulating sheath material had softened enough that the conductor rail clip lost its grip. Consequently, large sections of power-conducting rail were dropped or distorted. Rather than replacing this system with in-kind equipment, which was not economically justifiable and would have unnecessarily exposed personnel to contamination, it was replaced with a new flexible cable loop system. The cables, about 10● ft in length and much like long extension cords, provide power and control interconnections between the bridge and trolley. A similar cable bypasses the crane conductor system to supply three-phase power to the bridge.

Among the electrical equipment that was replaced in kind were 15 electrical relays and contacts in various control cabinets, which were corroded or malfunctioning. Five trolley and bridge accelerating resistor banks, which were also corroded, showed low insulation resistance or had open circuits. In Figure 10, damage to resistor bank insulation and windings can be seen.



The crane control pendant, which was superficially charred due to the hydrogen burn but found in subsequent tests to have retained its functional capabilities, was also replaced, as was the crane festoon, which was totally destroyed by the hydrogen burn. The control pendant, suspended from a cable, hangs at the 347-ft elevation in the Reactor Building and *provides a means to remotely operate the* crane in lieu of operation from the cab. The cable hangs from a trolley system at the walkway handrail. The pendant and cable may be trolleyed back and forth along the walkway by means of the festoon. This issue of the *Update* includes an article on the cable and connector evaluation program that discusses studies of the pendant cable.

While the cable was replaced in kind, the festoon was replaced with three extra-*flexible, flat, 12-conductor cables.* The control pendant was replaced with a lightweight, watertight, neoprene control station, which has all but two of the original control functions; the new unit does not have a warning bell push button or key operated on/off switch because of their relative unimportance to the primary role of the TMI-2 polar crane: head lift.

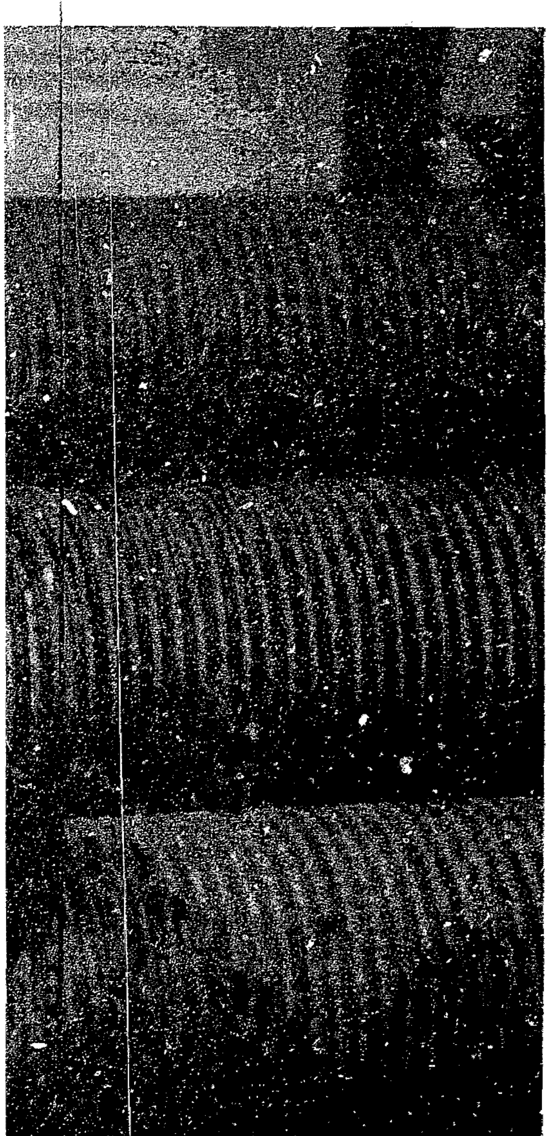


Figure 10. **Insulation corrosion and winding breakage in these polar crane resistor banks caused open circuits and low resistance.**

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

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

Once recovery of the electrical and mechanical components of the bridge, trolley, and main hoist was complete, they were tested without a load to verify their operating capabilities.

The remaining effort in the polar crane recovery project is full load testing of the polar crane. Scheduled for late 1983, the test calls for the hoist to lift about 210 tons so it may be certified at 170 tons—the approximate weight of the reactor vessel head. According to ANSI standards, the load rating of the crane can be no more than 80% of the maximum load the crane lifts during the test. A frame holding five missile shields, with an approximate total weight of 192 tons, as well as cables and associated lifting equipment will provide the weight.

The test load will be lowered, stopped and held by the hoist brakes, and finally lowered to the floor. Then it will be transported 10 ft out and back by the trolley and at least 10 ft from side to side by the bridge. The load will be lifted again after rotating the bridge 180 degrees. After recertification of the polar crane, reactor vessel head removal is scheduled for early 1984. □

Underhead Characterization Supports Reactor Vessel Head Removal

One of the early DOE TMI Reactor Evaluation Program activities involved the formation of a task group to evaluate and determine the best approaches to safely remove the TMI-2 reactor vessel head. This group, known as the Head Removal Task Group, included representatives from Babcock & Wilcox, Bechtel Corporation, EG&G Idaho, Inc., and GPU Nuclear.

Based on the data obtained during the quick look closed-circuit television (CCTV) examination, the axial power shaping rod test, the control rod drive mechanism (CRDM) uncoupling operations, and engineering evaluations, the task group recommended that the head be removed dry—without flooding the adjoining refueling canal. This dry method is essentially the same technique used during normal refueling. The task group also recommended that an alternative wet method, during which the refueling canal is flooded, be available as a backup should higher-than-expected radiation levels be encountered during head lift operations. Based on task group recommendations, the work described below was started to support reactor vessel head removal.

In order to confirm conditions that could be expected during and after head removal, engineers and technicians characterized the environment under the reactor vessel head. Project activities included visual or CCTV observations of the underhead and upper-plenum surfaces, radiation level measurements inside the vessel, and debris samples from the upper-plenum surface.

Table 2. **Quick Scan I Underhead Radiation Levels, December 16, 1982**

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

The first phase of underhead characterization began in December 1982 when an ionization chamber was lowered into the reactor through two access openings that were created when the CRDM lead screws were removed for the quick-looks. This underhead radiation survey, called Quick Scan I, gave engineers radiation level data at a core midradius location (E-9) and at core center location H-8. The detector obtained data, presented in Table 2, from the 327-ft elevation just under the head, down to the 322-ft elevation just above the top of the plenum. Preliminary analyses of these data, performed by GPU Nuclear, indicated that the radiation levels expected during head removal might be higher than originally estimated.

The second phase of underhead characterization, or Quick Scan II, began in August 1983, when the reactor vessel water level was lowered to about 2 ft above the top surface of the plenum. Technicians removed the 800-lb center CRDM to gain access to the space between the dome of the reactor head and the top of the plenum. Once the CRDM was removed, technicians installed a hollow manipulator tube between the CRDM service structure and the top of reactor head. The tube, attached to the CRDM standpipe flange, helped technicians guide instruments and surveillance cameras through the opening into the reactor head.

Because removal of the CRDM increased the size of the access opening through the head and the water level in the vessel was varied, engineers and technicians were able to use larger equipment and conduct more comprehensive studies of the underhead environment. These activities included CCTV examinations of the underside of the head and upper portions of the plenum, ionization chamber radiaton level measurements, thermoluminescent dosimeter (TLD) measurements, and sampling of debris on the plenum's top surface.

During the CCTV inspections, technicians were able to inspect portions of the plenum top surfaces and all the adjacent control rod guide tubes directly below the access opening. By manipulating and extending the camera boom, they were also able to inspect two areas near the outer edge of the plenum. A review group evaluated the video footage of the reactor upper plenum and found no visible evidence of distortion on the plenum assembly. No visible material floated on the water surface, and no distinct piles of debris were seen on the plenum surface around the access location. No mechanical debris or recognizable component pieces were seen.

The camera did reveal a uniform layer of loose, finely divided debris about 1 mm deep, over the area examined. The loose debris appeared to be flushable. The material observed on the inspected surfaces was of four different types. Some of the material consisted of finely divided, dark granules which appeared to be fairly heavy; these granules moved when disturbed by the camera manipulator, but settled quickly. A second type of debris consisted of very small, thin flakes of a light-reflective material; these were easily disturbed by water movement and settled slowly. The third type of material was a light-colored adherent layer deposited on horizontal and vertical surfaces; this did not move unless it was touched by the camera or manipulator. A few large very thin flakes of light material, up to 2.5 cm across, represented the fourth type of substance observed. These flakes were thought to be pieces of the adherent layer that had been dislodged by the camera or manipulator. The cameras, samplers, and other devices used were relatively uncontaminated when removed from the reactor.

After completing the CCTV examinations, technicians obtained two samples of the loose debris on the top surface of the plenum. One sample contained approximately 10 to 15 mg of material and had radiation readings of 900 mR/h gamma and 24 rad/h beta. The other sample contained about 20 to 40 mg of material and had radiation readings of 2 R/h gamma and 60 rad/h beta. Debris on a portion of one sample were tested for pyrophoric reaction by Battelle Pacific Northwest Laboratory's TMI facility. The tests demonstrated that the debris samples posed no pyrophoric hazards.

Ionization chamber radiation readings were made using the same basic technique used during Quick Scan I. The radiation data collected during Quick Scan II are currently being evaluated; however, preliminary reviews indicate that the radiation levels may be slightly lower than estimates based on the Quick Scan I data. In addition, multichip TLD strings were lowered into the reactor to provide an overall radiation level profile and verify ionization chamber measurements.

The reactor vessel water level was then lowered farther to expose the upper surface of the plenum. CCTV observations were again made, and radiation level measurements were obtained with the ionization chamber and TLD string. The review group reconvened to observe the video footage and this time noted that the camera was able to see more than half of the undersurface of the head and more than 10% of the plenum cover surface. Two lanes between guide tubes were inspected to the periphery of the plenum assembly and still no piles of debris were visible. The lighter adherent layer visible in the first inspection seemed to have cracked when it dried out. The white adherent layer appeared to be no more than 2 to 5 mils thick. Many tiny, highly reflective particles, approximately 3 mils in diameter, were visible on the horizontal and vertical surfaces. These may have been droplets of silver.

The underside of the head was clean, although a thin light coating visible on the underhead surface appeared to have flaked off in numerous places. Leadscrew support tubes had localized deposits of light material in many places. Most of the outside support tube surfaces appeared to be clean, but there were localized deposits of light material at the bottom end of several tubes.

The underhead characterization data are providing engineers with valuable information to help determine the best methods to protect workers from radiation exposure during head removal operations. The data will also form the basis for conducting plenum removal and eventual defueling of the damaged Unit 2 core. □

TMI Topics

With this issue of the Update we begin a column of TMI-2 news items of interest to the nuclear power industry, but which may not cover work conducted under the auspices of the DOE TI&EP. The TI&EP Information and Industry Coordination group periodically transmits such news items on Nuclear Notepad, at technical meetings, and through NOMIS, the Nuclear Operations and Maintenance Information Services. Readers desiring additional information on any of the news briefs should contact John Saunders or Jim Flaherty, TI&EP Information and Industry Coordination Group, (717) 948-1043.

TRANSMITTER CORROSION UNDER INVESTIGATION

A reactor coolant, pump-seal, return-flow transmitter was removed from the Reactor Building and sent to the INEL for testing. Although there was no visual evidence of corrosion or degradation on the interior surfaces of the device, transformer coil wires (in enclosed but not hermetically sealed housings) had separated at the solder terminals because of severe corrosion.

PRESSURE TRANSMITTERS DAMAGED BY WATER

Two core flood tank pressure transmitters in the Reactor Building during the TMI-2 accident were removed and examined as part of the DOE's Instrumentation and Electrical Research Program at TMI. The components evaluated are a Bailey Type BY and a Foxboro Model E11GM. The Bailey unit appears to have failed due to water leaking in and shorting the electronics. Water also leaked into the junction box of the Foxboro unit. A cable seal between the junction box and transmitter prevented further degradation. At the time of installation, these instruments were not required to be qualified to withstand a loss-of-coolant accident. However, in order to meet the intent of Regulatory Guide 1.97, Rev. 2, they must be able to survive long-term accident conditions.

It appears that flexible conduit, conduit fittings, and junction boxes may be a common instrument leakage pathway. Possible ways to prevent leakage include improved sealing and installation procedures and the use of drip loops where practical. Additional information on this examination is available in GEND-INF-029, TMI-2 Pressure Transmitter Examination Program Year-End Report: Examination and Evaluation of Pressure Transmitters CF-1-PT1 and CF-2-LT1.

NUCLEAR PLANT EQUIPMENT MAY NEED BETTER PROTECTION FROM WATER

TMI-2 equipment experienced malfunctions caused by water damage. This water may have come from either the Reactor Building spray system on the day of the accident, or from the 100% humidity conditions that existed in the building for nearly two years after the accident. Various methods of avoiding equipment water damage are being investigated. Some possible solutions involve shielding of electrical devices, placing instruments in areas that are not subject to water leakage, and mounting of instruments so that cable connections do not form leak paths.

COOLING SUITS FOR REACTOR BUILDING WORKERS

GPU Nuclear is currently using two types of body cooling devices for Reactor Building worker activities. A passive system, the ice vest, is used when mobility over large areas is required. The "vest" is a cotton garment with a number of pockets in which packets of ice sealed in plastic are placed, and is worn under the worker's protective clothing. The vest will keep the wearer cool for 1-1/2 to 2 h. The second body cooling device is an air-supplied vortex tube cooling system. The vortex suit is used for jobs where the worker will remain relatively stationary for long periods of time. The suit covers the body like a coverall and is fitted with tubing through which forced cool air is fed, cooling the worker. The cool air is supplied to the suit by an air line that can be adjusted by the wearer. Worker mobility is restricted by the air line, but the worker is able to stay cooler for longer periods. GPU Nuclear and the Electric Power Research Institute have investigated a number of body cooling devices and have information available on request relating to reduced heat stress and improved worker efficiency in high-temperature environments. □

B307-1283-3.5M

UPDATE

The TMI Update is issued by the EG&G Idaho, Inc., Configuration and Document Control Center at Three Mile Island Unit 2 under contract DE-AC07-76ID01570 to the U.S. Department of Energy. P.O. Box 88 Middletown, PA 17057. Telephone (717) 948-1050 or FTS 590-1050.

Authors .... G.R. Brown
B.A. Ettinger
C.J. Hess
T.J. Lewis
D.E. Scardena

Editor .... B.A. Ettinger

Production Editor .... G.R. Brown

Designer .... K. Stevens

DOE-TMI Site
Office Manager .... W.W. Bixby

EG&G Idaho TMI-2 Programs
Branch Manager .... H.M. Burton