TMI Unit 2 Technical Information & Examination Program UPDATE
Publication by EG&G Idaho, Inc. for the U.S. Department of Energy

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TMI Unit-2 Technical Information & Examination Program

UPDATE

April 15, 1980

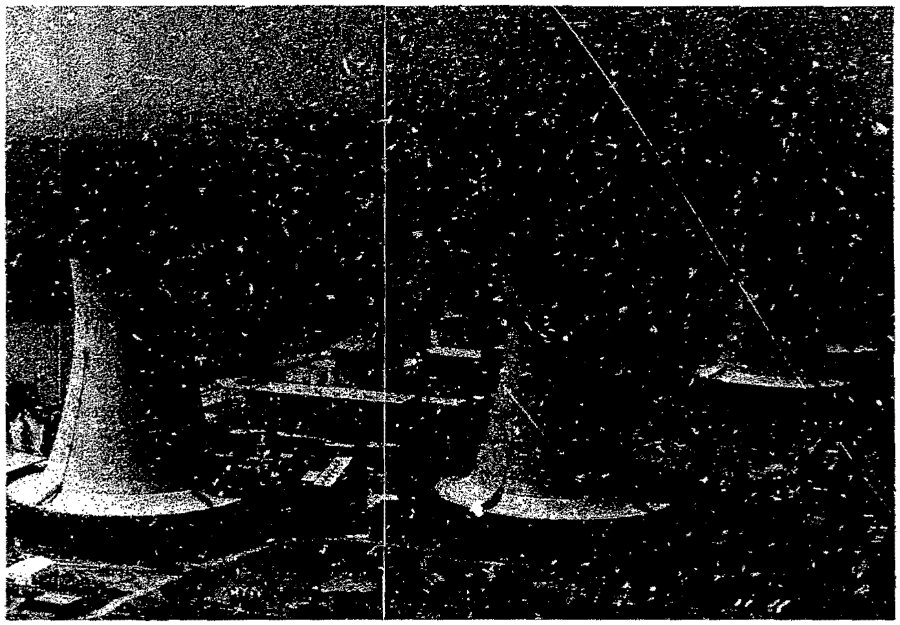
Published by EG&G Idaho, Inc. for the U.S. Department of Energy

TMI Offers Unique Opportunity

Researchers are taking advantage of the unique opportunities offered by the TMI Unit-2 accident that occurred on March 28, 1979. Damage to the reactor core and the release of fission products within the system give researchers the opportunity to:

- measure the performance of instrumentation, electrical, and mechanical equipment within the reactor containment building during and after the accident,
- determine physical damage to surfaces, components, and equipment resulting from radiation exposure,
- assess core damage for metallurgical and physical behavior of fuel, clad, and core components during and after the accident, and
- assess new technological developments for decontamination and the disposal of radioactive waste.

These activities will add to current knowledge on light-water-reactor behavior following accidents involving core damage. The results could lead to improvements in plant safety, reliability, regulation, and operation. Also, the information will benefit those engaged in the design, construction, operation, and maintenance of nuclear power plants.



Three Mile Island—Location of the nation's most severe commercial nuclear power plant accident.

TMI Unit-2 Technical Information and Examination Program Update

This first publication of the *TI & EP Update* introduces the TMI-2 Technical Information and Examination Program.

The *Update* is specifically designed to highlight data and information obtained as a part of the TMI-2 Information and Examination Program. Since this is the initial *Update*, our intent is to provide an introduction of the program. The *Update* will be issued as sufficient data or information is obtained to justify publication. Only summaries will be provided in the *Update*; however, more detailed information will be available in a data bank which is currently under development. In a later *Update*, a procedure for obtaining this information will be outlined. We hope these mechanisms satisfy requirements of all interested individuals and organizations for data and information from this program.

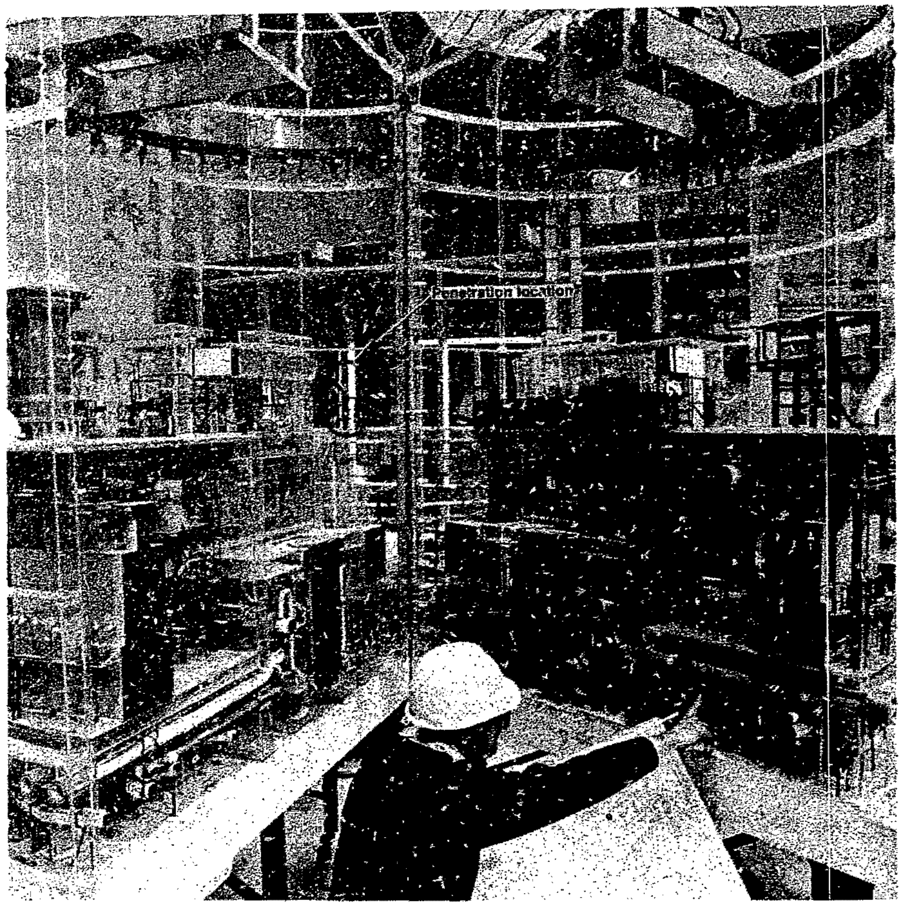
Interested individuals and organizations can obtain a complimentary subscription by filling out the form on the inside pages and mailing it to *TI & EP Update*, EG&G Idaho, Inc., P.O. Box 88, Middletown, PA 17057.

Participants Form Information and Examination Program; Seek Generic Data from Unit-2 Accident

Four groups, with a common interest in obtaining valuable generic information from the TMI Unit-2 accident, jointly established the TMI Unit-2 Information and Examination Program. The Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), the Electric Power Energy Research Institute (EPRI), and the General Public Utilities Company (GPU) signed a coordination agre ment on March 26, 1980, whic documents these common interests.

EG&G Idaho, Inc., has staffed tl Technical Integration Office (TIO which reports to Dr. Willis W. Bixb the DOE Manager of the TMI Si Office. The TIO is responsible for tl day-to-day management of the Infoı mation and Examination Program.

The TIO staff and their respectiv areas of responsibility are as follow: Harold M. Burton, EG&G Prograı Manager; Gregory R. Eidam, Radia tion and Decontamination Technic Coordinator; Robert E. Holzwortl Mechanical Systems and Rad Wast Technical Coordinator; James W Mock, Instrumentation and Electric Systems Technical Coordinatoı Dennis E. Owen, Fuels Technic Coordinator; Frank J. Kocsis, Con figuration and Document Contrc Technical Coordinator; Joseph R Kerscher, Planning, Scheduling, an Budgets Coordinator; Donna L Morris, Material and Contracts Cooı dinator; and Marilyn R. Rehbogen Secretary.



Model of TMI Unit-2 containment building shows penetration location.

Camera, Radiation Probe Explore Containment

Since the accident, the TMI Unit-2 containment building has been dark and inaccessible except through the eye of a small video camera.

On November 10, 1979, a nine-inch diameter hole was drilled through an inner flange of an existing spare penetration (see the photograph above), and a video camera, an associated strobe light, and a radiation probe were inserted into the containment through the opening. During that day, more than two hours of video taping was done. The camera, equipped with a zoom lens and capable of scanning 360 degrees, relayed good quality video tape information, but was limited in range and did not permit inspection of the water surface.

Radiation readings from the installed probe were taken on November 11, 1979. Gamma radiation levels were in the 3 to 5 rem per hour range, and beta radiation levels were in the range of 400 rems per hour.

At present, Metropolitan Edison Company is documenting the results and conclusions from the review of the tapes. Initial reviews do not show any structural damage. Final evaluation is forthcoming and preparations are being made for initial entry into the containment.

TMI Unit-2 Technical Information & Examination Program

UPDATE

Published by the Technical Integration Office, Configuration Document Control Section, EG&G Idaho, Inc. under Contract DE-AC07-76IDO1570 to the U.S. Department of Energy, P.O. Box 88, Middletown, PA 17057. Phone (717) 948-8486—FTS 590-3933.

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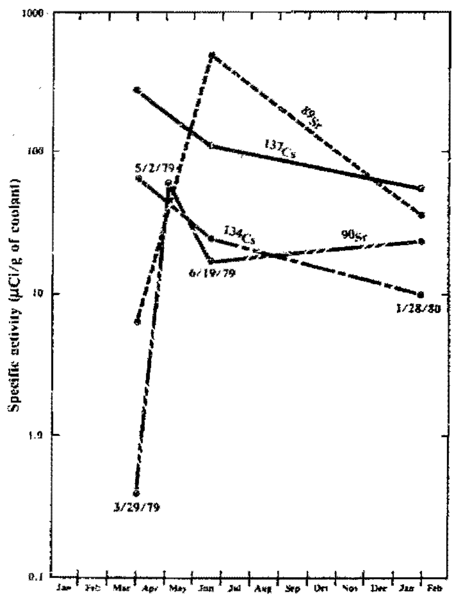
DOE-TMI Site Office

B&W Samples Reactor Building

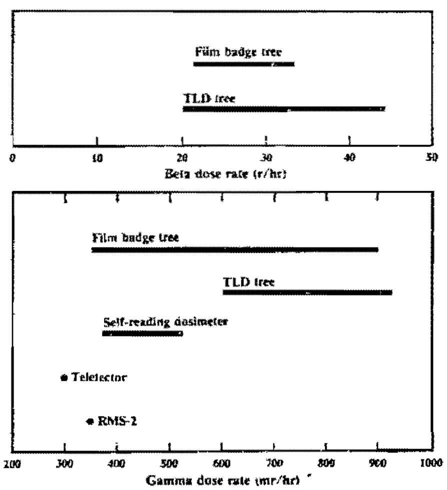
Reactor coolant has been sampled regularly since the TMI Unit-2 accident and then analyzed by Babcock & Wilcox for specific radioisotope activity. Data collected from the samples will be used in the fission product transportation and deposition task, part of the Technical Information and Examination Program. The graph below shows some sample results.

Babcock & Wilcox analyzed other samples in the reactor building such as the sump on October 20, 1979 (see the table to the right), and the air on February 13, 1980. The sample from the reactor building air documented specific radioisotope concentrations (e.g., ^{85}Kr activity of 1 µCi/ml, ^{134}Cs activity of < 7 x 10^{-6} µCi/ml, ^{137}Cs activity of < 3.2 x 10^{-5} µCi/ml).



TMI-2 reactor coolant samples reveal radioisotope concentrations.

The owner of TMI Unit-2, the General Public Utilities Company, measured the radiation in the reactor building on December 14, 1979, as one of the many preparatory steps for entry into the reactor, and to provide basic planning information for subsequent decontamination efforts. The measurements were performed through a shaft called Penetration R-626, using various instruments (see the chart below). The calculated dose rate to the skin, based on the observed beta dose in the building, lies within a range of 100 to 350 rad/hr.



Different instruments show dose rates inside the TMI-2 reactor building.

Reactor Building Sump Sample Analysis Results

Analysis		Result
Unfiltered:		
^{137}Cs	(µCi/g solution)	136
^{134}Cs	(µCi/g solution)	27
Filtrate:		
Na (ppm)		1250 ± 100
Cl (ppm)		10 ± 2
B (ppm)		1690 ± 40
pH		8.6 ± 0.2
^{90}Sr	(µCi/g solution)	4.8 ± 1.2
^{137}Cs	(µCi/g solution)	135
^{134}Cs	(µCi/g solution)	26
^{3}H	(µCi/g solution)	0.92
Gross Alpha	(µCi/g solution)	<1 X 10^{-6}
Gross Beta	(µCi/g solution)	149
Sr-89	(µCi/g solution)	37 ± 4
Filterable Solid (µCi/g solution):		
^{137}Cs		0.2
^{134}Cs		0.03
^{103}Ru		3.0 X 10^{-3}
^{140}La		8.0 X 10^{-3}
^{144}Ca		3.0 X 10^{-3}
^{95}Zr		1.0 X 10^{-3}
^{95}Nb		4.0 X 10^{-3}
^{54}Mn		7.0 X 10^{-5}

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Oak Ridge Analyzes "Cookie" from Containment Building

A disc (cookie) was cut from a shaft called Penetration R-626 in the TMI Unit-2 reactor containment building. Oak Ridge National Laboratory analyzed the 9-in. Type-304 "cookie" made of stainless steel.

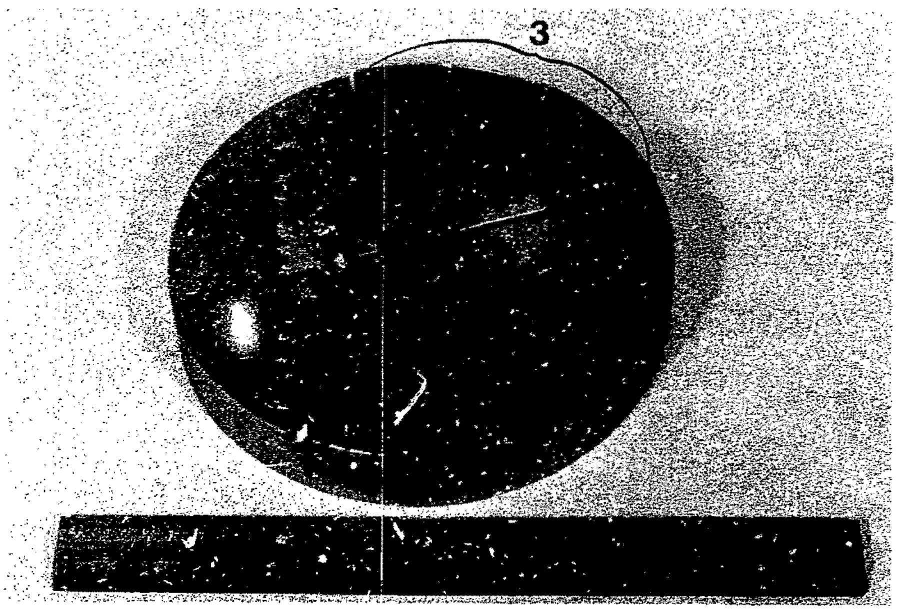
The test results indicated that significant amounts of surface contamination may remain following the decontamination process; however, the decontamination method described below reduced the background radiation levels due to surface contamination to about 1 to 2 mr/hr beta-gamma.

When Oak Ridge received the disc, the initial radiation readings were 80 mr/hr beta-gamma and 6 mr/hr gamma at 2 in. from the disc surface. See the table at the right for the analysis results.

The disc was cut into sections (refer to the photograph) for decontamination tests. The standard Bechtel Corporation Specification CP-952 decontamination series removed approximately 98% of the contamination from piece 3c. Wiping with dry cheesecloth removed approximately 38% of the activity from piece 3a, while wiping with wet cheesecloth removed 17% of the activity from piece 3b. The apparent inconsistency between the wet- and dry-cheesecloth methods may be due to nonuniform contamination levels on the disc surface.

Penetration R-626 Surface Contamination Results

Isotope	Total Activity on Disc (in µCi)	Average Contamination Level on Disc (in µCi/cm^2)
^{60}Co	0.019	6.09×10^{-5}
^{134}Cs	2.68	8.4×10^{-3}
^{137}Cs	12.7	4.0×10^{-2}



Penetration R-626 "cookie" is sectioned for decontamination tests.

TMI Unit-2 Technical Information & Examination Program
UPDATE
EG&G Idaho, Inc. • P.O. Box 88
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TMI Unit-2 Technical Information & Examination Program

July 31, 1980

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Containment Airlock Door Freed

The **TI&EP Update** *is specifically designed to highlight data and information obtained as part of the TMI-2 Technical Information and Examination Program (TI&EP). As space permits, the* **TI&EP Update** *may feature certain TMI-related articles which, though not part of the TI&EP, would be of general interest to the scientific community.*

A small pin, which acts as a safety device, apparently caused the malfunction of the TMI Unit-2 containment door locking mechanism that prevented entry into the containment. On May 20, a two-man entry team spent 13 minutes trying to turn the containment door locking wheel before halting the effort (see photograph). TMI officials report that the pin has been freed, and the locking mechanism now appears to be operating properly.

Following extensive evaluation, a small hole was drilled into the containment door behind the pin, which freed the pin and allowed it to return to its normal position. TMI officials believe that corrosion may have frozen the locking mechanism.

Proper operation of the locking mechanism has since been verified, and the containment was entered on July 23. Information obtained during the containment purge and early entry will be reported in a future issue of the *TI&EP Update*. TMI officials stress that the airlock door can still be shut and sealed.

TMI Unit-2 containment door-locking mechanism malfunction aborted first entry attempt

Groups Use TI&EP to Gather Unit-2 Generic Information

Four groups organized the TMI Unit-2 Technical Information and Examination Program (TI&EP) to gather valuable generic information about the Unit-2 accident. The four — the U.S. Department of Energy (DOE), the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and General Public Utilities Corporation (GPU) — compose the Coordination Group for the program.

EG&G Idaho, Inc., staffed the Technical Integration Office (TIO) for day-to-day management of the

Continued on following page

TI&EP Establishes Data Bank For Nuclear Community

The Three Mile Island (TMI) Unit-2 Technical Information and Examination Program (TI&EP) is establishing a data bank of material related to the TMI-2 accident. The data bank will include data, analytical reports, and design review documents produced since the accident. The information will be stored on the Zytron computer system at the Electric Power Research Institute Nuclear Safety Analysis Center at Palo Alto, California. The data bank information will be available to program participants and others and will benefit the entire nuclear community by enhancing nuclear plant safety and reliability. Initially, all information retrieval will be through the Technical Integration Office. Information distribution will be done on microfiche. Future issues of the *TI&EP Update* will include instructions for information retrieval and will list both new documents acquired and data developed.

Any information you may have which could be a useful input to the data bank should be sent to the *TI&EP Update*, EG&G Idaho, Inc., PO Box 88, Middletown, PA 17057. Any questions concerning the type of information needed should be directed to Frank Kocsis, Configuration and Document Control Coordinator, phone number (717) 948-8486, FTS number 590-3933.

Unit-2 Generic Information

Continued from page 1

program under contract to DOE. EG&G officials recently signed a contract with GPU to act as the interface between GPU and program agencies for work in the information and examination program. TIO will schedule work to be done and compile necessary documentation.

NRC licensing, inspection, and enforcement activities are unaffected by the contract. GPU and its contractors will perform all work within the Unit-2 facilities.

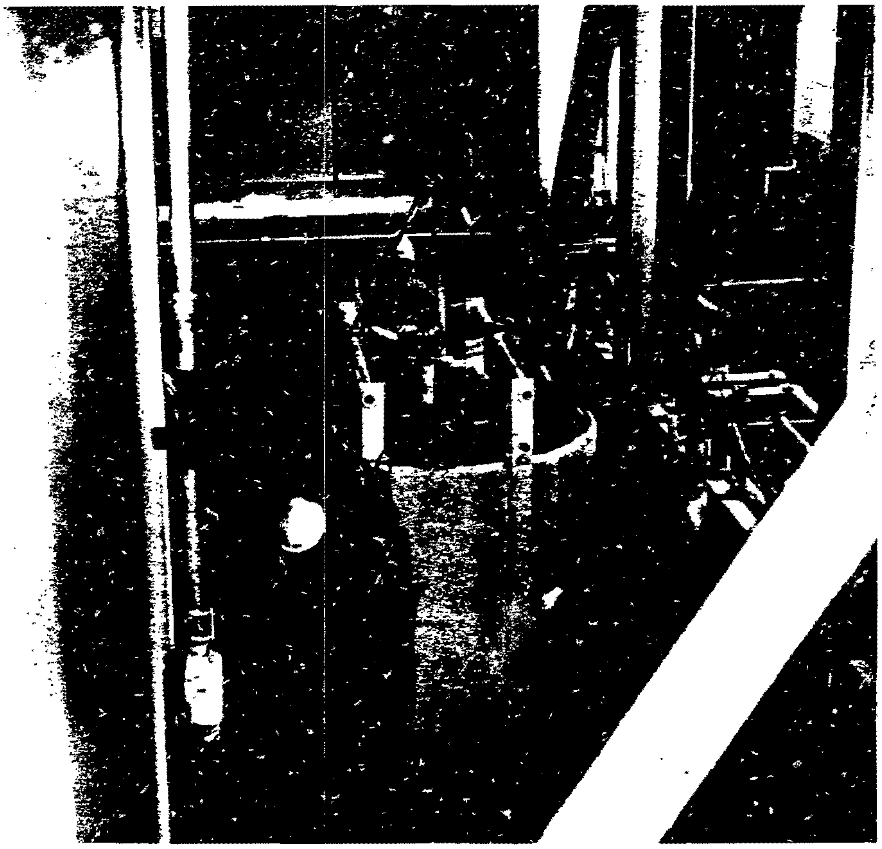
Analyses Completed on Containment Air Sample

EXXON Nuclear, EG&G, and GPU scientists recently completed analyses on air samples drawn from the Unit-2 containment building. The samples were taken in April, 1980, to provide data requested by the TMI Working Group. The samples were taken using a glove box and sampling apparatus installed in containment penetration 626. The analyses were completed on June 27, 1980. The results are presented in the table below.

Air sample analysis results

Isotope	Activity[a] (in microcuries per cubic centimeter)
tritium	$5 \pm 1 \times 10^{-5}$
carbon-14	$4 \pm 1 \times 10^{-7}$
iron-55	less than 6×10^{-11}
cobalt-58	less than 1×10^{-11}
cobalt-60	less than 1×10^{-11}
krypton-85	0.93 ± 0.07
strontium-89	$1.1 \pm 0.5 \times 10^{-10}$
strontium-90	$2.2 \pm 0.2 \times 10^{-10}$
ruthenium-103	less than 2×10^{-9}
ruthenium-106	less than 2×10^{-10}
iodine-129	$6 \pm 2 \times 10^{-11}$
cesium-134	$1.7 \pm 0.1 \times 10^{-10}$
cesium-137	$9.3 \pm 0.3 \times 10^{-10}$
uranium-235	less than 5×10^{-12}
uranium-238	less than 2×10^{-11}
plutonium-238	less than 8×10^{-12}
plutonium-239 and -240	less than 2×10^{-12}

(a) *Less than indicates below detectable limits for the analytical techniques available.*



Workmen changing EPICOR-II resin cask.

EPICOR-II Cleaning Waste Water

More than 365,000 gallons of contaminated water have passed through the EPICOR-II water treatment system as the first major step in cleaning up the Unit-2 facility at Three Mile Island.

The processing through a system of three large resin casks began last October with water from holding tanks in the auxiliary building. About 125,000 gallons remain to be processed.

The EPICOR-II system uses two filtering and demineralizing casks, each four feet in diameter by four feet high, and a final polishing cask that is six feet in diameter by six feet high. The water is decontaminated by filtration and ion exchange to extract strontium and cesium.

The resin casks are housed in a chemical cleansing building. The water moves into the system via shielded lines from the auxiliary building. If necessary, water can be cycled through the system a second time for additional purification.

A typical processing run handles about 17,000 gallons of water before the resin casks require changing (see photograph). Water samples are taken about every 1500 gallons at points before and after each filter. The processing rate averages 10 gallons a minute.

The water after processing is considered releasable under Environmental Protection Agency regulations, but is being stored in tanks on the island until a programmatic environmental impact statement is prepared. The spent resin casks also are stored in a shielded facility on the island.

TMI Unit-2 Technical Information & Examination Program

The *TI&EP Update* is issued by the EG&G Idaho, Inc., Technical Integration Office, Configuration Document Control Section, under contract DE-AM07-76ID01570 to the Department of Energy, PO Box 88, Middletown, PA 17057. Telephones (717)948-8586 or FTS 590-3933. W. W. Bixby is manager of the DOE-TMI Site Office. H. M. Burton is manager of the Technical Integration Office. D. M. Grigg is managing editor of the *TI&EP Update*.

Techniques Being Investigated for Early Core Damage Assessment

Insertion of an underwater television camera through a control rod drive mechanism nozzle may provide the first visual assessment of core damage in the TMI Unit-2 reactor vessel. The camera insertion is part of the potential early core damage assessment before the reactor vessel tophead is removed, according to Dennis E. Owen, Fuels Technical Coordinator of the Technical Integration Office, and George Kulynch of Babcock & Wilcox.

The core damage assessment may use three different approaches:

- Visual inspection of the core
- Temperature and flux mapping of the core
- Damage mapping of the core.

The visual inspection may use control rod drive mechanism (CRDM) and thermocouple penetrations in the reactor vessel head. Following removal of a CRDM, a radiation-hardened, underwater television camera may be inserted through the 2.5-inch-diameter opening in the CRDM nozzles, thus permitting visual inspection of the tops of the fuel elements. Visual examination of the peripheral areas of the core may be accomplished by inserting a borescope through some of the eight thermocouple penetrations. The thermocouple nozzles are on the outer perimeter of the vessel, where it is expected that the fuel may still be intact.

While the CRDM is removed, engineers could insert tooling to try to extract samples of core debris and determine whether any slumping of debris has occurred. Owen said the samples would be taken to a remote-handling hot cell for analysis. Analysis of samples would aid in planning the reactor core removal.

The temperature and flux mapping phase of the assessment could make use of instrument strings that run inside the reactor core. A small-diameter, swaged tube inside the strings, usually used for flux wires, provides a path for thermocouple insertion. The thermocouples would allow temperature measurements and locate physical blockages that might indicate coolant flow blockages within the core. Flux wires also are being considered for use in providing information about fuel distribution in the subcritical areas of the core.

The damage mapping phase of the assessment makes use of the instrument string guide tubes following withdrawl of the instrument string for analysis of the self-powered neutron detectors and other components. Both gamma and neutron detectors are under consideration for use in the instrument guide tubes to provide both radial and axial maps of the extent of reactor core damage, Owen said.

Additional detectors are being considered for insertion in the instrument guide tubes to check fuel redistribution, the extent of oxidation within the core, and mechanical strength of the core materials.

Local Residents Monitor Environmental Radiation

Each day in 12 Pennsylvania communities within a 5-mile radius of TMI Unit-2, specially trained residents take readings from radiation monitoring equipment located in municipal garages, firehouses, and sheds. These readings are part of the Citizens' Radiation Monitoring Program, a joint project of the U.S. Department of Energy (DOE), the Pennsylvania Department of Environmental Resources (DER), the Environmental Protection Agency (EPA), and Pennsylvania State University. Readings taken since May 15 show normal background radiation levels, according to G. R. Eidam, Radiation and Decontamination Project Coordinator for the Technical Integration Office.

Each day after the equipment readings are recorded, a DER employee collects the data for compilation. DER distributes the data each weekday to the General Public Utilities Corporation and local officials of the Nuclear Regulatory Commission, DOE, and EPA. The Pennsylvania governor's office releases the data to the news media.

Circumstances can alter the reading

Continued

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schedules, however. While krypton-85 was being purged from the TMI Unit-2 containment building, some communities chose to monitor the instruments either continuously or on an hourly basis. Residents of the communities can view and read the instruments at any time.

The program began early this year when DOE and DER representatives visited officials of 12 communities and explained the proposed monitoring concept. From a list of 50 people supplied by community officials, course organizers enrolled residents to attend 36 hours of radiation monitoring training presented by Pennsylvania State University faculty members over a 2-1/2-week period.

The course included a day of training at the Breazeale Nuclear Reactor Facility at Pennsylvania State University at State College, where the residents learned to take readings of argon-41 with the same equipment they would later use in their communities. They also participated in the calibration of the monitoring instruments, using a krypton-85 source inside a plastic tent.

The monitoring equipment includes a Ludlum 177 radiation monitor with an Eberline HP-260 two-inch-diameter probe, a Rustrak recorder, and a Lear Siegler Inc. gamma rate recorder. Eidam said the equipment selections were based on sensitivity for detecting beta-emitting radionuclides (i.e., krypton-85) and durability.

TMI Unit-2 Technical Information & Examination Program

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TMI Unit-2 Technical Information & Examination Program

UPDATE

October 29, 1980

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Third Successful Containment Entry Completed

This *TI&EP Update* highlights the venting of the containment and the first three containment entries at TMI Unit-2. This information, although not entirely a part of the technical information and examination program, is considered of general interest to the scientific community.

Five men completed the third successful entry into the Unit-2 containment building on October 16, 1980. They were able to stay longer than planned because radiation levels inside the building were lower than anticipated. The team completed all tasks planned for the entry.

The team members were Sam Griffith, 28, a health physics technician with Nuclear Support Services Inc.; Larry E. Eberly, 44, an instrument and control technician with Metropolitan Edison Company (Met-Ed); Guy E. Wise, 45, a Met-Ed machinist; Richard Croll, 28, a radiation-chemical technician with Met-Ed; and Peter Keegan, 27, a Met-Ed senior health physics technician. Griffith was also a member of the team that performed the second containment entry (see article on inside pages).

During the entry, Wise and Croll repaired the locking mechanism of a personnel airlock that is part of the equipment hatch (refer to location maps on the inside pages). The doors to this airlock have been shut since the incident on March 28, 1979. Other team members completed a radiological survey of core flood tanks, performed maintenance on two monitors that keep operators apprised of certain plant systems, and conducted a visual survey of the polar crane, the device used to lift the reactor head during refueling.

Also completed were five tasks supported by the Technical Integration Office (TIO). These were removal of a source range neutron detector preamplifier; removal of a spare parts monitor preamplifier; photographing areas of interest identified during previous entries; surveying radiation levels in the area around HP-RT-211, the radiation detector removed during the second entry; and removal of a

Continued on page 2

An entry team member takes beta radiation surveys just south of the enclosed stairway. Behind him are the ventilation ducts. Story on containment entry on inside pages.

Containment Entry Completed

Continued from page 1

section of cable that was connected to HP-RT-211.

The team recorded radiation readings of between 200 and 500 millirem (mrem) per hour on the 305-foot elevation, or entry level, and an average of 150 mrem per hour on the 347-foot elevation, or operating floor.

Wise and Eberly left the building after the first hour, staying twice as long as was planned. Keegan left with them when a camera malfunctioned. He was scheduled to join Croll and Griffith, who were inside for 90 minutes (30 minutes longer than scheduled).

Actual radiation doses to the team members were well below the 625 mrem limit set for the entry; they ranged from 200 to just over 450 mrem. By comparison, the company quarterly limit is 1250 mrem and the federal quarterly limit is 3000 mrem.

During the entry, the team wore cotton overalls instead of the heavier fireman's coats worn during the first two entries. They also wore battery-powered air filtration devices with positive-flow air masks, rather than the oxygen tanks worn during the first entry.

The entry preparations and control center activities were videotaped by the TIO. These tapes will be narrated by the entry team manager and will be available for training critiques and management briefing.

TMI Unit-2 Technical Information & Examination Program

UPDATE

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W.W. Bixby is manager of the DOE-TMI Site Office. H.M. Burton is manager of the Technical Integration Office. D.M. Grigg is managing editor of the *TI&EP Update*.

Operating Floor Radiation Measurements Taken During Second Entry

The entry team discovered extensive rusting on all of the metal around the power track.

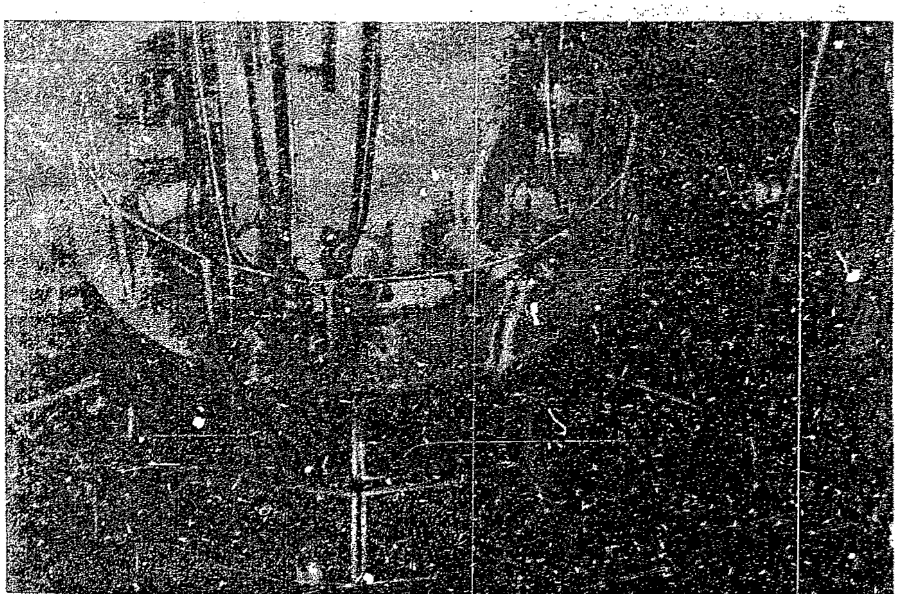
On August 15, 1980, a four-man team made the second entry into the TMI Unit-2 containment and visited the 347-foot level, or operating floor, for the first time since the March 28, 1979 incident. While there, they obtained radiation readings of 100 to 500 millirem per hour (mrem/hr). (See accompanying location maps.) The team members were Martin Cooper, William H. Behrle III, Sam Griffith, and Michael Benson. Benson and Behrle made the first containment entry on July 23, 1980.

The major priority tasks to be conducted were lighting both the 305-foot and 347-foot levels, surveying for radiation and surface contamination, and photographing the containment interior. Other tasks to be performed included protective covering, directional dose, and surface decontamination experiments and retrieval of selected items from within the containment for subsequent analysis.

After turning on the 305-foot and 347-foot level lights, the team obtained radiation readings and surface contamination samples in areas of the 305-foot level not surveyed during the first entry. Radiation readings on the 305-foot level included: north of the open stairwell, 2 rem/hr (location A); five to seven feet from the sump water (using a teletector), 40 to 45 rem/hr (location B); and at contact with the floor drain near the "A" core flood tank, 10 rad/hr and 3 rem/hr gamma (location C).

Floor swipes taken from the 305-foot level showed cesium-134 and cesium-137 concentrations of 6.6×10^{-2} and 4.07×10^{-1} microcuries per square centimeter ($\mu Ci/cm^2$) respectively under radiation monitor HP-RT-211 (location D) and of 3.8×10^{-4} and 2.3×10^{-3} $\mu Ci/cm^2$ respectively under the air coolers (location E). Scrape samples taken from the 305-foot level showed cesium-134 and cesium-137 sample activities of 8.8×10^{-1} and 5.25 μCi respectively near the open stairwell (location F) and of 2.6 and 16.1 μCi respectively near the air coolers (location G).

Radiation surveys taken as the teams moved up the enclosed stairwell (location H) showed an approximately



The walls of the transfer canal are clean. The shield tanks around the reactor vessel head are dry. On the reactor head, the cooling fans and associated electrical leads appear to be clean.

and water stains on equipment and floors of the 347-foot level, describing the conditions as similar to those found on the 305-foot level during the first entry. Officials said no significant structural damage was seen; however, elevated temperatures had partially melted a telephone housing, plastic rope, and some yellow plastic sheeting. Behrle reported seeing pieces of what appeared to be wood floating in the dark sump water that filled the containment below the ground level. An estimated 700,000 gallons of contaminated water are believed to be in the sump.

Experiments conducted by the teams included placing two trees of thermoluminescent dosimeters in the containment for protective covering and directional dose tests and wiping a portion of the 305-foot level floor

Continued on page 4

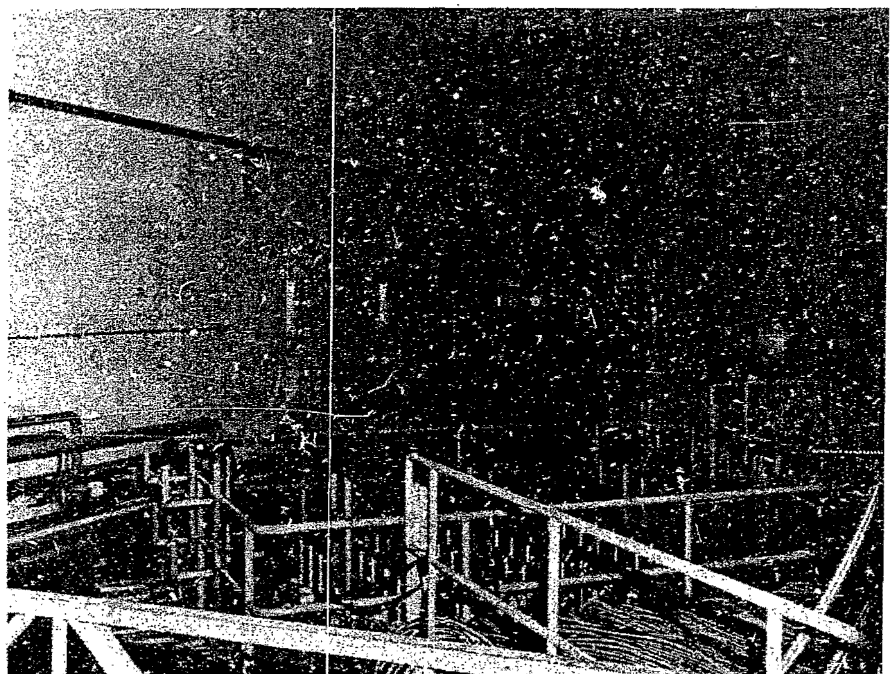
linear decrease from 3 to 5 rem/hr on the 305-foot level to 180 mrem/hr on the 347-foot level.

Radiation surveys on the 347-foot level revealed 600 mrem/hr at the decking outside the enclosed stairwell (location I) and 100 mrem/hr along the south containment wall (location J). Southeast of the head storage stand (location K), the readings increased to 400 mrem/hr. Other radiation levels measured on the 347-foot level included: fuel handling bridge (location L), 100 to 400 mrem/hr; 15 feet from the reactor head and stud bolts (location M), 125 mrem/hr; pressurizer spray line (location N), 2.5 rem/hr; over core flood tanks (locations O and P), 250 to 300 mrem/hr; and behind the enclosed stairwell (location Q), 50 mrem/hr.

Swipes taken on the 347-foot level yielded average cesium-134 and cesium-137 concentrations of 9.0×10^{-3} and 5.6×10^{-2} $\mu Ci/cm^2$ respectively on the floor and of 2.5×10^{-5} and 1.5×10^{-4} $\mu Ci/cm^2$ respectively on the walls. Strontium-90 concentrations were found to be 3.1×10^{-5} $\mu Ci/cm^2$ or less on the floor.

The teams took 67 photographs during the entry. The photographic survey on the 305-foot level showed more details of items identified from the first containment entry on July 23, 1980, and on the 347-foot level, it showed the general areas and structures of the operating deck, fuel handling bridges, D-rings, seal table, and reactor vessel head.

The team members reported rust



A view of the grill plate on top of the control rod drive mechanisms. The entry team reported it was dry and clean with no debris on it.

Gamma Exposures to Second Entry Team

Team member	Whole body dose (mrem)	Maximum extremity dose (mrem)
Cooper	140	210
Behrle	260	320
Benson	300	420
Griffith	165	270

Second Entry

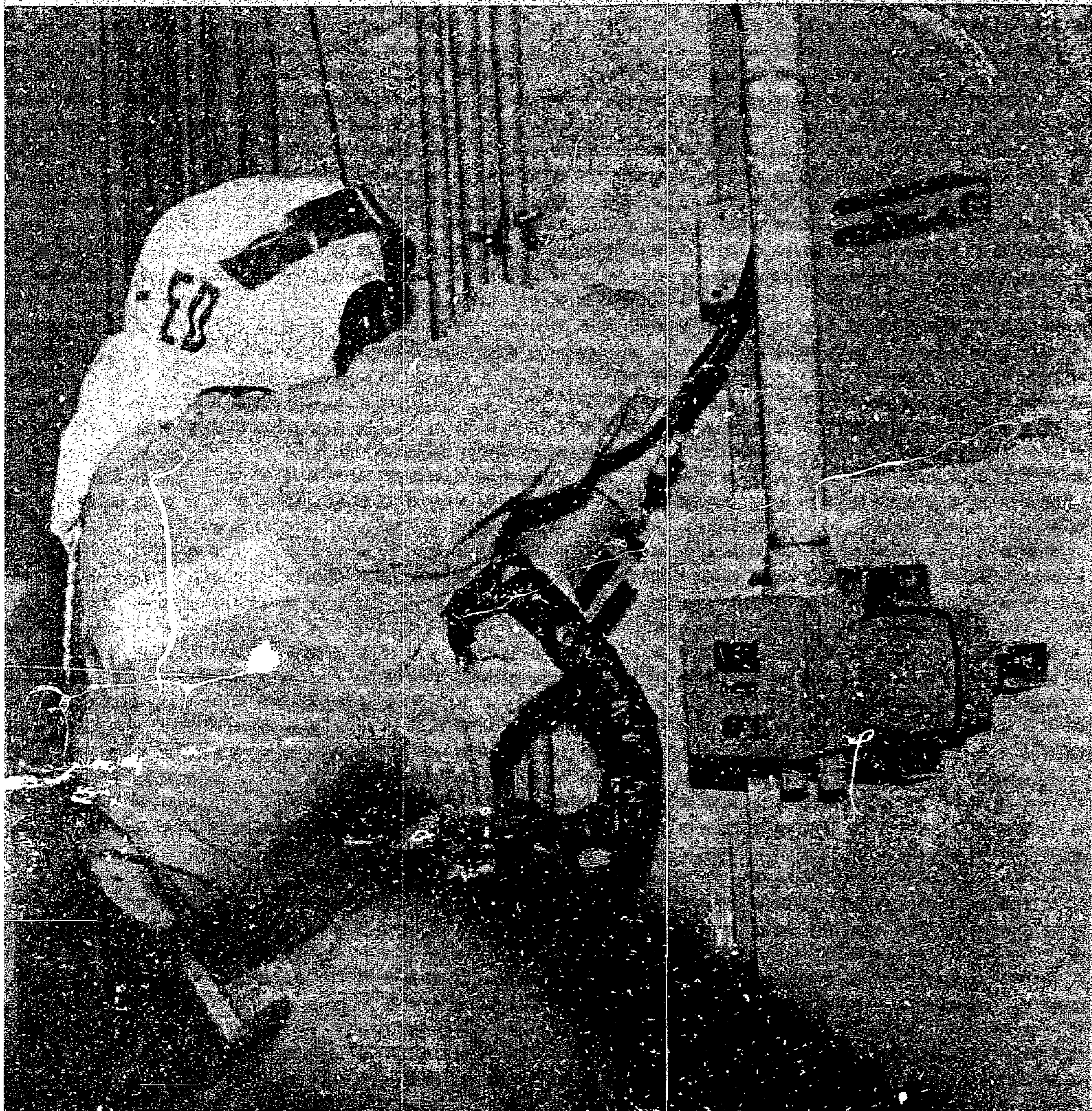
Continued from page 3

with a Masilin wipe for decontamination testing (location R). Swipes taken before and after the decontamination test showed the wipe removed approximately 90 percent of the loose surface contamination.

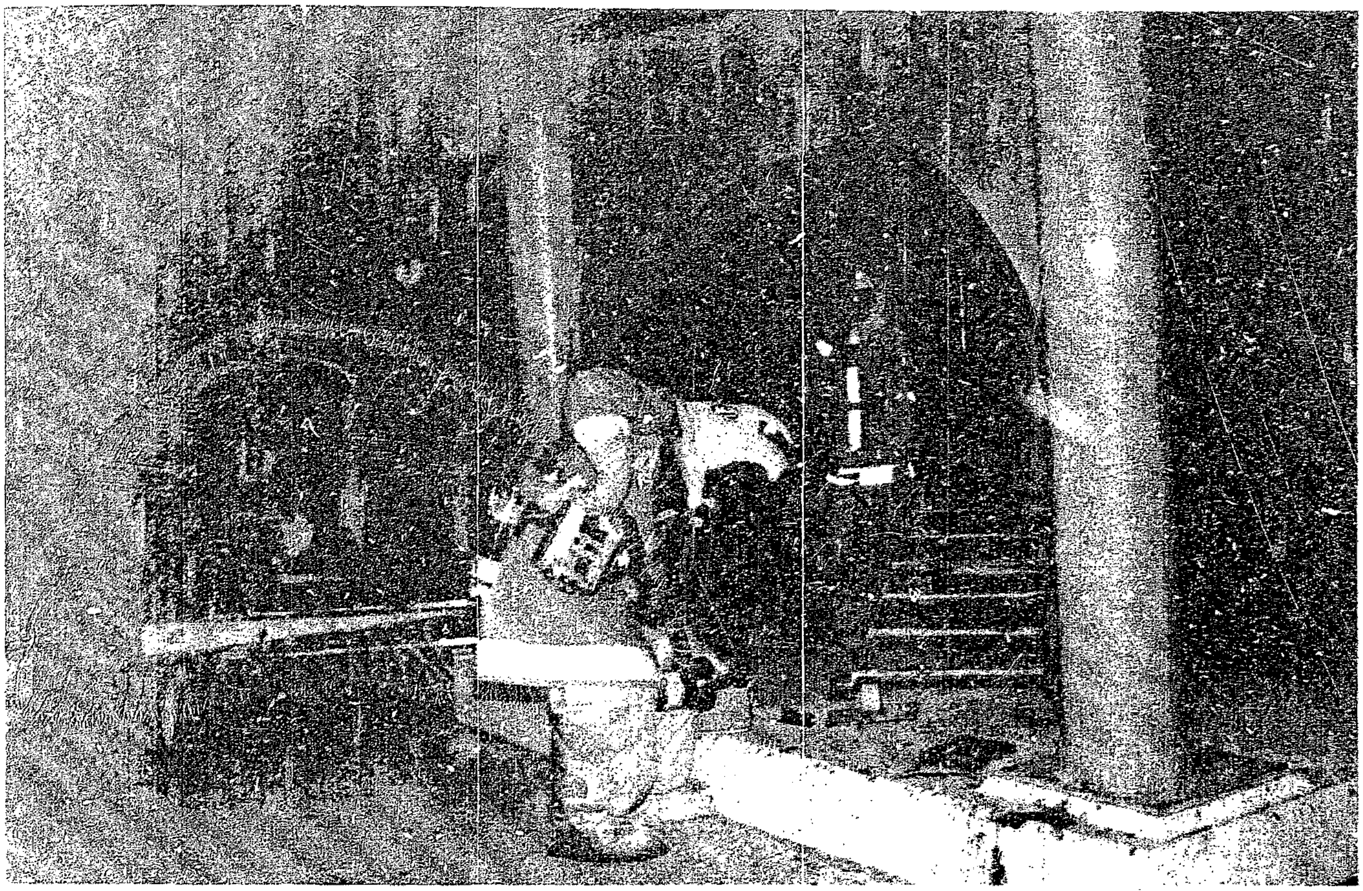
Items retrieved from the containment for subsequent analysis included a radiation detector (location D), a piece of glass (location S), a steel plate (location T), two metal covers, a funnel (location U), and four plastic pipe wrap ties.

Samples gathered during the second entry have been sent to the Department of Energy's Idaho National Engineering Laboratory for comprehensive analysis.

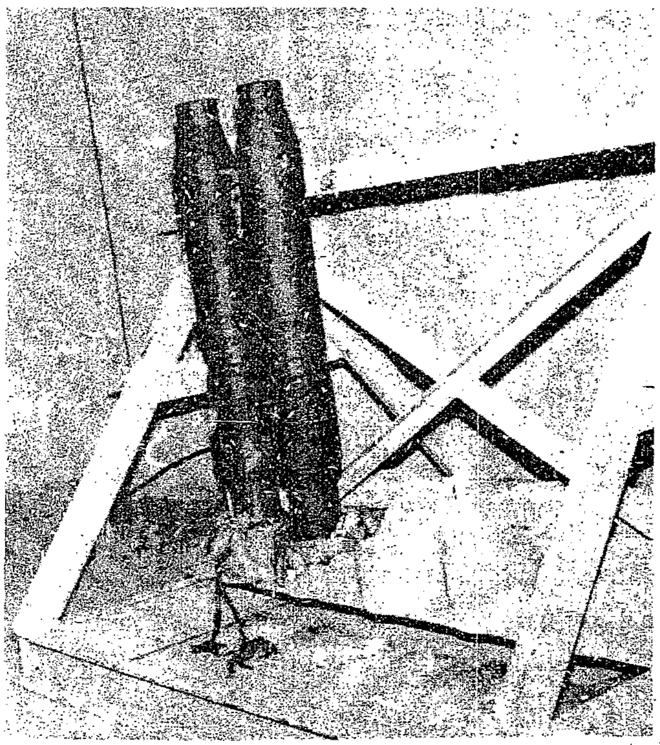
Gamma radiation exposures to the team members varied because of the tasks they performed and the amount of time they were inside the containment. Cooper and Behrle were in the containment just over 20 minutes; Griffith and Benson were in the containment about 40 minutes. The doses measured are presented in the accompanying table.



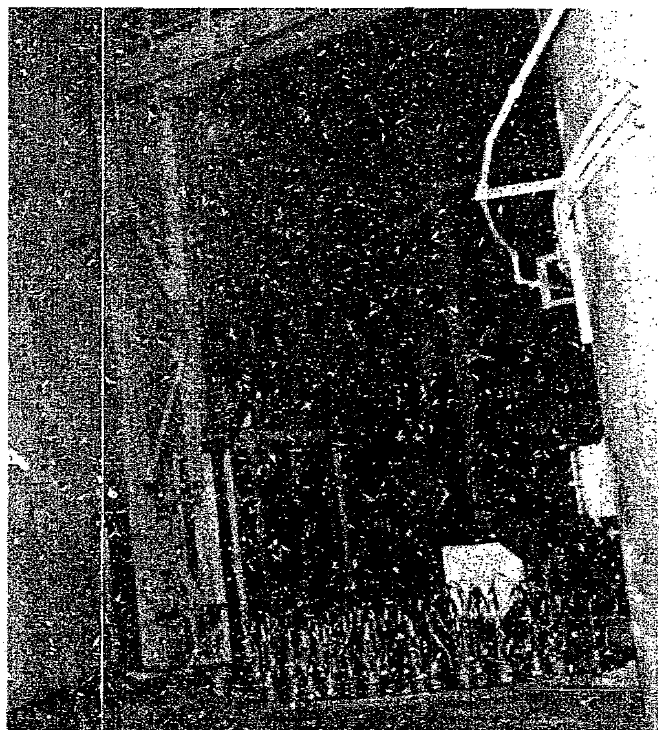
Martin Cooper works on removing radiation monitor HP-RT-211 from the cable. The detector was easily removed from the mounting bracket. The detector has been shipped to Sandia Laboratories for analysis.



A team member takes beta radiation survey at the bottom of the reactor cooler pump stand.



Two guide studs rest on the storage stands for reactor vessel studs. After the initial fueling efforts, the guide studs were wrapped in plastic. Melted plastic is at the bottom of the stand. In addition, a section of magenta and yellow plastic rope has melted to the stand base.



In the foreground, the incore instrumentation electrical connection are visible and appear undamaged. In the background, the steel structure for the fuel handling bridge is visible.

Containment Venting Releases 43,000 Curies of Krypton-85

A two-week project to vent krypton-85 from the Unit-2 containment building released an estimated 43,000 Curies of the isotope to the atmosphere between June 28 and July 11 of 1980. Monitoring by General Public Utilities Corporation (GPU) and federal agencies indicated the maximum offsite radiation doses during the venting were 4.34 millirem (mrem) to the skin and 0.044 mrem to the whole body. The maximum doses allowed by the NRC are 15 mrem to skin and 5 mrem to the whole body.

GPU officials attributed the difference between the actual release and the prerelease estimate of 57,000 Curies to deliberately conservative estimates of the containment volume and the amount of krypton trapped in the building. Original plans called for the venting procedure to take from two to four weeks using the reactor building purge system to effect the operation.

The venting began June 28 after the Nuclear Regulatory Commission (NRC) approved the operation. After four minutes of venting, however, radiation monitors sounded and officials halted the procedure.

Late in the afternoon of June 28, a five-hour test of venting rates began. The test helped engineers conclude that the radiation monitor alarm was erroneous. Venting was resumed at 3 p.m. June 29, using the hydrogen purge system to obtain a smaller air flow rate.

Engineers used the system for the next 10 days, admitting fresh air to the containment as krypton mixed with air was discharged through filtering mechanisms, past radiation monitoring devices, and up a 160-foot exhaust stack. Air flow rates through the hydrogen purge system reached a maximum of 565 cubic feet per minute (cfm).

On July 8, engineers switched the purging process back to the reactor building purge system at a flow rate of 1000 cfm. By this time, the krypton concentration had diminished sufficiently to allow a gradual increase of the purging rate to 18,500 cfm. About 27 Curies were vented in the final four hours of the process before engineers declared the venting completed.

The venting was stopped periodically during the two-week period for routine filter changes. Other stops occurred when weather conditions made venting undesirable.

Since completion of the venting, the utility has vented about 100 curies of krypton a month, which is permissible within NRC guidelines. The releases have usually occurred before teams go into the containment.

Two Engineers First to Enter Containment Since 1979 Incident

On July 23, 1980, William H. Behrle III and Michael Benson became the first persons to enter the Unit-2 containment building since the March 28, 1979 incident. During the entry, they visited the 305-foot elevation, or entry level, to conduct radiological and photographic surveys of conditions within the containment. (See accompanying location map.) While in the containment, they used two-way radios to communicate with the command center. Behrle and Benson are both engineers employed by Metropolitan Edison Company, a subsidiary of General Public Utilities Corporation (GPU).

While in the containment, the men took 29 photographs, made radiation measurements, and gathered six 00-square-centimeter swipe samples for subsequent contamination measurements. In addition, they retrieved a 5-gallon plastic bucket containing debris from the containment for subsequent analysis.

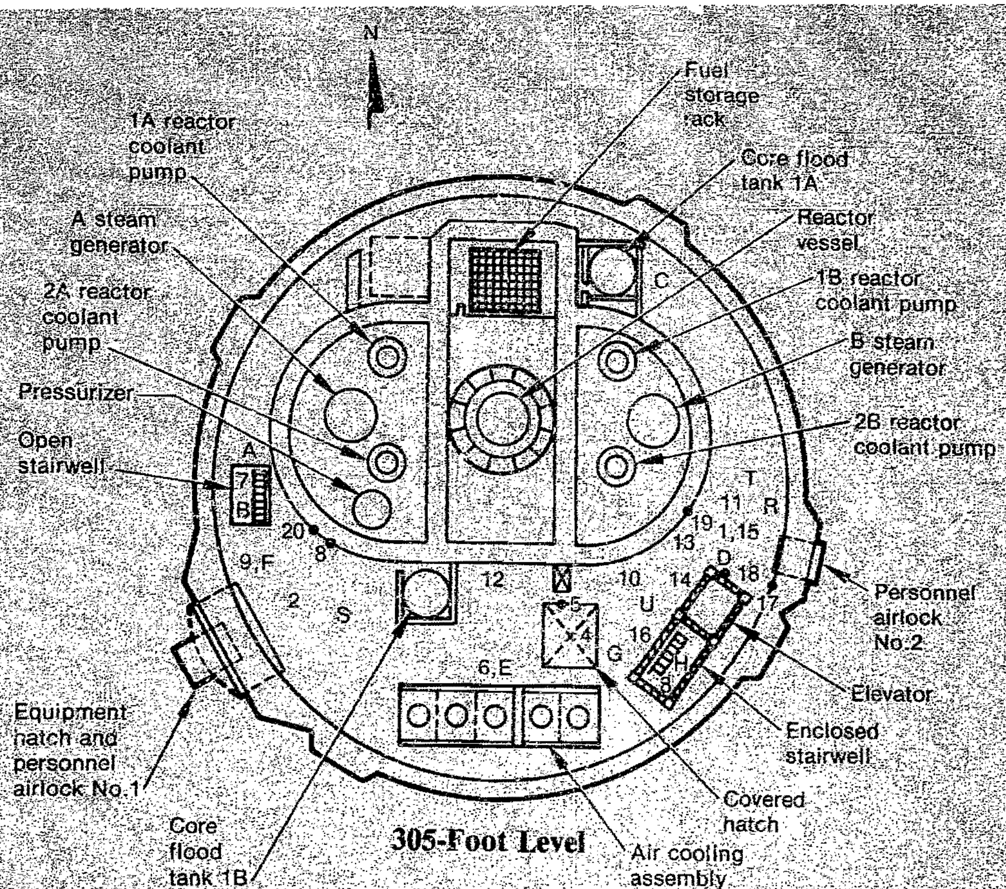
305-Foot Level Gamma Radiation Measurements

Location	Dose Rate (rem/hr)	Map location
Enclosed stairwell	8	3
Metal deck for covered floor hatch	10	4
Edge of metal-covered floor hatch	4	5
Air coolers	1.4	6
Top of open stairwell	18	7
D-ring and liner	0.4	8
Floor drains (range)	2 to 5	9,10,11
Core flood piping	3	12
Seal injection piping	3	13
Elevator door	3	14

Preliminary information gathered from the radiation measurements showed gamma radiation levels from 00 to 600 millirems per hour (mrem/hr) near the personnel airlock (location 1) and 700 mrem/hr at the equipment airlock (location 2). Other radiation measurements are given in the accompanying table. The general beta radiation levels in the area were 1 to 2 Rad per hour.

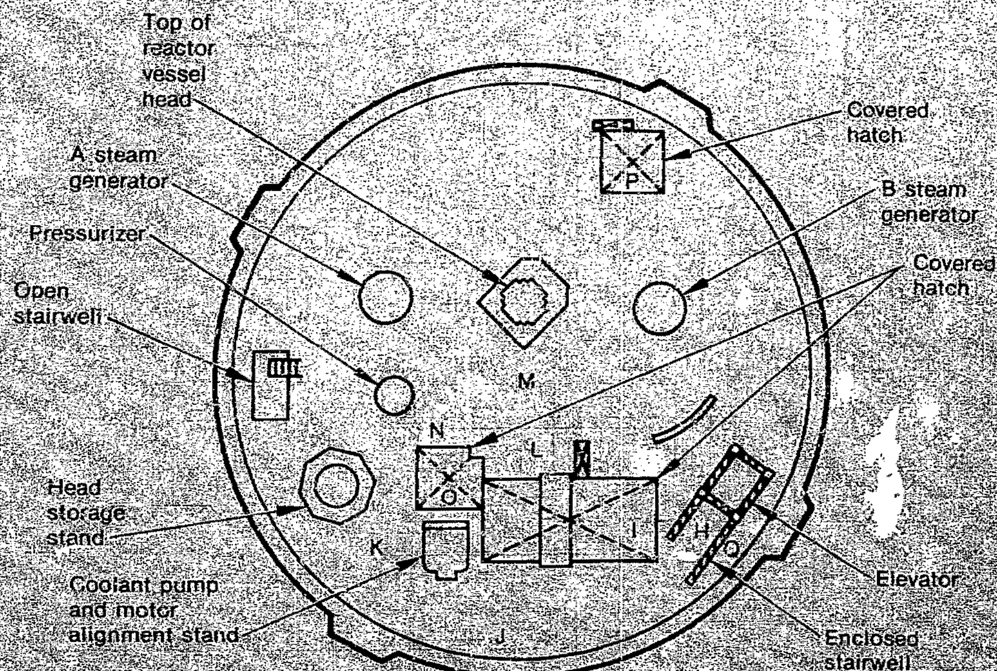
Preliminary results from floor swipes (locations 15 and 16) indicated the presence of cesium-134 and cesium-137 in concentrations of about 3×10^{-2} and 1×10^{-1} microcuries per square centimeter ($\mu Ci/cm^2$) respectively. Wall swipes (locations 17 through 20) indicated concentrations of the same cesium isotopes at about 2×10^{-5} and 4×10^{-4} $\mu Ci/cm^2$ respectively. Also detected in the wall swipes were radioactive isotopes of

Continued on page 8



305-Foot Level

Numbers indicate first entry locations
Letters indicate second entry locations



347-Foot Level

Continued from page 6
cerium, cobalt, antimony, and niobium in concentrations of 1×10^{-7} to 1×10^{-6} $\mu Ci/cm^2$. Technical personnel indicated that these elements were probably also present in the floor swipes but were undetectable because the high cesium levels masked their activity.

The men reported deposits of rust and dirt, colored orange and purple, on the floor. Some areas had watermarks that indicated apparent operation of the building spray system during or after the incident.

The men received whole body radiation exposures of approximately 190 mrem with a maximum extremity dose of about 220 mrem. No beta-dose measurements were taken. They wore two sets of anticontamination clothing under firefighter-type coats, pants, and boots. Self-contained breathing equipment supplied air for each man.

The first containment entry had been planned for May 20, 1980 but was aborted after the men were unable to turn the airlock door locking wheel. (See TI&EP Update dated July 31, 1980.) GPU officials attributed the failure to a malfunctioning locking mechanism. The door was later opened by drilling through a bulkhead to a locking pin and freeing the pin. The door still can be shut and sealed.

TMI Unit-2 Technical Information & Examination Program
UPDATE
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TMI Unit-2 Technical Information & Examination Program

UPDATE

March 11, 1981

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Entry Teams Take First Look at Reactor Head, Polar Crane

The first close look at the Unit-2 reactor vessel head and the polar crane revealed expected rust but no visible damage, according to members of the fifth containment entry team.

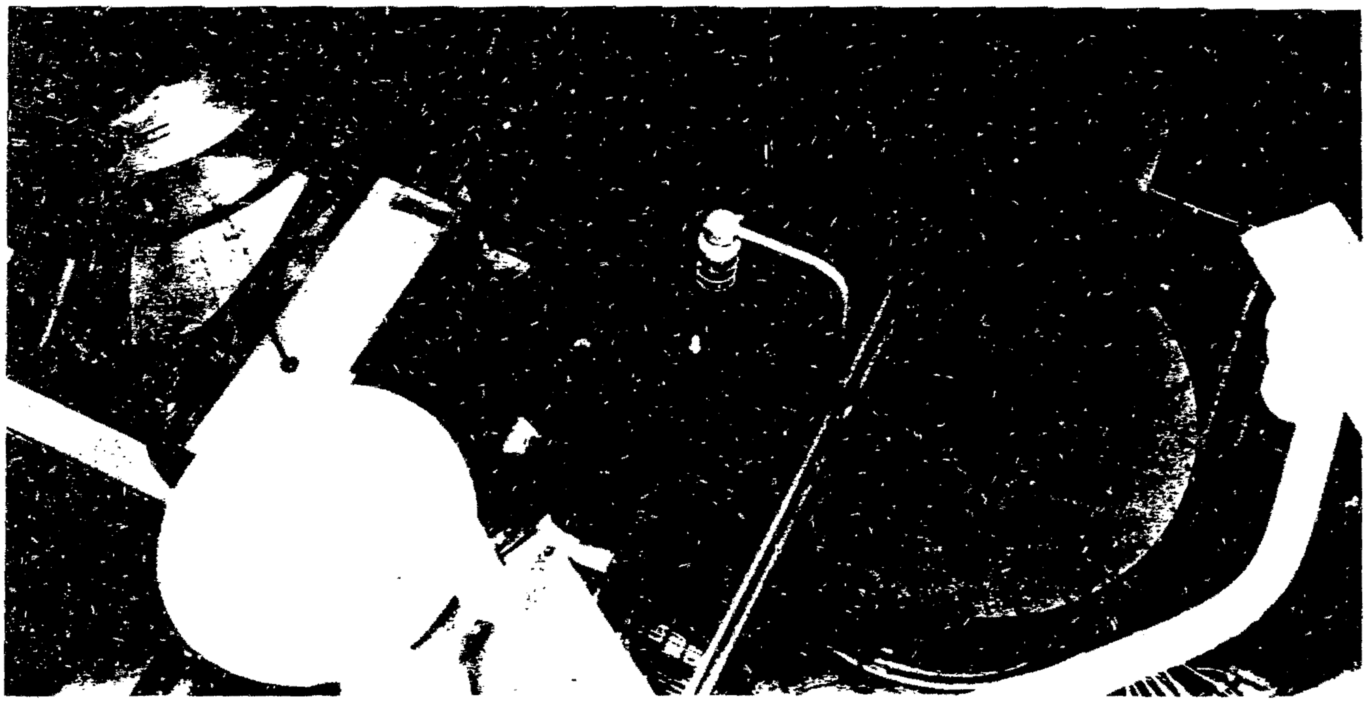
The fifth entry into the Unit-2 containment, conducted on December 11 by a team of 14 persons, was devoted to visual inspection of the reactor head and polar crane, additional radiation surveys, and tests of decontamination procedures. All tasks further photographic surveys of the containment interior. Results from the radiation surveys and decontamination tests are expected to be reported in a later issue of the TI&EP Update.

Gregory R. Eidam, a TIO project engineer who participated in the entry, reported that the crane, cables, and hook were rusty but appeared to be structurally sound. One section of copper conductor from the crane was found to have fallen to the 347-foot floor level. A more detailed examination of the polar crane, including motors and auxiliary equipment, is tentatively planned for the seventh containment entry to be conducted in early March.

Other team members reported that the area around the top of the reactor vessel appeared to be rusty. Water was found in the north neutron shield tanks; the south tanks were dry.

Four persons from GPU companies were occupied for more than an hour taking additional radiation surveys on the 305-foot floor level, examing floor penetrations for future sump sampling, and checking locations for television cameras scheduled for installation during the sixth containment entry (see article in this issue).

Continued on Page 2



View from the top of the D-ring looking down at the steam generator (left) and the reactor coolant pump (right). Most of the exposed metal surfaces show little sign of corrosion.

Entry Team

Continued from page 1

On the 347-foot floor level, other team members tested gross decontamination methods to determine the most effective techniques. The tests included water flushing at low pressure, water flushing at high pressure, decontamination solution with abrasive scrubbing and a low-pressure flush, strong decontamination solution with low-pressure flush, and a strippable coating. The temperature of all decontamination fluids was 150°F.

Among the entry team members was John Collins, deputy program director of the Nuclear Regulatory Commission's (NRC) TMI Site Office. Collins left TMI recently to take another NRC post with the Office of Inspection and Enforcement, Region 4, in Texas.

TMI Unit-2 Technical Information & Examination Program

March 11, 1981

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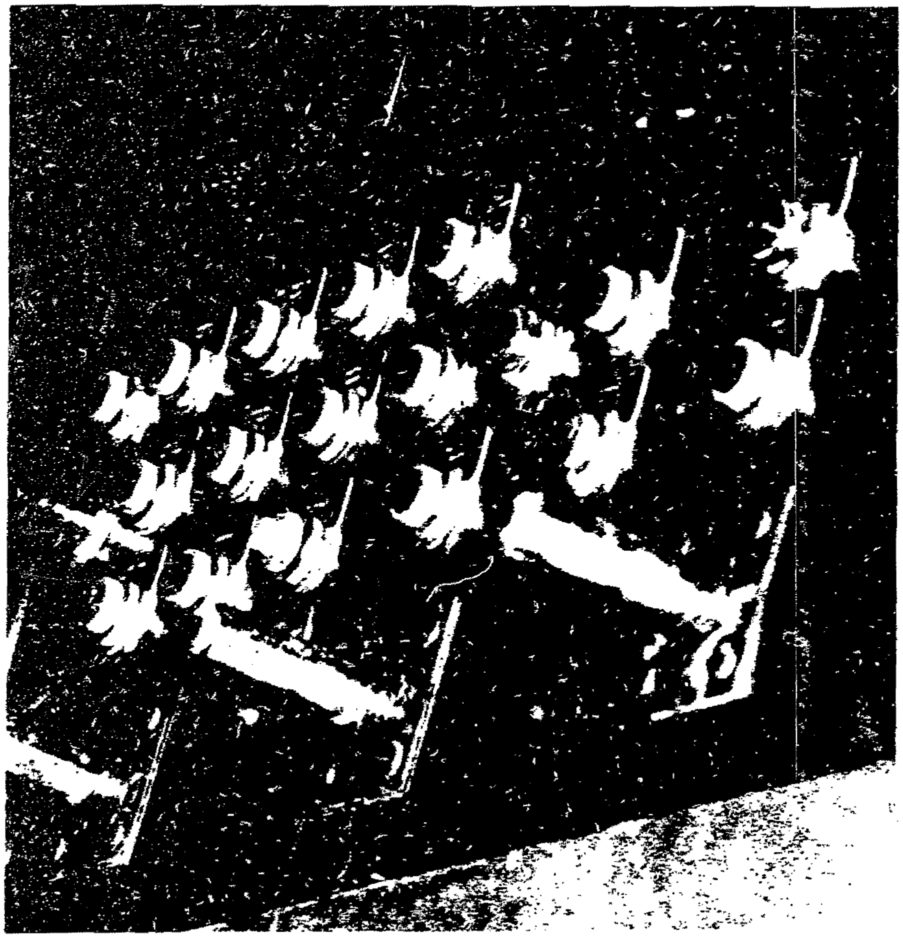
The *TI&EP Update* is specifically designed to highlight data and information obtained as part of the TMI-2 Technical Information and Examination Program (TI&EP). As space permits, the *TI&EP Update* may feature certain TMI-related information that, though not part of the TI&EP, would be of general interest to the scientific community.

W.W. Bixby is manager of the DOE-TMI Site Office. H.M. Burton is manager of the Technical Integration Office. D.M. Grigg is managing editor of the *TI&EP Update;* G.R. Brown is associate editor.



Exposed metal parts on the control panel for the auxiliary fuel handling bridge are extensively corroded. Plastic buttons and control handles are partially melted.

TIO Engineer Participates in Fifth Entry

Gregory R. Eidam, a Technical Integration Office Project engineer, helped inspect the Unit-2 polar crane during the fifth containment entry, conducted on December 11.

Eidam was the project engineer for the original installation and refurbishment efforts for the polar crane at the Loss-of-Fluid-Test (LOFT) Facility at the Department of Energy's Idaho National Engineering Laboratory.

"We took the stairs up (from the 347-foot floor level) and climbed the crane access ladder between the two crane box beams," Eidam said. "We took radiation readings and did a visual inspection. The crane looked rusty, as were the couplings and the rails. It appeared to be in structurally good condition."

Eidam remarked that one section of copper conductor (bus bar) from the crane had fallen to the 347-foot level floor.

Before climbing to the polar crane, Eidam helped photograph conditions inside the containment. He watched, as part of the buddy-system plan, when two other team members climbed down to inspect the top of the reactor vessel.

According to Harold Burton, Manager of the TIO, refurbishment of the Unit-2 polar crane is an extremely important activity of the TI&EP. "It represents a critical path activity to early examination of the reactor core and removal of the reactor vessel head, both key recovery program milestones," stated Burton.

Eidam is scheduled to join the seventh containment entry team in late February for more extensive mechanical and electrical inspections of the crane and its components.

Fourth Entry on Videotape

Twelve engineers and technicians, working in three shifts over a three-hour-and-forty-minute period, filmed an hour of videotape footage inside Unit-2 as part of the reactor building damage asessment during the fourth containment entry. They supplemented the videotape by taking 71 still photographs.

The November 13 videotape included 20 minutes of filming at the 305-foot floor level and 40 minutes at the 347-foot floor level. The tape was later edited to 20 minutes and narration was added.

The other activities by team members on the 347-foot floor level included moving the auxiliary fuel handling bridge for easier access to a ladder leading to the reactor vessel head area, testing decontamination methods, and conducting radiological surveys.

On the 305-foot level, gamma readings ranged from 200 millirem per hour (mrem/hr) in the northeast section of the containment to 3800 mrem/hr at the B core flood tank piping. Gamma readings on the 347-foot level ranged from 100 mrem/hr at the top of the east D ring to 1000 mrem/hr at a floor drain and 1500 mrem/hr at about one meter into the west D ring.

Smear samples taken on the 305-foot level as part of testing decontamination methods yielded the preliminary results shown in Table 1.

A scrape sample taken outside the decontamination methods test area yielded preliminary readings of 3.4 x 10^{-1} microcuries (μCi) of cesium-134 and 2.4 x 10^{1} μCi of cesium-137. A scrape sample from the test area following decontamination testing showed 5.0 x 10^{-2} μCi of cesium-134 and 3.6 x 10^{-1} μCi of cesium-137.

A paint chip found on the floor at the 347-foot level had results of 1.6 x 10^{-2} μCi of cesium-134, 1.1 x 10^{-1} μCi of cesium-137, and 4.8 x 10^{-3} μCi of strontium-90. The chip is believed to be from the containment building dome.

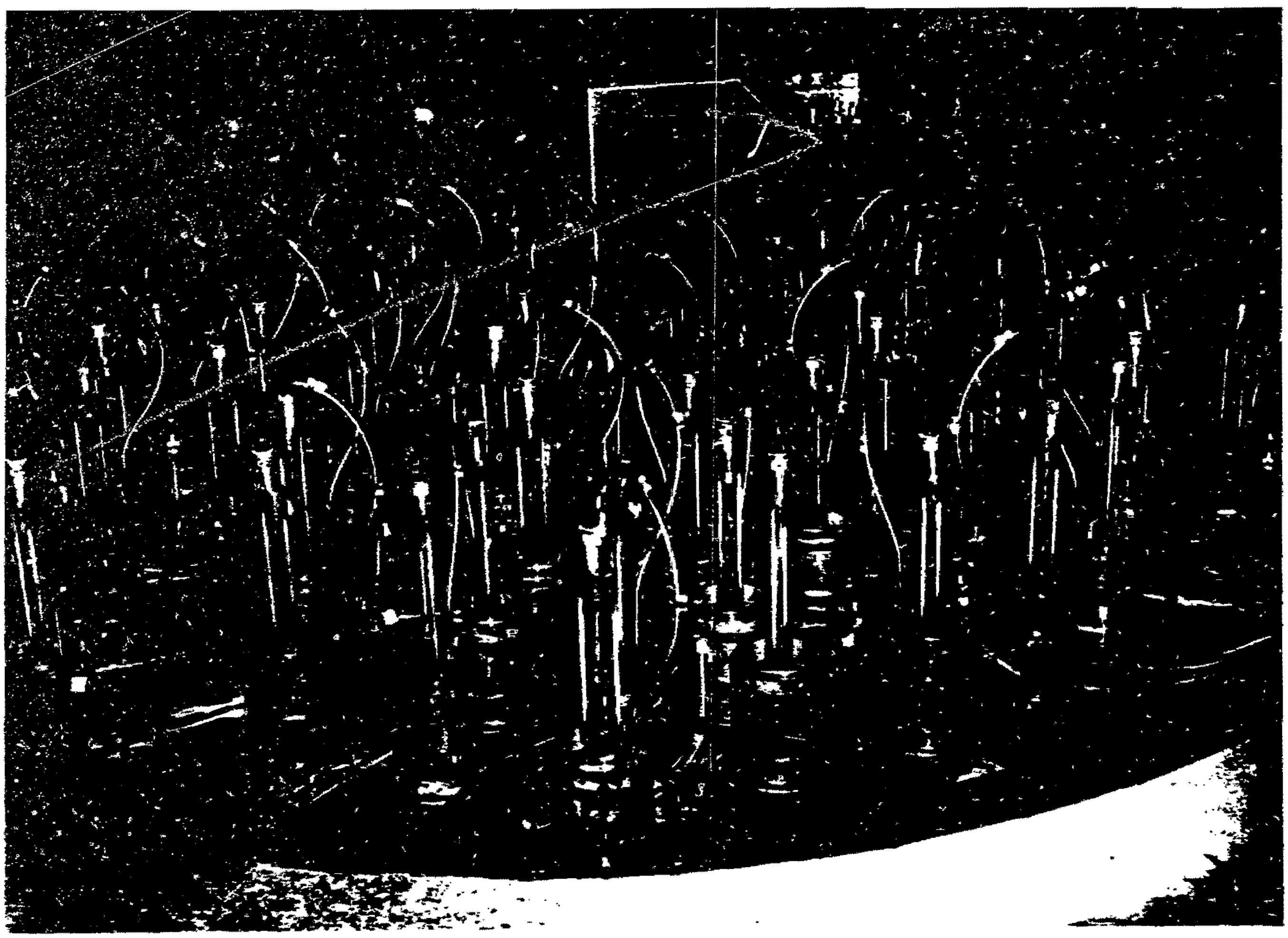
Continued on Page 4

The fourth entry was videotaped using a camera installed in containment penetration 626. This penetration was also used for visual examination of the containment interior and drawing reactor building air samples prior to the first entry (see *TI&EP Update* of April 15, 1980).

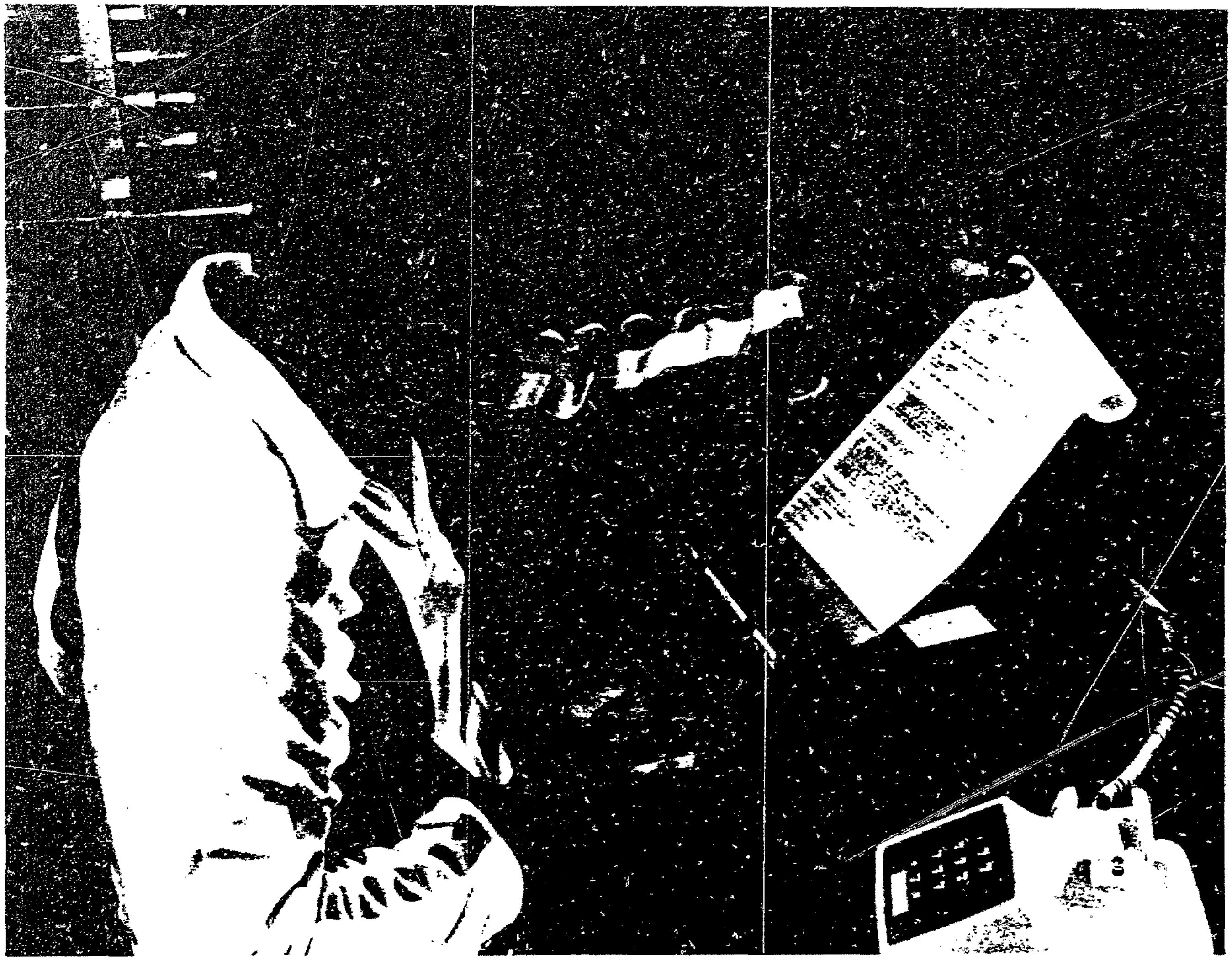
Fourth Entry *Continued from page 3*

Table 1. Preliminary Analyses of Smears Taken on the 305-Foot Level (in microcuries)

Treatment	Sample No.	Cesium-134	Cesium-137	Strontium-90
Before decontamination	1	1.7E0 ± 7.6E-3	1.2E1 ± 1.9E-2	2.3E-1 ± 5%
	2	1.3E0 + 6.6E-3	8.9E0 + 1.6E-2	5.7E-1 + 5%
After demineralized water wash	3	1.8E-2 ± 3.1E-4	1.4E-1 ± 7.6E-4	5.2E-3 ± 5%
	4	2.6E-2 + 3.7E-4	1.9E-1 + 8.9E-4	1.4E-2 + 5%
After Radiac wash	5	7.8E-3 ± 2E-4	5.8E-2 ± 4.9E-4	5.6E-3 ± 5%
	6	3.8E-3 + 1.4E-4	2.8E-2 + 3.4E-4	7.7E-4 + 5%
After Radiac scrub	7	8.5E-3 ± 2.1E-4	6.3E-2 ± 5.1E-4	3.3E-3 ± 5%
	8	5.1E-3 + 1.6E-4	3.6E-2 + 3.9E-4	1.2E-3 + 5%



Accumulation of boron crystals is evident around the base of the incore instrumentation seal table. Corrosion is visible on the exposed metal surfaces.



F.J. Kocsis III, TIO Information and Records Manager, uses a computer terminal to relay TMI data to nuclear utilities. The terminal is part of the Nuclear NOTEPAD computer conferencing system.

TIO Relays Data to Nuclear Utilities

A computer conferencing system is helping the TIO disseminate information to nuclear utilities about activities at TMI Unit-2.

Nuclear NOTEPAD, formerly known just as NOTEPAD, offers computer conferencing among 62 nuclear utilities with operating or construction licenses, the Nuclear Safety Analysis Center (NSAC)/ Institute of Nuclear Power Operations (INPO), and the TIO at Three Mile Island via either printout or video display terminals.

"Nuclear NOTEPAD users can receive from TIO the results from containment entries, pertinent news stories about TMI activities, and news releases from the GPU Public Affairs Office," said Frank J. Kocsis III, TIO Information and Records Manager.

"As more reports and data about the TMI-2 recovery become available, we will make announcements on NOTEPAD so that interested nuclear utilities may request it," Kocsis said.

Kocsis and Ronald Simard, Assistant Director for Information and Data Services at NSAC, set up the TMI NOTEPAD operation nine months ago. NSAC and INPO jointly found NOTEPAD. INFOMEDIA Corporation of Palo Alto, California, operates the computer conferencing system.

"NOTEPAD has been used extensively to share expertise among the nuclear utilities and gain information about Nuclear Regulatory Commission orders for design and operating changes in nuclear plants," Kocsis said.

NOTEPAD was organized in August, 1979, after the Kemeny Commission criticized the lack of information being shared among nuclear utilities.

The value of NOTEPAD, officials said, was demonstrated in February, 1980, when Florida Power Corporation's Crystal River Plant experienced problems and information sharing helped rectify the situation.

Cameras Installed in Unit-2

Technicians have installed eight black-and-white television cameras inside Unit-2 to provide continuous visual monitoring of conditions inside the containment building. Videotaping equipment, included as part of the system, will allow making visual records of future events within the containment and will assist the TIO in documenting damage within the reactor building.

The installation, performed as part of the sixth containment entry on February 3rd and 5th, included placement of four cameras each for overlapping coverage on the 305- and 347-foot levels of the building. Technicians transported the cameras into the containment on the first day of the entry, and installed them on the second day.

The night-vision cameras, dubbed "moon landers" because of the white environmental housings atop tripods, are linked to control consoles in the Unit-2 control room and the entry command center.

Each camera has a zoom lens with a one-billion-to-one dynamic range so that they will operate in extremely low light levels, according to James W. Mock, the TIO project engineer who worked with General Public Utilities Corporation on the installation. Auxiliary lights can be attached to the cameras if necessary.

The camera housings include windshield wipers to remove condensation that accumulates in the high humidity of the containment. Each housing also has positions, both inside and outside, for attaching radiation dosimeters.

Mock explained that for camera control from the consoles, each camera unit has redundant receivers for remote operation of lens zooming, panning, and tilting. Cables especially manufactured for the project carry the signals from the containment to the monitor screens on each console. Operators at the console can see transmissions from each camera and select a desired view for recording on videotape.

Krypton-85 Venting Final Results

Final results of effluent monitoring done during the two-week venting of krypton-85 from the Unit-2 containment showed that about 44,132 Curies were released. The venting occurred from June 28 to July 11, 1980 and was described in the *TI&EP Update*, October 29, 1980.

Analyses indicated the reactor building originally contained about 44,600 Curies of krypton. In addition, the project vented an estimated 1.3 Curies of tritium, 5.5×10^{-6} Curies of cesium-137, and 5.72×10^{-9} Curies of strontium-90.

Radiological monitoring by the General Public Utilities Corporation and the Environmental Protection Agency confirmed that detectable offsite releases of radioactive material were well within the technical specifications set for venting by the Nuclear Regulatory Commission (NRC).

Since July, the utility has vented about 100 Curies of krypton-85 a month, which is permissible within NRC guidelines. The releases are usually made before entries into the containment building.

GEND Group Hosts First International Seminar

More than 100 persons from 21 countries learned about the progress of the TMI Unit-2 research effort and plans for future work during a two-day seminar in Washington, D.C.

The seminar, hosted by the GEND group -- General Public Utilities Corporation, Electric Power Research Institute, United States Nuclear Regulatory Commission, and United States Department of Energy -- was the first directed to an international audience.

"The emphasis of the presentations was the plans for carrying out research programs and selected results to date in three areas -- instrumentation and electrical survivability, decontamination and dose reduction, and radioactive waste processing," said Willis W. Bixby, DOE Site Manager at TMI.

The GEND group sponsors the Three Mile Island Technical Information and Examination Program to obtain valuable generic information from the Unit-2 accident.

The November 21 and 22 seminar, Bixby said, included presentations on nine major task areas:

- Instrumentation and electrical survivability
- Fission product transport and deposition
- Decontamination and dose reduction
- Radioactive waste handling
- Data bank
- Mechanical component survivability
- Early core damage assessment
- Core deposition studies
- Fuel and core component examination program

Participants also had the opportunity to watch a 20-minute, narrated, color videotape of the fourth entry into the Unit-2 containment.

The GEND group plans to be "quite flexible in working with individual nations or organizations within nations," Herbert Feinroth, DOE director for the System and Safety Evaluation Division, told the seminar. Suggestions for participation included purchase of technical reports and direct involvement in development of nuclear waste disposal programs.

Five countries -- Germany, Italy, Spain, Sweden, and Taiwan -- already have representatives at TMI to work with the utility. Representatives from other countries gave positive responses to the possibility of participating, but none have made committments as yet.

"Each government or organization at the meeting was encouraged to participate in a manner considered suitable to its needs," Bixby said.

HP-RT-211 Analysis Results

Sandia National Laboratories has completed analysis of radiation detector HP-RT-211, which was removed from the containment building during the second entry (see *TI&EP Update* of October 29, 1980). The major preliminary findings are:

- the total radiation dose estimate for the detector is lower than previous estimates,
- equipment having electrical connectors should be oriented so that the connector is shielded from direct spray or the connector shell should be potted to reduce the possibility of water and contaminant intrusion, and
- with slight adjustment or modification, some instruments may still provide correct and usable information despite partial failure.

The cause of the instrument failure was confirmed to be a 163-ohm short circuit between the collector and emitter of a 2N-3906 transistor, which was part of the detector output circuit.

Scanning electron microscope photographs showed a catastrophic punch-through between the collector and emitter. Scientists postulated that the failure occurred when the containment building spray system actuated during the TMI incident, shorting the signal and 600-volt pins in the backshell of the connector joining the detector to its cable. (No other radiation detector failed in this manner.)

Although the failure caused the instrument to indicate low, instrument data recorded on a stripchart appear to be proportional to actual radiation levels in the containment. Efforts to reconstruct the radiation profile time history are continuing.

Six transistors, two pieces of Teflon tubing, and a buna nitrile "O" ring were removed from the detector for use in estimating the total gamma radiation dose. Data collected by exposing like devices to known radiation doses enabled scientists to estimate the total dose at the detector location within the containment to be between 0.7E5 and 3E5 rads.

Gamma spectroscopy and radiochemical analysis of the outside detector surfaces indicated the presence of cesium and strontium contaminants but no transuranics (to the minimum detectable limits of the instruments used). The top horizontal surface had cesium-134 and cesium-137 gamma levels of 0.048 and 0.305 microcuries per square centimeter respectively. The levels on the sides and bottom were lower by a factor of four. Beta-gamma film profiles of the contaminant distributions showed localized hotspots. The "O" ring seal prevented contaminants from entering the inside of the detector.

TMI Unit-2 Technical Information & Examination Program

UPDATE

November 30, 1981

Published by EG&G Idaho, Inc. for the U.S. Department of Energy

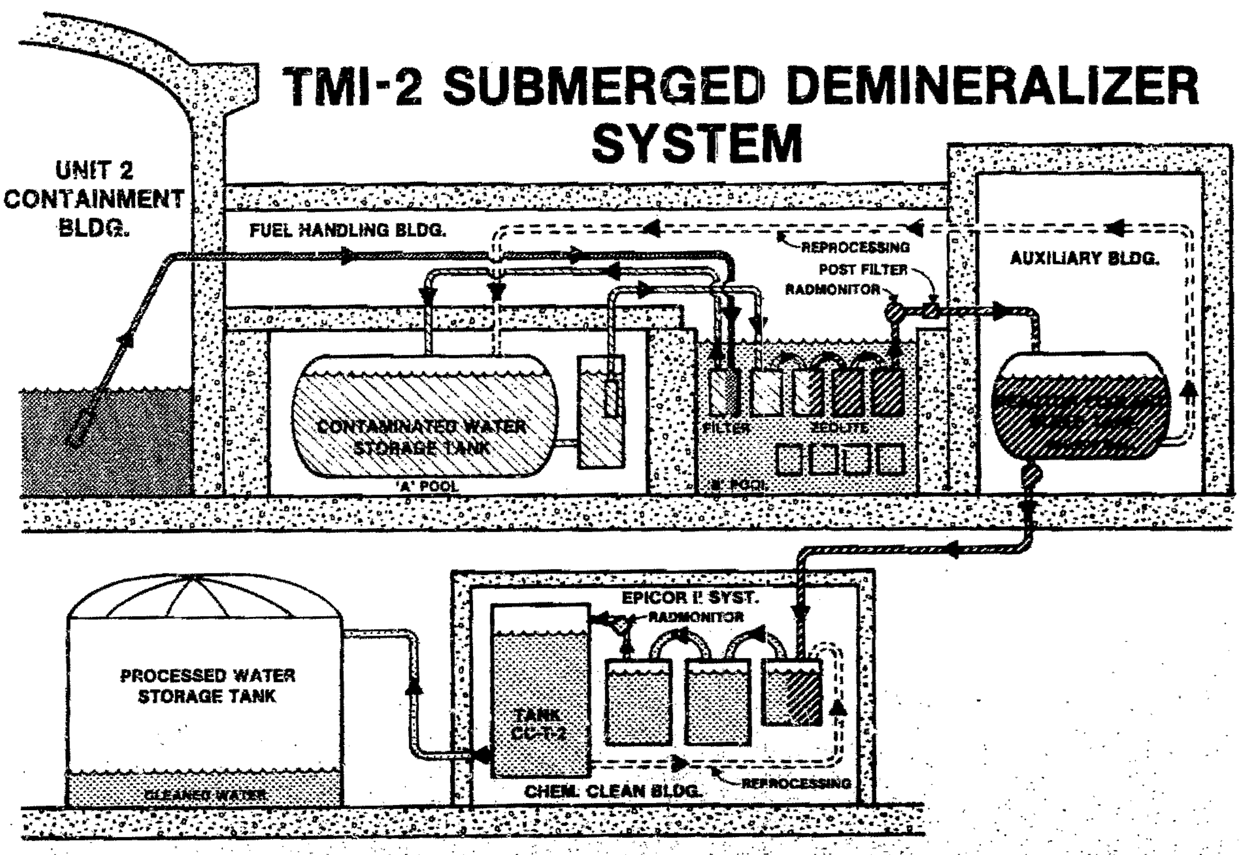
Submerged Demineralizer System Processes Contaminated Water

A major stage in the recovery and cleanup of the damaged TMI-2 reactor began in July with testing of an ion exchange water treatment process known as the Submerged Demineralizer System (SDS). The SDS will process 100,000 gallons of coolant in the reactor coolant system, and 600,000 gallons of more highly contaminated water from the basement of the Unit 2 containment building. Together with the containment entry program (see article, this issue) the cleanup of this water is one of the two major ongoing projects of the recovery program.

Removal of the contaminated water from the reactor building will significantly reduce the levels of direct and mobile airborne radiation present in the building. The water is a source of direct radiation to plant personnel who must go into the reactor building during containment entries to maintain plant systems. The removal of contaminated water, which entered the reactor building through an open relief valve during the accident more than two and a half years ago, is a necessary step in the recovery effort.

The SDS is an ion exchange process, similar to the EPICOR II system used earlier in the recovery effort to treat

Continued on Page 2



Submerged Demineralizer System being used to process contaminated water from the basement of the TMI Unit 2 containment building.

Demineralizer System

Continued from Page 1

500,000 gallons of water in the less contaminated auxiliary building (see July 31, 1980 *Update*). The SDS differs from the EPICOR II in two major ways. First, the SDS operates under water in the Unit 2 spent fuel pool adjacent to the reactor building. This underwater operation protects plant workers from the high radiation levels.

Second, the SDS uses an inorganic material called zeolite, rather than the predominantly organic resins used in EPICOR II, to absorb the fission products from the water. The SDS zeolite is composed of 40 percent Linde ionsiv zeolite-A-51 and 60 percent Ionsiv zeolite-IE-96. The inorganic zeolite can accommodate loadings in excess of 20,000 curies per cubic foot, while resins in the EPICOR II system normally accomodate loadings less than an average of 40 curies per cubic foot. The ion exchange process in the SDS effectively removes more than 99 percent of the fission products, primarily cesium and strontium, from the contaminated water. The ion exchange media are expected to produce a decontamination factor in excess of 30,000 for cesium and 250 for strontium. Tritium, a fission product also present in the water, can not be filtered out because of its structural similarity to the hydrogen component of the water molecule.

Radioactive water from the basement of the reactor building is being pumped into the SDS in the spent fuel pool by means of a suction pump floating on the surface of the contaminated water (see accompanying figure). The water passes through preliminary filters and on to the zeolite resin canisters. After it passes through the zeolite canisters, the water can be stored in tanks, or can be processed further with the EPICOR II system. The further processing through EPICOR II is expected to produce an average decontamination factor in excess of 100 for cesium and strontium.

SDS processing of contaminated water generates radioactive waste in the form of filters and ion exchange resins laden with radioactive concentrates. This waste will be temporarily stored in the fuel pool adjacent to the demineralizer system or in specially constructed containers on site. The Department of Energy (DOE) plans to ship the SDS resins to DOE facilities for research, development, and testing purposes.

The processed water from the SDS still contains concentrations of tritium in excess of 0.8 microcuries (μCi) per milliliter. Until the Nuclear Regulatory Commission (NRC) approves final disposition of this water, it will be stored in various tanks on site. Potential on-site use of the water may include pumping it back into the reactor building to protect workers from residual radiation on the building's basement floor, and using the water in decontamination activities and in various plant systems. Most of the water to be processed will be stored in two specially constructed 500,000 gallon tanks on site.

Actual implementation of SDS testing was not possible until the third week of June, when the NRC approved its use. Approval was contingent upon the conduct of an NRC study of the environmental consequences of the entire recovery program. This study, now completed, concluded that the cleanup program as planned can proceed with little risk to plant personnel or the public. NRC approval of the SDS allowed plant technicians to begin test processing of water through the system. Approximately 150,000 gallons of less heavily contaminated water from the auxiliary building were processed during the summer of 1981, and processing of the reactor building sump water began in September.

By the end of the third week in October, the SDS system had processed approximately 123,000 gallons of reactor building sump water. The system processing rate is about 5 gallons of water per minute. Recovery program estimates indicate that it will take about four to five months to process the contaminated water in the TMI Unit 2 containment building.

The SDS cost $11 million to design and build. Its use will be a significant step toward the cleanup of the damaged TMI Unit 2 nuclear plant. SDS operation will also provide generic information to the nuclear industry regarding processing of high specific activity liquids.

First Multilevel Sample Taken

EG&G engineers developed a unique multilevel sampling device to obtain representative liquid and sludge samples from the 600,000 gallons of highly contaminated water in the TMI-2 containment building basement. The device, called Water and Sludge Sample Device (WSSD), was designed to obtain eight simultaneous 150-milliliter samples at four different levels. This technique will allow scientists and engineers to determine the extent of stratification and flocculant dispersal patterns in the eight and a half feet of water in the containment basement.

The lightweight aluminum device was designed for easy operation under adverse conditions inside the containment building. Unique features of the WSSD include the following:

- Acquires eight simultaneous samples at four levels, two near the liquid surface, two at the mid-level, two near the bottom, and two sludge samples
- Minimizes stratification disturbance
- Minimizes losses of flocculant during sampling by rapid sample acquisition
- Minimizes losses of entrained gas in the sample
- Permits exact sample locations to be determined relative to the basement floor
- Provides known sample volumes
- Ensures that outside of sample bottle remains contamination-free by using watertight bottle housings
- Allows immediate visual observation of the sample

In addition lightweight design of the WSSD permits operation by one individual.

The lower portion of the WSSD is shown in the accompanying photograph. Evacuated sample bottles are placed septum down into the shield base to engage with an O-ring to form the lower watertight seal. Installation of the shield cap over the shield base

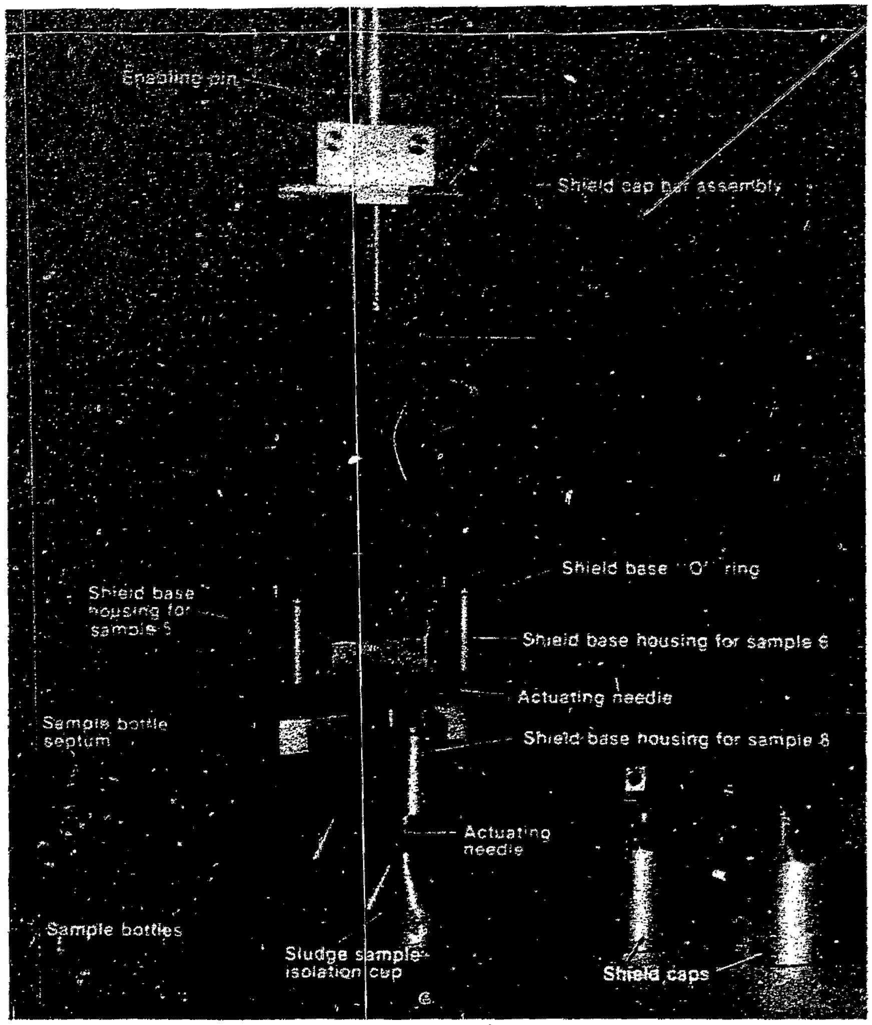
Continued on Page 3

Sample Taken

Continued from Page 2

engages the outer shield base O-ring to complete the watertight anticontamination seal of the containment housing around each bottle. The shield caps are securely locked into position by a fast-acting shield cap bar ratchet assembly. The locking bar ratchet assembly permits rapid unlocking and removal of the shield caps to minimize personnel operating time and radiation exposures. The sludge sample isolation cup at the base of the WSSD traps an area of sludge on the basement floor, and maintains the actuating needle suction point as close to the floor as possible in order to ensure representative sampling.

After the WSSD is lowered into the proper position in the containment basement, an enabling pin is removed, and sample acquisition is initiated by plunger action. This action drives the actuating needles through the sample bottle septums, causing liquid or sludge samples to be rapidly forced into the evacuated bottles. When the WSSD is raised, the bottles retract from the needles and the self-sealing septum prevents any loss of sample material. The WSSD is raised from the basement to the 305-foot elevation, and technicians remove the shield caps. This exposes the sample bottles which,

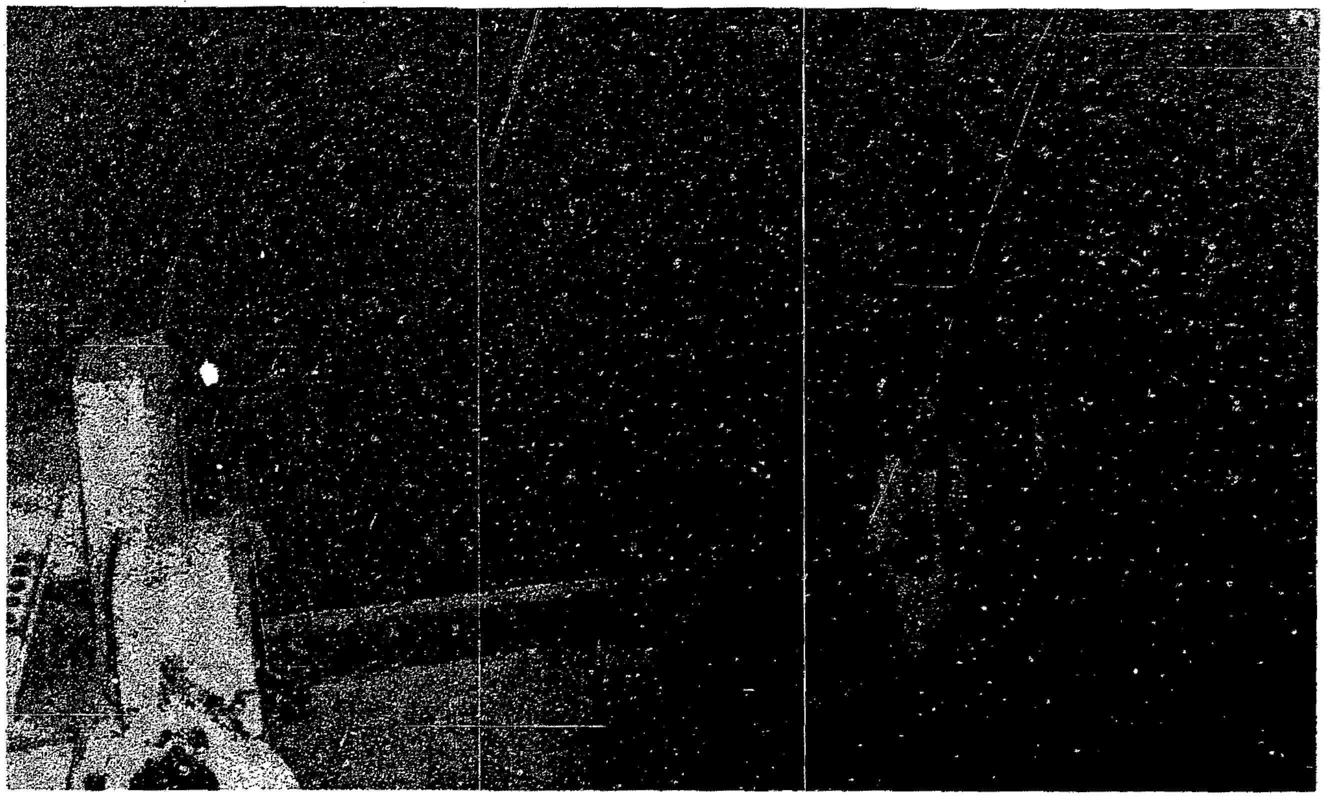


Lower Portion of the Water and Sump Sampling Device

Table 1. TMI-2 reactor building basement water sample analyses results[a]

Sample Number:	1	3	6	8		
Nuclide	(µCi/ml)	(µCi/ml)	(µCi/ml)	Slurry (µCi/ml)	Supernate (µCi/ml)	Particulate (µCi/g solids)
^{54}Mn	ND[b]	ND	ND	>2E-04	NA[c]	ND
^{60}Co	>6E-04	>3E-03	>2E-03	>8E-04	NA	1.7 ± 0.2E+01
^{90}Sr	5.0 ± 0.2E+00	5.4 ± 0.2E+00	5.2 ± 0.2E+00	NA	NA	8 ± 2E+02
^{90}Sr	5.4 ± 0.5E+00	5.2 ± 0.5E+00	5.1 ± 0.5E+00	NA	5.3 ± 0.5E+00	7.8 ± 0.8E+02
^{106}Ru	ND	ND	ND	>4E-04	NA	ND
^{125}Sb	>3E-02	>3E-02	>3E-02	>5E-02	NA	4.5 ± 0.2E+02
^{129}I	5.5 ± 0.7E-06	5.4 ± 0.7E-05	3.8 ± 0.5E-06	NA	2.5 ± 0.5E-06	NA
^{134}Cs	1.85 ± 0.01E+01	1.84 ± 0.01E+01	1.86 ± 0.01E+01	1.87 ± 0.01E+01	NA	1.79 ± 0.04E+02
^{137}Cs	1.43 ± 0.01E+02	1.42 ± 0.01E+02	1.43 ± 0.01E+02	1.44 ± 0.01E+02	NA	1.29 ± 0.01E+03
^{144}Ce	ND	ND	ND	>8E-03	NA	7.6 ± 0.6E+01
	(µg/ml)	(µg/ml)	(µg/ml)	(µg/ml)	(µg/ml)	(mg/g solids)
^{235}U and ^{239}Pu	<1E-02	<1E-02	<1E-02	NA	NA	8.8 ± 0.9E-02
^{238}Pu	4 ± 1E-08	NA	NA	5 ± 1E-07	NA	5 ± 1E-07
^{239}Pu	2.2 ± 0.7E-04	NA	NA	2.6 ± 0.5E-03	NA	2.9 ± 0.6E-03

a. Concentrations as of 6-1-81.
b. ND = not detected.
c. NA = not analyzed.



The preliminary polar crane inspection was the first in a series of inspections to determine the general condition of the crane and to conduct overall area damage assessments and radiation surveys.

Preliminary Inspection of Polar Crane Complete

A four-man team performed the preliminary inspection of the TMI-2 reactor building polar crane during containment entry 13. The inspection included opening and inspecting the drive train and main hoist gear boxes, conducting motor winding resistance checks, performing visual inspections of the motors' internals through their inspection ports, and conducting overall area damage assessments and radiation surveys. This inspection was the first in a series of detailed inspections to determine the general condition of the crane and to provide early assessments of which components may require replacement.

The polar crane work is an essential part of the TMI-2 recovery and R&D efforts, since the crane is required to remove missile shields and the reactor vessel head. The two major areas of R&D interest include electrical and mechanical component survivability. The Technical Integration Office's Instrumentation and Electrical program will focus its efforts on determing the survivability of such components as limit switches, motors, loadcells, and control cabinets. The Electric Power Research Institute's Mechanical Components program will focus its efforts on determining the survivability of such components as reduction gears, cable drums, and wire rope. These efforts will not only provide the data necessary for GPU to determine the extent of reburbishment required, but will also contribute valuable information to overall understanding of the the reactor building environment during the accident.

Sample Taken

Continued from Page 3

free of external surface contamination, are loaded into lead-shielded shipping containers.

The WSSD was successfully used on May 14, 1981 during containment entry 10 to obtain eight TMI-2 containment basement water samples. The samples were shipped to the Idaho National Engineering Laboratory for analysis and archiving. Four samples were archived, and the preliminary analytical results of the other four are shown in the accompanying table.

Each of the four samples analyzed was taken at a different level relative to the basement floor. Sample 1 was taken at 84-3/4 inches above the floor; sample 3 was taken at 47-3/4 inches; sample 6 at 5-3/8 inches; and sample 8 at the basement floor itself. Sample 8 contained solids as well as liquid, and both of these were analyzed in the study. The table contains data from nuclide analyses conducted for the gamma emitters (cesium-137 and -134), for the beta emitter (strontium-90), for the x-ray emitter (iodine-129), and for fissile material. In addition the presence of cerium-144, antimony-125, and cobalt-60 were observed and were quantitatively measured where possible. All data gathered in these analyses are currently undergoing further detailed analysis.

TMI Containment Entry Highlights

A total of eight successful containment entries have been completed since the last issue of the *Update*. Following are highlights of the key tasks performed during these entries:

Entry 6

This entry was conducted over a two-day period, February 3 and 4, 1981. Eight closed circuit TV cameras were installed; decontamination tests and photographic surveys of damage were conducted on the 347-foot elevation (see photo); radiation surveys were made; and various samples of paint chips and other material were obtained from the 347-foot elevation.

Entry 7

This entry was conducted over a three-day period, March 17, 18, and 19, 1981. Three one-liter samples and one 150-milliliter sample of the water in the containment building basement were obtained using a roto-flex pump. A zeolite column was installed and operated to obtain five gallons of processed effluent from the basement water for the Submerged Demineralizer System (SDS) development data (see SDS article, this issue). Detailed radiation and photographic surveys were conducted in the in-core instrumentation tunnel area to support the sump surface suction plan for the SDS. In addition, the first radiation surveys were conducted at the top of the CRDM service structure (see photo).

Entry 8

This entry was conducted on April 8, 1981. A photographic survey and a general reconnaissance of the area at the 305-foot elevation open stairwell were conducted. Closed circuit TV cameras numbers 4 and 7, which were installed in entry 6, were repositioned (see photo), and the power source for camera number 7 was changed.

Entry 9

This entry was conducted on April 30, 1981. The cover of penetration R-561 was removed in preparation for entry 10 decontamination testing. GPU's SDS sump pump was installed through the open stairwell, and a photographic history of the pump installation was made. A photographic survey of electrical penetrations R-504 and R-509 was made for the Instrumentation and Electrical program. A radiation survey of zeolite resin columns used in entry 7 was conducted. The scaffolding used to install the closed circuit TV cameras in entry 6 was dismantled.

Entry 10

This entry was conducted on May 14, 1981. Safety equipment was installed on accessible portions of the polar crane. Radiation and smear surveys were conducted on the control rod drive mechanism service structure internals. Entry team members obtained six water and two sludge samples from the containment basement using EG&G's Water and Sludge Sampling Device (see article, this issue). They also performed the first large-scale

Continued on Page 6

Technicians entering the TMI-2 containment building in one of a series of manned entries into the building.
(GPU Nuclear Photo by Don Shoemaker)

Entry Highlights

Continued from Page 5

decontamination experiment on the 305-foot elevation using an initial spray mist and a combination of low pressure and high pressure sprays. In addition, a post-decontamination experiment cleanup was performed.

Entry 11

This entry was conducted on May 28, 1981. Areas where large-scale decontamination experiments were conducted during entry 10 were protected from recontamination by using contamination control areas and procedures (see photo). Entry team members completed installation of polar crane safety equipment, transferred a portable gamma spectrometer into the reactor building, and obtained three floor scans on 305-foot elevation which included three area spectra and three background spectra. Technicians replaced radiation monitor HP-R-213 on the 347-foot elevation with a new instrument, and replaced the GAI-tronics paging telephones on the 305- and 347-foot elevations with new ones. The team members also performed radiation and photographic surveys of the pilot-operated relief valve and other general areas within the east D-ring (or biological shield) and connected SDS hoses to the R-626 penetration.

Entry 12

This entry was conducted on June 25, 1981. Closed circuit TV camera number 4 was replaced, and the connectors on camera number 7 were repaired. Entry team members performed maintenance and modification tasks on lighting panel LPR-3A and the GAI-tronics telephone system, they installed temporary lighting in the enclosed stairwell, and they performed smear surveys on the walls at the 305- and 347-foot elevations. Loose samples of peeling paint were obtained at the 305-foot elevation near core flood tank B, at an electrical box at the 347-foot elevation, and from the containment dome on the floor north of the open stairwell at the 347-foot elevation.

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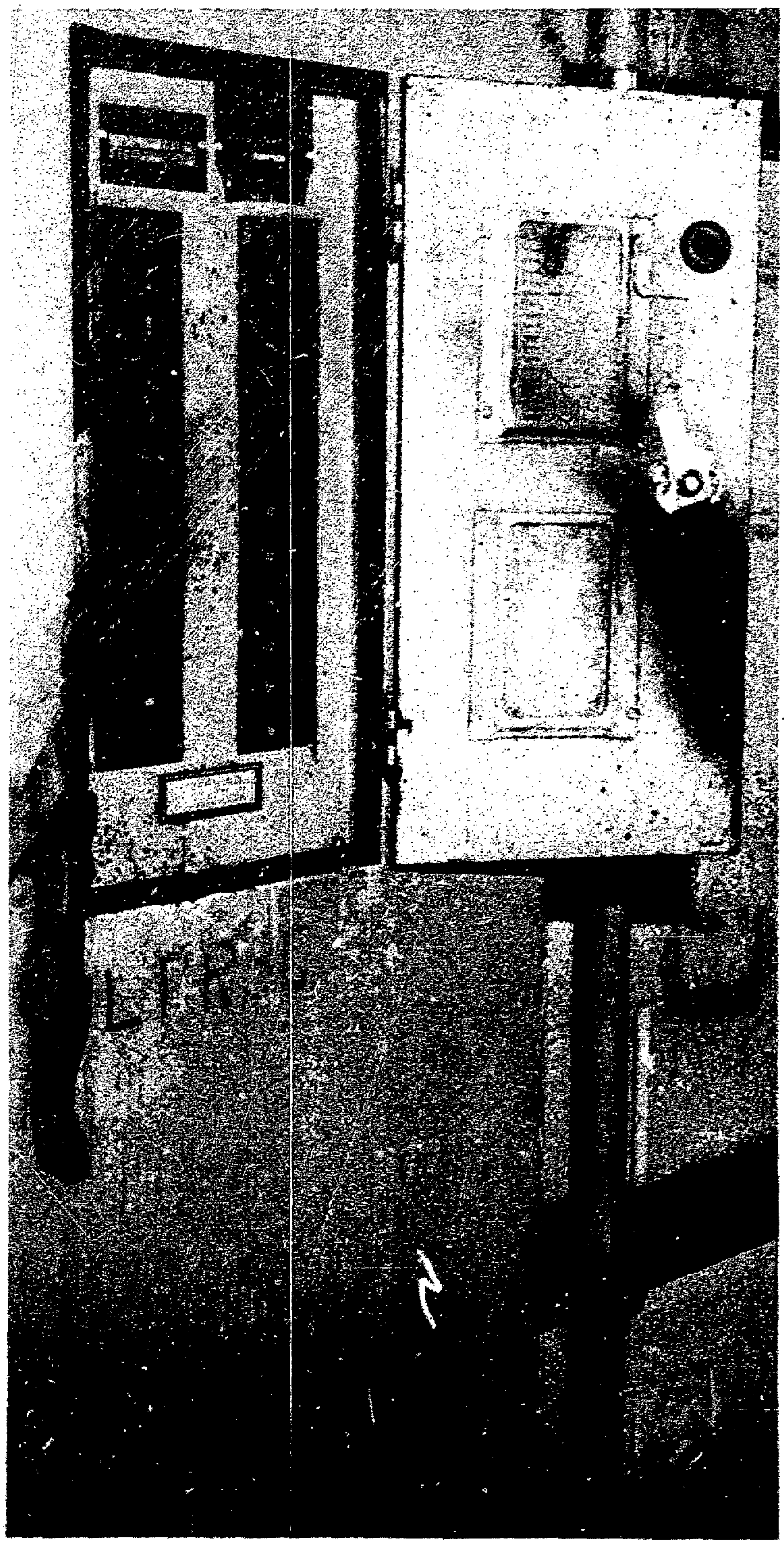
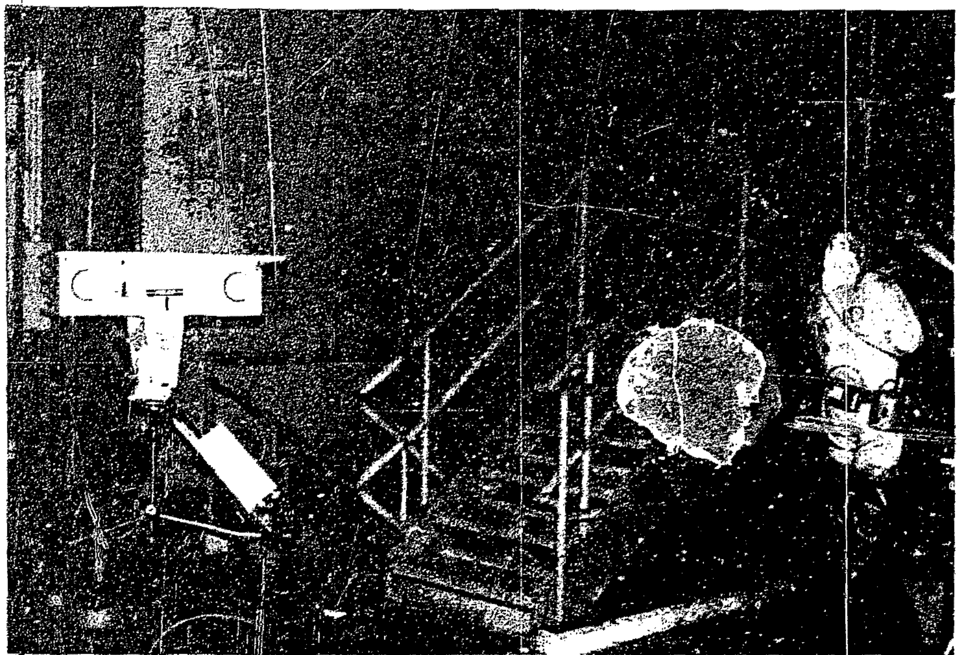
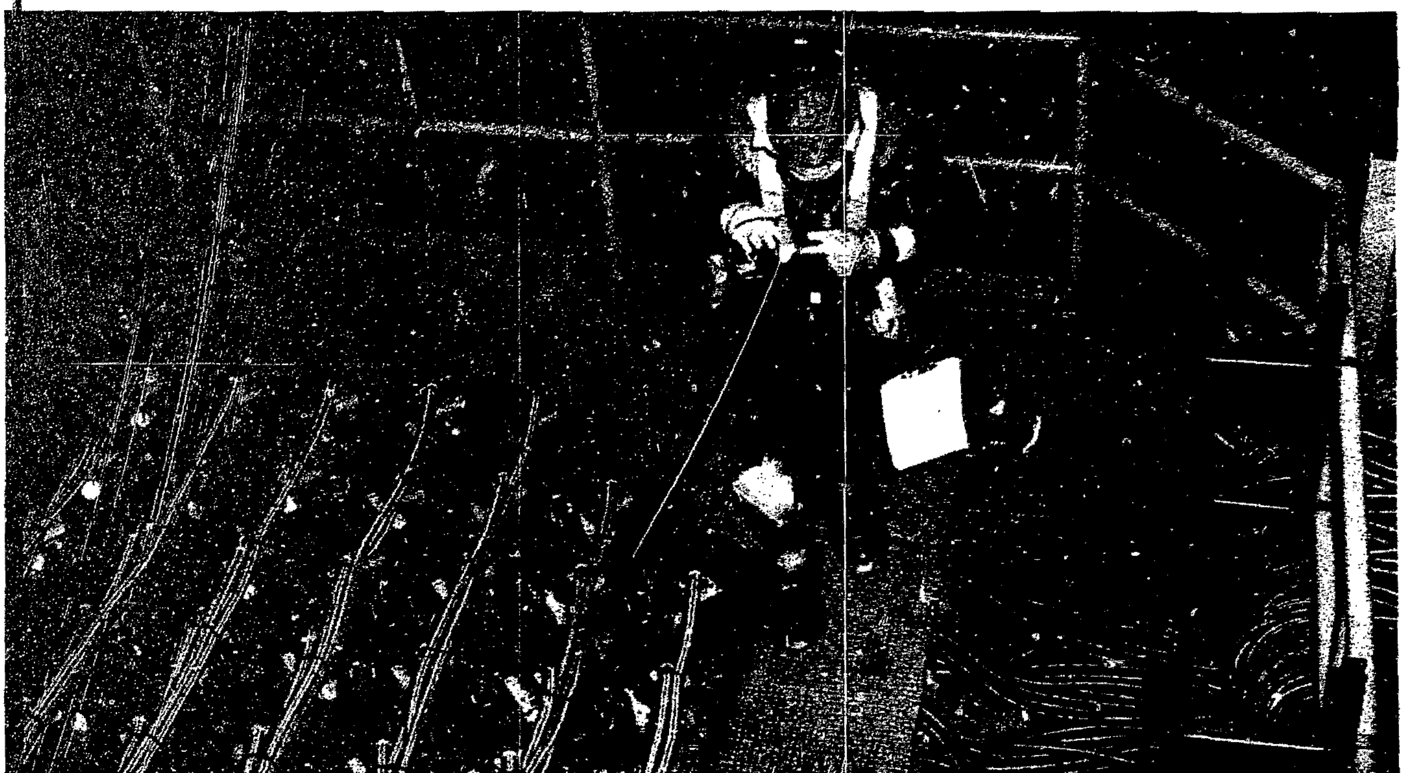




Closed circuit television camera number 4 at open stairwell on the 305-foot elevation. *(TIO Photo Entry 8)*



Physics technician performing the first radiation survey on the top of the control rod drive mechanism service structure. *(TIO Photo Entry 7)*

lighting panel LPR-3C on elevation.
to Entry 6)

Technicians remove outer set of protective shoe covers at contamination control area on the 305-foot elevation. *(TIO Photo Entry 11)*

Entry Highlights
Continued from Page 7

Entry 13
This entry was conducted on July 1, 1981. This entry was made to perform radiation surveys and to complete the polar crane inspection which had to be aborted during entry 12 due to problems with personnel airlock no. 2.

Entry 14
This entry was conducted on July 23, 1981. Closed circuit TV camera number 2 was replaced and connectors on camera number 8 were repaired. Team members obtained a 150-milliliter sample of the water under personnel airlock number 1 and a sample of the white crystal accumulation on the floor of the 347-foot elevation by the in-core instrumentation seal table. Water samples from the neutron shield tanks could not be obtained because the tanks are empty. Photographic surveys were taken of the air coolers and some selected instruments. Beta and gamma radiation and smear surveys were conducted on the reactor vessel service structure and the refueling pool floor. Entry team members removed core flood tank transducers CF1-PT4 and CF2-LT4 for analysis and installed a continuous air monitor and an area radiation monitor. In addition, Radiological Engineer Della Loggia became the first woman to enter the TMI-2 containment since the accident.

Entry 15
This entry was conducted on August 27, 1981. Spectra from scans of the floor on the 305-foot elevation were obtained by gamma spectrometry. Overhead beta and gamma and smear surveys on core flood tanks 1A and 1B and on platforms on the east side of the reactor building were also obtained. In addition, a remote radiation survey of the deep end of the refuel pool, a smear survey on the mezzanine, and a survey around the open stairwell were also obtained. One entry team inspected the steam generator cleaning line and obtained several photographs of the area. Another team replaced closed circuit TV camera number 8 with a new camera box and installed new wires and a control box on camera number 5. Reactor building nitrogen pressure alarm switch NM-PS-1454 was replaced, and flow transmitter MU-10-FT1 was removed. The last entry team performed air cooler inspections and took photographs of the motors. They also obtained some thermocouple readings.

Entry 16
This entry was conducted on September 24, 1981. Two teams photographed various penetrations and inventoried the defueling tools in the building. One of the teams inspected the air cooler fan motors, removed three fan motor covers, and obtained one resistance temperature device reading. The third, team obtained a sump water sludge sample at the open stairwell using EG&G's Water and Sludge Sampling Device (see article, this issue). The fourth team performed containment characterization surveys on the 347-foot elevation.

Department of Energy Ships EPICOR II Resin Canister To Research Facility

An EPICOR II resin canister from a contaminated-water treatment system at TMI-2 was shipped to Battelle Columbus Labs (BCL) in West Jefferson, Ohio on May 19, 1981. The Department of Energy (DOE) coordinated the shipment of the resin canister to the Ohio research facility.

The canister, a prefilter referred to as the PF-16 liner, is one of a total of more than 50 highly loaded EPICOR II liners used in processing contaminated water in the auxiliary building at the damaged Unit 2 reactor. DOE will sponsor research to determine the condition of the highly loaded resins and liners after they have been stored for long periods. The selected liner was used March 3 and 4, 1980 to process 8250 gallons of contaminated water from the auxiliary building. The PF-16 is one of the most highly radioactive resin liners used in the EPICOR II system, with a loading of approximately 1300 curies of cesium-137 and strontium-90.

Loading the EPICOR II resin canister for shipping.

Researchers at BCL have begun a variety of tests on the liner. These tests include resin sampling analysis, gas and liquid sampling analysis, visual examination of the liner, and various other studies of its chemical and radiological makeup. The tests will continue over several months; analytical results on these studies will be published in future issues of the *Update* as data become available.

BCL analysis of the PF-16 liner will contribute to the development of technology for storing, processing, and disposing of contaminated resin liners. Some specific goals of the program include acquisition of data for:

- Developing short-term storage requirements for such liners
- Developing storage canisters and disposal requirements for permanent burial
- Determining the effects of long-term storage on these resins and canisters
- Developing other options for processing the resins

The PF-16 liner was shipped to Battelle in a high integrity shielded cask mounted on a low-boy tractor trailer (see photo). The liner is 48 inches in diameter and 60 inches high, and contains approximately 32 cubic feet of ion exchange media. It was shipped in a licensed type B cask, 92 inches high and 85 inches in diameter. The cask walls consist of two one-inch layers of steel separated by three and one half inches of lead. The cask was designed to resist extreme environmental stresses such as fire and immersion in water.

Although the PF-16 liner was the first highly loaded resin canister to leave the island since the 1979 accident, General Public Utilities (GPU) shipped 22 low level radioactive resin canisters from the EPICOR II system to a burial site in Hanford, Washington between April 22 and June 28, 1981. The last canister shipment arrived in Hanford on June 30, 1981. The radiation levels of these shipments were lower than those of routine low level wastes from other nuclear power plants.

PF-16 liner leaving TMI for characterization at Battelle Columbus Labs. *(GPU Nuclear Photo)*

TMI-2 GEND Reports Available to the Public

In the continuing effort to distribute information about the TMI-2 cleanup and recovery effort to the nuclear industry, twelve reports on various aspects of the Technical Information and Examination Programs (TI&EP) have been published. A brief description of each of these reports is offered below, along with the formal report title, its number, and its date of publication. These reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, Tennessee 37830.

GEND Planning Report. GEND 001, published October 1980. The report describes overall plans for the Technical Information and Examination Programs as established by the GEND group: General Public Utilities, the Electric Power Research Institute, the Nuclear Regulatory Commission, and the Department of Energy.

Facility Decontamination Technology Workshop' November 27-29, 1979, Hershey, Pennsylvania. GEND 002, published October 1980. This report provides a record of decontamination and dose reduction activities at other facilities. The report is in the form of published proceedings of the decontamination technology workshop.

TMI-2 Information and Examination Program Technical Integration Office Annual Report. GEND 003, published February 1981. The annual progress report discusses activities conducted under the DOE portion of the TI&EP during FY-1980.

Interim Status Report on Personnel Dosimetry. GEND 004, published June 1981. Dosimetry studies documented in this report surveyed available dosimeter systems, set up a prototype system, and compared the prototype with the commercial systems.

Characterization of the Three Mile Island Unit 2 Reactor Building Atmosphere Prior to the Reactor Building Purge. GEND 005, published May 1980. Samples of the TMI-2 containment atmosphere taken prior to the krypton-85 venting were analyzed for radionuclide concentrations and for gaseous molecular components. The sampling procedures, analysis methods, and results are summarized in this report.

Three Mile Island Unit 2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling. GEND 007, published May 1981. The report summarizes TMI-2 core damage analytical assessments performed by reconstructing the sequence of events, by estimating the amount of hydrogen generation, and by evaluating the amount of fission products released.

Report on Citizens Radiation Monitoring Program. GEND 008, published July 1981. The Citizens Radiation Monitoring Program developed a system for citizens to independently measure radiation levels in and around their communities. The report describes the program and its results.

Measurements of I-129 and Radioactive Particulate Concentrations in the TMI-2 Containment Atmosphere During and After the Venting. GEND 009, published April 1981. The report discusses the equilibrium concentration and species distribution during and after the reactor building krypton-85 venting. Concentrations of iodine-129, krypton-85, cesium-134, cesium-137, and strontium-90 were measured during the venting operation and are reported here.

In-Vessel Inspection Before Head Removal (Conceptual Development). GEND 010 PHASE I, published August 1981. This first phase of a three-part report deals with conceptual development of the core inspection project. Concepts are described for internal inspection of the reactor vessel and fuel assemblies prior to removal of the reactor vessel head.

In-Vessel Inspection Before Head Removal (Tooling and Systems Design). GEND 010 PHASE II, published July 1981. This Phase II report discusses designs of the concept, procedures, and tooling descriptions presented in the Phase I report. Preliminary procedures for beginning the work are also presented.

Canister Design Considerations for Packaging TMI-2 Damaged Fuel and Debris. GEND 011, published October 1981. This document reviews requirements and provides design concepts for a standardized canister for packaging damaged fuel and core debris.

TMI-2 Reactor Building Purge—Kr-85 Venting. GEND 013, published March 1981. A comprehensive technical report is presented on the total effort involved in decontaminating the reactor building atmosphere by venting the contained krypton-85 to the environment.

Accountability Study for TMI-2 Fuel. GEND 016, published May 1981. The Accountability Study considers problems of identifying, measuring, and accounting for TMI-2 fuel in its present condition and as it is removed from the core and examined. The study identifies methods which will provide a material balance equal to the pre-accident balance.

Technical Integration Office Reorganized

The Technical Integration Office at DOE's Three Mile Island Site Office has been reorganized to accomodate the expanded research effort of the program over the next several years. The program has been separated into two major areas of activity: the Data Acquisition Program and the Waste Immobilization and Reactor Evaluation Program. Both of these programs fall under the overall title of the TMI-2 Technical Information and Examination Programs, or TI&EP.

Data Acquisition Program activities will include the Configuration and Document Control Center, the Instrumentation and Electrical Program,

Continued on next page

the Radiation and Environment Program, the Off-Site Core Examination Program, and the Radwaste Technology Development Program. The Waste Immobilization Program will conduct zeolite and resin disposition studies; and the Reactor Evaluation Program will conduct core damage assessment, reactor disassembly studies, and fuel and core storage and disposal research and development. The reorganization became effective October 1, 1981 with the start of the new fiscal year.

TMI Unit-2 Technical Information & Examination Program

UPDATE

The *TI&EP Update* is issued by the EG&G Idaho, Inc., Configuration and Document Control Center at Three Mile Island under contract DE-AM07-76IDO1570 to the Department of Energy, P.O. Box 88, Middletown, PA 17057. Telephone 717 948-1050 or FTS 590-1050.

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TMI Unit-2 Technical Information & Examination Program

UPDATE

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Middletown, PA 17057

Inside:

- *Submerged Demineralizer System*
- **First Multi-Level Sump Sample**
- **Preliminary Polar Crane**
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- **TIO Reorganized**
- **DOE Ships EPICOR II Resin Canister**

TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 3, Number 1 November 1, 1982

Initial Quick Look Conducted to Assess Unit 2 Core Damage

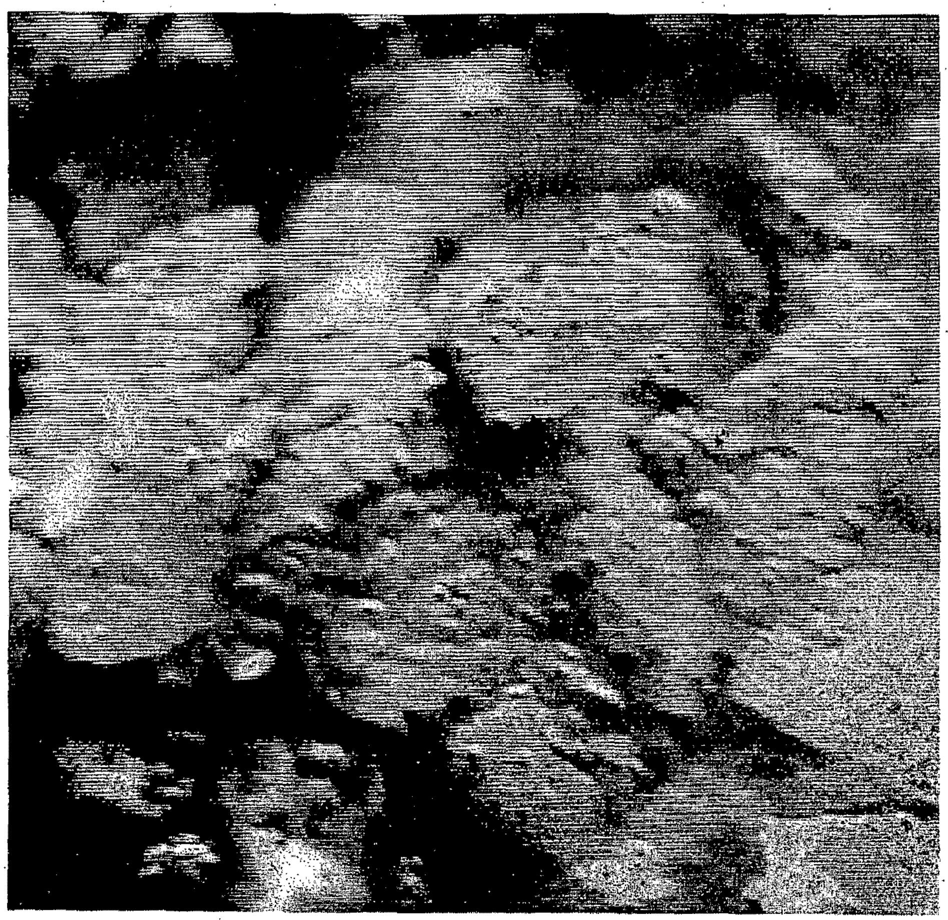


Figure 1 **Closed-circuit television closeup view of the rubble bed.**

Technicians uncoupled and removed the leadscrew from a control rod drive mechanism (CRDM) and on July 21, 1982, inserted a miniature radiation-resistant television camera down through the motor tube. As a result of this effort, research and recovery engineers now have black and white videotapes of the internals of the damaged TMI Unit 2 reactor.

In the TMI-2 operations, the leadscrew was removed from the center CRDM. The television camera inserted through this access extended to between 4-1/2 and 5 ft below the bottom of the plenum into the top of the core region, before reaching what is described by GPU Nuclear as the surface of a "rubble bed." While it is too early to speculate about the condition of the entire core until further examinations have been made at other core locations, this initial "quick look" did confirm predictions that the damage would be extensive at the upper center region of the core. *Figure 1* shows the television view of a small area of the "rubble bed." *Figure 2* shows a portion of a control rod spider resting on top of the "rubble bed."

The small area scanned with the camera showed that at least the top 5 ft of the fuel assemblies in the center region had become a bed of rubble. However, the upper plenum structure appeared to be intact and not significantly damaged. Throughout the underwater inspection, flakes of light-colored material frequently swirled in front of the camera lens during the movement of the camera, especially in the upper plenum region.

The 24-ft-long leadscrew that was removed to provide camera access will be shipped to the Idaho National Engineering Laboratory (INEL) for examination by EG&G Idaho, DOE's contractor for the TI&EP. The metallurgical structure of the leadscrew will be examined at the INEL in support of efforts to determine accident temperatures.

Published by EG&G Idaho, Inc., for the U.S. Department of Energy

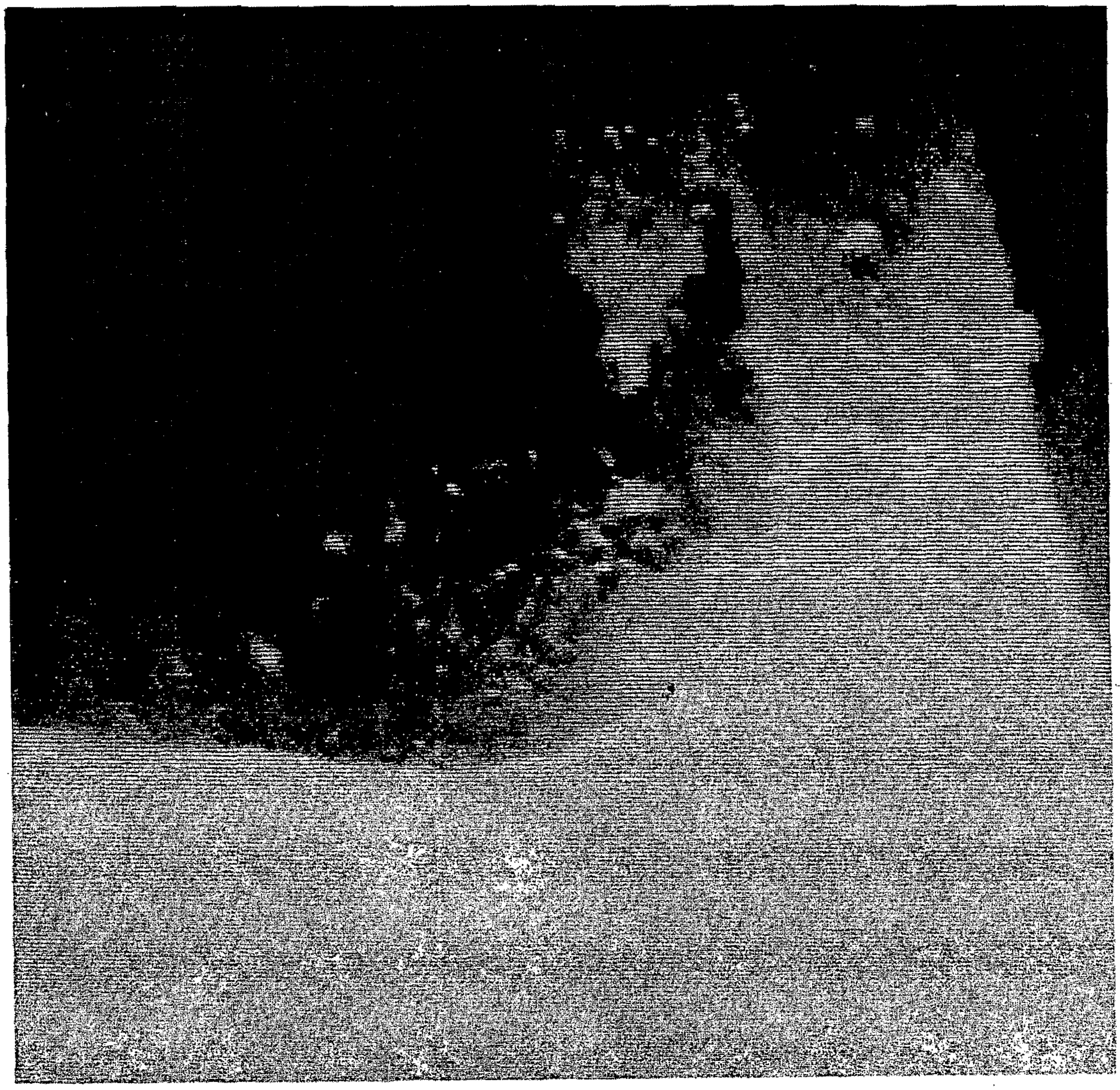


Figure 2 **Closed-circuit television closeup view of a CRDM spider resting on top of the rubble bed.**

During the quick look, data samples of vented gas and reactor coolant liquid were obtained and are being analyzed. Altogether, the observations and samplings are providing early data on:

- The relative quantity and distribution of core debris in the plenum assembly
- Thermal distortion or other structural damage in the plenum assembly
- The condition of the core, particularly relative to debris bed formation
- The physical condition of control rod couplings.

The assessment will offer the first concrete information about core damage, providing a benchmark for previous core damage estimates that have varied widely. Further careful examination of the videotape and results of the sample analysis, when considered together, will offer the first direct assessment of TMI-2 core damage and will provide some of the information necessary to engineer core removal tooling, canning, shipping, and damage assessment examination facilities.

Axial Power Shaping Rod Test Meets with Success at TMI-2

From June 23 to 25, 1982, the Department of Energy and General Public Utilities Nuclear Corporation combined forces with a number of contractors and support engineering firms to conduct the first Axial Power Shaping Rod (APSR) insertion tests at the damaged TMI-2 reactor. The eight 12-ft-long APSRs control efficient use of the fuel during normal plant operations. They are not part of the reactor safety system and so did not insert automatically into the core at the time of the accident, as did the other 61 control rods.

The APSRs have been positioned approximately 3 ft above the full "in" position since the accident; the insertion tests determined the mechanical motion of the rod drive systems. Two rods inserted the full 3 ft into the core; two of them inserted to within about 7 in. of the full "in" position; two rods moved in less than 7 in.; and two did not move in at all, although their drive rotor assemblies did latch and unlatch properly and showed minor rotational movement. Because normal rod movement procedures were not considered possible as a result of accident damage, the TI&EP-supported program operated the APSRs using an auxiliary power supply and control devices totally independent of the normal operating controls.

The APSR test has two objectives. First, engineers will be able to gain insight into the extent of core and upper plenum damage by studying the data obtained during the tests. This knowledge will be factored into plans for subsequent inspections, head and upper plenum removal, and core removal. Second, test planners wanted to insert the APSRs as far into the core as possible to facilitate head removal operations. The APSR leadscrews must be uncoupled prior to head removal, and the uncoupling is most easily accomplished when the APSRs are fully supported at the bottom by a resistance greater than the downward force needed to uncouple the leadscrews.

Insertion testing of an individual APSR followed a basic sequence. First, using the auxiliary portable service power supply, the APSR leadscrew was latched to the roller nut of the drive rotor assembly. Next, engineers attempted to slowly move the APSR assembly a total of 3/16 in. outward. They then attempted to slowly insert the assembly 3/8 in. into the core. If initial inward motion succeeded, engineers "jogged" the assembly inward, instead of inserting it rapidly, to provide maximum motor torque and control on the leadscrew.

During each step, acoustical and electrical outputs provided evidence of APSR movement or jamming. Emissions from recently installed acoustic monitors on the APSR drive mechanisms helped to verify the mechanical motion of the rod drive system. Engineers were able to correlate noise during APSR movement with relative positions of the leadscrew to upper plenum brazement plates.

Table 1 **Positions of APSRs after insertion testing**

Rod Number	Location in Core[a]	APSR Position[b] (%)
62	F-4	5.2
63	L-4	18.8
64	N-6	25.0
65	N-10	0.0
66	L-12	4.2
67	F-12	1.1
68	D-10	22.9
69	D-6	26.1

a. See Figure 3 for map of the core.

b. Percentage of rod not inserted into core.

Electrical output from the drive motors indicated electromagnetic "pole slippage" whenever the APSR encountered an obstacle and the leadscrew stopped turning. This pole slippage occurred when the position of the rotor turning the leadscrew would fall out of synchronization with the electrical field of the driving stator. The lack of synchronization indicated to engineers that the APSR was no longer traveling down into the core. When jamming occurred at any juncture in the test sequence, controlled increases in stator current and axial force were made, from the minimum of 500 lb up to the maximum 1400 lb of force, the normal operating force for APSRs. In this way each APSR was either completely inserted or driven in until supported by a resistance greater than the 1400-lb downward force the auxiliary power supply could provide.

The distance from full APSR removal (100%) to full rod insertion (0%) is approximately 139 in. Before the testing, all the rods were at 25%, with approximately 35 in. extending above the top of the core. When testing ended on June 25, the positions of the eight APSRs were as shown in *Table 1*. The configuration of maximum to minimum inserted rods did not follow any observable pattern, although further study may show some relationship between APSR movement and apparent internal core damage indicated by other data gathered at TMI-2. *Figure 3* shows the locations of the APSRs and the 61 control rods in a cross-sectional view of the TMI-2 reactor vessel internals.

Although the TMI-2 APSR insertion tests were successful, it is not possible to draw firm conclusions about the condition of the entire core from these test results. Engineers will study data from the sound recording and electrical monitoring devices to build a clearer picture of conditions in the core. The APSR test is a major step in an extensive inspection and examination program planning for safe removal of the fuel from the TMI-2 reactor.

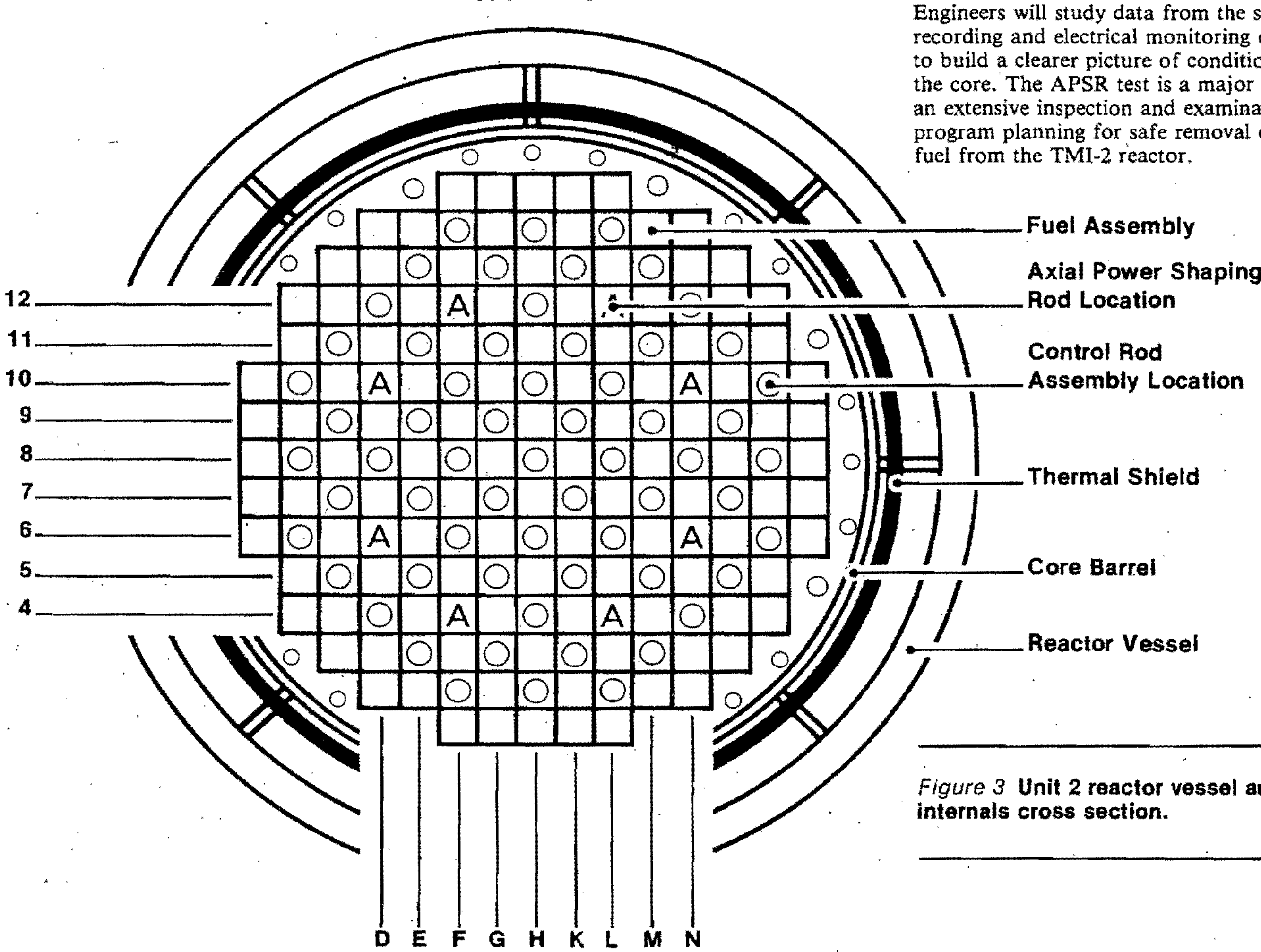
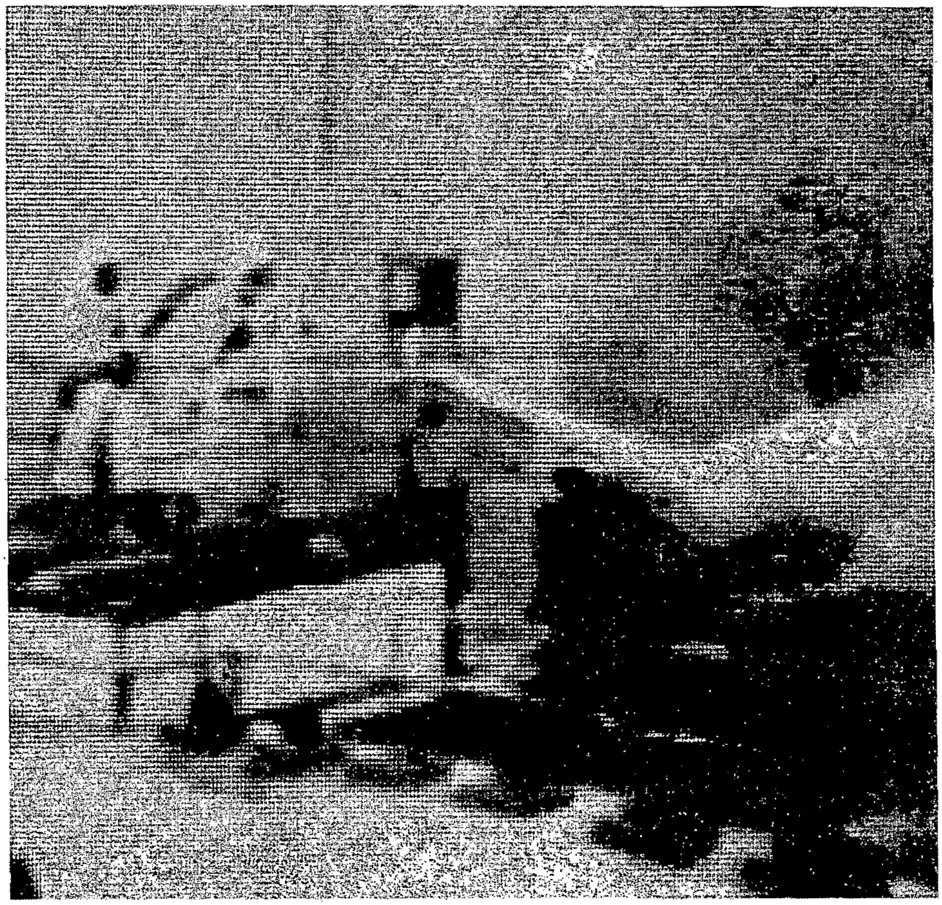


Figure 3 **Unit 2 reactor vessel and internals cross section.**

Gross Decontamination Techniques Tested in Experiment at TMI-2

Preliminary data results from the gross decontamination experiment completed March 24 in the TMI-2 Reactor Building look encouraging. Beginning with the first preparation entry in September 1981, the 6-month, 43-entry experiment culminated in 3 weeks and 11 entries of actual testing of several decontamination methods on a variety of surfaces. The purpose of the experiment was to determine the effectiveness, safety, application rates, and efficiency of several gross decontamination techniques.

The predominant technique used in the decontamination experiment was low- and high-pressure water spraying of floors and walls—hydrolasing—a process not so severe as to strip the epoxy paint from surfaces nor drive and embed contaminants into them, yet forceful enough to dislodge contaminant-bearing particles of rust and other debris. *Figure 4* shows technicians hydrolasing the top of the D-rings. Water was sprayed over surfaces from a blaster lance (see *Figure 5*) with various size fan nozzles under pressures ranging from 2,000 to 6,000 psi and temperatures from ambient to 140°F. The portable high-pressure pump and temporary hot water heater system located outside of the Reactor Building are capable of supplying water heated to 140°F at flow rates up to 25 gpm and pressures up to 10,000 psi (see *Figure 6*).



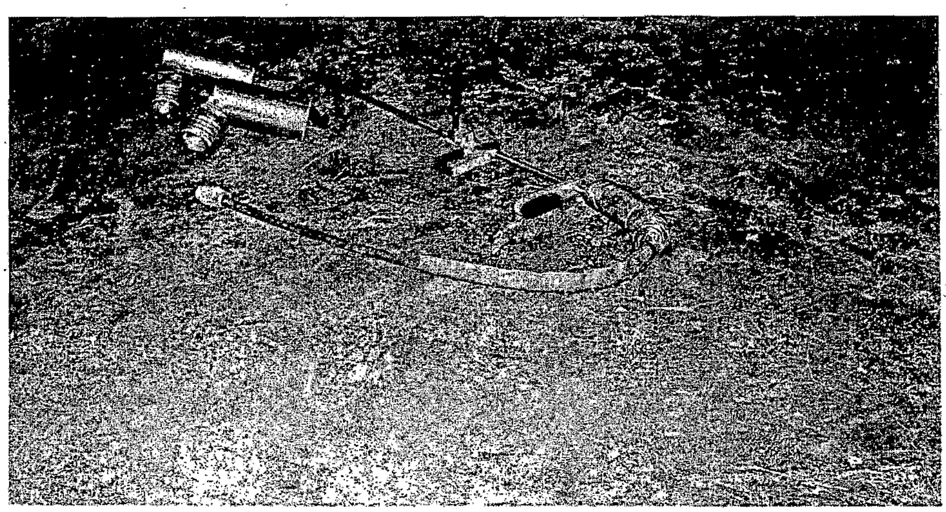


Figure 4 **Technicians hydrolasing the top of the D-rings.**

Figure 5 **Hydrolasing blaster nozzle and safety equipment.**

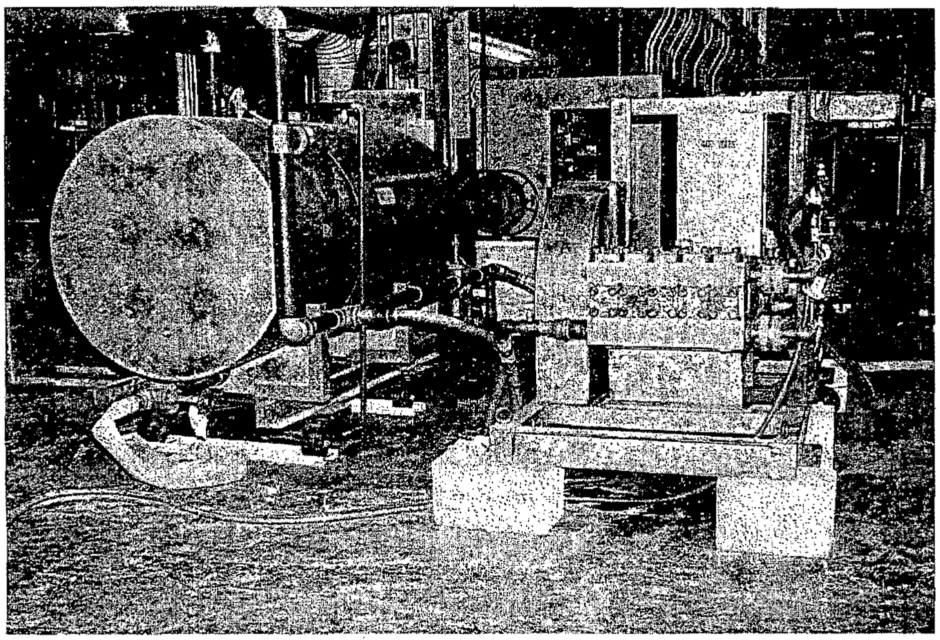


Figure 6 **Gross decontamination experiment hot water heater and high pressure pump.**

Low-pressure flushes preceded the high-pressure hydrolasing in an attempt to wash the bulk of contaminant-bearing particulates and debris into drains that empty into the Reactor Building basement. The wash water was then processed by the Submerged Demineralizer System. The water used in the experiment was decontaminated accident-generated water, which was processed by the EPICOR II system in the early months after the accident. The wash water was borated and also contained trace amounts of tritium.

Initially, surfaces were misted by a fine water spray to stabilize loose contamination, thereby minimizing airborne contamination caused by the force of the high-pressure spraying. This procedure was suspended when airborne contamination levels indicated only a slight increase during periods of spraying—whether surfaces were misted or not. Of more concern than airborne activity caused by spraying was the problem of recontamination by the inadvertent splashing of areas previously washed. In order to reduce the magnitude of recontamination caused by spraying, detailed procedures and spray sequences were developed.

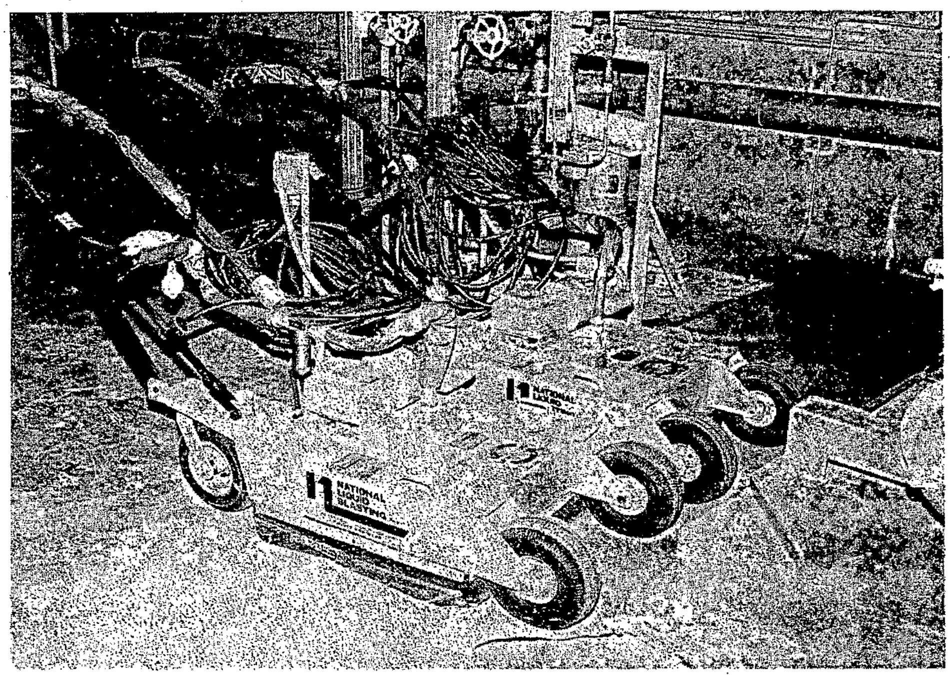


Figure 7 **Gross decontamination experiment spinjet sprayers.**

Other methods in the gross decontamination included the use of wheelmounted spinjets, mechanical floor scrubbers and detergents, and strippable silicon coatings. The spinjets look much like domestic lawn mowers with water jets instead of blades (see *Figure 7*); they provide easy mobility and an even spray application over floors. The mechanical scrubbers are similar to floor buffers, but are equipped with abrasive pads, apparently highly effective in removing contaminant-bearing rust. However, because the chemical detergents used with the scrubbers might have affected the functioning of the Submerged Demineralizer System, a specially designed vacuum gathered the detergents into barrels after scrubbing. This wet/dry vacuum picked up contaminated material without contaminating the basic internals of the vacuum equipment, using a series of special throwaway filters. The mechanical scrubbers proved to be a very effective decontamination technique and will undergo further testing using zeolite and abrasive pads without detergents.

The strippable silicon coatings were applied in a liquid from the seeped into pores and cracks and around uneven surfaces yet dried into a highly self-binding sheet that could be pulled up in one intact piece, holding radioctive particles (see *Figure 8*). They were especially effective on surfaces that would be otherwise inaccessable. Each method proved effective for its special use in the overall decontamination experiment.

Areas included in the gross decontamination experiment were the reactor building dome, the 500-ton polar crane, the walkways on top of the two D-rings, the refueling canal, and the top of the reactor vessel missile shields. Also decontaminated were large tools, equipment, and floor surfaces on the operating deck or 347-foot elevation, and overhead areas, walls, and floors on the entry or 305-foot elevation.

Figure 8 **Technician removes strippable coating from Reactor Building floor.**

Aiding transport of equipment and personnel during the experiment were a scissors lift platform and a "spider" lift installed during predecontamination entries. The scissors lift is a device to lift equipment and personnel approximately 19 ft off the reactor building entry elevation to permit decontamination of overheads and walls. The spider lift, named for its ascent and descent on a cable, allows personnel and equipment direct access from the 347-foot elevation to the polar crane. The lift is needed because the crane is not parked in its normal position and thus prevents normal access. *Figure 9* shows technicians using the spider lift to ascend to the polar crane.

Other activities performed in support of the gross decontamination experiment were the wrapping for protection of approximately 55 instrumentation and electrical components, the removal of selected radiation monitors, and the acquisition of pre- and postdecontamination data. Pre- and postexperiment data acquisition included:

- Placing and collecting of approximately 100 thermoluminescent dosimeters at selected Reactor Building locations (see article this issue)
- Obtaining some 200 loose particulate, concrete, metal, cable, and damaged item samples
- Collecting gamma spectrometer measurements
- Measuring contact beta and gamma radiation levels using a portable radiation instrument
- Conducting comprehensive smear surveys
- Collecting air samples
- Performing area characterization photographic surveys.

The overall general area dose levels in the building showed only small median reductions as a result of the experiment because of various high radiation sources such as the water remaining in the basement (see article this issue). However, the preliminary results of the experiment show that floor surface smearable contamination levels could be reduced to the order of 10^4 dpm/100 cm^2 of ^{137}Cs or less by the use of water in a combination of low- and high-pressure application. As anticipated, the hotter the water and the higher the pressure, the better the results from hydrolasing; however, recontamination by splashing also increases. Use of mechanical floor scrubbing with detergents could further reduce the floor surface smearable contamination to the order of 10^3 dpm/100 cm^2 of ^{137}Cs. Smearable surface contamination was reduced in a range of 64 to 93%.

The experiment has shown that consistent reductions in smearable levels in the 90% range are achievable with the appropriate sequence and application of techniques. The data now being evaluated are augmented by photographs and videotapes of the decontamination activities, which will contribute to the first comprehensive documentation of gross decontamination in a commercial reactor facility.

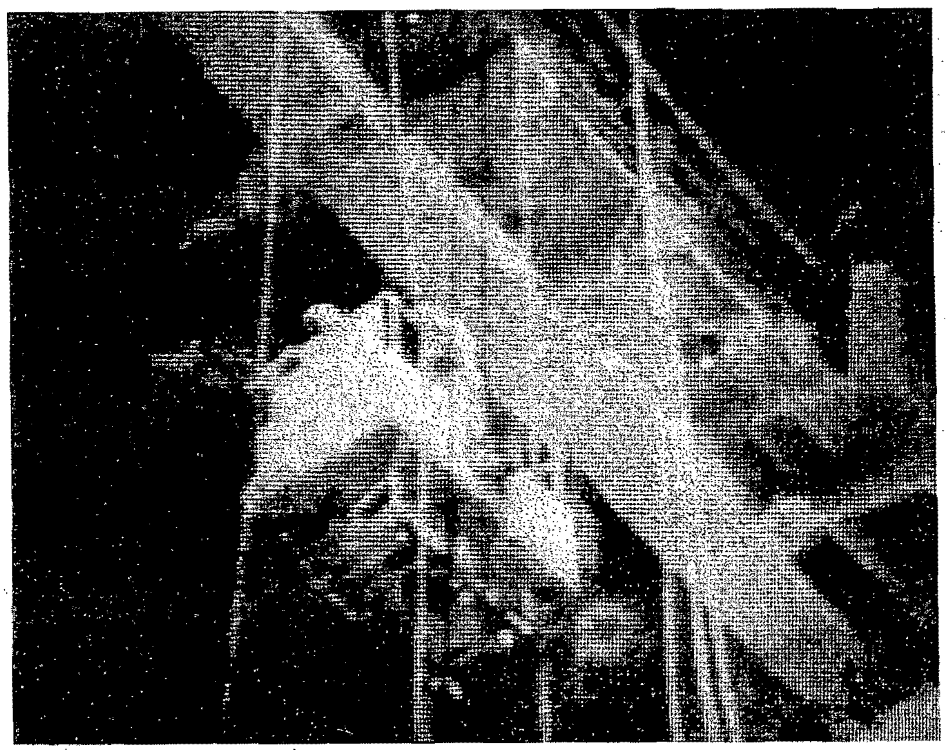


Figure 9 **Technicians ascending to the polar crane on the spider lift.**

Dose Levels Reduced in TMI-2 Reactor Building

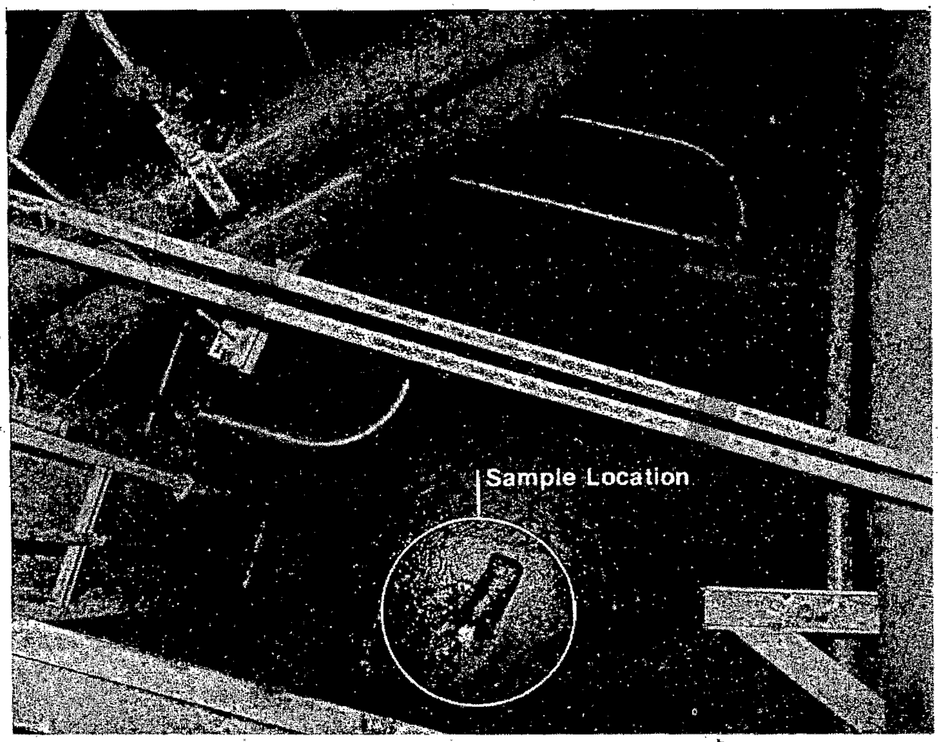


Figure 10 **Basement floor one level below 305-foot elevation after sample was taken.**

Removing contaminated water from the basement of the TMI-2 Reactor Building eliminated a major source of high radiation that contributed to fields found in the building as a result of the March 28, 1979 accident. This water removal, in conjunction with the TI&EP-supported gross decontamination experiment, (see article in this issue) reduced radiation dose levels substantially. The dose rate reductions will, among other benefits, allow cleanup workers to remain in the Reactor Building environment for longer periods of time.

In September 1981, GPU Nuclear technicians and engineers began pumping water from the Reactor Building basement using a surface suction technique. A submersible pump attached to a raft removed approximately 600,000 gal for processing through the Submerged Demineralizer System water processing system. Removing the 600,000 gal lowered the water level from 8-1/2 ft down to approximately 6 in., leaving an estimated 30,000 gal. At this level, the pump's snorkel tube touched the floor causing the pump to tilt and lose suction.

While removing the 600,000 gal of water did seem to reduce levels slightly, the remaining 30,000 gal were still emitting radiation into the building. Before and during the initial removal, the shielding effects of the water itself prevented radiation emissions from below the top 16 to 18 in. of water from reaching the surface to be measured by detectors. As water was being pumped out of the basement, radiation readings taken at the 305-foot elevation decreased slightly, primarily because the readable source was moving farther away. Since the detectors had only been able to measure the top "layer" of the water at any given time, the dose levels from the remaining water were still 2/3 of the original readable dose. Basement walls, composed of unpainted cinder blocks and cement that readily absorbed radioactivity, were no longer shielded by water and also contributed to dose level readings.

In March 1982, GPU Nuclear, assisted by TI&EP personnel, began the gross decontamination experiment. The experiment was designed to reduce contamination on selected surfaces in the Reactor Building. Depending on the decontamination techniques used and the particular areas involved, preliminary experiment results showed that dose rates had dropped significantly at several locations. However, the overall radiation dose levels remained near the pre-experiment levels. This was due in part to the contaminated water still in the basement and in part to other sources which as yet have not been identified.

In May 1982, entry team personnel installed a specially adapted jet pump into a depression in the basement floor under the reactor vessel to remove the remaining 30,000 gal of basement water. Two draindowns with this pump were required to draw off the remaining water for processing through the Submerged Demineralizer System (SDS). After completion of the last draindown, an engineer from one of the entry teams visually inspected the basement. By shining a light onto the floor from the first landing below the 305-foot elevation he could see a silt-like material on the floor covered by 1/4 to 1/2 in. of water. From the bottom landing, he scooped up a sample of this silty material and placed it into a container for later analysis. *Figure 10* shows the sample location on the basement floor.

Removal of the last 30,000 gal of water significantly reduced radiation dose levels inside the Reactor Building. Two fixed-point surveys that best represent the impact of the contaminated water were made at the open stairwell leading to the basement, and at a hatchway exposed to the basement. Readings taken at these points before and after removal of the last few inches of water showed reductions from 4000 to 2200 mR/h and from 6000 to 2500 mR/h, respectively.

Purifying the contaminated basement water was accomplished in two stages using ion-exchange technology. Initially, the SDS, containing inorganic zeolite ion-exchange media, effectively removed 99% of the fission products (See article in November 31, 1981 *Update*). The products, primarily ions of cesium and strontium, were chemically exchanged for ions of sodium. EPICOR II, a system containing organic and inorganic ion-exchange media, removed 99% of the fission products remaining in the SDS effluent.

Table 2 **Curies of radioactive isotopes removed from Reactor Building basement water down to a level of about 1 ft**

Element	Ion-Exchange System SDS (Ci)	EPICOR II (Ci)
^{137}Cs	210,000	2
^{134}Cs	22,600	2
^{90}Sr	7,331	19
Other[a]	--	23

a. Primarily ^{125}Sb, ^{144}Cm, ^{60}Co.

Table 2 shows the number of curies removed from the Reactor Building basement water down to a level of about 1 ft. Because of a chemical structure similar to water, tritium passes through the ion-exchange systems. The liquid effluent from the EPICOR II system, at this point, contained about 1,800 Ci of tritium. This tritiated water remains stored on the island in two 500,000-gal tanks and will be used for further decontamination work.

Additional calculations done after completion of the surface suction phase show that a total of 320,000 Ci were removed from basement water. The final volume of water removed from the basement will remain in SDS storage tanks until reactor coolant system water cleanup is completed.

The processing of about 100,000 gal of water from the reactor coolant system (RCS), a project underway since May 17, 1982, continues the dose reduction operations in TMI-2. Concentrations of radioactivity are continually reduced by a "feed and bleed" dilution process; water is "fed" into the RCS in 50,000-gal batches, while equivalent amounts are "bled" off. This method ensures that the damaged reactor core is always covered with water. The first batch contained about 45% of radioactivity measured in the system liquid. Each subsequent batch reduces the radioactivity concentration by a factor of about two. Original predictions were that seven "feed and bleed" batches (350,000 gal) would be needed to clean RCS water. The SDS is expected to adsorb about 15,000 Ci of cesium, strontium, and other elements. At the end of three batches, the SDS had processed 150,516 gal and removed 9,562 Ci of radioactivity from the RCS.

Completion of this project will reduce radiation dose levels in the Reactor Building even further. The fact that workers will be allowed to remain in the Reactor Building for longer periods of time, consistent with ALARA principles, will accelerate the successful defueling of the damaged reactor core.

Hydrogen Burn Damage Studies Continue at TMI-2

About 10 hours into the accident at TMI-2, the Reactor Building pressure stripchart recorder indicated a pressure spike of about 28 psig in the Reactor Building. Together with associated high temperatures and pertinent gas readings, this pressure spike led industry experts to conclude that an undetermined amount of hydrogen ignited and burned in the building. The apparent hydrogen burn is of interest as a resource for studying hydrogen generation mechanisms during a loss-of-coolant accident such as the one at TMI-2.

Two research efforts of interest to the TI&EP will provide information to aid the industry in resolving concerns associated with licensing plants under the new rules established by NRC following the TMI-2 accident. The research efforts will also provide information to aid the TI&EP's Reactor Evaluation Program in assessing damage to the TMI-2 core. Estimates of how much hydrogen burned during the accident can be compared with the known preaccident inventory of hydrogen. Researchers can then calculate how much zircaloy cladding had to oxidize to produce the hydrogen; which burn damage indicates existed in the building at the time of the accident. This zircaloy oxidation estimate provides one reliable basis for assessing the extent of damage to the TMI-2 core.

The first research effort, conducted by Dr. J. O. Henrie of Rockwell Hanford Operations, involves a preliminary study of instrument readings taken during the accident. According to Dr. Henrie's work, as much as 390 kg of the 450 kg of hydrogen present in the reactor building may have burned. As part of an overall data qualification being conducted, accident data obtained from other instruments will also be examined. All data will be qualified to assess how accurately they represent the actual phenomena observed in the plant. Specific data include Reactor Building pressure and temperature, steam generator secondary-side pressure, and building atmospheric samples. The qualification process will also assess the calibration history, accuracy, range, response time, and sample rates recorded by plant instrumentation. This evaluation will allows researchers to assign confidence intervals to instrument output and uncertainty intervals to all data.

Figure 11 **Damaged 55-gal barrels on the 347-foot elevation.**

The second study, conducted by N. J. Alvares and D. G. Beason of Lawrence Livermore National Laboratory and G. R. Eidam of the TI&EP Technical Integration Office (TIO) was published as a GEND report (GEND-INF-023, Volume 1). Entitled *Investigation of Hydrogen Burn Damage in the TMI-2 Reactor Building*, the study concentrated on analyzing the effects of the hydrogen burn in order to identify possible burn flame paths and areas of localized damage.

Reactor Building entry photographs taken as cleanup efforts at the damaged plant continue have been the primary source material for this second study. The authors studied photographs from the first 15 Reactor Building entries to attempt to determine possible flame paths, concentrations of damage evidence, and some possible temperature estimates for different Reactor Building locations. Most of the damage occurred on the operating deck, or 347-foot elevation, in the north, south, and east quadrants. The polar crane region exhibited burn damage also, while very little damage was noted on either the 305-foot (entry level) elevation or in the west quadrant of the 347-foot elevation. This study theorized that some areas received more damage than others because air flow from ventilation systems affected the path the burn took through the building.

Discussed below are some examples of the damage evident in the TMI-2 building caused by the hydrogen burn. Damage caused by a sudden increase in pressure appears to be restricted to areas near the elevator and enclosed stairwell complex on the 347-foot elevation. The elevator door and stairwell door both were bent out of shape, indicating a buildup of pressure to levels the door materials and structure could not withstand. These distorted doors, as well as the crushed or imploded barrels pictured in *Figure 11* could both have been affected by a pressure pulse, or possibly by heating followed by rapid cooling.

Indications of thermal damage, such as charring, melting, or actual burning of material were present almost exclusively in the north, south, and east quadrants of the 347-foot elevation. Wood items such as scaffolding, the plywood backing of a telephone table near the south wall, and boxes in the northeast quadrant all charred, in some places, heavily. However, no damage at all was evident on a wooden fence frame in the west area near the open stairwell.

Paper and fabric in the north, east, and south areas showed evidence of scattered local thermal damage. The paper maintenance manual on the fuel handling bridge pictured in *Figure 12* burned and crumpled. A fabric rag inside the head storage stand charred heavily, while a rag lying only a few feet away indicated no damage at all.

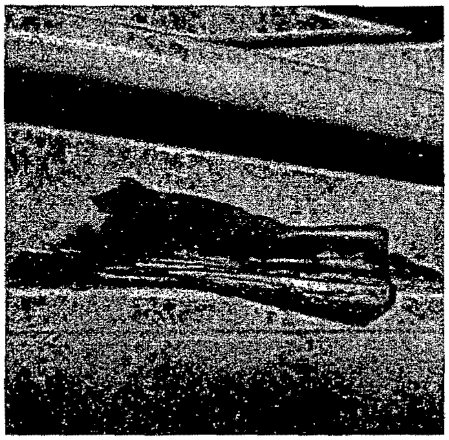


Figure 12 **Burned and crumpled maintenance manual on the fuel handling bridge.**

Plastic and polyethelene materials demonstrated the most dramatic evidence of thermal damage. Control panel buttons on the auxiliary fuel handling bridge melted out of shape, and telephone cord wire softened and lost its shape. A telephone on the 347-foot elevation, pictured in *Figure 13*, deformed from the heat; experimental results and manufacturer's data indicate that temperatures above 221°F would cause a telephone to deform under its own weight.

The locations of the damaged items throughout the 347-foot elevation might indicate that the burn followed a pathway caused by air flowing from the building air cooling system. Two loss-of-coolant accident (LOCA) vents convey a major portion of cooling system air to overhead regions of the Reactor Building on the south wall. (Both vents were to the right side of the polar crane position during the accident.) Circulation patterns during normal cooling operations draw air from the 347-, 305-, and 282-foot elevations and exhaust through the D-rings (personnel shields). During the accident the LOCA vent dampers were automatically opened following Reactor Building isolation. Since no record exists of operators manually closing the LOCA vent dampers, efflux from the air coolers may have moved through the LOCA dampers and discharged to the upper containment regions during much of the accident sequence. Cooling fan flow directions may account for burn patterns on the 347-foot elevation, and also on the polar crane in the Reactor Building.

The polar crane above the 347-foot elevation appeared to have widespread thermal damage, but no evidence of pressure-caused damage. The study described the damage as uniform, "as though all burned and melted materials were engulfed in flame or hot gas for a short period." In the crane operator's cab, the operator's chair and the instrument panel buttons were melted and charred. Thermal damage also was observed on bus bars (some of which fell to the 347-foot elevation), bus bar insulation, labels, hose, and ceiling paint. Thermal damage to polar crane components appears uniform. Discharge of the air coolers through the LOCA ducts may have been a primary dispersal mechanism of hydrogen and air to the polar crane region.

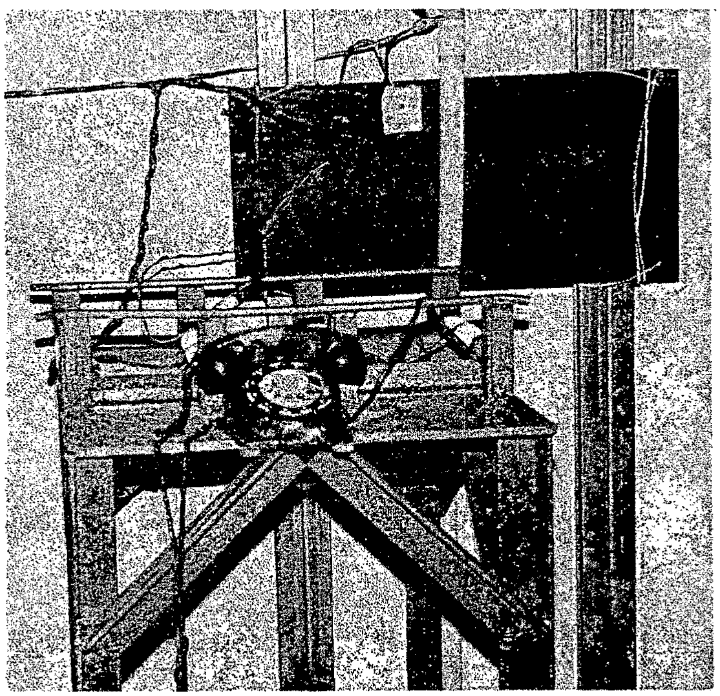


Figure 13 **Deformed telephone on the 347-foot elevation.**

There are almost no indications of hydrogen burn on the 305-foot elevation. One floor plate in front of the air coolers on the 305-foot elevation moved slightly from its normal position, possibly as a result of a slight pressure pulse in the basement region below the 305-foot elevation. On one telephone the cord coil relaxed, possibly as a result of the heat emitted from the enclosed stairwell nearby; the elevator control buttons also melted, possibly as a result of hot gas emission from the elevator shaft.

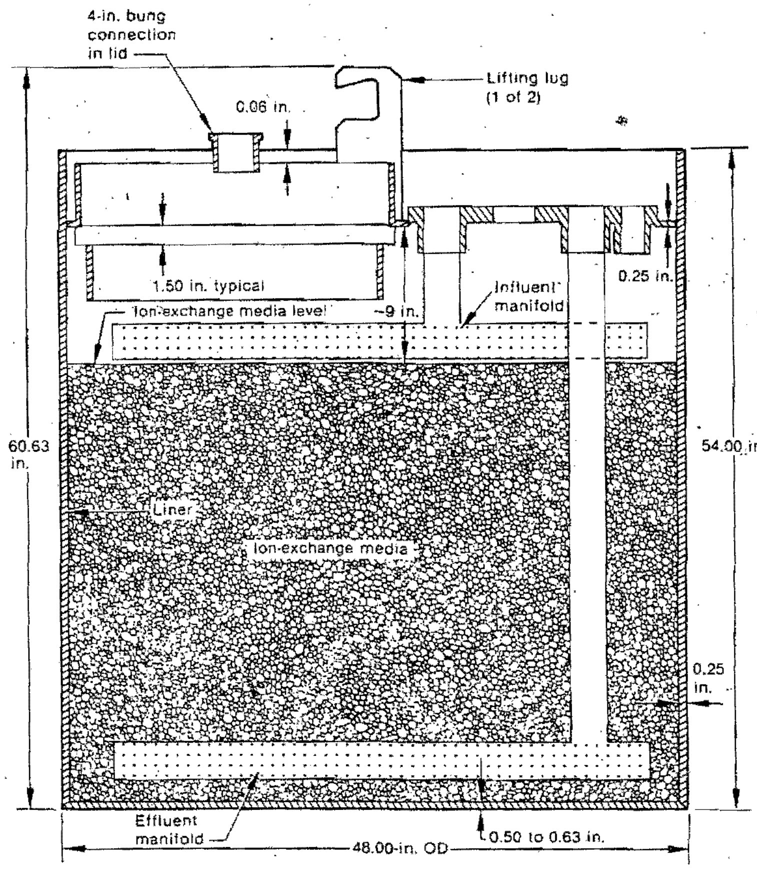
The two research efforts discussed above will provide information to aid the industry in understanding hydrogen burn control mechanisms. Assessing the extent of damage attributable to the burn, evaluating the building pressure and temperature response, and correlating the extent of damage with the amount of hydrogen burned will all contribute to a more complete understanding of the incident at TMI-2.

Characterization of EPICOR II Resin Canister PF-16 Complete

As part of DOE's research and development program, Battelle Columbus Laboratories (BCL) completed characterization studies on an ion-exchange media canister used to process TMI-2 accident water. Preliminary results indicate that the content of the ion-exchange media canister characterized was not extensively degraded as a result of being loaded with accident-generated waste. This characterization will contribute to the technology required for safe storage, processing, and ultimate disposal of highly contaminated ion-exchange media.

The canister studied is one of the 50 ion-exchange media prefilter canisters used in the EPICOR II water processing system at TMI-2. The EPICOR II system processed approximately 500,000 gal of highly contaminated accident-generated water that accumulated in the Auxiliary and Fuel Handling buildings during the accident. The processing generated prefilter canisters, such as the one sketched in *Figure 14*, highly loaded with predominately ^{137}Cs and ^{90}Sr. In order to determine what effects exposure to accident-generated wastes might have on this type of ion-exchange media and container, one of these prefilters, PF-16, was selected to undergo characterization. This liner was used on March 3 and 4, 1980 to process 8,250 gal of water from Reactor Coolant Bleed Tank "A." The PF-16 was considered one of the most likely prefilter liners to demonstrate deterioration because of its relatively high loading of 1,250 curies and low residual pH of 2.79.

The PF-16 was shipped to BCL on May 19, 1981 (See article November 30, 1981 *Update*) where characterization tasks were performed. After completing acceptance radiological surveys and cask internal gas sampling, technicians removed the shipping cask lid and hoisted the liner into the heavy element hot cell using a shielded transfer and storage device pictured in *Figure 15*.



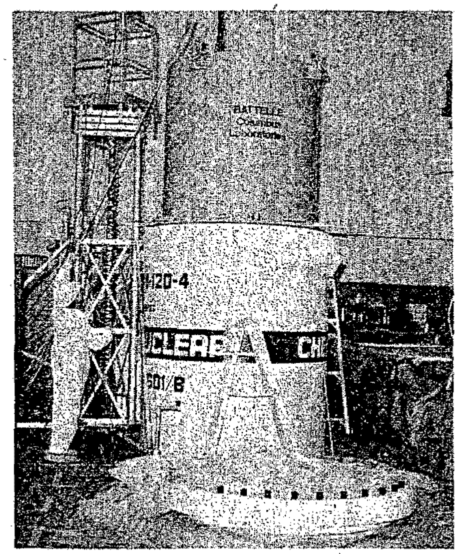


Figure 14 **Cutaway sketch of EPICOR II prefilter liner.**

Figure 15 **BCL transfer and storage device atop the PF-16 shipping cask.**

One of the initial characterization tasks conducted was a visual inspection of the liner external surfaces. This inspection was performed by viewing the liner directly through the hot cell window and by using the in-cell television camera with an external monitor. All external surfaces appeared to be clean and in good condition.

Technicians then obtained two gas samples from the area inside the liner above the ion-exchange media, using an evacuated sample chamber that was fixed over the liner vent plug by an electromagnetic device. A third sample was drawn in a similar manner from the bottom of the liner effluent tube.

Scientists analyzed the gas samples using mass spectrometry and gas chromatography. The test results shown in *Table 3* indicate that Samples 1 and 2, obtained through the vent plug, were enriched with hydrogen and carbon dioxide but were oxygen depleted. These samples also contained slightly higher than normal (compared with air) concentrations of nitrogen and carbon monoxide and several small quantities of hydrocarbons. The third sample, obtained from the bottom of the effluent tube, had gas concentrations very close to that of air with no large concentrations of such heavy combustible gases as methane.

After removing the liner manway cover, technicians lowered a television camera through the opening and visually examined the liner internal surfaces. The protective coating on the vertical inner surface appeared to be blistered yet intact, as did the underside of the liner top plate. No visible corrosion was evident when a portion of the ion-exchange media was removed to view the liner/media interface. The manway cover, which did not have a protective coating on the undersurface, was quite rusted. The surface of the ion-exchange media was dark, crusty, cracked, and caked with a white material believed to be boron deposits.

The PF-16 is believed to contain inorganic zeolites and three types of organic ion-exchange media—cation, anion, and mixed bed. Since the actual composition of the media is considered proprietary by EPICOR Incorporated, characterization of the PF-16 required examination and sampling to identify such basic items as the types and ratios of the media in the liner. A 2-in. Lucite tube with a basket retainer and a stainless steel cutting tip allowed BCL personnel to obtain a core sample of the ion-exchange media.

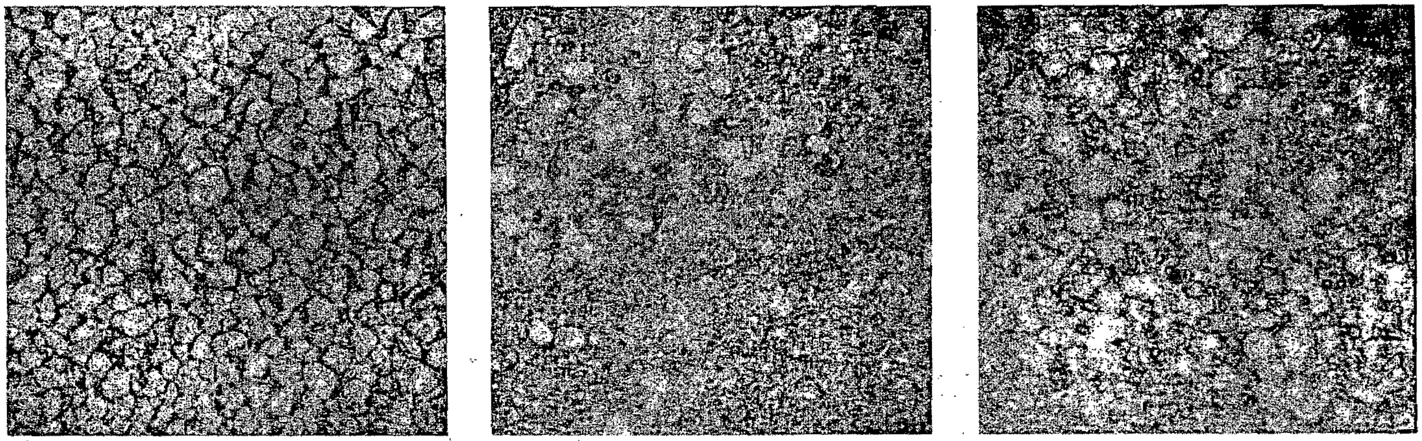


Figure 16 **Top, middle, and bottom regions of ion-exchange media core sample (30x).**

Table 3 **PF-16 gas analysis**

	Vent Plug		Effluent Tube
	Sample 1	Sample 2[a]	Sample 3
	Volume Percent		
Carbon dioxide	5.52 ± 0.06	5.27 ± 0.06	0.30 ± 0.03
Argon	0.96 ± 0.05	0.96 ± 0.05	0.94 ± 0.05
Oxygen	0.20 ± 0.02	0.30 ± 0.05	20.2 ± 0.2
Nitrogen	80.6 ± 0.4	81.2[b] ± 0.5	78.0 ± 0.4
Carbon monoxide	0.2 ± 0.02	—	0.004 ± 0.001
Hydrogen	12.4 ± 0.2	12.2 ± 0.02	0.5 ± 0.05
	Parts per Million by Volume		
Methane	500 ± 2.5		45 ± 5.0
Ethylene and Acetylene	0.7 ± 0.1		0.1
Ethane	42 ± 4		4 ± 1.0
Propylene	0.1		0.1
Propane	6 ± 1		1 ± 0.2
Isobutane	0.6 ± 0.1		0.4 ± 0.1
n-Butane	0.1		0.1
Hydrogen sulfide	20		20
Carbonyl sulfide	10		10
Sulfur dioxide	10		10
Unknown compounds	20		20

a. Not subjected to detailed analysis.
b. Includes carbon monoxide.

Table 4 PF-16 residual liquid chemistry analysis

pH	5.3 ± 0.1 at 27°C
Conductivity	30 μmho/cm at 27°C
Acidity	1.2 meq/ml at pH 7.0
Total residue upon evaporation	3.1 ± 0.1 mg/ml

Component	Concentration
Sodium	< 2000 ppb
Iron	34 ppb
Phosphorus	< 110 ppb
Zinc	88 ppb
Magnesium	< 20 ppb
Calcium	100 ppb
Aluminum	110 ppb
Boron	1.12×10^6 ppb
NH_4	0.8 μg/ml
SO_4	5.2 μg/ml
NO_3	< 0.3 μg/ml
Chlorine	3.0 μg/ml
Total organic carbon	61 μg/ml
Total Kjeldahl nitrogen (TKN)	0.48 μg/ml

Table 5 Residual liquid radiochemistry analysis

Component	Concentration (μCi/ml)
Gross beta/gamma	$1.77 \pm 0.01 \times 10^{-2}$
Gross alpha	$5.9 \pm 0.01 \times 10^{-4}$
Strontium 89/90	$5.2 \pm 0.1 \times 10^{-4}$
Antimony 125	$7.94 \pm 0.42 \times 10^{-4}$
Cesium 134	$1.32 \pm 0.02 \times 10^{-3}$
Cesium 137	$1.308 \pm 0.005 \times 10^{-2}$
Plutonium 238, 239, 240	$< 1.0 \times 10^{-4}$
Uranium 238	$< 1.0 \times 10^{-4}$

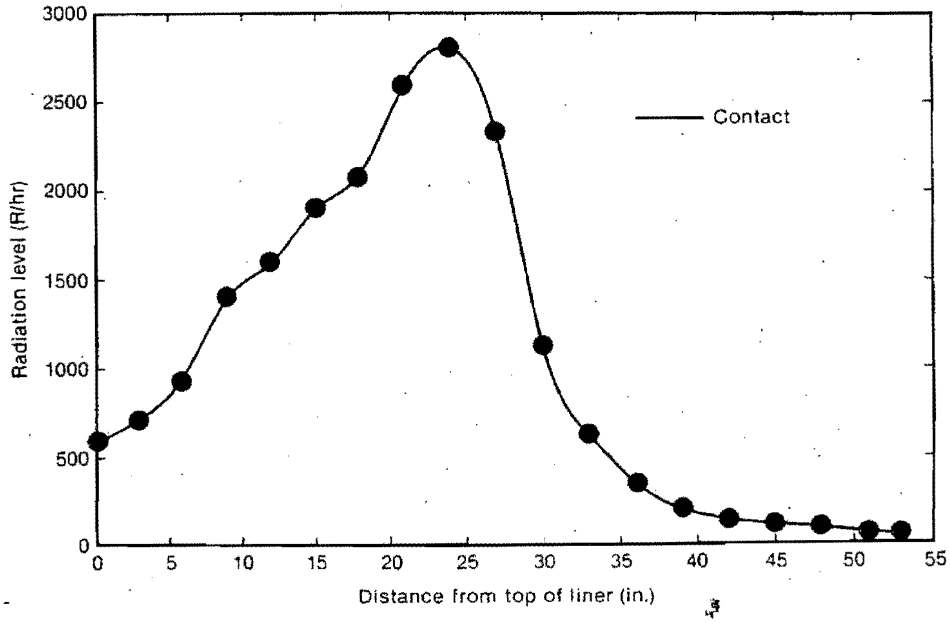
Scientists performed preliminary examination of the core sample using the hot cell stereo microscope at approximately 30x magnification while the sample remained encased in the Lucite tube. The three well-defined regions that were observed are shown in *Figure 16*. The top region, presumed to be an inorganic medium, consists of free-flowing, dry, granular, and irregularly shaped particles. The middle region consisted of regularly shaped, spherical, translucent particles, while the bottom region consisted mostly of opaque and translucent particles. Some opaque agglomerates were present in the bottom region, but may have been caused by moisture and particles adhering to the Lucite tube.

Radiochemical analysis of portions of the core sample yielded preliminary data on the distribution of radionuclides throughout the three regions in the liner. The top region was apparently quite effective in removing the cesium from the contaminated water, as most of the cesium was located in that region. The strontium was less effectively removed by the top region and was more uniformly distributed throughout the other two regions.

From these preliminary examinations of the media core sample, scientists at BCL concluded that the ion-exchange media did not appear to be significantly degraded by radiation. This fact was later confirmed when ion-exchange media integrity examinations were performed using electron microscopic scanning. Deterioration of the ion-exchange media by radiation appears to be minimal even in the regions of the highest activity loading near the top of the media. It should be noted however, that some media surface cracking and spalling was observed in the bottom layer of the core sample. Since this region is farthest from the high activity area, this degradation may be an effect of either high moisture content in the region or of chemical attack.

Following the first core sample, BCL personnel obtained a sample of the residual liquid from the bottom of the core sample hole. Technicians performed comprehensive chemical and radiochemical analyses of the liquid; the results are shown in *Tables 4* and *5*. The only significant chemical species present in the liquid was 1.12×10^6 ppb of boron, an element not effectively removed from contaminated water by the ion-exchange media. The liquid sample exhibited a very low ion content. This indicates that no significant amounts of either corrosion products or ion-exchange media degradation products are present in the liquid. The liquid sample also exhibited a relatively neutral pH of 5.3, which would not be expected to present a corrosion hazard to the liner steel. In addition, the radiochemical analysis indicates no significant release of radionuclides from the resin matrix, even though the liner contains approximately 1,250 Ci of activity.

Figure 17 **Plot of PF-16 contact gamma radiation readings.**



External gamma scans were performed to determine the relative deposition of gamma-emitting radionuclides throughout the liner. Most of the activity was concentrated in the top 3 to 6 in. of the ion-exchange media bed. Technicians performed gamma spectroscopy at the location of the peak activity and, consistent with the radiochemical analysis of the core sample, found that ^{137}Cs and ^{134}Cs contributed most of the gamma activity. The maximum external radiation readings were 2800 R/h on contact, 1000 R/h at a distance of 1 ft, and 410 R/h at 3 ft. *Figure 17* shows a plot of the contact external gamma scan.

After resealing the liner manway cover, technicians conducted gas generation tests. The test results clearly demonstrated that oxygen depletion and hydrogen generation mechanisms exist in the liner. While these data indicate that such mechanisms exist, the data could not be used to quantify gas generation because of the liner leak rate. The leak rate was detected after BCL conducted pressurized leak testing, and may have been caused when the liner was pressurized for the leak test.

BCL personnel performed a number of other characterization tasks before they shipped the liner to the Idaho National Engineering Laboratory. These tasks included measurement of ion-exchange media water content; measurement of the liner internal dose rates; determination of ion-exchange media pH; and measurement of the liner temperature profile. The TI&EP published specific results of all BCL's PF-16 characterization tests in GEND-015, *Characterization of EPICOR II Prefilter Liner 16.*

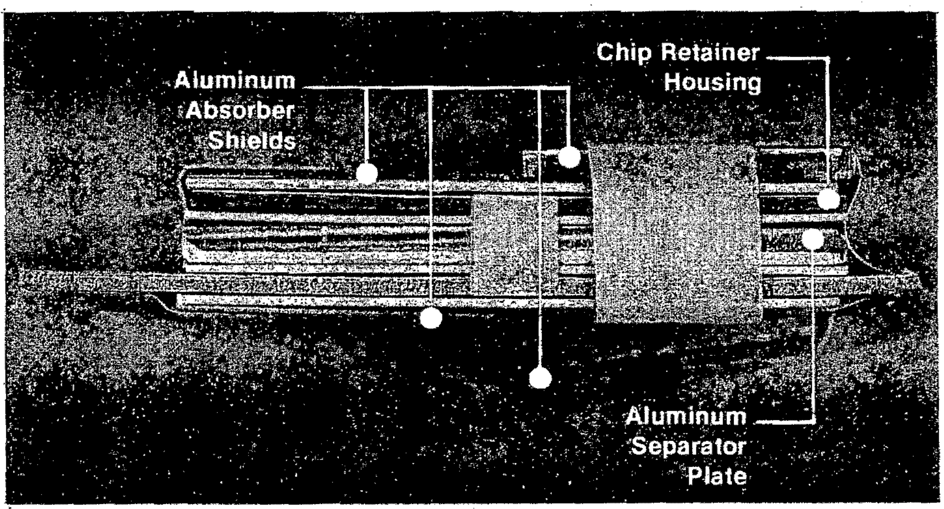
The characterization of PF-16 provided reseachers with valuable information about the behavior of highly loaded ion-exchange media and yielded baseline data for the development of safe handling, storage, and disposal techniques. One of the most important results of this characterization work is the fact that the PF-16 liner and its ion-exchange media suffered no extensive damage as a result of being loaded with accident-generated waste.

Multichip TLDs used in TMI-2 Reactor Building Characterization

The TI&EP is conducting experiments using a multichip thermoluminescent dosimeter (TLD) developed by Battelle Pacific Northwest Laboratories (PNL). These experiments, part of the TI&EP Radiation and Environment Program's Reactor Building characterization and radiation mapping effort, are designed to measure gross beta-gamma fields at various locations throughout the building. The experiments include:

- Placing TLDs at pre- and postdecontamination experiment sample locations on the 305- and 347-foot elevations to collect the radiation data required to measure the gross decontamination experiment effectiveness
- Placing TLDs at selected locations around dome area radiation monitor HP-R-214 prior to the monitor's removal to collect the radiation data required for HP-R-214 failure studies
- Suspending TLDs on "trees" through four 305-foot elevation floor penetrations to collect the radiation data required for Reactor Building basement characterization.

Figure 18 **TLD side view.**



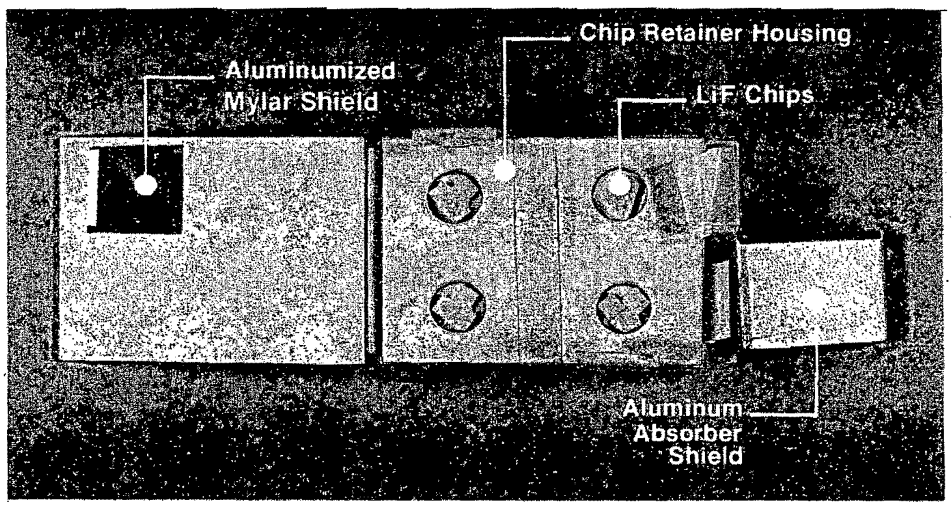


Figure 19 **TLD center section**

Each dosimeter contains 24 LiF chips that are oriented so that 12 chips face the front and 12 chips face the back. The front and back sections are separated by a 0.125-in.-thick aluminum separator plate (see *Figure 18*). The chips are clustered in groups of three under four different thickness absorber shields (see *Figures 19* and *20*) encased with two wraps of 0.005-in. aluminized Mylar and a 0.005-in. anticontamination plastic bag. The laminated construction of the shields allows the use of varying thicknesses of shielding over each cluster of chips. There are three aluminum absorber shields and one thin aluminized Mylar film shield. The thickest aluminum shield is 0.125-in. thick and prevents all beta radiation from penetrating to the TLD chips. The other two aluminum shields are 0.020- and 0.032-in. thick and allow only those beta particles with sufficently high energy levels to penetrate the aluminum to reach the TLD chips. The aluminized Mylar shield is thin enough to allow virtually all except very low energy beta particles to penetrate to the TLD chips. This design provides the capability for determining the relative beta energy distribution as well as providing an accurate measurement of the gross gamma field.

These TLDs offer many distinct advantages for TI&EP researchers. Among their advantages is the unique capability to measure background radiation levels on one side of the dosimeter while simultaneously measuring the radiation emitted from the surface in contact with the opposite side. This capability was used to measure radiation levels before and after the gross decontamination experiment at predetermined locations on the 305- and 347-foot elevations where concrete spalling or metal samples, smear surveys, RO-2A portable radiation instrument, and portable gamma spectrometer measurements were taken. The directional capability of this TLD can also be employed to determine the relative location of high radiation sources when the TLD is suspended in a stationary position such as in the Reactor Building basement characterization.

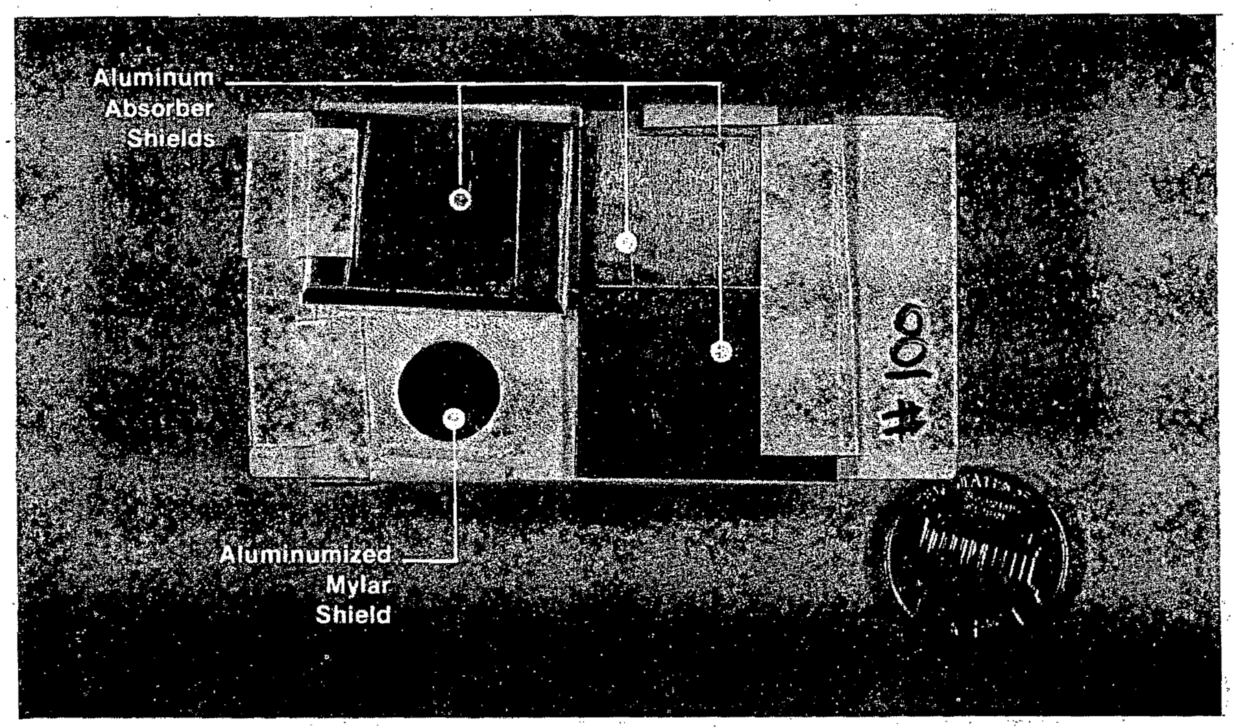


Figure 20 **TLD top view.**

The data obtained from these multichip TLDs will provide TI&EP researchers with valuable information to aid in accurately determining the effectiveness of gross decontamination techniques, and will provide an additional source of information for reactor building characterization work. In addition, the data may also provide an additional resource for determining worker requirements in high beta-gamma fields such as those at TMI-2.

Prototype Gas Sampler Developed for TMI-2 Waste Shipments

In support of the TMI-2 TI&EP, EG&G Idaho, Inc., engineers developed special tooling to sample gases from highly radioactive ion-exchange media canisters (liners). The tool, shown in *Figures 21* and *22*, is called a prototype gas sampler (PGS). To ensure safe shipment of liners from TMI for research and disposition, the PGS was designed to remotely remove and reinstall liner vent plugs, capture any gases released, and purge liners of combustible gases with an inert gas.

TMI-2 technicians detected a combustible gas mixture in an EPICOR II ion-exchange media liner as they prepared the liner for shipment to an off-island facility during March of 1981 (see PF-16 characterization article, this issue). Although the liner was vented prior to shipment, preliminary gas sampling upon arrival at Battelle Columbus Laboratories indicated the generation of combustible gas while also indicating a depletion of oxygen. Based on these results, GPU Nuclear and TI&EP personnel decided that EPICOR II liners should be sampled for gas, vented, and purged with an inert gas (if necessary) before they were shipped.

After considerable preliminary evaluation, EG&G Idaho engineers at the Idaho National Engineering Laboratory (INEL) decided that the best method for accessing the gas-containing area within the liners would involve removal of the 2-in. pipe plug from the liner vent port. Under TI&EP direction, these engineers designed, fabricated, and tested the PGS. Following tests at the INEL, EG&G Idaho delivered the sampler to GPU Nuclear at TMI-2, where it will be used on liners housed in storage modules at the Solid Waste Staging Facility.

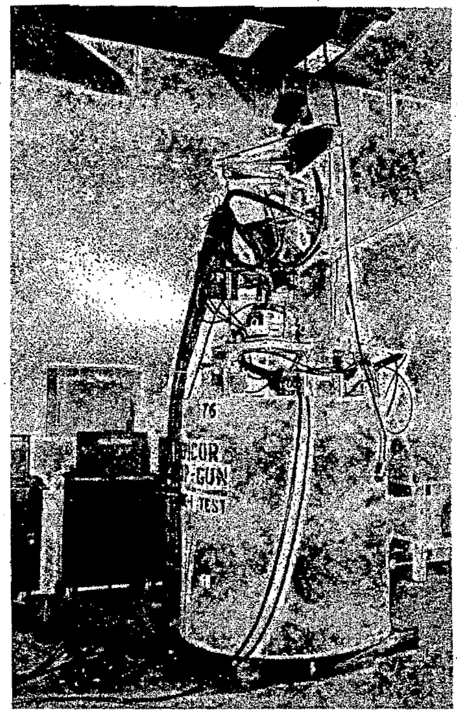
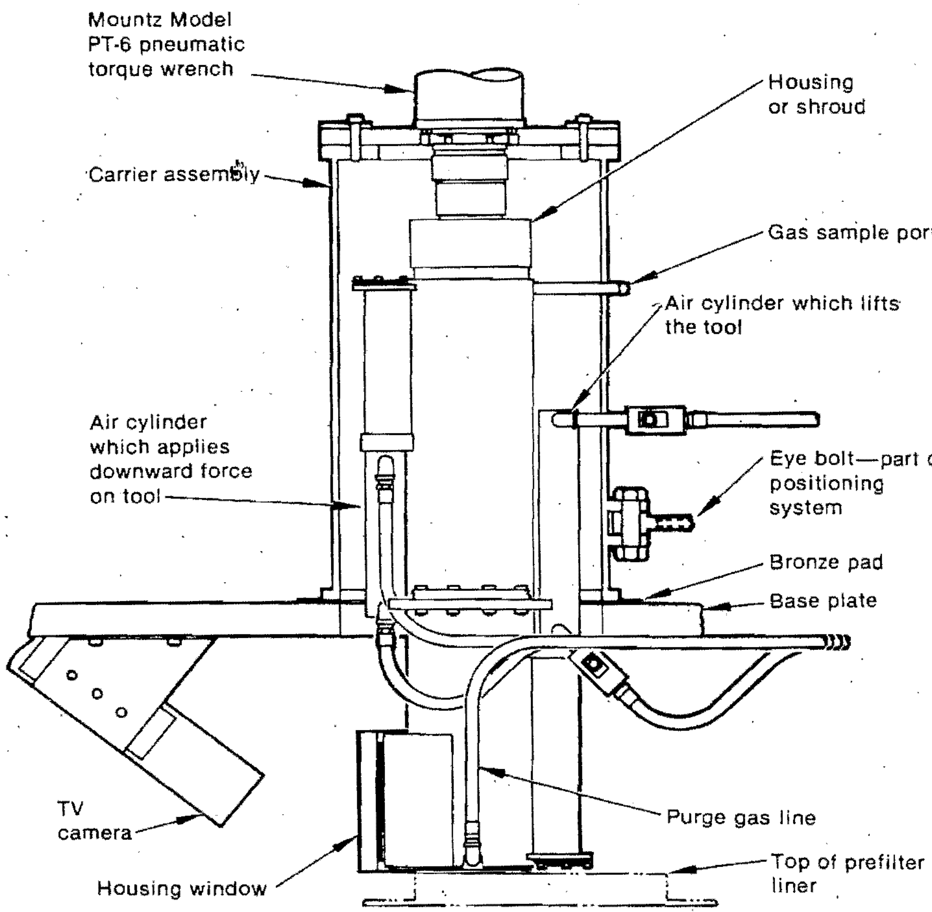


Figure 21 **Prototype gas sampler installed on EPICOR II liner.**

Figure 22 **Cross-sectional view of the prototype gas sampler.**



Major components of the PGS system include a portable, 18-in.-thick concrete shielding structure (blockhouse) and a remote support facility that is the command center for all operations. The sampler, a pneumatically operated device, consists of a platform, sample housing, position and rotation drive assemblies, and a closed-circuit television monitoring system. To reduce the possibility of combustion, PGS surfaces that have relative motion are made of nonsparking material and the sampler is electrically grounded to the liner. A lifting fixture attaches the sampler to the hoist system of the blockhouse. A 100-ft umbilical cable carrying television camera signals, power, lighting, compressed air, and gas handling lines connects the PGS to the command center.

Using a mirror-window arrangement in the blockhouse, initial alignment of the PGS over the liner vent port is accomplished using the liner lifting lugs as indexing guides. The position of the vent plug relative to the lifting lugs varies from liner to liner, therefore precise positioning of the sampler drive shaft over the vent plug is made using air-driven threaded adjustments on the sampler. An adjustment range of ± 1 in. in all directions from the nominal position is provided at a rate of approximately 1 in. per minute. PGS operators monitor final alignment of the tool tip to the vent plug with the closed-circuit television system mounted on the PGS.

The tool tip is designed to secure the plug with sufficient force to lift it free of the vent port but with small enough force to allow the tool to be disengaged after reinstallation. In addition, a downward force can be applied to the tool to force it into the plug.

After the tool is engaged in the liner vent plug (see *Figure 23*), the PGS is lowered until the shroud around the tool is sealed against the liner top. The PGS's weight, 850 lb, is sufficient to maintain the seal should liners reach maximum anticipated pressures of up to 19 psig.

The plug removal drive system consists of a pneumatic torque wrench and a ball bearing spline. The torque system is capable of producing 2500 ft-lb of torque with a maximum unloaded speed of 5 rpm. The liner lifting lugs serve to dissipate rotational forces that result from torque applied during unthreading of the vent plug.

The ball bearing spline allows the drive shaft to move vertically during unthreading and threading. In addition, two air cylinders mounted on the housing allow the tool tip to be raised to lift the plug clear of the port after unthreading or lowered to reinsert the plug.

With the television system, operators monitor indexing marks on the shaft to determine direction of drive rotation as well as the number of revolutions in the threading and unthreading sequences. Two air cylinders operate a cable mechanism to change direction of drive rotation, as required.

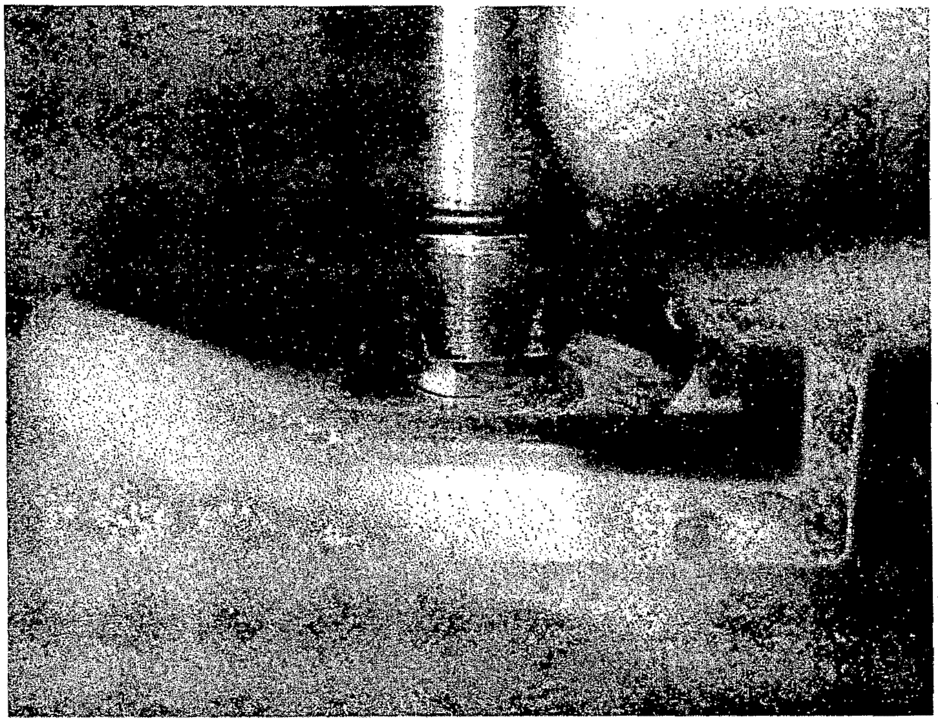


Figure 23 **View through shroud window of tool tip engaged in liner vent plug.**

After PGS operators unscrew the plug and lift it clear of the port, liner gases can pass into the shroud and through gas-handling lines to the command center. Upon completion of venting, nitrogen is used to "sweep" the sampler and liner to remove any gases before the plug is lowered back into the port and tightened, thereby resealing the liner. The storage module is ventilated through a HEPA filter unit and the PGS assembly is removed. The liner can now be retrieved from the storage cell and placed into a shipping cask.

Functional testing at TMI demonstrated that the PGS can be effectively used to vent and inert EPICOR II liners to ensure their safe shipment from TMI.

First SDS Liner Leaves TMI for Vitrification Testing

On May 21, 1982 the U.S. Department of Energy (DOE) shipped the first radioactivity-bearing ion-exchange media liner from the Submerged Demineralizer System (SDS) at Three Mile Island. The liner was part of the system that processed more than a half-million gallons of radioactive water from the TMI-2 Reactor Building where the water accumulated as a result of the accident in March of 1979. DOE shipped this liner to Battelle Pacific Northwest Laboratory (PNL) at Richland, Washington, for testing and disposition research.

At PNL, the liner, loaded with approximately 13,000 Ci of radioactive fission products, will undergo experiments on the feasibility of vitrifying the radioactive inorganic ion-exchange media zeolites. In the vitrification process the zeolites and glass-forming chemicals are fed into a canister in a furnace where the mixture is heated, causing vitrification. After the mixture cools, the canister serves as the container for the final waste product, a glass column considered to be a stable form for the SDS zeolite waste.

PNL has done extensive research into waste material vitrification. During 1981, PNL conducted four nonradioactive demonstrations of the zeolite vitrification process on behalf of the TI&EP Waste Immobilization Program. The nonradioactive demonstrations proved the feasibility of vitrifying inorganic zeolites in tests with nonradioactive cesium and strontium.

On the basis of the nonradioactive tests, the SDS liner shipped in May and another two SDS liners yet to be shipped will undergo radioactive vitrification demonstration tests at PNL. The three radioactive demonstrations scheduled for 1982 and 1983 will further establish vitrification's technical feasibility as a disposition option for TMI's highly loaded radioactive wastes.

Industry Benefits from Electrical Equipment Survivability Information

Charge converters used in the loose parts monitoring (LPM) system within the Reactor Building at TMI-2 apparently failed shortly after the accident, and the metal oxide semiconductor (MOS) field-effect transistors that caused the failure should not be used in high radiation fields such as adjacent to a nuclear reactor. This is the report from M. B. Murphy of Sandia National Laboratories where analysis of the charge converters has been done in support of the TIO Instrumentation and Electrical Equipment Survivability Program. Murphy presented his data at the TMI-2 Programs Seminar in San Francisco during December 1981.

Both Rockwell International, which supplied the LPM system, and Endevco, which supplied the charge converter used in the system, conducted independent examinations on the converters. Both examinations verified that failure would occur in the MOS field effect transistor in the converter from excessive radiation at dose levels of approximately 10^5 rad. Although at TMI-2 the charge converters were mounted in areas where the radiation doses during normal plant operation would be well below the damage threshold, the radiation release inside the Reactor Building during the accident was high enough to cause failure.

Degradation of the Endevco charge converters has also been observed at Tennessee Valley Authority's Sequoyah 1 plant where they were mounted within 10 ft of the transducers, thus putting them in high radiation areas near the reactor vessel and steam generators. The MOS transistors were damaged after less than 1 year of reactor operation. The charge converters used at TMI-2 and Sequoyah 1 were Endevco models 2652M4 and 2652M3, respectively. *Figure 24* is a photograph of a failed TMI-2 charge converter with the covering sleeve cut away for removal and testing of suspect components.

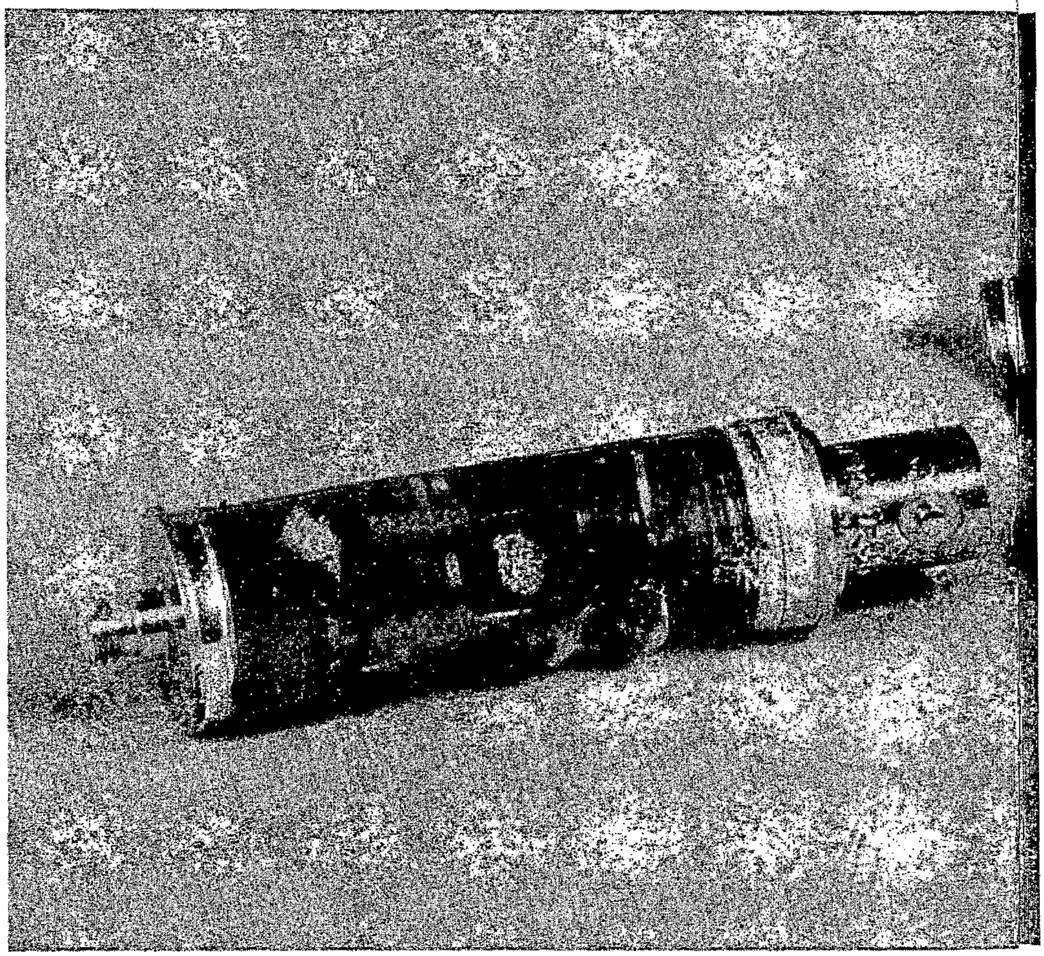
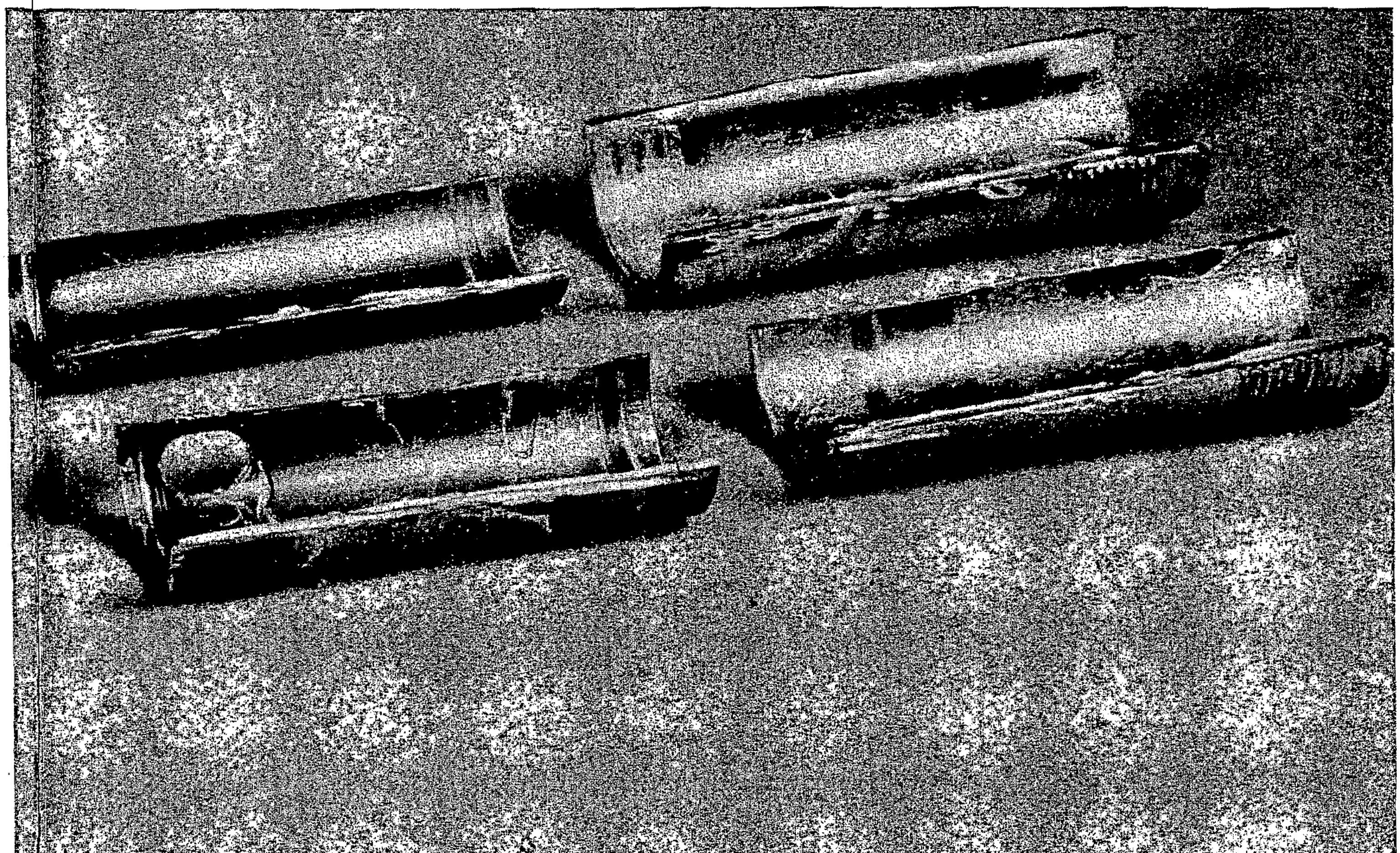


Figure 24 **Failed charge converter after disassembly.**



Radiation damage to the MOS transistor did not cause a sudden failure, but rather caused its gradual deterioration. Bias adjustments could appear to correct for deterioration of the transistor, meaning that the system might not have responded correctly to a loose part noise after adjustment.

Deterioration of the MOS transistor can be detected remotely by measuring the d.c. converter. This voltage is normally 13.5 V, supplying 7 mA to the 2652M4 charge converter. Respective voltage and current for the 2652M3 are 18 V and 9 mA.

In a letter to the TIO expressing appreciation that useful recovery information is being passed on to industry, Rockwell International reported development of a charge converter using junction field effect transistors. Three designs were tested in their gamma radiation facility, exposed at a rate of 10^6 rad/h. One design operated at an exposure greater than 10^7 rad. Run-to-failure tests were made at 10^6 rad/h with this successful circuit and a duplicate. Both operated at exposures in excess of 10^7 rad. Rockwell will offer these units as replacements in their existing systems, and will incorporate them into any future LPM systems.

TMI-2 GEND Reports Available to the Public

B094-1182-2.5M

Between November 1981, when the last *Update* was published, and September 1982, seven formal reports were published by the TMI-2 Technical Information and Examination Program. The title, GEND number, date of publication, and a brief description of each is presented below. The reports are available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161.

Color Photographs of the TMI Reactor Containment Building for Entries 1, 2, 4, 5, and 6. GEND-006, published February 1982. A collection of all 308 photographs taken during the first six entries, arranged in sequence and produced in color. The photographs are accompanied by maps indicating location in the Reactor Building of each subject.

Examination Results of the TMI Radiation Detector HP-R-0211. GEND-014, published October 1981. An analysis of the first piece of electrical equipment removed from the Unit 2 Reactor Building, including cause of failure and recommendations to the industry.

Characterization of EPICOR II Prefilter Liner 16, GEND-015, published August 1982. Description of the characterization work and analytical results from completed study of the PF-16 liner.

Response of the SPND Measurement System to Temperature During the Three Mile Island Unit 2 Accident. GEND-017, published December 1981. A discussion of why the SPND Measuring System did not indicate accurate fuel rod temperatures during the accident.

Nondestructive Techniques for Assaying Fuel Debris in Piping at Three Mile Island Unit 2. GEND-018, published November 1981. An evaluation of the four major categories of nondestructive techniques for assaying fuel debris in the primary coolant: ultrasonics, passive gamma ray, infrared detection, and remote video examination.

Controlled Air Incinerator Conceptual Design Study. GEND-021, published January 1982. A conceptual design study for a controlled air incinerator facility for incineration of low-level combustible waste at TMI-2.

TMI-2 Information and Examination Program 1981 Annual Report. GEND-022, published April 1982. An overview of work accomplished in the TI&EP Data Acquisition, Waste Immobilization, and Reactor Evaluation programs from October 1980 through December 1981.

Zeolite Vitrification Demonstration Program Characterization of Nonradioactive Demonstration Product. GEND-025, published September 1982. A laboratory analysis of the glass product made when nonradioactive ion-exchange media were vitrified. The media were loaded with nonradioactive cesium, strontium, and other fission products to simulate the actual condition of radioactive TMI-2 ion-exchange media (from the Submerged Demineralizer System) to be vitrified later in 1982.

UPDATE

The **TI&EP 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.

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TMI Unit 2 Technical Information & Examination Program

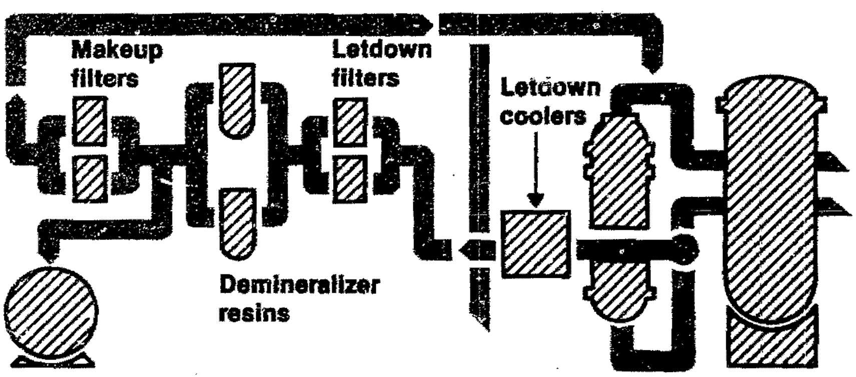
UPDATE

Volume 3, Number 2 August 15, 1983

Resin Characterization Supports Waste Removal Efforts

Assisted by Westinghouse Hanford Company (WHC), TI&EP and GPUNC engineers began planning for the removal of ion exchange resin from the makeup and purification system demineralizer vessels. Classified as abnormal wastes (those not routinely generated at nuclear power plants), the demineralizer resins have the potential for research and development work in the area of waste disposal technology.

Figure 1. **Makeup and Purification System which maintains RCS quality and chemical limits.**



During normal reactor operations, the makeup and purification system, shown in Figure 1, maintains reactor coolant quality and chemistry within prescribed limits. After the start of the accident on March 28, 1979, reactor coolant system (RCS) letdown flow was directed through the filters and demineralizers for at least 18-1/2 hours before the flow stopped. The two demineralizer vessels, each located in a separate cubicle (designated A and B) on the 305-00 elevation of the Auxiliary Building, were bypassed sometime after letdown flow was lost and have since remained isolated from the RCS.

Using demineralizer drawings and accident operating histories provided by GPUNC, plans were developed to assess the largely unknown status of the demineralizers, and to outline a suitable cleanup strategy. Because high radiation levels prevented recovery personnel from entering the cubicles, a remotely operated miniature transport vehicle called the Surveillance and Inservice Inspection Robot or SISI was designed and equipped by WHC for entry into the demineralizer cubicles to obtain preliminary characterization information. SISI is shown in Figure 2.

During these exploratory entries into the demineralizer cubicles, SISI also provided engineers with video observations of the cubicle interiors. The videotapes verified as-built equipment conditions, and showed piping and equipment to be in satisfactory condition. An evaluation of the equipment from the videotapes was useful in defining a resin removal approach, the most desirable option being that of using existing in-plant equipment and piping.

In order to confirm the presence of fuel in the vessels and to determine if the amount was at or near the critical level of 70 kg, several independent measurement techniques were used. Solid-state track recorders (SSTRs) which provide a record of tracks of fission products generated by neutron-initiated fissions in the ^{235}U contained in the SSTR were lowered alongside the A vessel. Using SSTR data, 1.7 ± 0.6 kg of uranium were estimated

Published by EG&G Idaho, Inc., for the U.S. Department of Energy

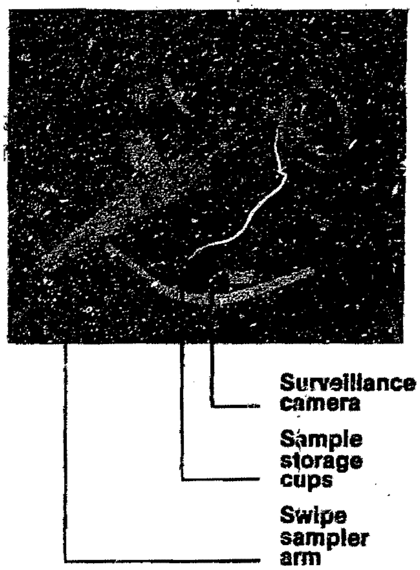


Figure 2. **Remotely operated transport vehicle developed for exploratory work in the demineralizer cubicles.**

	Demineralizer A		Demineralizer B	
Gas	**(8 psig)**	**(4 psig)**	**(8 psig)**	**(4 psig)**
^{85}Kr μCi/cm^3	2.1E-2	1.9E-2	9.9E-2	1.0E-1
H_2%	6.9	7.2	78	74
O_2%	<0.3	<0.3	<0.2	<0.2
N_2%	86.3	90.4	10	11
Other%	6.8	2.4	12	15

Table 1. **Onsite demineralizer vessel gas sample analysis**

to be in the A vessel. While SSTR data confirmed the presence of fuel within the vessels, the data are unable to help verify the locations of the fuel.

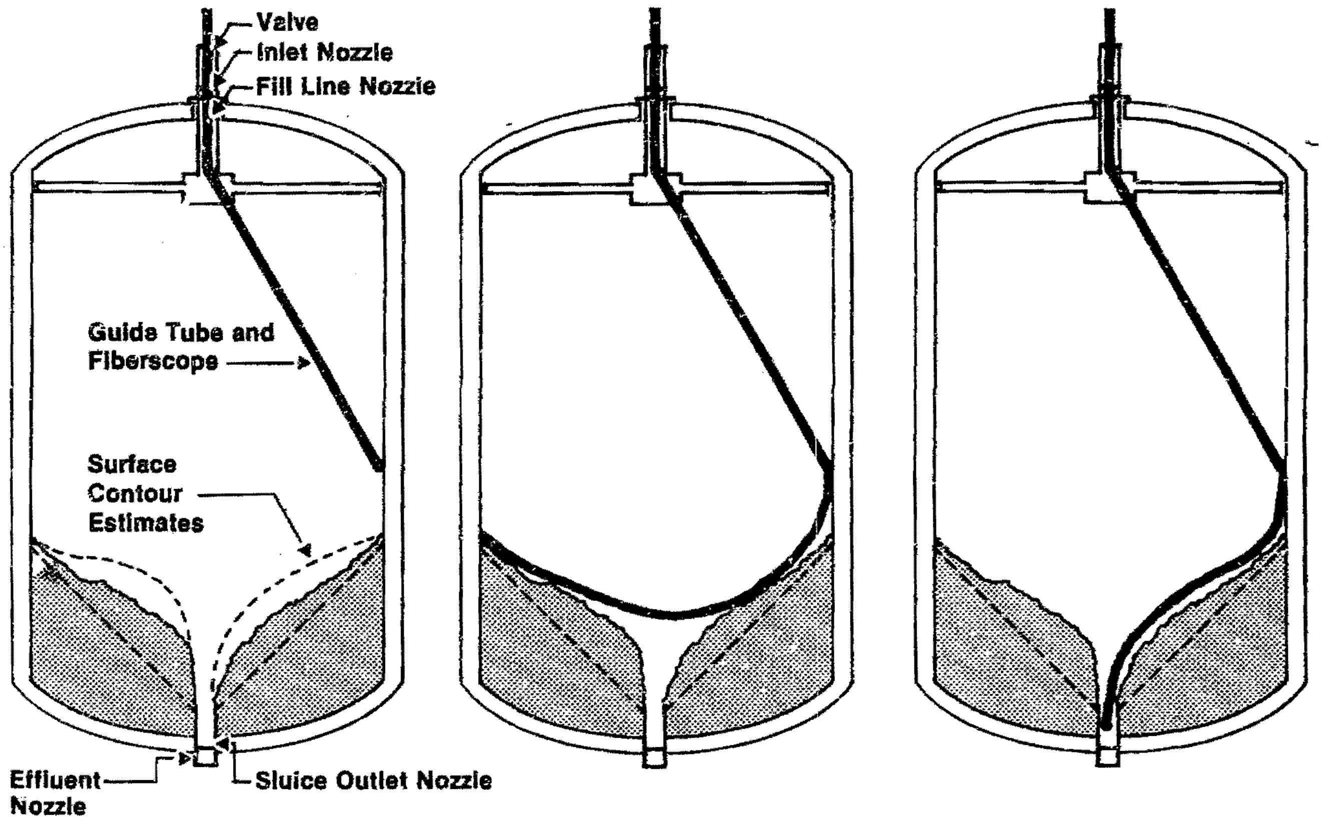
To determine fuel location and obtain additional information on fission product content, WHC placed a silicon-lithium gamma spectrometer system inside the cubicles. The detector can "see" the high-energy gamma ray associated with the ^{144}Pr daughter of the fission product ^{144}Ce. Cerium-144 is valuable as a fuel tracker because it is known to have chemical properties similar to uranium and is generally known to stay within the fuel matrix. Analysis of the A vessel's spectral data estimated fuel content to be 1.3 $\pm$ 0.6 kg of uranium. The vessel was also estimated to contain about 6000 Ci of ^{138}Cs, the most prominent fission product.

The gamma spectroscopy showed that the activity for ^{137}Cs and for the fuel tracker, ^{144}Ce/Pr, peaks at two feet from the bottom on the far side of the A vessel. This profile suggests there is no water above the top of the resin bed. If the estimates of location of the top of the resin bed are accurate, then the resin volume is one-half of the volume originally installed in the vessel. This finding is consistent with Pacific Northwest Laboratory (PNL) nonradioactive resin irradiation tests that showed a similar volume reduction for resin exposed to 1.7 x 10^9 rads, the dose GPUNC estimated the resins received as a result of the accident. Although no quantitative fuel estimates could be made for the B vessel, the one data point obtained indicates less fuel but more fission products than for the A vessel.

The characterization of the makeup and purification demineralizers culminated with sampling and analyzing vessel gases, liquids, and resin itself. Results of the gas sample analysis performed by the on-site chemistry department confirmed predictions concerning the composition of the gases that have been trapped and generated in the demineralizer vessels since the accident. Due to the vessel's high radiation levels, radiolysis of the vessel water resulted in high amounts of hydrogen and a substantial quantity of nondiatomic gases. The amount of oxygen was low due to an oxygen scavenging reaction with the resins. A comparison between the nondiatomic gases analyzed by WHC and the PNL resin irradiation tests suggests that the resins in both demineralizer vessels were wet when irradiated.

In early March 1983, engineers inserted a vacuum pickup probe through the diaphram valve and resin fill line into the B vessel and produced the first high dose rate sample. The sample solution varied from amber to dark brown in color, but with very little solids evident.

In April 1983, TI&EP engineers completed a successful examination of demineralizer A vessel using a 50-ft long, radiation-tolerant, fiberoptic scope. The scope, inside a polyethylene guide tube, was pushed into the vessel through the resin fill line and passed easily through the resin fill line diaphragm valve. The fiberoptic scope and guide tube paths are detailed in Figure 3. Observations by TI&EP personnel during the fiberscope inspection concluded that the A vessel contains a bed of resin with a crust of boron crystals coating the top of the bed. The center of the bed has a large void that appears to be above the resin sluicing outlet line. The resin in the bed is agglomerated and amber colored below the crystalline crust.



Using mechanical probes and vacuum sampling system, a 10-g solid sample of the A vessel resin was obtained. This sample had radiation readings of 3 rad/h beta and 150 R/h gamma. The mechanical probe inserted into the B vessel found the resin bed approximately 1 ft. below the top of the water and 18 in. thick. Estimates of the resin and water levels in the B vessel are shown in Figure 4. Samples from various depths in the resin bed were resulting in a 75-ml slurry with approximately 50 ml of solids. Radiation readings taken without shielding at the top of the sample shipping container were 40 rad/h beta and 800 mR gamma.

Oak Ridge National Laboratory (ORNL) will do the chemical and radiochemical analyses on the resin samples. Results of the resin sample analyses will be reported in subsequent issues of the Update as the information becomes available. With this characterization information, TI&EP and GPUNC will be able to determine compatibility of the resin and comptability of any resulting liquid waste with the Submerged Demineralizer System (SDS) ion exchange processing system.

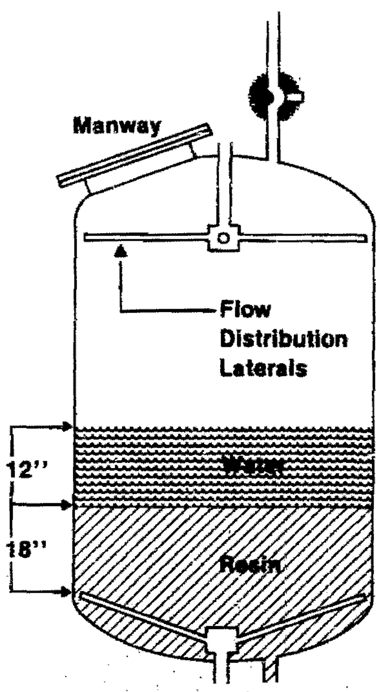


Figure 3. **Pathways of fiber optic horescope during examination of demineralizer A vessel internals.**

Figure 4. **Estimates of resin and water levels in B vessel.**

Video Inspections Support Reactor Building Basement Characterization

Significant effort is being expended toward overall characterization of the TMI-2 Reactor Building. These efforts support dose reduction tasks, fission product transport and deposition studies, Reactor Building damage assessments, and eventual cleanup of the basement by providing information necessary to determine decontamination techniques.

As a result of the TMI-2 accident, contaminated water flooded the Reactor Building basement. Approximately 640,000 gal of water collected in the basement and remained until September 1981. At that time, the Submerged Demineralizer System (SDS) and the EPICOR II ion exchange system were put to work to remove and decontaminate the bulk of the basement water. By May 1982, nearly all of the basement water had been removed and processed.

Initially, characterization efforts in the basement centered aroung sampling and analyzing the standing water and solids from the basement floor. Analysis results indicate the $^{134,137}Cs$ and ^{90}Sr are the major radionuclides with ^{90}Sr found predominately and after water removal in the solids. In August 1982, prior to a decontamination water flushing of the basement wall, beta and gamma radiation measurement began using thermoluminscent dosimeters (TLD). TLD "trees," each containing four TLDs spaced 5 ft apart on a cord, were lowered into the Reactor Building basement from the ground or 305-00 elevation. The preliminary TLD data indicate the basement walls, up to approximately 8 ft, and the floor area are the principal sources of gross beta and gamma radiation. The degree of radionuclide penetration into the concrete as a result of the standing water is a major area of interest to the recovery project.

Visual surveys, taken with closed circuit television (CCTV) cameras and reported on by Reactor Building work crews during task debriefing sessions, are helping researchers develop a graphic record of building damage. At greatly reduced man/rem exposures over in-person inspections, a color CCTV with remotely operated functions for focus, zoom, iris, and pan-tilt operations, was lowered into the Reactor Building basement. The camera surveyed the outside of the Reactor Coolant Drain Tank, the area below Core Flood Tank A, and the area below the equipment hatch. The camera surveys showed no signs of physical damage resulting from the accident, except some corrosion of carbon steel. All systems appeared intact; however, further quantitative testing may reveal internal damage. Deposits or "bathtub rings" left on the walls by changes in level of postaccident basement water are evident. The solids on the basement floor, which are considered to be one of the major contributors to dose rates in the Reactor Building, appears evenly distributed, thin, and loosely settled in the small area the camera surveyed. However, in subsequent surveys done in spring 1983, a number of bare sponts were observed in some areas of the floor.

The special capabilities of the camera system allowed observation otherwise unavailable, of a malfunctioning motor-operated valve located on the sampling line from Steam Generator B. This valve must be opened to drain the steam generator, a necessary operation prior to reactor vessel head lift. Following the camera inspection, engineers concluded that the pin connecting the valve motor stem to the valve was broken and the valve must be bypassed in order to drain the steam generator. The best points in the sampling line for cutting and installing the valve bypass were selected using the camera.

Upon completion of this series of video surveys, the camera was replaced because of radiation damage to the camera system due to the high radiation fields close to the basement floor. A manually-operated camera was assembled using off-the-shelf components. Because radiation dose rates in the area of Core Flood Tank A are relatively low (60 to 80 mR/h) compared to other areas of the 305-00 elevation, entry personnel were able to manipulate the telescoping boom and pan-tilt mechanism for the camera.

From a 30-in. manway, and a penetration near the Reactor Building's seismic gap, shown in Figure 5, technicians manipulated the camera

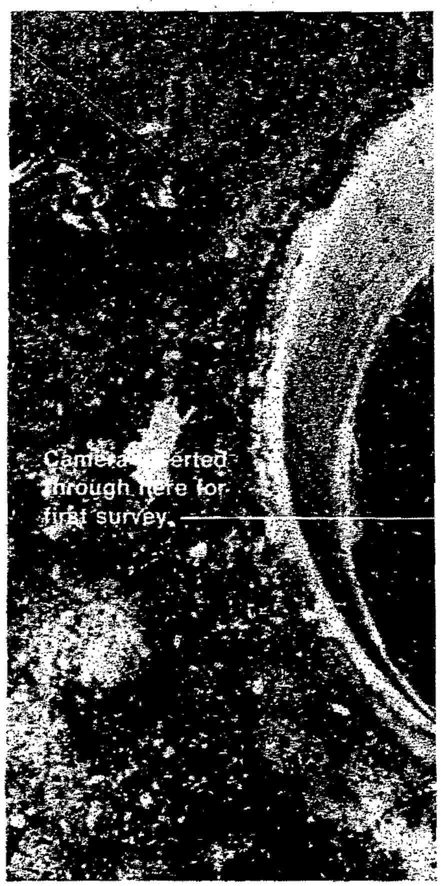
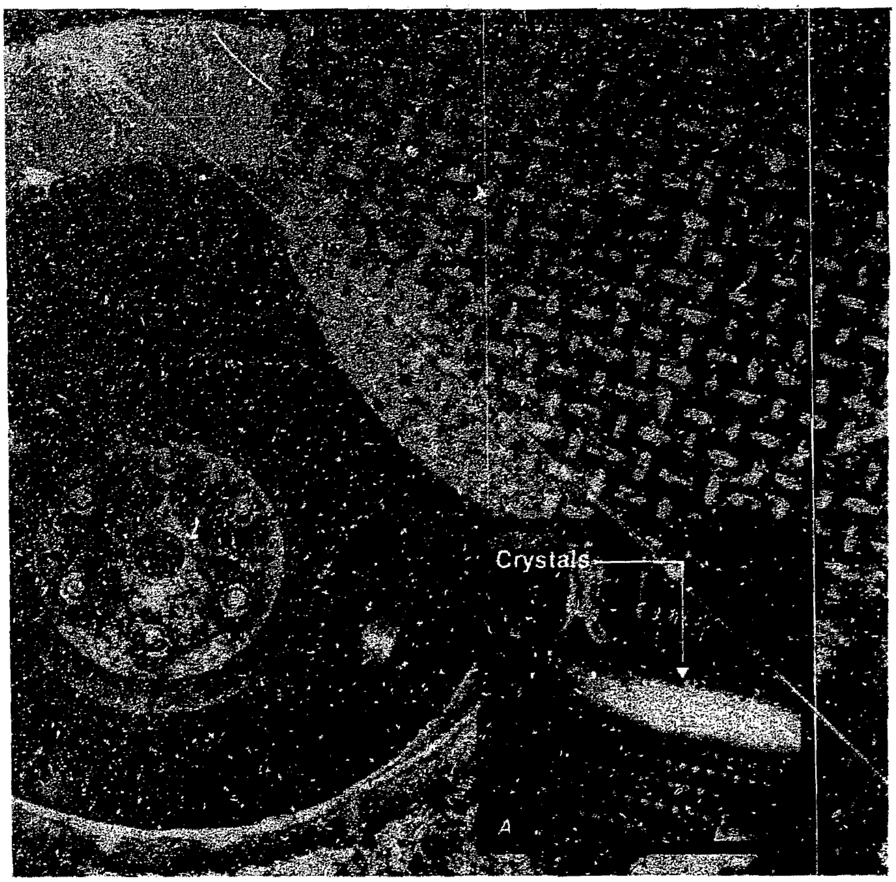
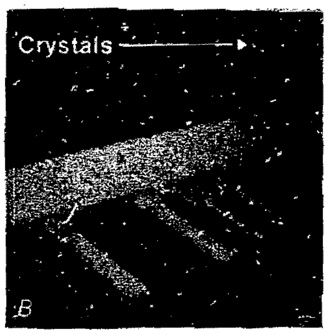


Figure 5. **The 30-in. manway in the 305-foot elevation that provided access to the reactor building basement for camera surveys.**





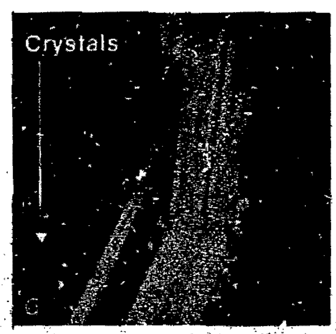


Figure 6.

A. The top of the sump inlet trash rack in the Reactor Building basement. Exidence of extensive rusting is present on metal surfaces and boric acid crystals can be seen on piping.

B. A cable tray located approximately six feet below the ceiling contains galvinized collte conduit. Boric acid crystals can be seen on pipe section above the cable tray.

C. I-beam support and pipe below ceiling show evidence of boric acid crystals.

through pipe and equipment congested pathways to gain access to the basement area near the sump inlet trash rack which is located in the northwest corner of the Reactor Building basement. Confirming information from earlier surveys, no visual evidence of physical damage to structures or equipment was found, but there is extensive rust and corrosion on carbon steel surfaces. The top of the sump inlet trash rack is shown in Figure 5. Solids and sediment deposition on the basement floor is not uniform. An estimated 50% of the floor area surveyed was covered with a thin layer of sediment or sludge.

Turning the camera toward the ceiling of the basement, the surfaces of pipes, conduits, electrical cables, cable trays were examined. Heavy deposits of agglomerated boron crystals were seen. A cable tray located approximately six feet below the ceiling is shown in Figure 6. As the camera rubbed or bumped surfaces and equipment, a "snow storm" of this loose debris fell from surfaces near the ceiling to the basement floor. Additional evidence of boron crystals is seen in Figure 6. The presence of this type of debris has added a new component to baseline cleanup and recovery consideration. The crystalline boron material, that is believed to have originated primarily as a precipitate out of accident water and decontamination water sprays, represents a potential source of airborne contamination.

The basement walls, support columns, and equipment items appeared relatively clear of the bathtub rings noted in the earlier surveys except for what appeared to be the remnants of two partially washed away rings on one support column. This could attest to the effectiveness of a high-pressure water spray washdown of the basement walls.

The camera surveys of the Reactor Building basement have contributed to the overall understanding of the postaccident condition of this area. Integrating the visual information with the preliminary radiological and chemical studies will add a new dimension to the characterization effort. Additional surveys, planned in preparation for additional radiological and chemical studies, will assist recovery engineers in determining the most beneficial locations for sampling the basement floor sludge, for additional radiation measurements, and for accessing basement equipment.

Vitrification of Radioactive Liners Completed

Submerged Demineralizer System (SDS) liners from TMI-2's zeolite ion exchange media water cleanup system are being used in Department of Energy (DOE) waste disposition research and development programs at a DOE national laboratory in Washington State. Three liners were shipped to the Pacific Northwest Laboratory (PNL) during 1982 and 1983 where their contents were successfully immobilized as vitrified glass logs.

Table 2. **Leach rates of low activity and nonradioactive glass logs ($g/cm^2/day$).**

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

PNL has been studying vitrification as an effective method for immobilizing the high specific activity radioactive material. In the vitrification process, zeolites (which contain silicates and many of the basic constituents needed to make glass) are mixed with glass-forming chemicals and are fed into a canister in a furnace, where the mixture is heated to approximately 1050°C. When the mixture cools, the canister becomes the container for the final waste product, a glass column that is a stable form for the SDS zeolites.

In four tests on nonradioactive liners conducted in 1981, PNL demonstrated the effectiveness of the process. Then, in May 1982, the first radioactive TMI liner arrived at PNL for vitrification. This liner, D10015, loaded with 13,000 Ci of radioactive cesium, strontium, and daughter products, was one of the least radioactive liners from TMI.

PNL technicians fed a mixture of D10015 zeolites and glass formers into the vitrification in-can melter system shown in Figure 7. Vitrification produced an 8-in.-diameter, 7-ft-long glass log that was extensively monitored and tested after it had cooled. Glass core samples of the log were taken from the top, middle, and bottom of the glass and subjected to leach rate tests. In Table 2, those leach rate test results are shown along with results obtained during tests on a nonradioactive vitrified log. The test results are comparable with existing standards for vitrified nuclear wastes and they indicated that the glass successfully trapped the radioactive contaminants.

Following vitrification of the contents of liner D10015, all the components used in the system were analyzed in preparation for vitrification of the two highest loaded SDS liners. All tests, including analytical studies of the performance of the off-gas system filter and measurements of the effects of vitrification on canister wall thickness and smoothness, indicated that the system maintained its integrity while functioning as designed to vitrify the radioactive zeolites.

In January 1983, a highly radioactive liner loaded with almost 113,000 Ci of cesium and strontium plus daughter products arrived at PNL from TMI. This liner, number D10012, was the first radioactive liner to be shipped with catalysts to recombine radiolytic gases, (as described in the related article on SDS wastes). The D10012 zeolites were vitrified in stages over a period of weeks. In the first vitrification run, a portion of the D10012 zeolites was mixed with glass formers to produce a 190-kg mix. This mix tumbled for one hour in the mixer feeder vessel shown in Figure 7 until the mixture was homogenized. It was then fed at a rate of 10 kg/h into the canister in the 1050° C furnace where vitrification occurred. Following vitrification, the mixture was heat soaked for four hours. Once cooled, the canister contained a solid glass log, approximately 6.5 ft long and 8 in. in diameter.

When another radioactive liner, D10016, loaded to 112,000 Ci, arrived at PNL from TMI, the balance of zeolites from D10012 was vitrified in a second vitrification run together with part of the zeolites from D10016. The remaining D10016 zeolites were then vitrified in a third canister. The three canisters, produced through vitrification of the highly loaded zeolites, are currently undergoing characterization and leach rate tests similar to those performed on the low-level liner vitrified in May 1982 (see Table 2). Preliminary test results indicate that the vitrification system performed extremely well, proving that highly loaded zeolites can be successfully immobilized as glass logs.

Figure 7. **In-can melter system in use at Pacific Northwest Laboratory to vitrify ZDS zeolites.**

Radiolytic Gases Recombined in SDS Waste Liners

The Department of Energy's TI&EP at TMI-2 facilitated recent shipment of highly loaded radioactive waste canisters from the Island by developing a system to prevent formation of combustible gas mixtures in the canisters. The gas mixtures were formed because of radiolytic gas generation in the canisters containing radioactive zeolite ion exchange media.

The canisters, called liners, were used in the Submerged Demineralizer System (SDS) to process accident-generated water predominantly contaminated with radioactive cesium and strontium. Over one million gallons of water flowed through the SDS from the Reactor Coolant Bleed Tanks, the Reactor Building basement, and the Reactor Coolant System, and resulted in curie loadings of up to 113,000 Ci including daughter products in some liners. The Department of Energy (DOE) agreed to take 19 of these liners for research and development work (See vitrification article in this issue).

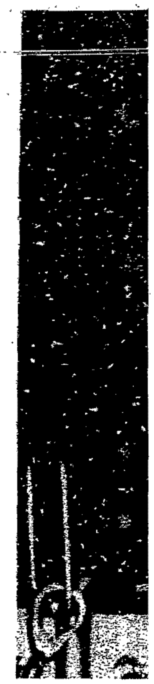
While GPUNC and the DOE TI&EP were preparing to ship the liners to DOE laboratories, technicians determined that the highly loaded liners were generating hydrogen and oxygen gases at rates which could produce unsafe concentrations during shipment. Technicians calculated gas generation rates of up to 1.1 liters per hour by monitoring the used liners both to assess the rate of increase in liner pressure and to analyze the composition of gases being generated. The data indicated that each liner's gas generation rate was proportional to both its curie loading and the amount of water remaining in it.

The TI&EP assembled a task force of technical experts to develop a solution to the radiolytic gas generation problem in SDS liners. After evaluating a list of possible solutions, the task force decided to test an approach in which catalyst pellets are placed inside each liner to recombine the radiolytic hydrogen and oxygen into water. Catalyst recombiners had been used successfully in homogeneous solution research reactors to recombine hydrogen and oxygen over long periods of time. The task force concluded that conditions for catalyst use in the SDS liners would have to be modified for successful application of the technique at TMI. Water would have to be removed from the liners to prevent possible catalyst submersion in the event of a shipping accident involving liner inversion, since catalyst action is inhibited when the pellets are submerged in water. Water removal was also expected to help reduce the radiolytic gas generation rate since those rates depended on the liner water content as well as the curie loading.

In compliance with task force recommendations for use of catalysts in SDS liners, Westinghouse Hanford Company developed a vacuum outgassing system to remove residual water from the liners. Vacuum outgassing removes water by reducing the pressure below the vapor pressure of water at ambient temperature. The residual water then boils off at room temperature. In performance tests, the vacuum outgassing system successfully removed 10 lb of water per day from a nonradioactive liner.

Rockwell Hanford Operations (RHO) conducted laboratory tests during the late spring and early summer of 1982 to evaluate the use of catalyst recombiners in SDS liners. They selected Englehard Type D platinum-palladium catalysts for the TMI studies. RHO performed these tests on a nonradioactive SDS liner at three different liner pressures and in upright and inverted positions to simulate the possible conditions under which the catalysts might have to perform during shipping. The tests were conducted with gas generation rates of up to 3 liters per hour, more than twice the rate (1.1 liters per hour) observed in the highest loaded liner at TMI. To comply with federal shipping regulations, the catalysts would have to maintain hydrogen concentrations in the liners below 4% by volume or oxygen below 5% by volume. All tests confirmed that the catalysts would successfully recombine gases produced at more than twice the maximum gas generation rate observed at TMI.

Actual vacuum outgassing and catalyst addition would have to be performed at TMI from a remote location in order to protect workers from the high radiation in the SDS liners. RHO designed a combination vacuum outgassing and catalyst addition tool to allow TMI technicians to perform both functions remotely. When using the tool for vacuum outgassing, technicians connect the tool's 1-1/2-in. diameter pipe to the SDS liner vent port through which residual water can then be removed. Tests using the tool to add catalysts to the liner concluded that the pellets could be added remotely through the vent ports to a filter assembly inside each liner. Figure 8 shows a technician carefully pouring the catalyst pellets into the portal on one end of the tool. The Johnson screen filter assembly, with an area of 770 mm^2, is located below the vent port and can hold 236 g of the platinum-palladium catalysts. From the Johnson screen assembly, shown in Figure 9, the catalysts experience enough gas flow to successfully recombine the radiolytic gases.



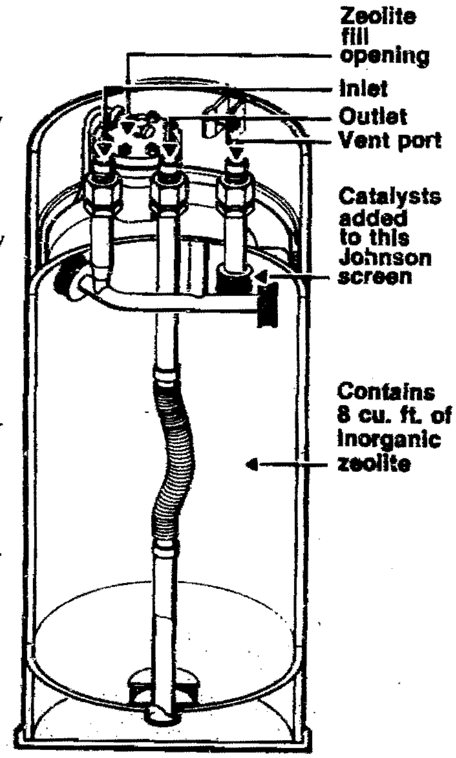


Figure 9. **Cutaway view of an SDS liner showing Johnson Screen to which catalysts are added.**

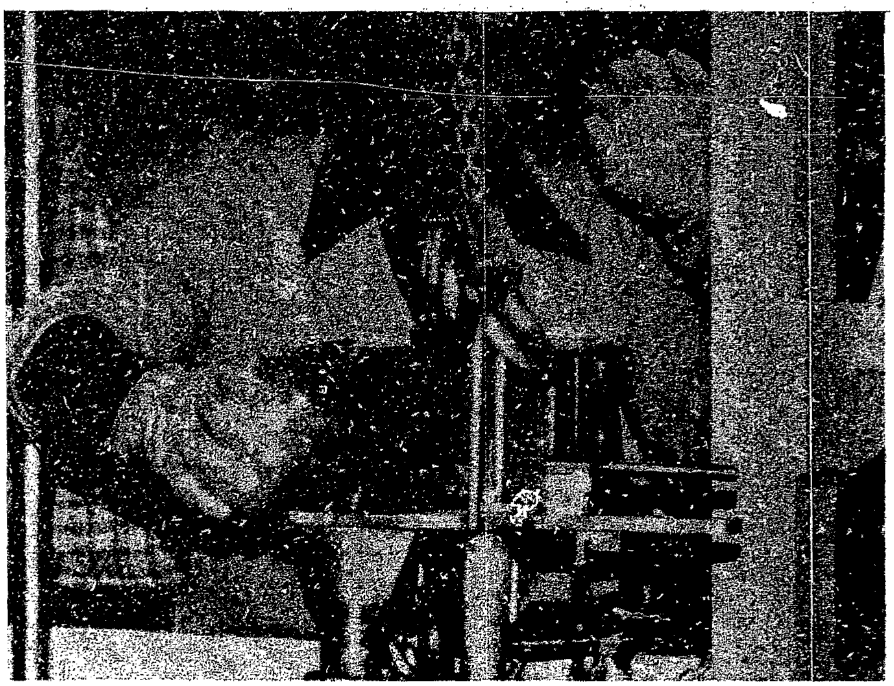


Figure 8. **Technician adds catalyst pellets to SDS liner through catalyst addition portal.**

Once all testing on nonradioactive liners proved the viability of the suggested techniques, tests were conducted at TMI on the most highly loaded radioactive liner, D10012, to observe the process under actual conditions. During the demonstration, the vacuum system performed as expected and the catalysts worked successfully to recombine the radiolytic gases. As part of the demonstration, the pressures in radioactive test liner D10012 were then monitored during a 14-day observation period. Monitoring confirmed that the catalysts were effectively recombining the radiolytic gases.

Since December 1982, the combined vacuuming outgassing and catalyst recombiner approach has been used in preparing all SDS liners for shipment. The test liner D10012 left TMI for Pacific Northwest Laboratory on December 31, 1982. When the shipment arrived at PNL on January 3, 1983, PNL sampled the liner gases through the liner's vent hose. The results, shown in Table 3, indicate that the catalyst controlled hydrogen concentrations below 4% as required by federal regulations that will be used for safe shipment. Shipments have since proceeded smoothly and on schedule so that by the end of May 1983, 9 of 19 liners will use for research had been shipped to a DOE research laboratory at Richland, Washington.

Gas	Composition (vol %)
Nitrogen	83.2
Hydrogen	2.1
Oxygen	12.3
Carbon dioxide	1.3
Argon	1.1

Table 3. **Gas sample results of Liner D10012 after shipment.**

EPICOR Waste Canister Shipments Continue Ahead of Schedule

When the Department of Energy first prepared to ship ion exchange media canisters from the EPICOR II water processing system off TMI in 1981, the list of canisters to be shipped numbered 50. Now, two years later, less than 10 remain to be shipped. As shown in Figure 10, shipments of these canisters are proceeding ahead of schedule. By the end of July 1983, all of the original 50 EPICOR II canisters will have been shipped from TMI.

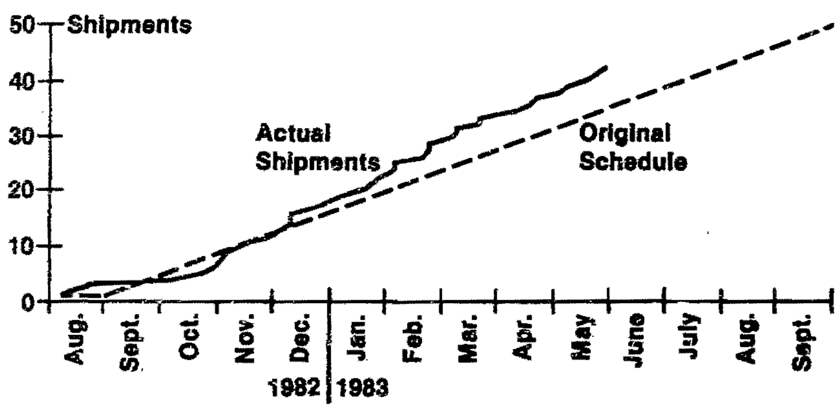


Figure 10. **Actual shipments of EPICOR liners are proceeding ahead of original projections, with completion in July, two months ahead of schedule.**

The canisters are prefilters from the EPICOR II water processing system at TMI-2, which decontaminated 500,000 gal of accident water from the TMI-2 Auxiliary and Fuel Handling buildings. The curie loadings on the canisters after processing accident water range from a low of 160 Ci to a high of 2200 Ci.

The first canister left the Island in May 1981 for characterization studies at Battelle Columbus Laboratories, where researchers concluded that the canister had suffered minimal damage as a result of exposure to the radioactive ion exchange media it contains. The liner then continued on to the Idaho National Engineering Laboratory (INEL) for further characterization. After that first shipment, regular shipments to the INEL began in October 1982 and have continued at a rate of three to six a month. At the INEL, researchers are studying the short- and long-term effects of ionizing radiation on various types of ion exchange media and on the canisters containing those media.

The characterization studies performed at the INEL will contribute to the development of technology needed to safely store, process, and ultimately dispose of the contaminated ion exchange media. Two disposition options for these canisters currently under examination are (a) ion exchange media solidification in a cement or polymer and (b) media isolation in a high-integrity container. Future Updates will discuss these disposal options and characterization studies as progress is made.

Information and Industry Coordination Serves Needs of Nuclear Industry

In support of TI&EP's overall goal of distributing information to industry, the Information and Industry Coordination Group (I&IC) was formed in late 1982 to collect and distribute technical information learned from the accident at TMI-2. Systems, which are already serving the nuclear industry, have been used by I&IC to receive and distribute information. Notepad, managed by the Institute of Nuclear Power Operations, is primarily designed for architectural and consulting firms, and the utility companies. NOMIS (Nuclear Operations and Maintenance Services), managed by NUS Corporation, for U.S. nuclear power utilities including GPU, is intended for maintenance and operations personnel and has the advantage of a mandatory feedback system.

The I&IC Group determines which audience needs the information to be distributed and sends it for transmittal to NOTEPAD or GPU as a member of the NOMIS network. Another responsibility is to review all incoming Notepad and NOMIS bulletins to determine if there are concerns to which the DOE TI&EP can respond. I&IC can then communicate with the persons requesting the information or can tailor information notices so that the proper people can be reached.

Many times, the I&IC Group will contact the manufacturers or users of certain instruments when specific problems with the instruments in an accident environment are encountered. If generic problems are encountered and neither Notepad nor NOMIS is well suited for dissemination, I&IC may publish the information through the TI&EP's established GEND reporting system, in trade articles, or make a presentation to the approporiate audience. For example, the I&IC Group has given presentations to IEEE meetings and has provided information on request to several utilities about heat stress.

I&IC is constantly upgrading, expanding, and tailoring the program to contribute to the needs of industry. For more information about I&IC, contact John Saunders or Jim Flaherty at (717) 948-1043.

Results of Quick Look Examinations Provide Damage Assessment

After months of extensive planning, preparations, and training, engineers and technicians conducted a series of visual examinations inside the damaged TMI Unit 2 reactor. The examinations, called a quick look, were conducted over a three-week period in July and August 1982. Although the quick look was limited in scope, it provided engineers and researchers with concrete evidence of the actual condition of the reactor core and upper internals. This information forms a basis for evaluating early accident damage assessments, performing future core damage research, and developing the necessary plenum and fuel removal tooling in preparation for reactor vessel head removal and ultimate defueling of the damaged reactor core.

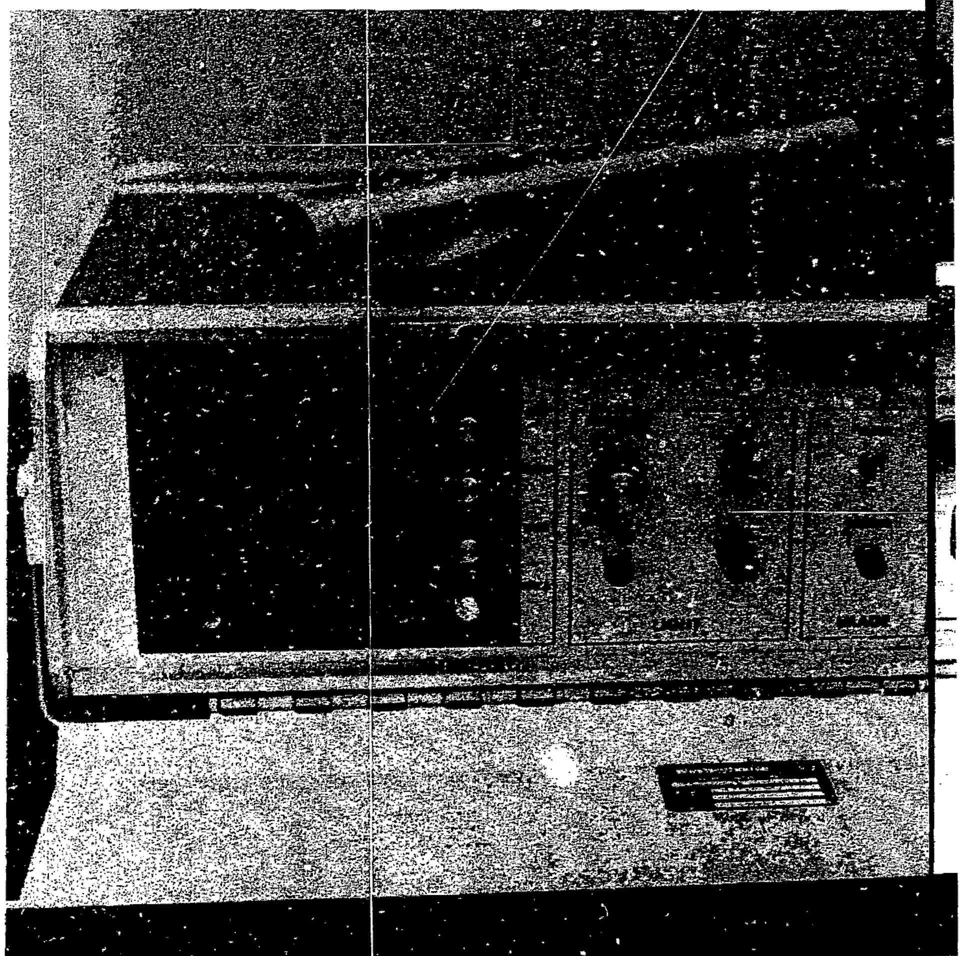
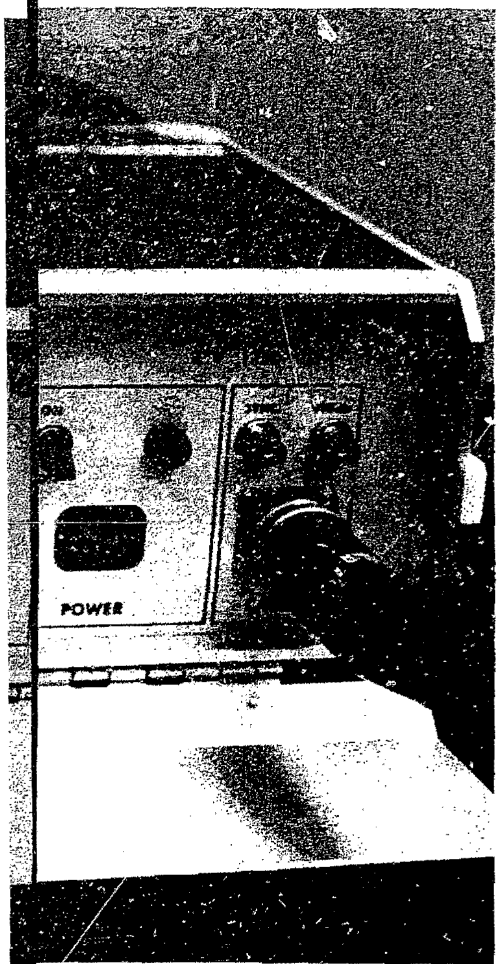


Figure 11. **Quick Look inspection camera and control unit.**



The primary objective of the quick look was to inspect the control rod guide tubes, a portion of the upper grid, the top of the fuel assemblies, and—if the fuel assembly upper end fittings were missing—the reactor core itself. A small, radiation-resistant, closed-circuit television (CCTV) camera (see Figure 11) was lowered through an opening created by the removal of a control rod drive mechanism (CRDM) leadscrew (see Figure 12). Because of the size constraints of the opening, the camera was manipulated using its power cable and a separate articulating cable attached to the tip. During the series of quick look examinations, the reactor internals were examined at three locations, which were selected to provide a composite picture of the reactor conditions: core center, midradius, and near the outer edge. The results of these separate examinations are discussed below.

Although the three examinations required the use of slightly different procedures because of the varying conditions at the inspection locations, the same basic sequence of events occured at each location. The inspections began with technicians lowering the camera through the CRDM motor tube into the reactor plenum to the general vicinity of the tenth support plate. Following preliminary inspections in the areas of the tenth support plate and the upper end fitting of the fuel assembly directly below the access opening, the technicians manipulated the camera to perform detailed inspections of the plenum components and adjacent fuel assemblies.

During the quick look, visibility was limited by water turbidity and the intensity of available light. These conditions caused the effective visibility range to vary from as little as 3 in. to a maximum of 24 in. from the camera lens.

The detailed examination of the reactor plenum assembly revealed that, overall, the plenum appeared to be intact and relatively undamaged. The interior surfaces of the CRDM guide tubes examined appeared to be in good condition. Flakes of debris were observed on the top of nearly every horizontal surface; these flakes measured approximately 1/8 in. in diameter or less and formed layers, some to a depth of 1/16 in. The thickness of the layers increased on surfaces closer to the core. These layers apparently were loosely deposited, because the motion of the camera in the water often disturbed the flakes. The undersides of horizontal surfaces and the faces of vertical surfaces were clean and free of loose debris. The vertical surfaces of the CRDM guide tubes, split-tubes, and C-tubes were relatively free of debris near the top of the plenum, but had some slight deposit of material in the lower portion. The bottom end of one of the split tubes appeared to have evidence of minor metal removal. However, some of the C-tubes only inches away were undamaged. All of the support plate brazements that were inspected appeared unbroken, free of distortion, and generally undamaged.

At the core center position, the entire upper end fitting was missing, as were all adjacent end fittings. The grillwork from the midradius upper end fitting was completely missing as was its control rod spider, spring, and spring retainer. The grillwork on each of the other upper end fittings visible from this location was present but partially melted and suspended from the plenum grid plate. One section of grillwork also had other identifiable components, such as a spacer grid, stubs of control elements, and partial fuel rods, suspended from it.

The insides of the midradius upper end fitting were scanned using the camera's right angle lens. The end fitting appeared to be in its normal position with respect to the grid structure. Metal chips and debris were found in the small space between the center tabs on the end fitting and the grid. In addition, some areas of the top portions of this upper end fitting have the appearance of having been cut by a torch, while adjacent areas appear to be in the as-manufactured condition.

The fuel assembly upper end fitting and spider assemblies were found in their normal positions at the outer-edge inspection location and one adjacent location. This indicates that the upper end fittings and the fuel assemblies in these locations were sufficiently intact to support the spiders.

Because the entire upper end fitting at core center location and the end fitting grillwork at midradius location were also missing, access to the active core region was possible. This examination revealed that a void exists in the upper central portion of the core. The void extends from the bottom of the plenum to the top surface of a rubble bed, approximately 5 ft below the bottom of the plenum and radially outward to just beyond the midradius inspection point. This void was

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

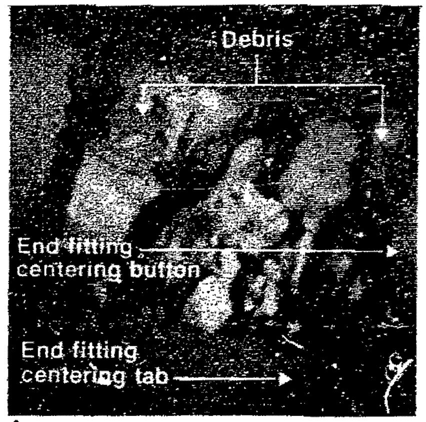
B. Damage to E-9 upper end fitting looks like metal after it has been cut by a torch.

C. Control rod element stub in upper end fitting grill work.

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

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

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



A

B

C

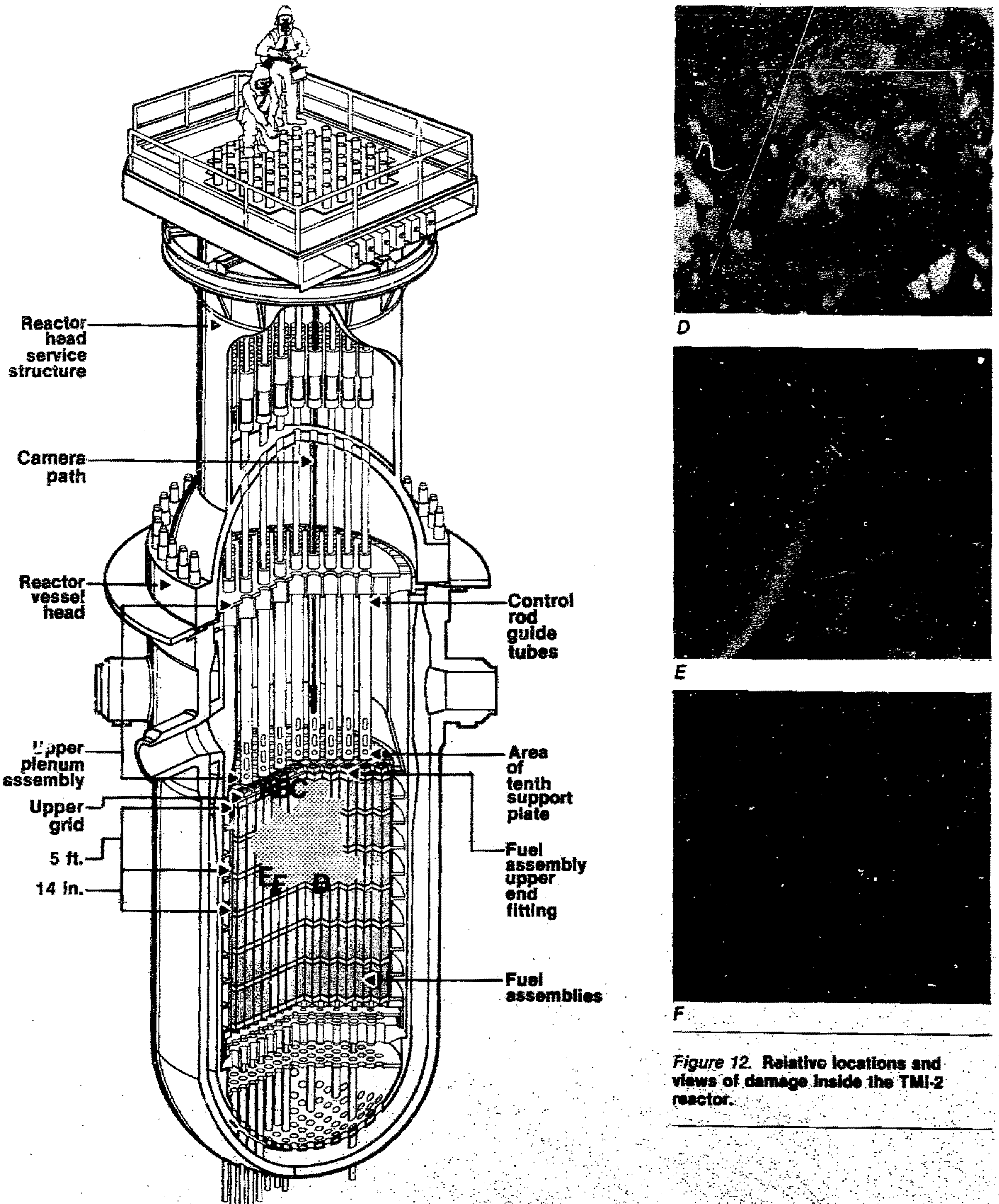


Figure 12. **Relative locations and views of damage inside the TMI-2 reactor.**

formed by the redistribution of fuel from central fuel assemblies. The rubble bed in the central region consists of fine granular particles, angular in shape, and approximately 1/8 in. in size. No recognizable shapes could be identified other than a portion of a control rod spider assembly. Engineers believe that this is the core center spider assembly which fell into the rubble bed when its leadscrew was uncoupled to provide access for the quick look camera. The general appearance of the rubble bed in the midradius region was considerably different than that at the core center location. In the midradius region, the rubble bed was comprised of much larger pieces and numerous recognizable shapes. Stubs of fuel rods were also observed protruding upward from the rubble and a forest of rods could be seen looking radially outward toward the west edge of the core. These rods and stubs were suspended from the remains of the upper end fittings that were still in place.

Probing of the rubble bed at core center and midradius inspection locations completed the quick look examinations. Technicans inserted a 1/2-in.-diameter steel rod into the reactor vessel through the CRDM guide tube until it came in contact with the rubble. The rod was then rotated and allowed to penetrate the debris to a depth of 14 in., where it was stopped by an unyielding obstruction. The rod penetrated the rubble bed to the same depth at both locations.

The results of the quick look examinations, when taken together with other core damage estimates, provide engineers with a more accurate description of core damage and demonstrate that work in and around the reactor itself can be conducted safely and efficiently. Engineers reviewing the quick look data have concluded that a number of the Unit 2 fuel assemblies sustained considerable damage, causing the formation of a void area and a rubble bed. This rubble bed consists of loose material and is not a fused mass. There was some evidence of partial melting of nonfuel material in components with melting points much lower than uranium oxide fuel; no evidence of melted fuel pellets was found. Engineers also concluded that the plenum assembly appeared to be substantially undamaged. The information and experience gained during the quick look provide a solid basis for conducting future recovery activities, including reactor head removal, plenum removal, and safe defueling.

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UPDATE

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TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 4, Number 1 December 15, 1983

New Tool Maps Shape of Damaged Core Internals

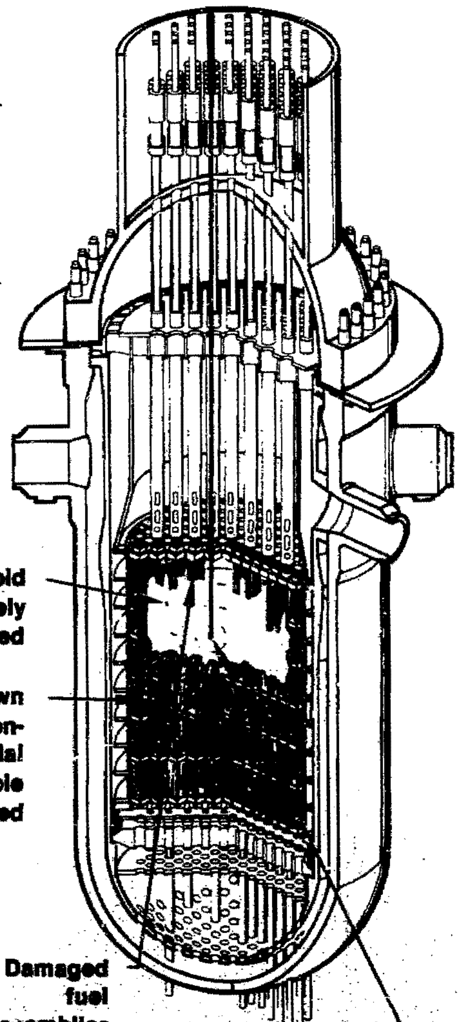

Figure 1. **The CTS sonar mapping device was lowered through a lead screw opening into the damaged TMI-2 core to obtain information on the shape of the core void area.**

A specially developed tool was used in August and September to ultrasonically determine the profile of the void area in the upper region of the damaged TMI Unit 2 core. The tool and its related components, together called the Core Topography System (CTS), were designed and built by U.S. Department of Energy (DOE) contractor EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL) to measure the size and shape of the cavity inside the damaged Unit 2 reactor. This cavity, discovered during camera inspection of the Unit 2 reactor in June 1982, resulted from fracturing and relocation of the upper portion of the core during the accident.

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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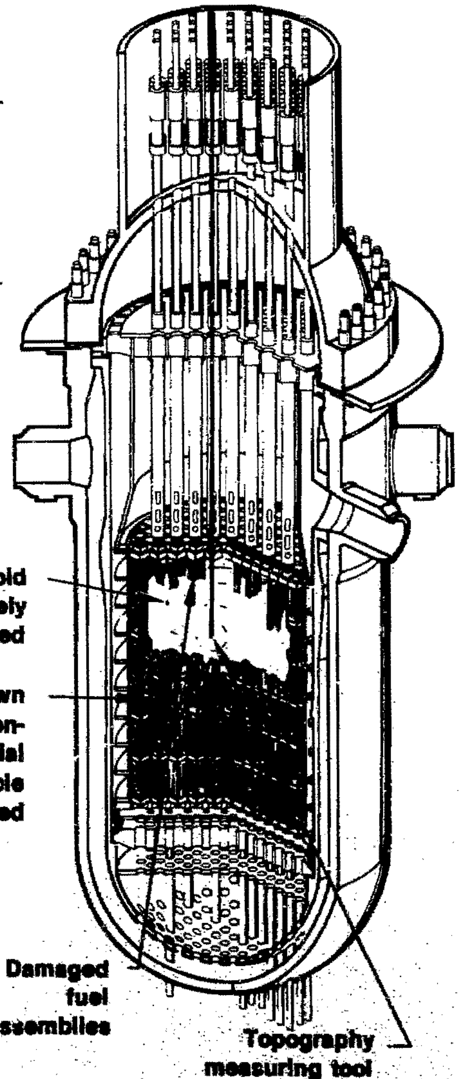

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The CTS consists of a rigid metal boom, by which the measurement devices

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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. □

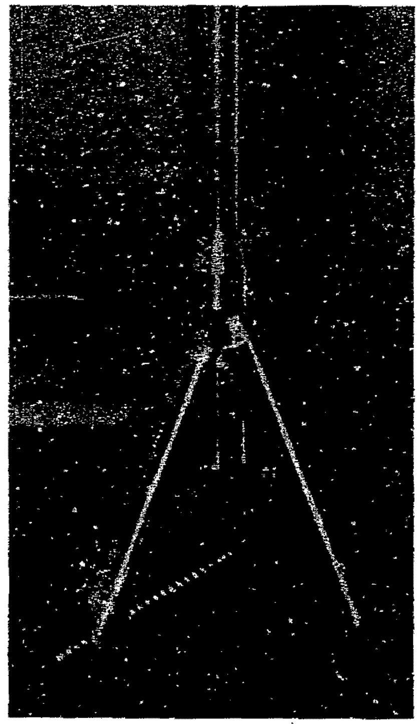


Figure 2. **Seen in its open position, this clamshell sampler was used to obtain debris from the surface of the rubble bed inside the TMI-2 core.**

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. □

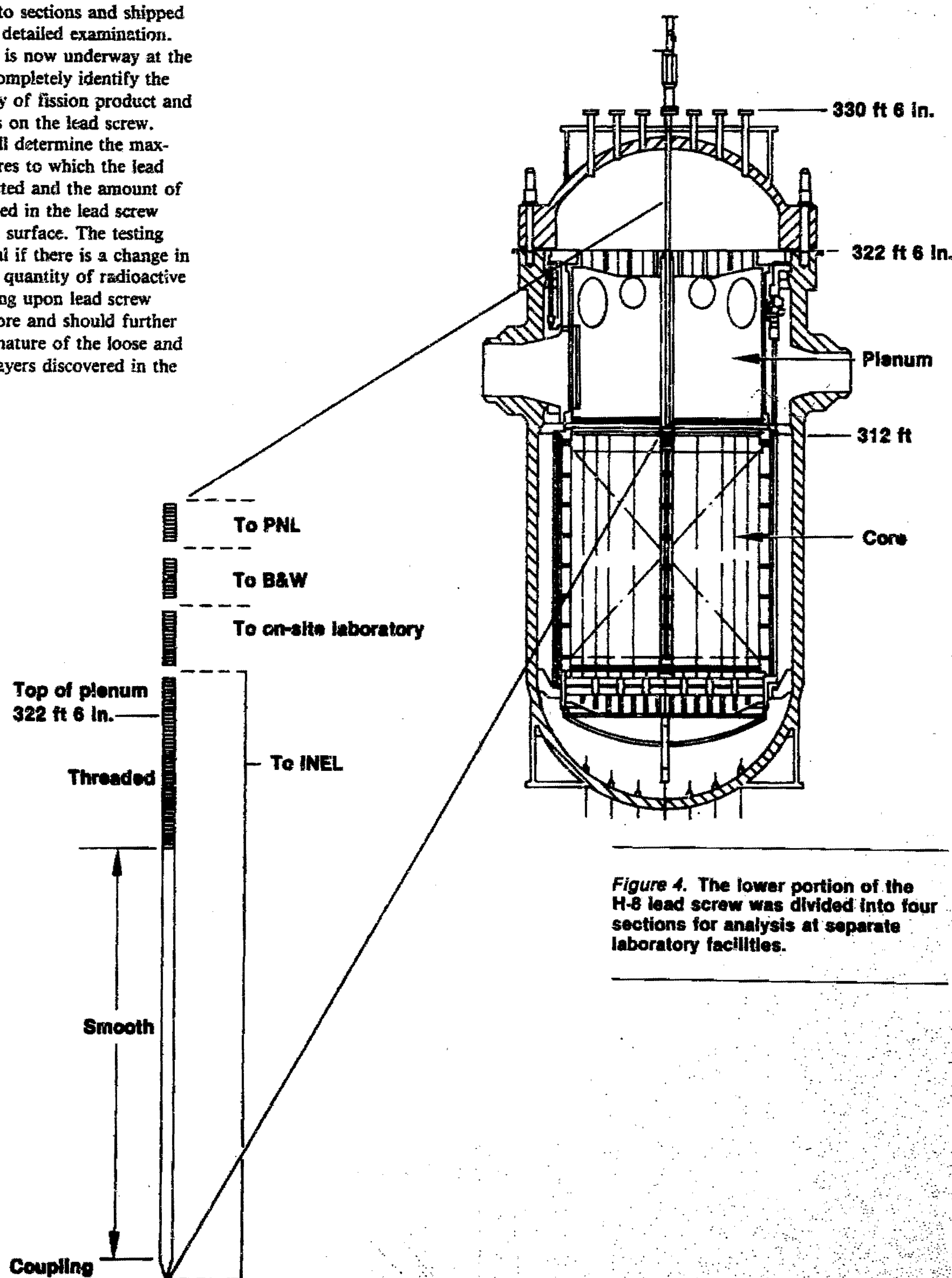


Figure 4. **The lower portion of the H-8 lead screw was divided into four sections for analysis at separate laboratory facilities.**

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.

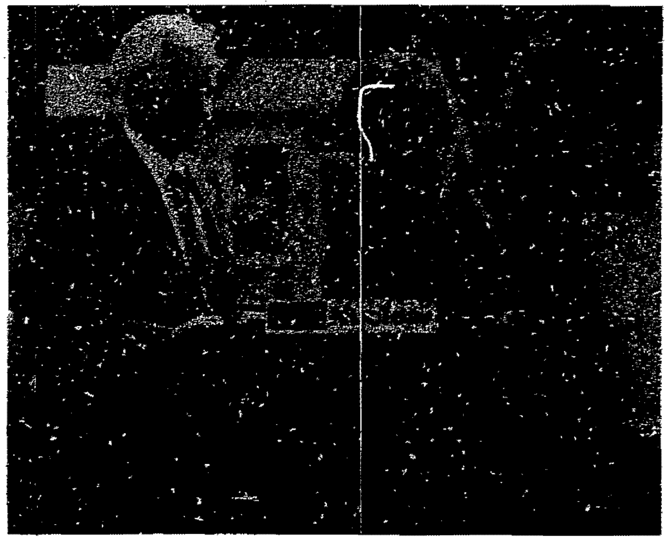
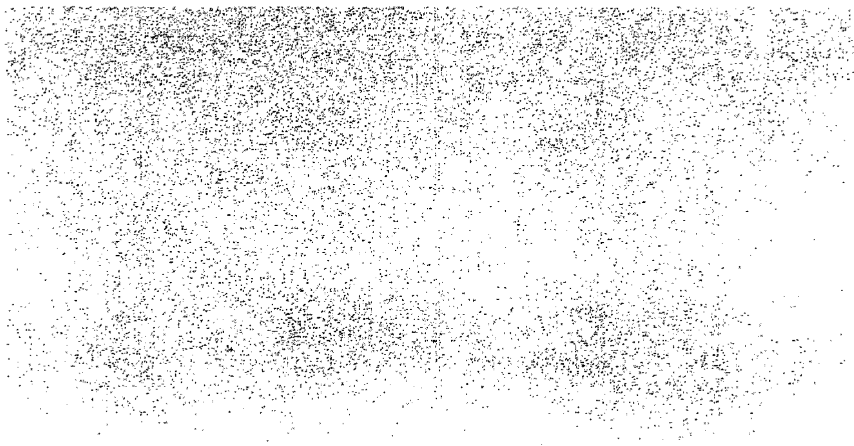


Figure 5. **Dr. Shelby Brewer, DOE Assistant Secretary for Nuclear Energy, accepts the last SDS liner used to process accident waste water while GPU Nuclear President Robert Arnold looks on.**

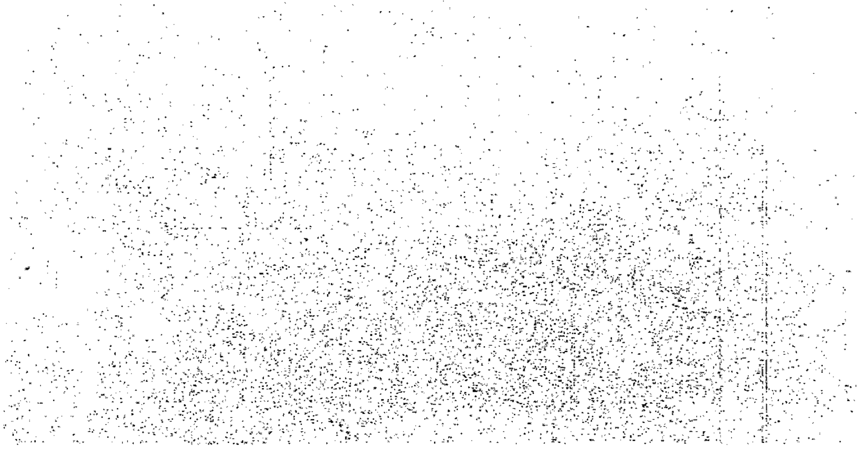
Of the 19 SDS liners that DOE is accepting for research and disposition projects, only six were not used in basement water processing. These six liners, used to process Reactor Building decontamination water and reactor coolant system water, remained on the island following the August ceremonies there, 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. □



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. □



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.

Figure 6. **A new HIC, still wrapped in shipping material, sits on a lowboy 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. □

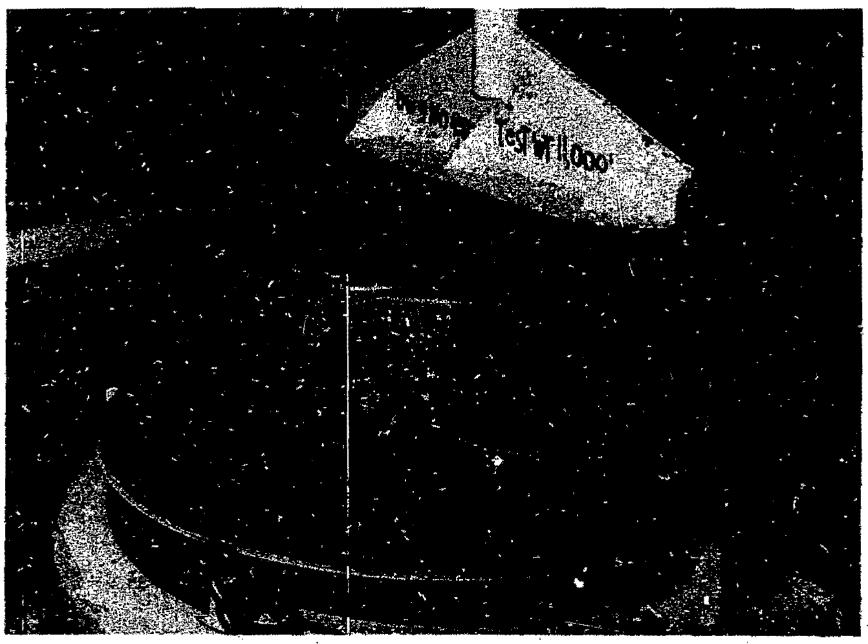


Figure 7. **A nonradioactive EPICOR liner is lowered slowly into the HIC.**

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 $\mu Ci/g$ in the A vessel solid sample to 16.9 x 10^3 $\mu Ci/g$ 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	0.88 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.

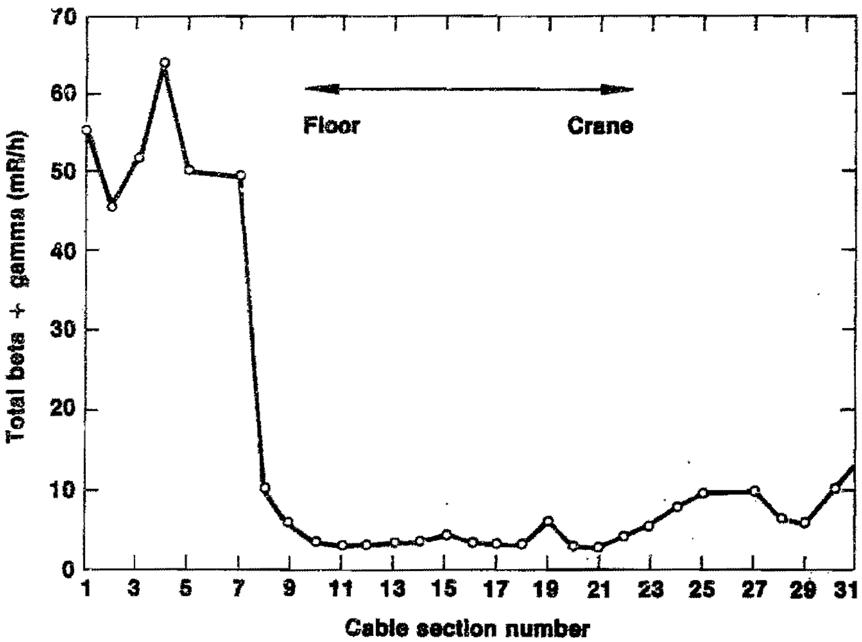


Figure 8. **Radiation levels on the polar crane cable were a function of location along the length of the cable from the crane to the floor. Radiation levels decreased as the distance of the cable from the floor increased.**

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

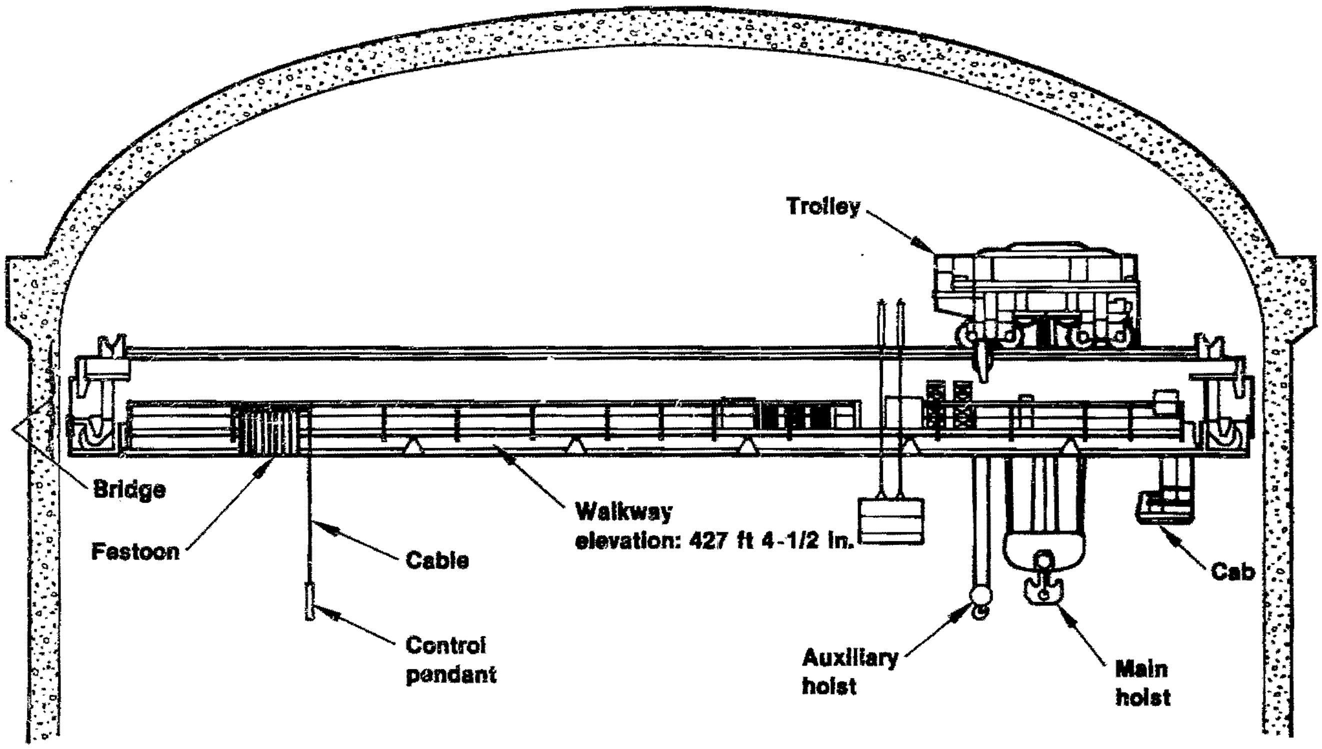


Figure 9. **Major components of the TMI-2 polar crane were refurbished during recent work preparing for reactor vessel head removal.**

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.

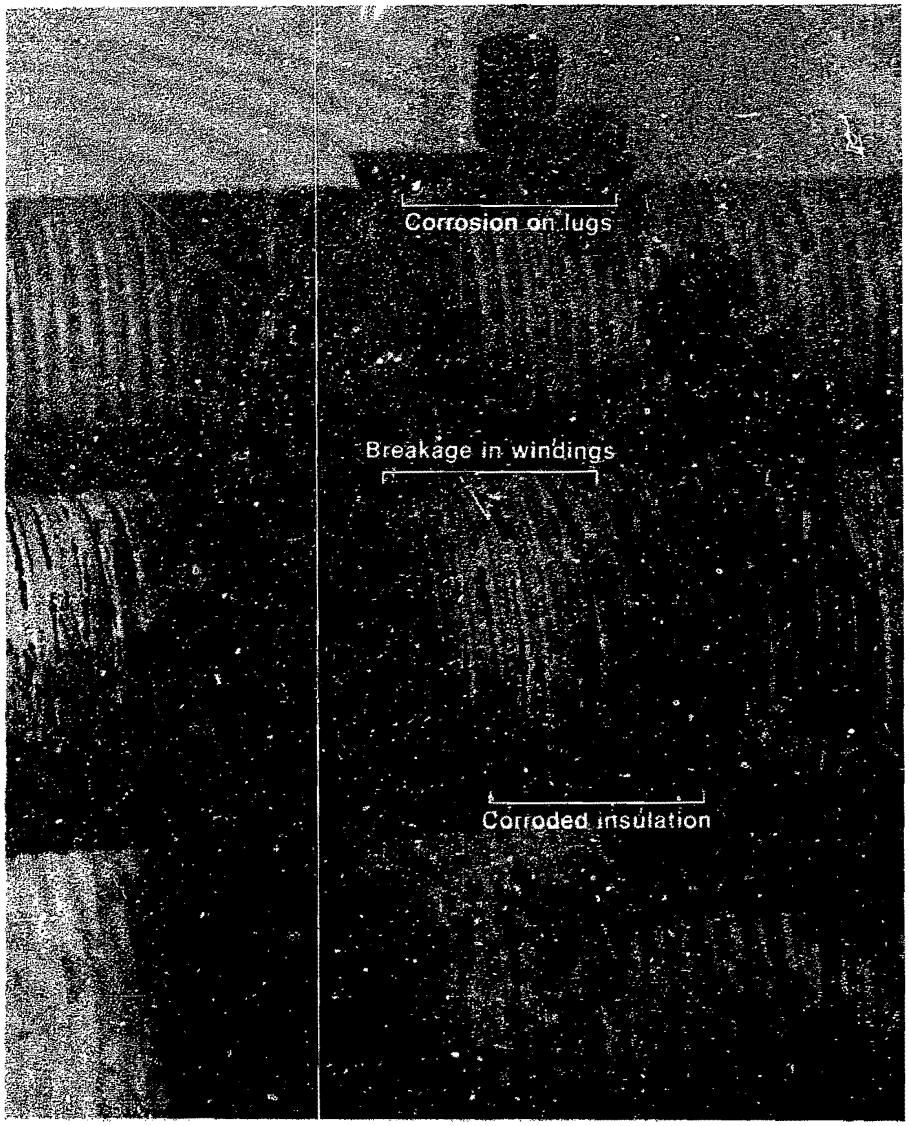
Because of the polar crane's strategic importance in removing the reactor vessel head, GPU Nuclear and the Polar Crane Task Group decided inspections, refurbishments, and tests should be aimed at restoring functional capabilities of strictly the bridge, trolley, and main hoist mechanisms using pendant 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 100 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.

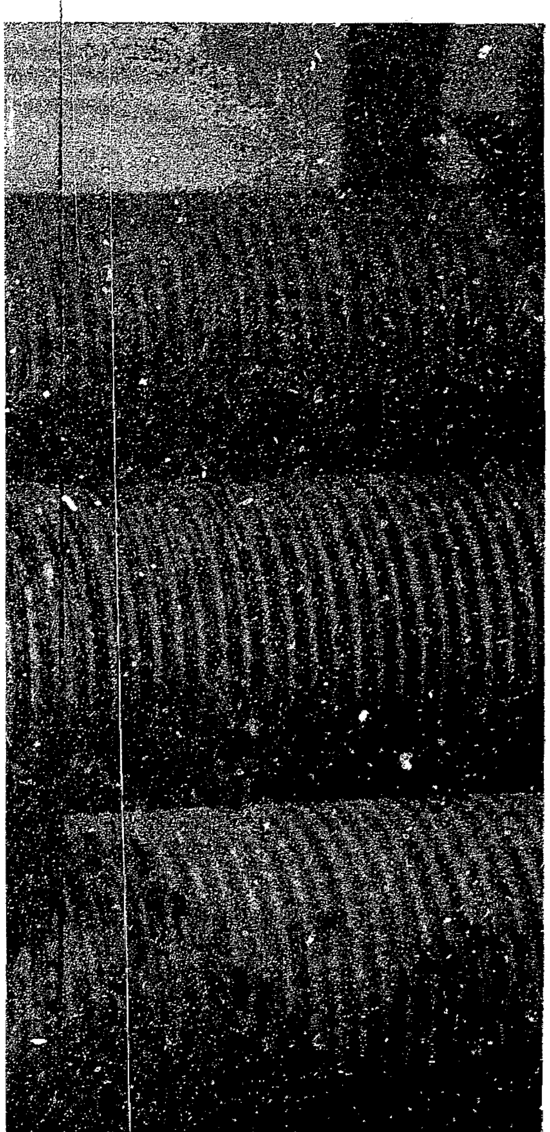


Figure 10. **Insulation corrosion and winding breakage in these polar crane resistor banks caused open circuits and low resistance.**

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) 948-1045.

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. □

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UPDATE

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TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 4, Number 2 June 15, 1984

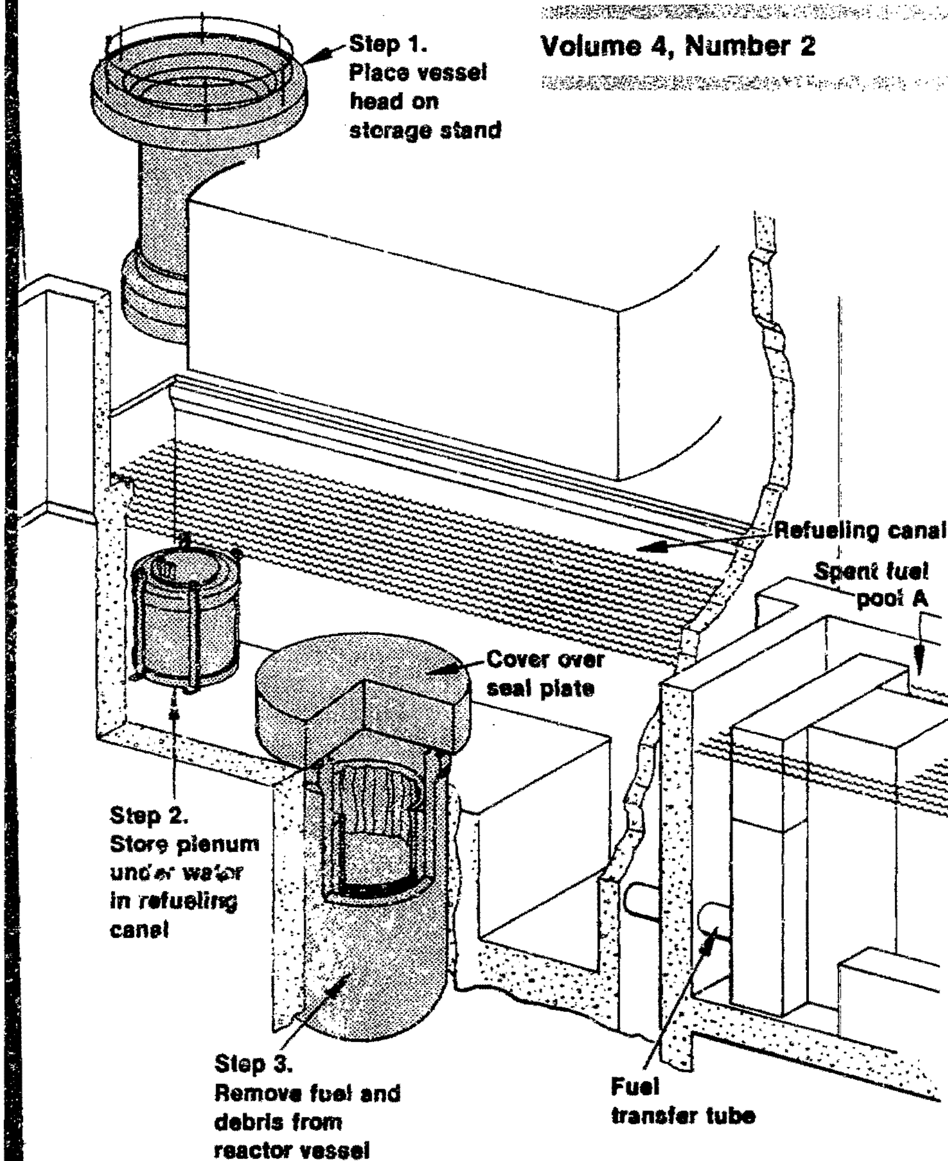

Figure 1 **This schematic illustrates the three major steps toward defueling Unit 2.**

TMI Moves One Step Closer to Cleanup with Head Removal

This summer, workers at Three Mile Island (TMI) will take another major step toward defueling the Unit 2 reactor vessel by removing the vessel's head and placing it on the storage stand. After several tests, engineers gave the go-ahead to *remove the head dry—without flooding* the refueling canal. These tests included radiation surveys, sampling, and video inspections of the underside of the head to determine radioactivity, analyze particle makeup, and look for debris that could make removal difficult. *Figure 1* illustrates the three remaining major steps toward disassembly of the reactor at TMI-2: placing the head on the storage stand, storing the plenum under water in the refueling canal, and removing fuel and debris from the vessel.

While these activities, and all of the vital projects in between, . .e scheduled to take the Technical Information and Examination Program (TI&EP) well into 1987, General Public Utilities Nuclear Corporation (GPU Nuclear) and U.S. Department of Energy (DOE) personnel have done much to bring the program to its present stage, discovering along the way new aspects of accident recovery and answering a number of important questions about the nature and impact of the accident.

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In preparation for head removal, the service structure and refueling canal both were decontaminated, and the lifting and handling equipment was tested. As part of a complete safety evaluation, engineers conducted a head-drop analysis, verifying that the components below could withstand such an impact. Looking for debris that could make head removal difficult, analysts took debris samples from the top of the plenum assembly and found the materials there were nonpyrophoric.

Two final projects in preparing the head for removal are currently underway. Workers are installing the canal seal plate, which will close the gap between the mouth of the reactor vessel and the refueling canal, and they are installing a temporary defueling water cleanup system as a precaution, should an unexpected event require the refueling canal to be flooded for shielding during head lift.

As *Figure 1* illustrates, the second major step toward disassembly is moving the plenum assembly to the shallow end of the refueling canal, where it will be stored under water until its removal. The plan is to first jack the plenum up from its current seated position and then lift it from the reactor vessel using the polar crane.

In 1983, major advances were made in designing tools to inspect and remove the plenum. Four 60-ton hydraulic jacks are being custom designed for the job. From a central pumping station, four operators will hand-pump the jacks, lifting the plenum slowly and in small increments so it may be continuously monitored from top to bottom to check for clearance and prevent jamming. Engineers are already aware from video inspections of the plenum's underside that partial fuel assemblies still hang from the plenum. After initial jacking, workers will knock off these stubs using a tool that is now in the design stages.

Once the plenum assembly is lifted, engineers will wrap it in a large, semi-rigid plastic bag, called the transfer contamination barrier, and place the entire assembly in the shallow end of the refueling canal.

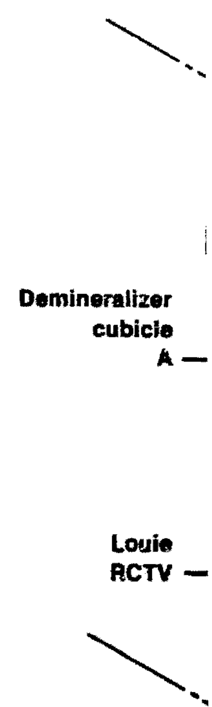
Figure 1 indicates that the final major step to the TMI-2 reactor disassembly is removal of fuel and debris from the reactor vessel. GPU Nuclear and TI&EP engineers have been concentrating on a number of aspects of this task, including a major defueling water cleanup system, fuel removal tools, and canisters in which to ship the fuel and debris.

The extensive defueling water cleanup system to be installed essentially consists of two subsystems, one to provide water filtration and processing for the reactor vessel and contamination barrier, and a second to provide water filtration and processing for the refueling canal and spent fuel pool A *(see Figure 1)*.

Clearly, a key element to defueling is the fuel removal tooling, to which engineers and designers have been devoting much of their efforts. Westinghouse Electric Corporation, selected as contractor to design and fabricate these tools, suggested two designs for consideration. The first design—a manual, remote defueling system—reflects the technical specifications previously established for the necessary mechanical equipment and vacuum and separation equipment. The vacuum would remove fine materials as small as 10 μm and debris as large as fuel pellets. The system would then separate the debris from the water and load the debris into canisters.

Westinghouse's second proposed design also meets the technical specifications, but it is a more automated approach to defueling. The system would use robotic arms to position a vacuum hose and load into a shredder materials too large for vacuuming. All of the debris then would be pumped in a slurry out of the reactor vessel for separation and canister loading.

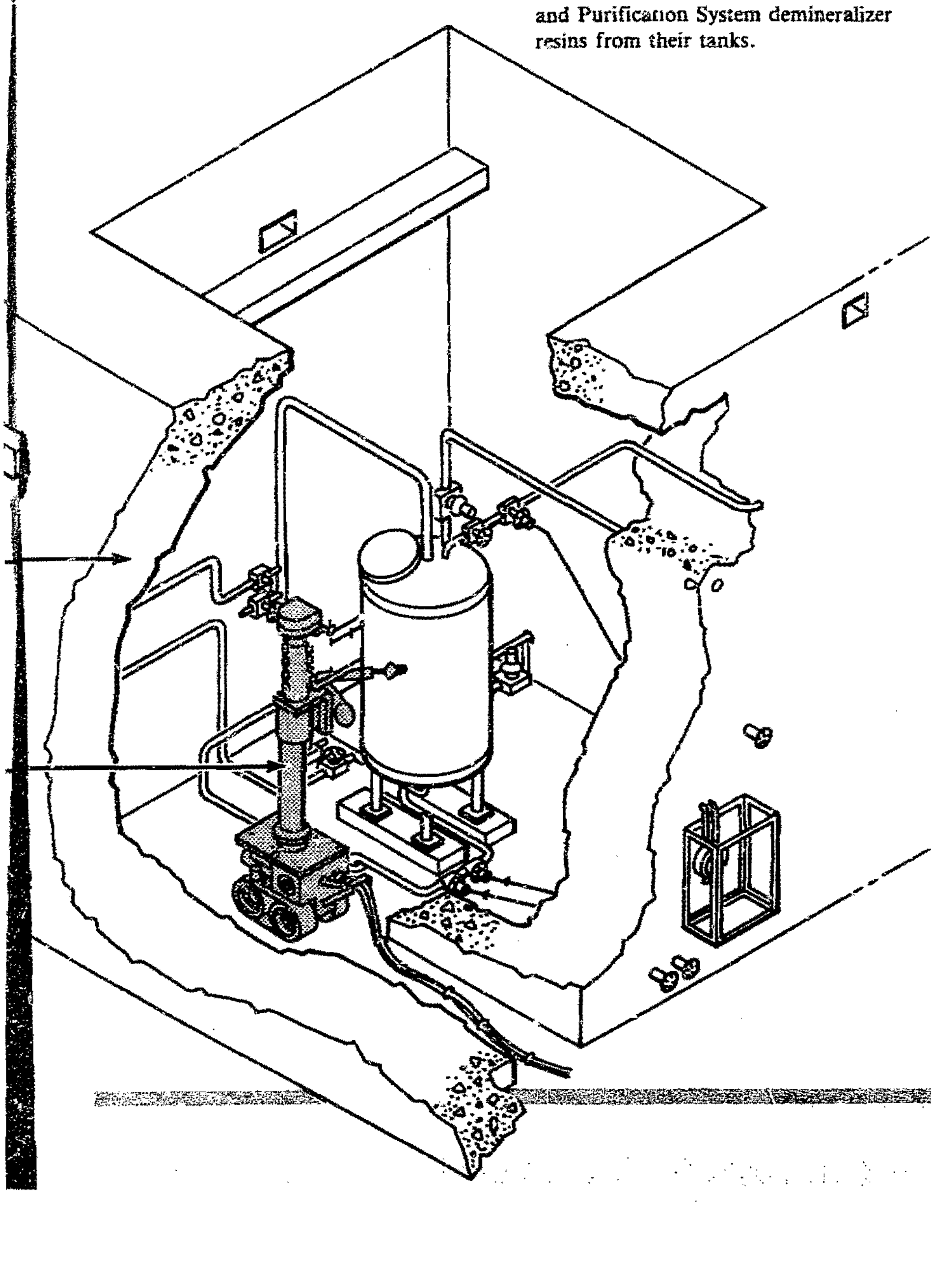
After discussing options for fuel canisters, TI&EP and GPU Nuclear representatives selected a design wide enough to hold intact cross sections of fuel assemblies. This canister is, however, shorter than anticipated because engineers do not expect to be shipping any full length fuel assemblies; few if any appear to be intact. A benefit to using this canister design is the potential cost savings; government-owned rail shipping casks may be available for transporting the canisters containing the remains of the Unit 2 core from TMI. □



Robots Reduce Worker Radiation Exposure

Many work areas in the Unit 2 plant contain large amounts of radioactivity and many surfaces are highly contaminated. While workers are protected with anticontamination clothing and monitored with personnel dosimeters, the effort to keep exposure rates low is a challenging one. Researchers are finding that robots can take the place of humans for many tasks in high radiation areas. A look at two robots, destined for use at TMI, illustrates their useful role in reducing human radiation exposures.

"Left-handed Louie" comes to TMI from Westinghouse Hanford Operations after a long history of remote-control work in a variety of environments. Louie's official name is the Remotely Controlled Transporter Vehicle, or RCTV, but since an anonymous technician during the 1950s scrawled "Louie" on the robot's arm, the RCTV has been known by its more humanlike name. Louie will assist in a crucial waste handling operation at TMI, that of monitoring the removal of the Makeup and Purification System demineralizer resins from their tanks.



During the accident, reactor coolant water flowed through the Makeup and Purification System for nearly 18 hours and deposited significant amounts of contaminants on the resins in the purification system demineralizer tanks. Extensive efforts to determine exactly how contaminated these resins are have shown that cesium activity levels in the resins may far exceed known values for any other accident-generated waste in the plant. Research work reported in both the August and December 1983 issues of the *Update* obtained dose rate readings of 3000 R/h in the A cubicle, and 1000 R/h in the B cubicle.

The contaminants in the resins are soluble fission products. They will be removed through a series of rinsing and elution steps conducted remotely through specially adapted plant systems. Dose rates inside the purification resin tank cubicles are much too high to permit human workers to enter routinely, but the resin tanks must be repeatedly monitored to see if the rinsing and elution steps are decontaminating the resins. Left-handed Louie will be used for the job. As shown in *Figure* 2, Louie will carry a gamma radiation detector into a resin tank cubicle and hold the detector in a predetermined place up against the tank wall. Technicians seated at control panels outside the cubicles can then monitor gamma radiation levels.

Figure 2 **Louie will enter the demineralizer cubicle to monitor gamms activity in the resin tank during cesium elution.**

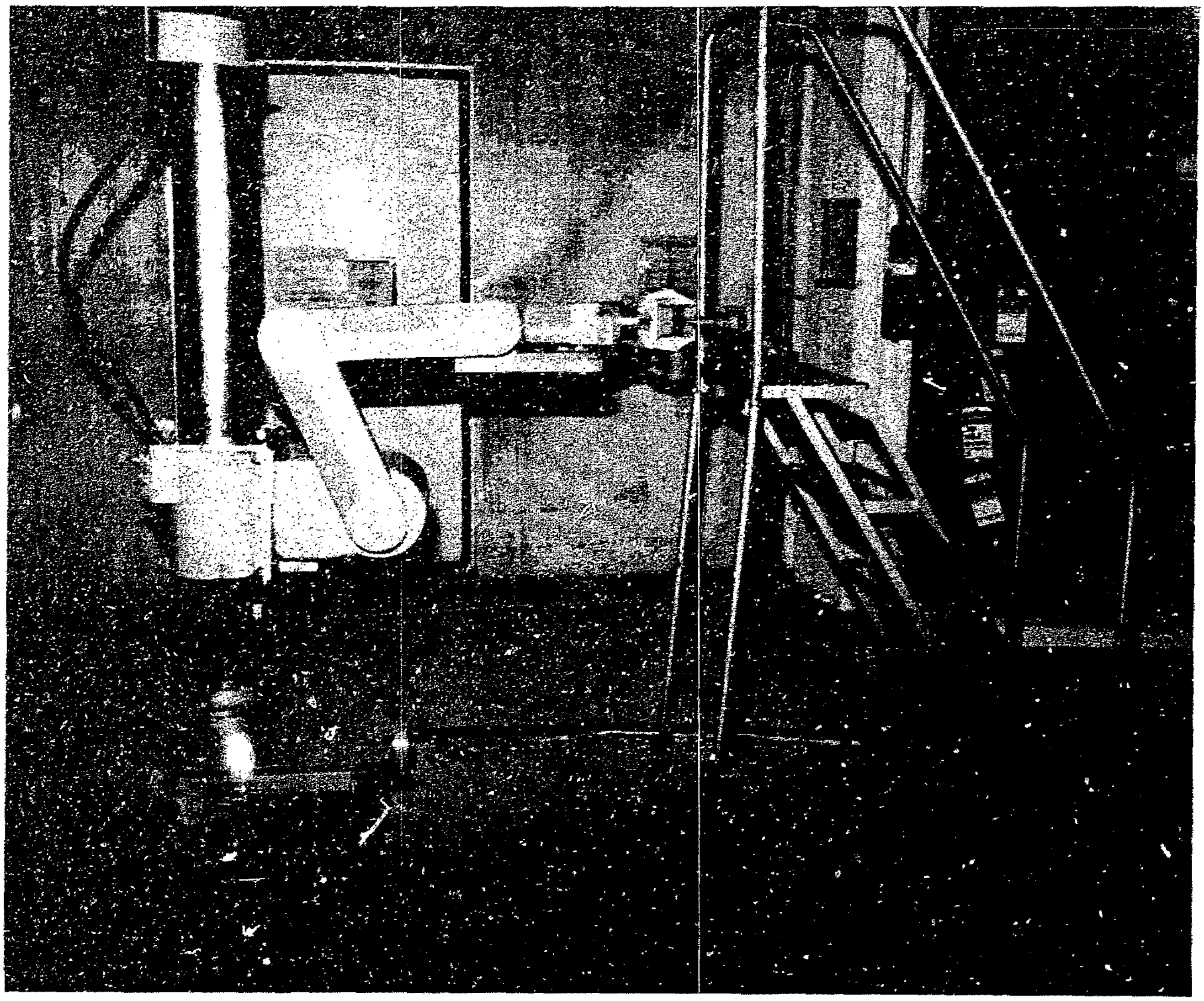


Figure 3 **Louie's 1.3-m manipulator arm will help monitor cesium levels in a high radiation area.**

Louie's components include a transporter base, a telescoping tube, a manipulator arm, three cameras, lights, and a 30-m-long control cable. Pictured in *Figure 3*, Louie's telescoping tube and manipulator arm have a combined vertical reach of 2.9 m, which is ample height to reach nearly to the top of the 3-m-high demineralizer tanks. Its manipulator arm can reach as far as 1.3 m horizontally in any direction. Louie's strength will not be needed for its demineralizer work, but the telescoping tube can hoist up to 450 kg, and the manipulator arm can lift and maneuver up to 68 kg.

Louie's cameras are radiation hardened, and will be the operating technician's "eyes" in the cubicles, not only for positioning Louie and the gamma monitor correctly, but also for observing conditions inside the cubicles during the rinsing and elution processes. The 30-m cable connecting the transporter to the control console outside the cubicles weighs nearly 23 kg.

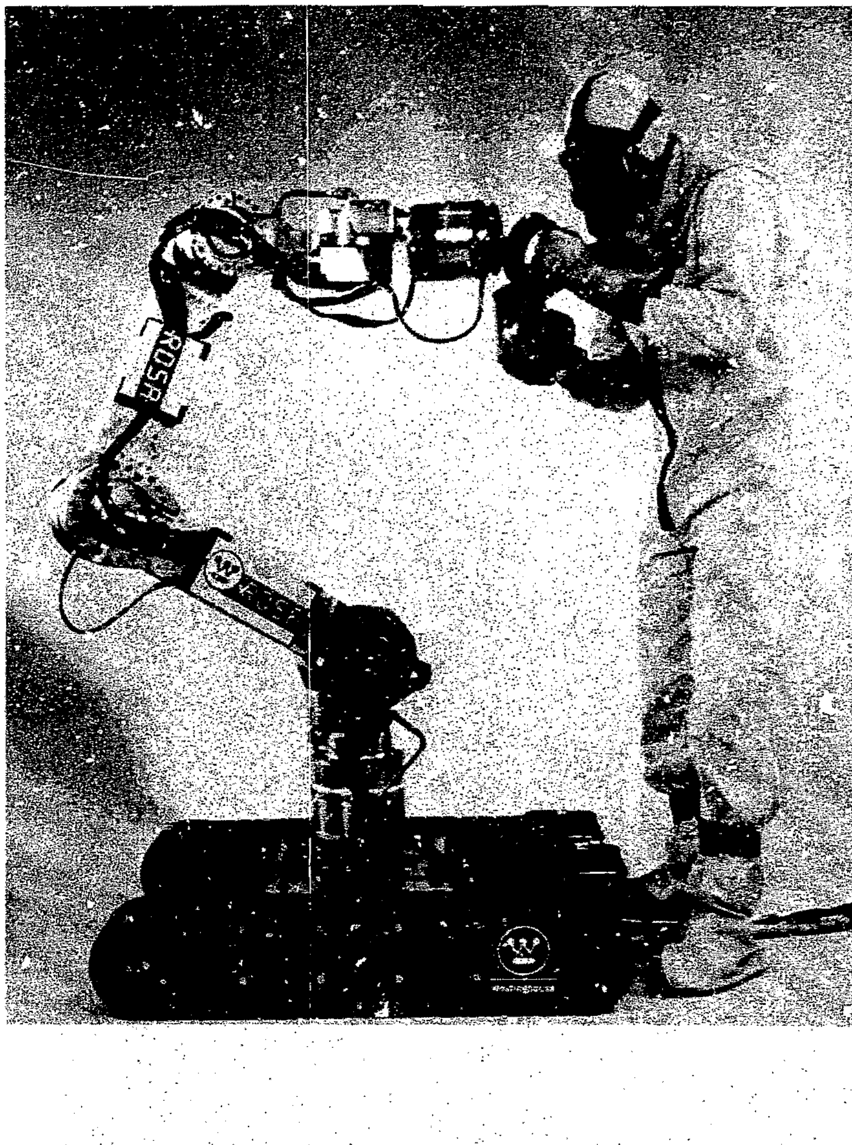
This robot is a simple one, whose every movement must be dictated by the operating technician. It has no ability to interpret commands or repeat activities on its own. Classed as a "master-slave" robot, it is basically a hot cell manipulator arm on a transporter. But its simplicity does not diminish its usefulness to the cleanup task in which it will serve. The radiation monitor Louie will carry will help determine when the ^{137}Cs concentrations in the demineralizer resins have been reduced to the lowest practical level. Operating technicians will conclude the rinsing and elution process when the gamma detectors show no change in readings after two consecutive elution cycles. Louie could be exposed to dose rates of up to 3000 R/h in the course of the cesium elution process. Its use will greatly reduce the human exposure risks involved in the important task of eluting and removing the purification system resins from the plant.

Robots may play a key role in the defueling of the reactor vessel, scheduled to begin in late 1986. A remotely controlled tool positioning system called ROSA might be used in place of long-handled tools handled by human workers in the Reactor Building. ROSA is a Westinghouse Electric Corporation robot; its name is an acronym for Remotely Operated Service Arm. It can be used in connection with a Westinghouse-designed defueling system to remove all debris from the damaged Unit 2 reactor.

Research has shown that the accident reduced much of the original core to a bed of rubble. The top 1.5 m of the core is now a large cavity, with a volume of approximately 9.5 m^3. Into this cavity, cleanup engineers will lower a variety of special equipment. Small, loose debris will be vacuumed out of the core into shipping canisters located inside the Reactor Building. Large debris will be reduced to a smaller size and then removed to shipping canisters. Material fused to the rubble bed will be cut loose with cutting shears, chisels, and drills. Items to be removed intact for research studies will be placed into debris baskets hanging inside the core. ROSA can do the lifting, rearranging, and positioning required for these special defueling activities.

Westinghouse's ROSA, shown on a transporter in *Figure 4*, is a manipulator arm with humanlike articulation at a "shoulder," "elbow," and "wrist." Its flexibility comes from its modular design. The arm consists of six segments, each of which is a self-contained unit with a motor, gear train, brake, and position indicators. Constructed of lightweight alloys, the arm is very strong for its 54-kg weight. When fully extended to its 2-m length, ROSA can lift 23 kg; when the arm is close to its base, it can lift more than 90 kg. With special adjustments, ROSA may be able to lift even more weight. ROSA's arm joints are all electric, but are completely watertight so the arm will work well in the flooded vessel and canal.

Figure 4 **ROSA's "shoulder," "elbow," and "wrist" move to commands issued through a computer.**



A 180-m umbilical cable will connect ROSA to its control computer, to be located in a trailer outside the Reactor Building. This computer, containing several microprocessors operating in parallel, is ROSA's brain. For manual control, an operator sits at the control console, issuing commands with a joystick. The computer translates these commands into arm movements.

Unlike the movements of such master-slave robots as Louie, ROSA's movements do not need to be dictated step-by-step. The computer calculates the arm joint movements required along six different axes to move the arm in the manner indicated by the joystick. The calculations are performed in a fraction of a second, and the arm responds almost immediately.

ROSA's computer can be preprogrammed with complete instructions for any standard task. The arm can also be taught to perform a specific activity. An operator first leads ROSA through the activity with the joystick and then instructs the robot to repeat that activity, unsupervised. ROSA's memory records all the information necessary to repeat the activity indefinitely. Both the preprogramming and the teachability will be useful for repetitive, tedious tasks.

Westinghouse developed ROSA in the early 1980s and has used ROSA in steam generator repair work and inside reactor vessels. ROSA would remain the property of Westinghouse during its use at TMI. Indeed, it would be accompanied by a Westinghouse technical adviser and operated by Westinghouse technicians. But the robotic arm can far outwork its human counterparts. While they are restricted to eight-hour days, ROSA can work without rest or maintenance for 1500 nonstop hours. ROSA's flexibility, strength, and ease of operation can help to complete the defueling operation safely and efficiently.

Louie and ROSA can complete vital cleanup tasks along the road to cleanup of the TMI-2 plant. They are designed to be versatile, and each could be used for jobs not discussed here. GPU Nuclear is working with the Electric Power Research Institute and Carnegie-Mellon University to develop another robot for performing characterization and cleanup tasks, and the potential for using robots in other areas in the future continues to be examined. These machines can play a useful role in both simple and complex jobs as they help keep human exposure as low as is reasonably achievable.□

Source Term Assessment Continues at TMI

The TI&EP continues to make progress in its studies of source term, recently completing visual inspections of the TMI-2 Reactor Coolant Drain Tank (RCDT) and investigating the concentration of radioiodine and tellurium in the reactor core and building basement. In a related effort, GPU Nuclear engineers took concrete core samples on the 305-ft and 347-ft elevations to assess radionuclide penetration into the concrete.

The RCDT was the major pathway for the release of accident water into the Reactor Building basement. In video inspections of the tank below the vent line, analysts saw a dark, particulate sediment of less than 0.32 cm thick, nonuniformly distributed on the bottom. Personnel also saw particles larger than they predicted would have been released through the pressurizer to the drain tank; they calculated a maximum particle size of 30 μm.

During each of these inspections, samples of the water and sediment were extracted for analysis. By studying the samples, personnel at the Idaho National Engineering Laboratory (INEL) will determine the quantity of fuel and fission products released to the RCDT. The resulting data will also support ongoing analyses of fission product mass balance and source term—the concentration and distribution of radionuclide activity.

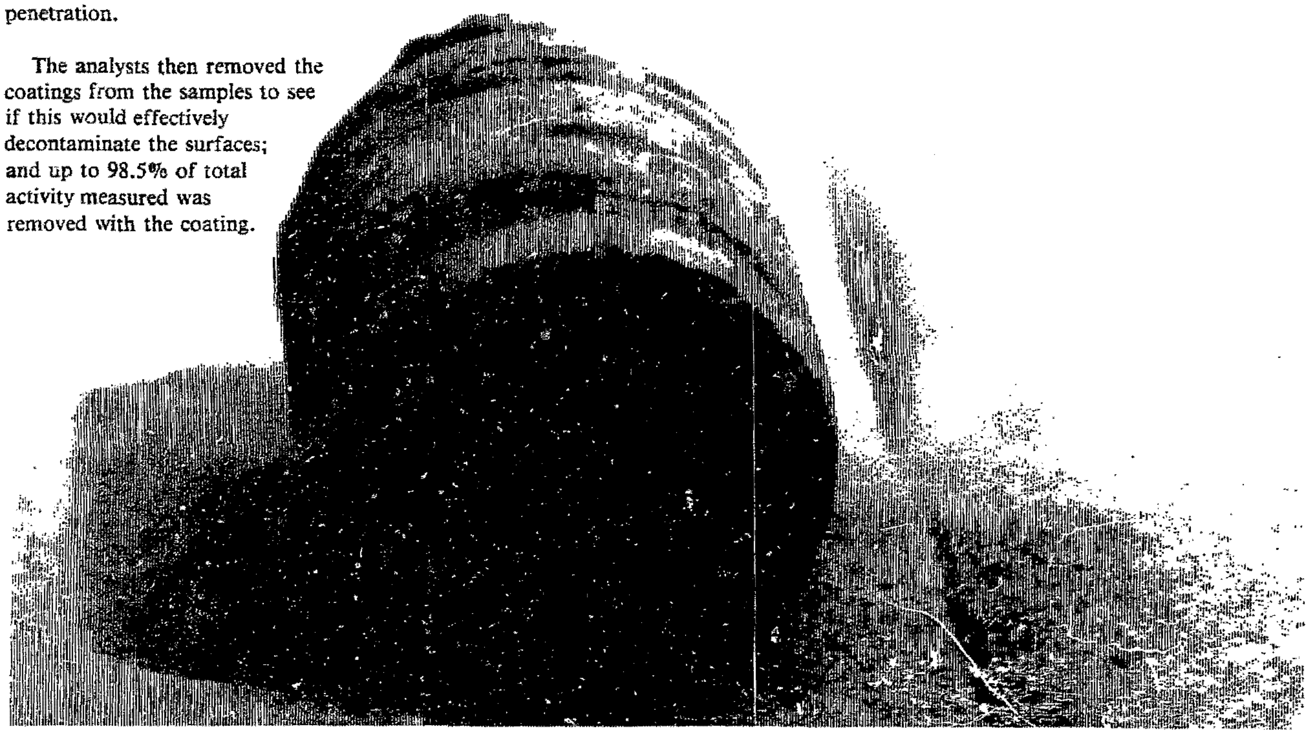
In defining the source term in Unit 2, program personnel are placing special emphasis on determining the concentration of ^{129}I and ^{130}Te in liquid and solid samples collected from the reactor core and Reactor Building basement. Data collected to date indicate that the fraction of radioiodine released from the core to the basement is significantly less than the fraction of radiocesium released to the basement. Basement sediment samples are being analyzed to determine if the radioiodine precipitated. Lead screw and core grab samples are being analyzed for ^{130}Te—whose behavior is important because it is a parent of iodine—to determine if plateout or scavenging of the tellurium significantly reduced the amount of radioiodine released from the core.

To assess the extent to which radionuclides migrated into the concrete in the building, engineers obtained 17 samples of concrete from the 305-ft and 347-ft elevations. (The basement concrete was not included in this study because of its inaccessibility.) Tests on these large core samples, obtained using a concrete boring tool, resulted in one significant finding: that the majority of radionuclides released from the Reactor Coolant System (RCS) into the Reactor Building environment were trapped in the concrete's surface coatings.

Most concrete surfaces in TMI-2 are protected with epoxy-based, nuclear-grade coatings, making an otherwise porous concrete resistant and easier to decontaminate. Pictured in *Figure 5* is a sample of concrete taken from the D ring wall at the 305-ft elevation. To its right is the sample's autoradiograph, which documents the coating's ability to absorb radionuclides and prevent them from penetrating the concrete. Only where the coating was scarred and the unprotected surface was exposed to contaminated water for an extended period of time did analysts discover significant radionuclide penetration.

The analysts then removed the coatings from the samples to see if this would effectively decontaminate the surfaces; and up to 98.5% of total activity measured was removed with the coating.

While analysts are not saying widespread coating removal in TMI-2 is necessary, it may be beneficial in areas where long-term personnel operations are planned. A GPU Nuclear report suggests devising other equally effective decontamination methods as alternatives to coating removal.□



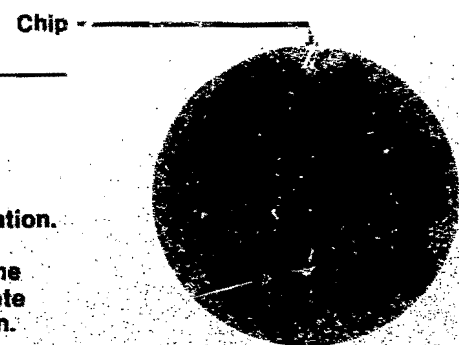


Figure 5 **This concrete core sample, whose coating was accidentally chipped during removal, was taken from the D ring wall at the 305-ft elevation. Its autoradiograph, at right, shows dark shading where the coating protected the concrete from radionuclide penetration.**

Hydrogen Burn Study Answers Questions About Its Cause and Damage

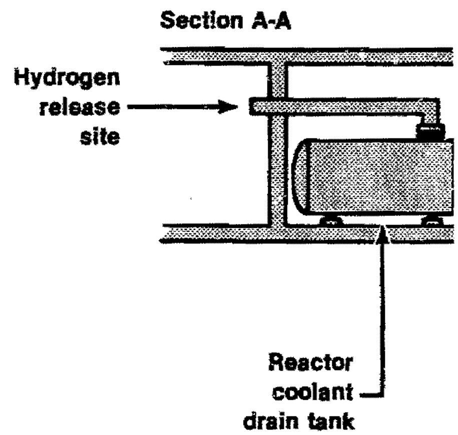
After two years of research and analysis, TI&EP engineers completed their studies of the hydrogen burn that occurred at TMI, answering a number of questions generated from the event and gaining a better understanding of its cause and its damage.

Overall, analysts concluded that the hydrogen burn caused little damage to the Reactor Building itself and no damage to safety systems. The damage that was found, for the most part, was fully consistent with TI&EP expectations.

The burn occurred about 10 hours into the accident, after the reactor core overheated and the zircaloy cladding that encased the fuel reacted with steam, liberating large quantities of hydrogen to the building, where the gas later ignited. The estimated pressure rise time for the event was 10 to 15 seconds, but most of the 28-psig pressure increase occurred in the final 3 to 6 seconds.

Evidence, such as charred paint and cables, indicates a flame rose from the building basement to the dome, where it remained until quenched.

The precise location and ignition source of the flame are unknown and may always be so, but evidence indicates the burn originated on the building's west side, in the basement. Among the candidate sources of the burn is the electrical equipment on two Motor Control Centers that tripped at the time of the event. Access to the 282-ft level will provide further insight into the ignition source and location.



Location of hydrogen release

Calculations by several researchers indicate that about 370 kg of hydrogen were in the Reactor Building at ignition, of which about 320 kg burned. A hydrogen and steam mixture was released into the building through the discharge duct from the RCDT, as illustrated in *Figure 6*. The hydrogen then rose through openings and became well mixed throughout most of the building by way of the ventilation system, with an average concentration being 7 to 8%, depending on water vapor concentrations. Somewhat leaner hydrogen concentrations existed in the unventilated enclosed stairwell and elevator hoistway, while richer concentrations existed immediately before the burn in the vicinity of the release site and open stairway.

Since the electrical equipment was operating in the presence of a flammable mixture for one to seven hours, analysts questioned why ignition did not occur earlier. They developed three possible explanations. First, releases of steam early in the accident generated relatively high concentrations of water vapor, thus raising the lower flammability limit of hydrogen in air. Second, sparks occurring earlier may have been too weak to cause an ignition. And third, it is possible that earlier ignitions did actually occur, but without any significant flame propagation away from the ignition site.

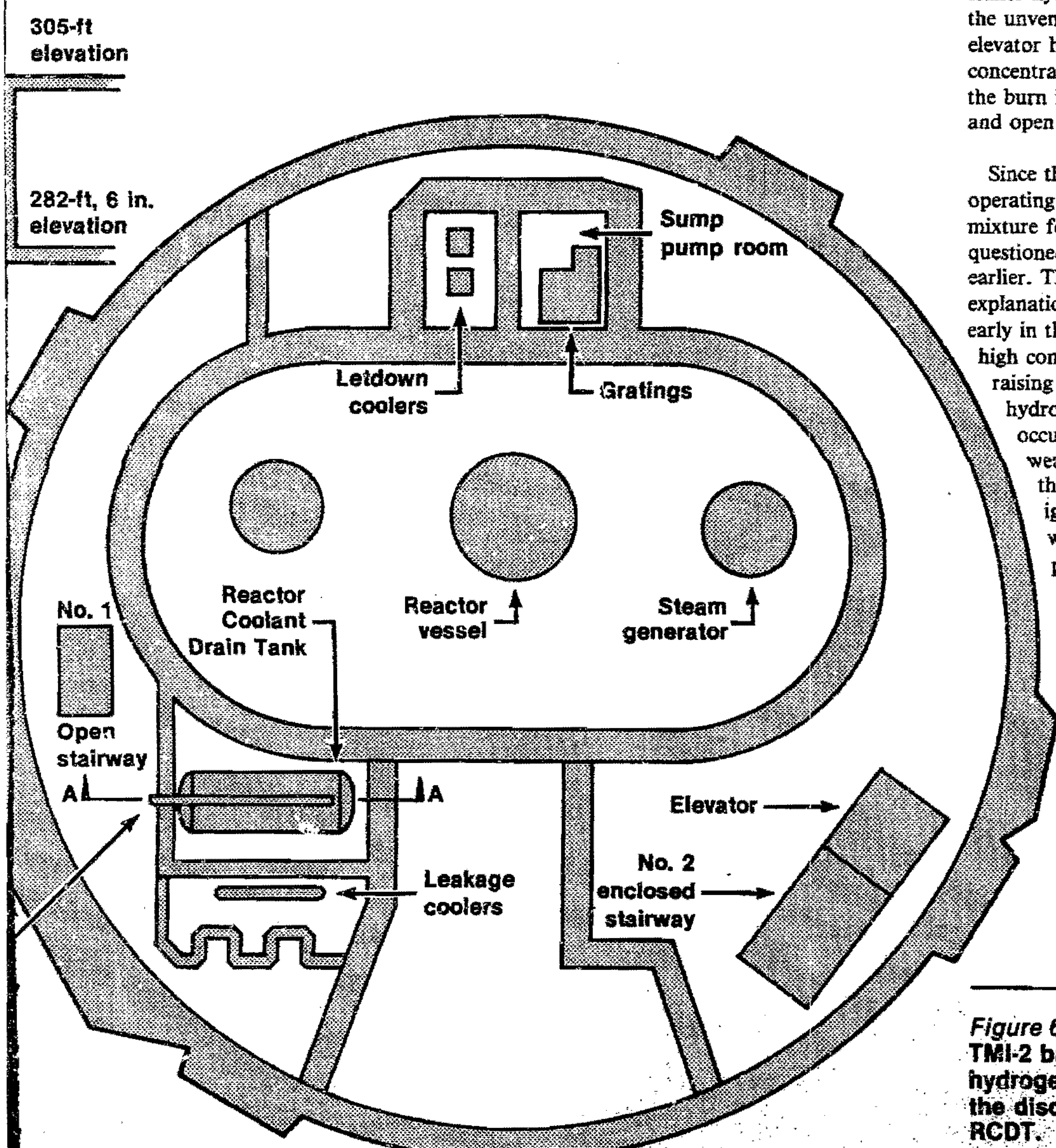
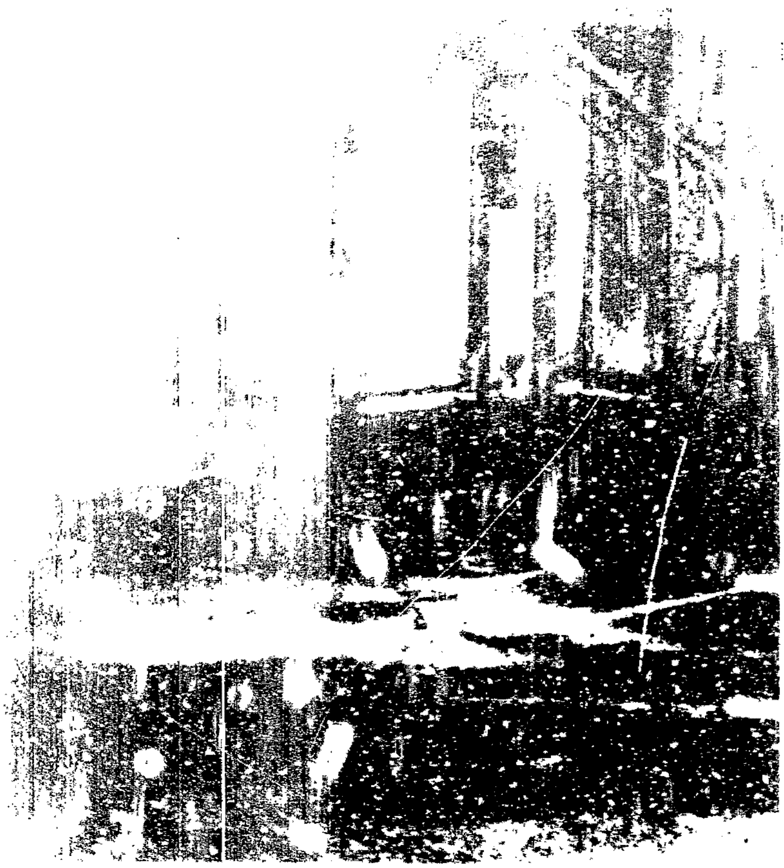


Figure 6 **As this drawing of the TMI-2 basement indicates, the hydrogen was released through the discharge duct from the RCDT.**



TMI Program Highlights Available on Videotape

The TI&EP has released a videotape program for loan without charge. "1983 in Review: A DOE TMI-2 Programs Brief" is a 16-minute program highlighting the accomplishments of 1983 at TMI. The program covers the shipment of the last processing container used to decontaminate accident water, experiments to immobilize the contaminated contents of this and other containers, the study of the hydrogen burn event, tests to determine cable and connection performance in accident and postaccident environments, sampling of the rubble bed in the core, and fabrication of a clear plastic model of the cavity in the damaged reactor, among other achievements.

This videotape may be obtained by contacting Kim Haddock, EG&G Idaho, Inc., TMI Site Office, P.O. Box 88, Middletown, PA 17057, telephone FTS 590-1019 or (717) 948-1019.

DOE Studies of Ion Exchange Media Focus on Gas Generation and Resin Degradation

Through DOE research into the accident at TMI, laboratory personnel have learned a great deal about organic and inorganic ion exchange media, both of which have been part of the cleanup process there. Researchers analyzed the media in the EPICOR II prefilters, Makeup and Purification System demineralizers, and Submerged Demineralizer System (SDS) liners, focusing specifically on two major concerns: radiolytic generation of combustible gases and resin degradation.

Organic ion exchange resins were used alone or in combination with inorganic zeolites in the 50 EPICOR II prefilters that decontaminated accident water from the Auxiliary and Fuel Handling buildings. Research on the EPICOR II ion exchange media focused specifically on radiolytic generation of combustible gases, resin degradation, and liner integrity. Laboratory scale studies have shown that at doses of about 10^7 rads and more, residual water in the organic resin decomposes and the resin itself degrades. Both processes generate gases which could lead to combustible gas mixtures in containers during handling, shipping, or storage.

The radiolytic decomposition of water produces hydrogen, which accumulates in sealed containers, and oxygen, which is consumed during a reaction with the organic resin and water in the container. The hydrogen gas generation rate is dependent on the radiation dose to the water and organic resin: the rate increases in a nearly linear relationship with an increasing curie content *(see Figure 8)*. Internal pressure in a sealed container will at first decrease sharply as oxygen in the air inside the container is consumed by a chemical reaction. The dose delivered to the organic resin acts as a catalyst for this reaction. When all the oxygen has reacted, the ongoing radiolytic process builds up hydrogen gas concentrations in the container, causing an increase in container pressure above atmospheric pressure.

At TMI, potential hazards associated with this hydrogen gas buildup in sealed containers led DOE researchers to develop a tool to solve the problem. A gas sampler with a remotely operated support facility safely vented, sampled, and purged gases in waste containers. The remote operation reduced both man-rem exposure and potential hazards associated with hydrogen gas.

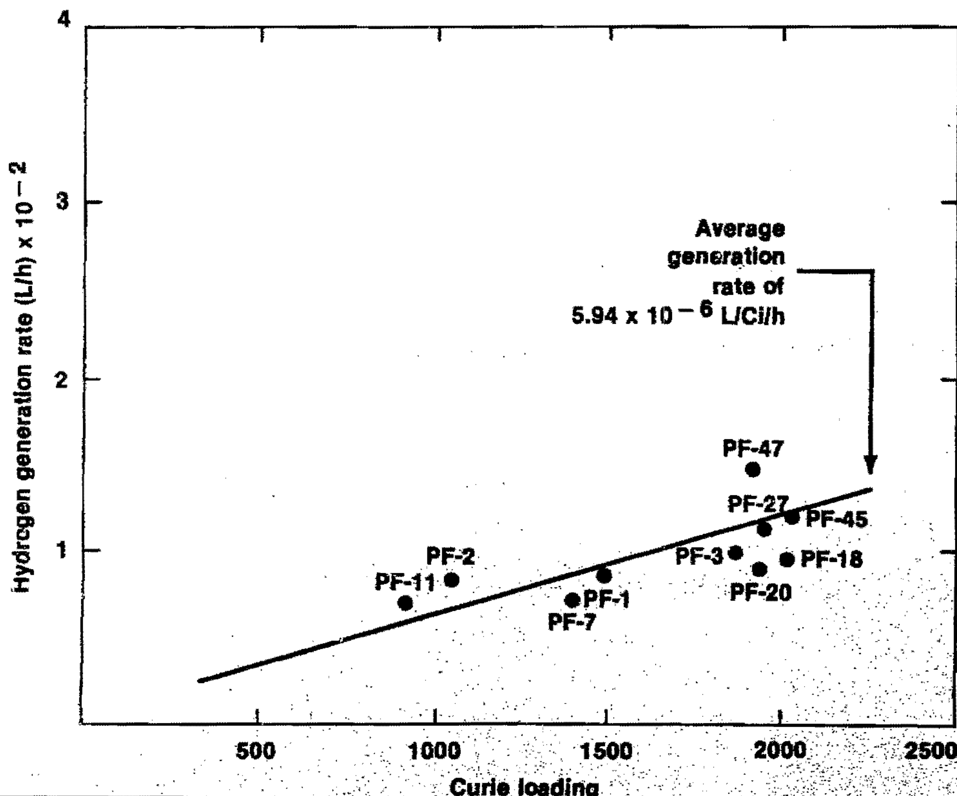


Figure 8 **This graph presents hydrogen generation rates versus curies for EPICOR II prefilters.**

In a closer look at resin degradation, analysts characterized the two worst-case EPICOR II prefilters and found that the pH of the ion exchange media was acidic and became significantly more so from top to bottom of the bed. But resin degradation, such as surface cracking, spalling, and fragmentation due to radiation exposure appeared to be minimal. This minimal degradation was observed at the bottom of the bed, far from the highest activity loading. Thus, it likely was the result of high moisture content or chemical attack. Analyses also demonstrate no significant leaching of nuclides from the resin and essentially no threat to liner integrity. And results of metallurgical studies suggest that the liners can be disposed of safely for more than 300 years in high-integrity containers without any threat to the environment.

Studying the two Makeup and Purification System demineralizers, analysts focused on resin degradation due to irradiation and high temperatures. After five years of radioactive decay, demineralizer A contains approximately 5000 Ci of cesium, and demineralizer B holds about 14,000 Ci of cesium. To date, these resins have absorbed a dose of more than 10^9 rads. In addition, fission product decay heat likely produced temperatures of up to 811 K. As expected, this combination of high temperatures and radiation dose caused the demineralizer resins to degrade; the bed size reduced by about 70%.

Samples of both demineralizer beds were sent to Oak Ridge National Laboratory for analysis. Demineralizer A was found to be dry, with agglomerated, black resin fragments, indicating temperatures in excess of 672 K and extensive radiation damage. Analysts have not concluded whether this vessel's resins are sluicible, but they have confirmed that demineralizer B resins likely can be sluiced because they remained under water.

As expected in resins irradiated to the high levels found in these two vessels, radiolysis produced hydrogen and oxygen gases. Analysts say the hydrogen generation and oxygen depletion mechanisms observed in these resins were the same as in the EPICOR II system ion exchange media. Before sluicing, the cesium will be removed by elution from the resins and processed by the SDS zeolites, whose abilities to handle such high curie loadings has been proven.

SDS was the third system at TMI in which an ion exchange medium was used. The inorganic zeolites in these liners effectively processed about 3785 m^3 of contaminated water from the reactor coolant bleed tanks, the RCS, and the building basement.

Below a dose of 10^9 rads, inorganic zeolites do not suffer radiation degradation; the only effect of the extremely high curie loading is radiolytic generation of hydrogen and oxygen gases. Focusing their studies on this concern, analysts have found that gas generation rates are approximately proportional to curie loading. Also, the oxygen depletion mechanism found in organic resins does not occur with inorganic zeolites to the same degree. The measured fraction of hydrogen was greater than stoichiometric, but oxygen was generated in sufficient quantities to form a combustible mixture in a sealed container.

Zeolite liners in storage at TMI exhibited gas generation rates of nearly 1 L/h. Rockwell Hanford Operations developed a catalyst recombiner and vacuum outgassing system to solve the gas generation problem. Vacuum outgassing removed residual water, and palladium catalyst pellets recombined the radiolytic gases generated from both the residual and chemically bound water so the liners could be safely shipped from the Island.□

Core Topography System Data and Photos Give First Accurate Picture of Core Void

As reported in the December 15, 1983, issue of the *Update*, engineers lowered a sonic sensing head into the TMI-2 core and collected nearly 500,000 data points to acoustically map the size and shape of the cavity inside the damaged reactor.

Since then, these individual data points have been reconstructed into the three-dimensional clear plastic model pictured in *Figure 9*. This model is the first accurate map of the upper portion of the core, which experienced considerable damage during the accident.

The model shows in some places the void extends all the way to the edge of the core. The suspended materials in the model are axial power shaping rods, which were driven in after the accident, and stubs of fuel assemblies. After studying the results of the core topography system, engineers have determined the cavity volume to be approximately 9.5 m^3. At the deepest point, the cavity drops about 2 m, as measured from the underside of the plenum.

***Figure 9* Mike Martin, senior project engineer, explains the making of this clear plastic model of the core void, developed from nearly 500,000 data points.**

These findings are also supported by videotapes of the region between the plenum and rubble bed. *Figure 10* shows the stubs of fuel assemblies hanging unsupported from the underside of the reactor plenum. Data from the sonar mapping device indicate these segments typically are 5 to 25 cm long. A close shot, seen in *Figure 11*, reveals an exposed fuel rod plenum spring. In *Figure 12*, the core former wall is clearly visible and appears in most places to be undamaged. Just in one area on the east side of the reactor does the core former wall appear to bow outward by about 6 cm.

Figure 10 **Stubs of fuel assemblies hang from the underside of the plenum assembly.**

Figure 11 **This fuel rod broke off close to the end fitting, exposing the fuel rod plenum spring.**

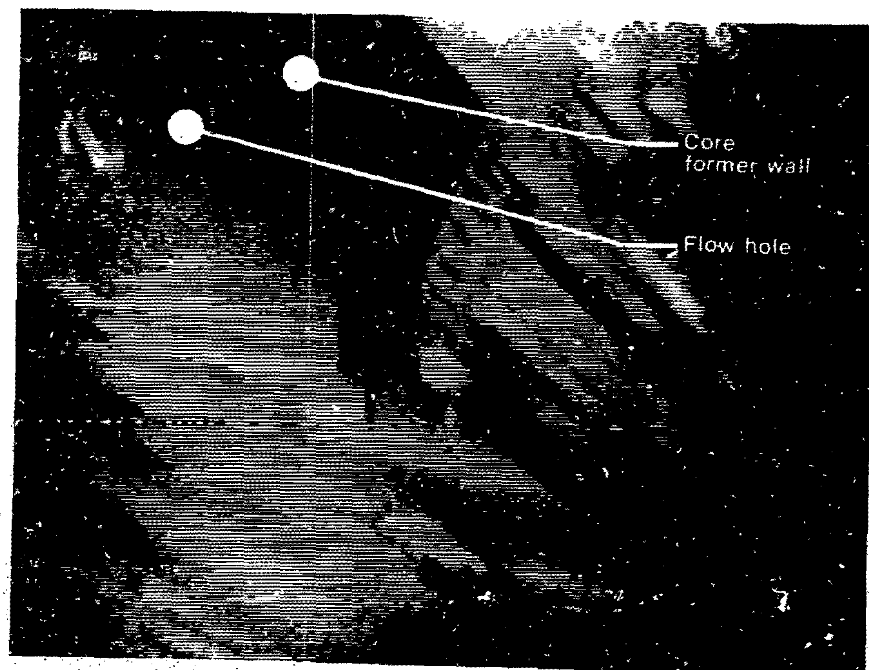
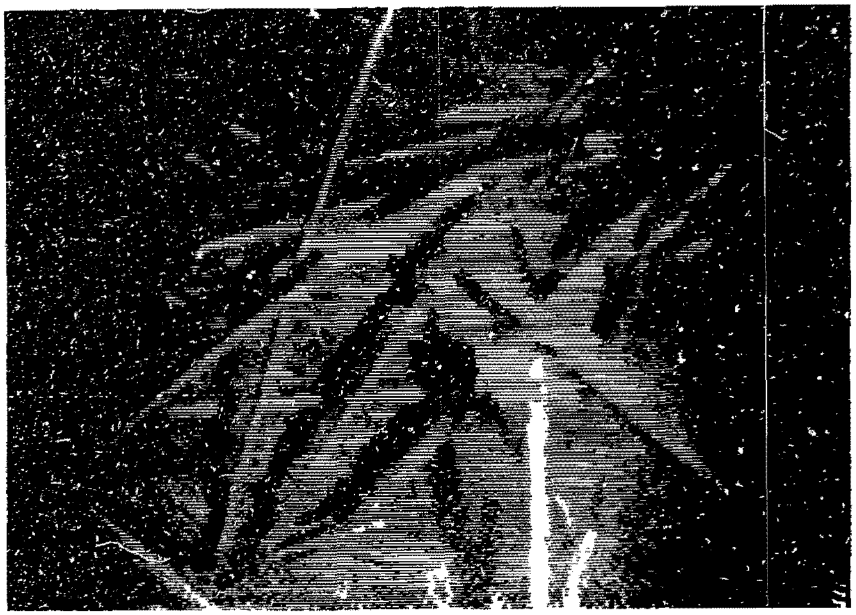


Figure 12 **Although the void extends, in most places, to the edge of the core, the core former wall appears undamaged.**



Only the top 1.5 m of the nearly 4-m-tall core are visible to analysts at this time; the area below is still unknown. But looking down, the video camera has provided some clear photographs of the rubble bed. In *Figure 13*, fractured fuel rods lay like pickup sticks on the surface of the gravel-like rubble bed. Fuel rod plenum springs are also visible there *(see Figure 14)*. And contrary to earlier conclusions based on limited visual observations, the surface of the rubble bed is uneven. In one region of the rubble bed, engineers saw a "valley," which they speculate was created during the accident by upward water flowing through the rubble bed.

Figure 13 **Fractured fuel rods lie scattered across the surface of the rubble bed.**

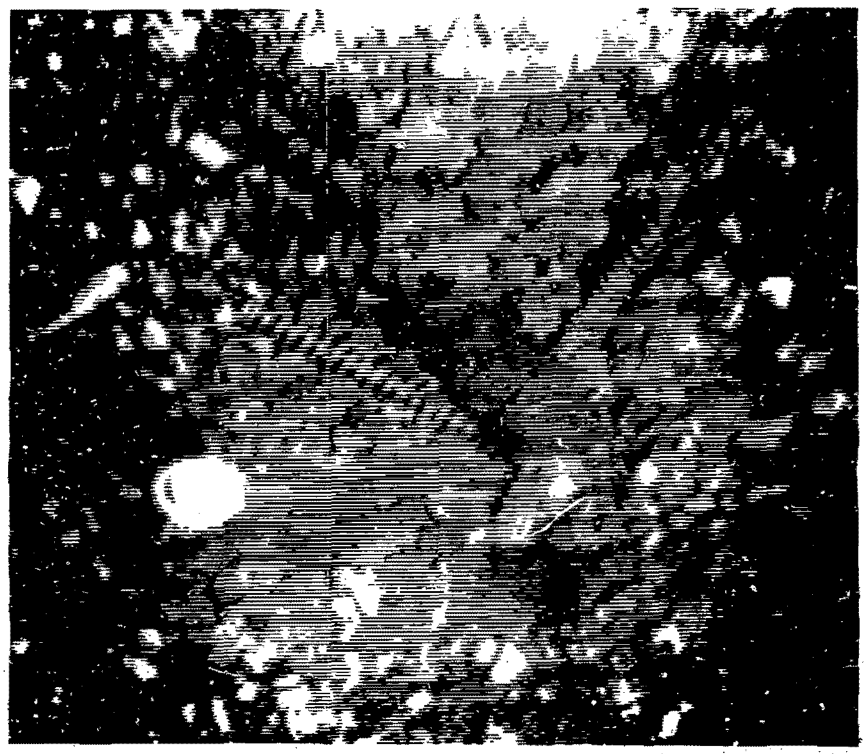


Figure 14 **In a closer shot, fuel rod plenum springs are also visible on the rubble bed surface.**

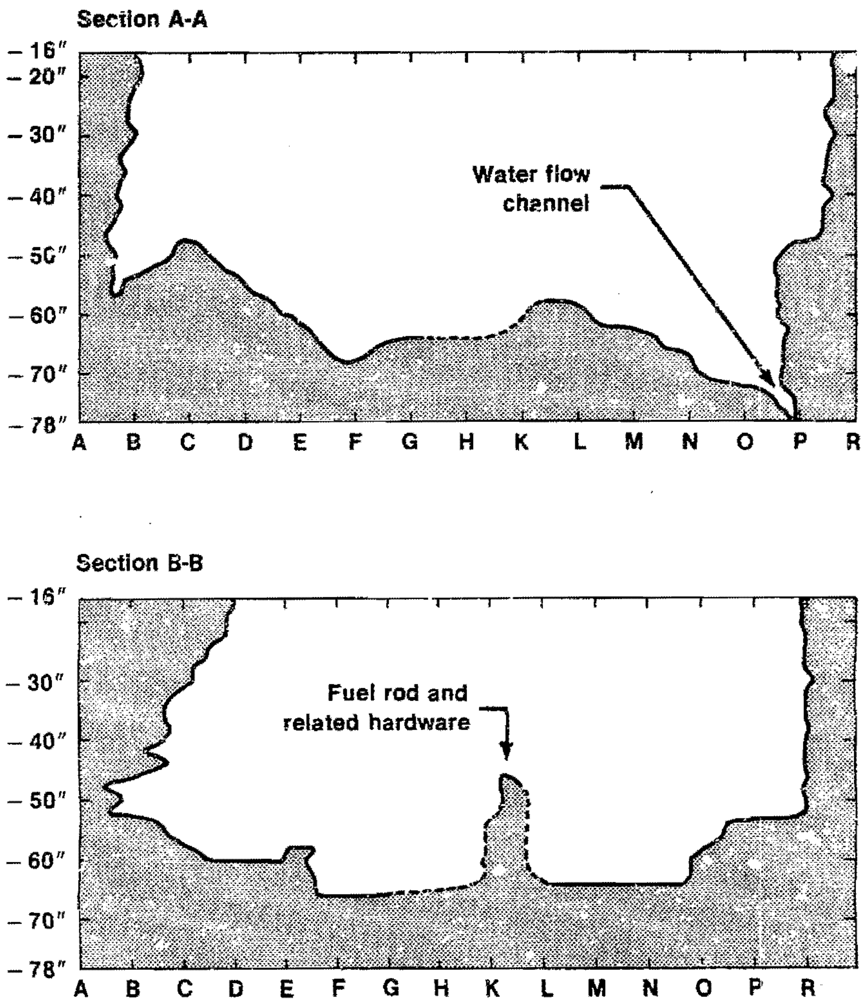


Figure 15 **Sections A-A and B-B are topographical plot cross sections of the TMI-2 core void.**

Using the data collected from the sonic sensing head, engineers also were able to plot cross sections of the void. Two such cross sections are presented in *Figure 15*. The solid lines in these drawings were produced from actual data, while the dashed lines are approximate locations obtained by extrapolation or interpolation. (A few areas of the rubble bed were not within the range of the acoustic transducers.)

Section A-A of *Figure 15* shows the deepest point of the void, at Position P, where water possibly flowed, as explained earlier. The bottom of this narrow channel is about 2 m into the core. In Section B-B, the data points plotted in the center of the cross section jut up into the void. As videotapes of this region confirm, this protrusion is a partial fuel rod and its associated hardware, which fell from the plenum into the rubble bed.

As a result of this information, analysts have concluded that 135 of the total 177 fuel assemblies in Unit 2 are broken, thus creating the void in the top 1.5 m of the core. Of the remaining 42 assemblies, 19 are more than 50% intact (including the two possible fully intact assemblies), and 23 are less than 50% intact. Again, these numbers strictly pertain to the known 1.5 m of the core. This and other sonar topographic data are still being evaluated and will be useful in planning for plenum and fuel removal.

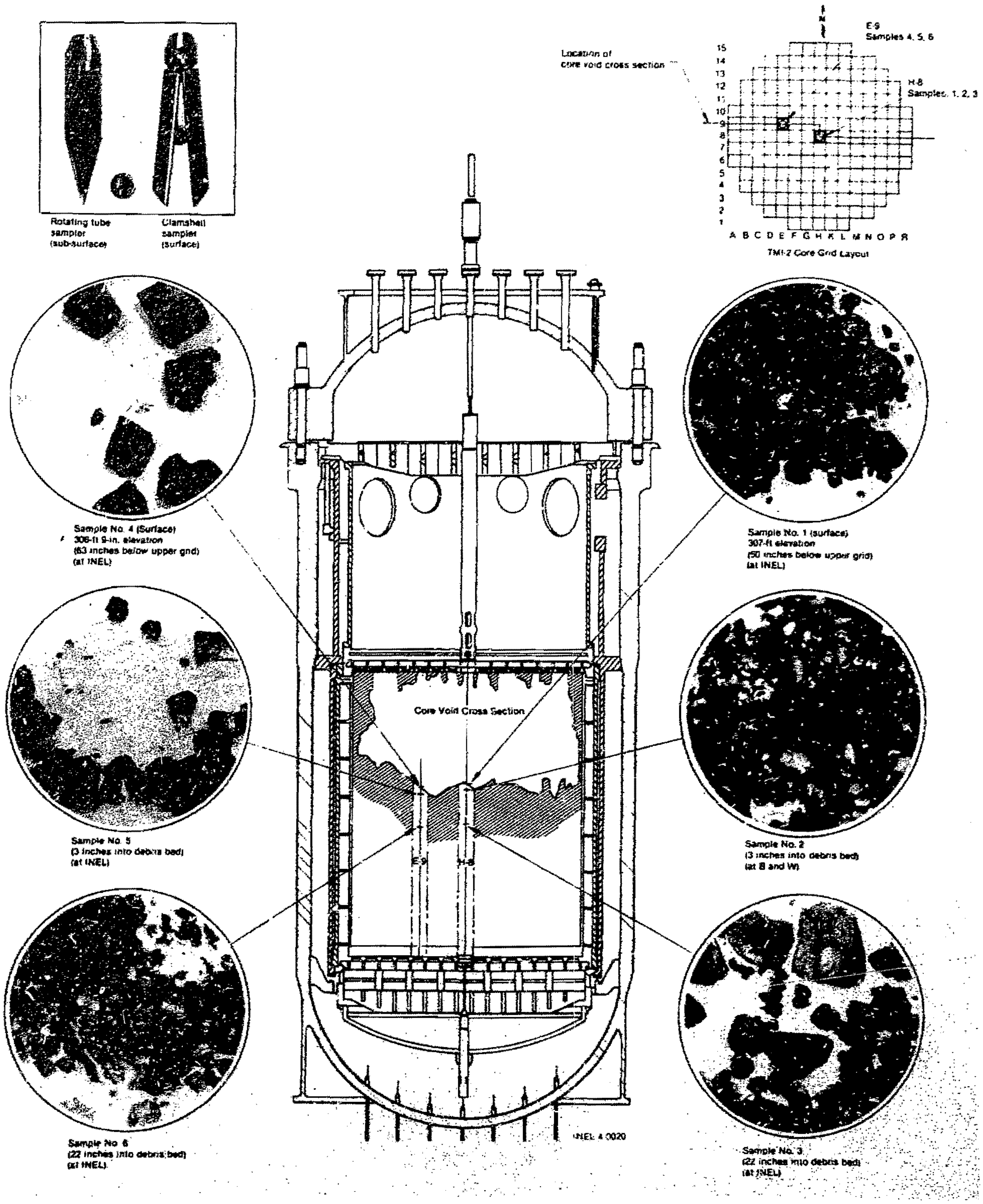
Engineers plan this year to answer some questions about the unknown region below the currently defined void by analyzing more samples taken from deep in the rubble bed.□

First Samples of Core Debris Analyzed

These six samples shown at right were the first obtained from the damaged TMI-2 reactor core. After their removal last fall, five of the samples were sent to the INEL and the sixth to Babcock & Wilcox laboratories in Lynchburg, Virginia, where they were weighed and photographed.

At the INEL, analysts removed eight particles from the five samples for gamma spectroscopy and fissile determination. The core debris consisted of fuel pellet fragments, shards of cladding or guide tubes, and particles which appear to be glazed with previously liquid material. Five of the eight particles were primarily fuel. While ^{144}Ce, ^{106}Ru, and ^{154}Eu appeared to be associated mostly with the fuel, ^{137}Cs and ^{125}Sb were released from the fuel particles. Based on a limited analysis, the released cesium and antimony appear to be present on other core materials. Radiation levels for the five samples, using a teletector from 2.5 cm away, ranged from 3 to 36 R/h gamma. Particle sizes ranged from about 0.6 cm to a fine debris. One surface sample consisted of 13 large chunks of material.

Future work will address the chemistry of the debris in all six samples. The information from the debris samples will aid in assessing the tools and procedures required to defuel the TMI-2 reactor. In addition, the data will be used to define the behavior of a commercial reactor core under the accident conditions found in TMI-2. □


Rotating tube sampler (sub-surface)
Clamshell sampler (surface)
Location of core void cross section
E-9 Samples 4, 5, 6
H-8 Samples 1, 2, 3
15 14 13 12 11 10 9 8 7 6 5 4 3 2 1
A B C D E F G H K L M N O P R
TMI-2 Core Grid Layout
Core Void Cross Section
E-9
H-8
Sample No. 4 (Surface) 306-ft 9-in. elevation (63 inches below upper grid) (at INEL)
Sample No. 1 (surface) 307-ft elevation (50 inches below upper grid) (at INEL)
Sample No. 5 (3 inches into debris bed) (at INEL)
Sample No. 2 (3 inches into debris bed) (at B and W)
Sample No. 6 (22 inches into debris bed) (at INEL)
Sample No. 3 (22 inches into debris bed) (at INEL)
INEL 4 0020

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 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.

TEST PROVES CRANE CAN LIFT VESSEL HEAD

The Unit 2 polar crane successfully passed a load-lifting test on February 29 when it lifted and maneuvered a 214-ton load in the Reactor Building. The load test indicates the crane can safely lift and remove the 170-ton reactor vessel head. The polar crane was damaged as a result of the 1979 accident and underwent nearly a year of inspections and repairs. Engineers first tested the crane without a load to assess its structural integrity and its mechanical and electrical functions. The load test was delayed one year because of concerns about the safety and feasibility of the test. The Nuclear Regulatory Commission investigated the safety of the test and concluded in November 1983 that it could be conducted without risk.

The crane stacked four 40-ton missile shields from over the reactor vessel and one 32-ton missile shield from over the reactor system pressurizer into a steel lifting frame. This stack, together with the lifting frame and assorted rigging assemblies, constituted the test load. The load was lifted, lowered, stopped and held by the hoist brakes, and then lowered to the floor. Then the load was transported 3 m out and back by the crane trolley and approximately 3 m from side to side by the crane bridge. The bridge was rotated 180 degrees and the load was lifted again. As a result of the load test, the polar crane will be rated capable of safely lifting and removing the reactor vessel head, a major step on the road to completing the cleanup.

NEW TLDs HANDLE BETA FIELDS

DOE TMI programs working with Pacific Northwest Laboratory have developed a multielement thermoluminescent dosimeter (TLD) capable of correcting for energy distribution in beta radiation fields. These devices have been used in the TMI recovery program to characterize beta and gamma fields in the Reactor Building while [illegible] exposure to ionizing radiation. This new TLD system may aid the industry in [illegible] protection, and [illegible]

MONITORS MAY NOT BE SUITABLE FOR REACTOR BUILDING USE

Investigations of failed radiation monitors at TMI-2 have revealed that the causes of failure are not unique to an accident environment. Failures experienced by monitors at TMI could occur in similar nonqualified monitors in operating plants, indicating that these monitors may not be suitable for reactor building use.

Incorrect installation instructions supplied by vendors and installation problems are among the factors associated with radiation monitor failure at TMI. Investigations of one monitor revealed that the cable connector attaching the cable to the monitor was not screwed on properly. This improper seal allowed water to enter the connector and accumulate on the connector pins, causing a short circuit. The situation was aggravated by the fact that the entire monitor was installed upside down. Ambiguous instructions in the manufacturer's installation manual led technicians to install the monitor with the cable connection facing the ceiling. The manufacturer has since corrected the manual.

The manufacturer advises against subjecting monitors to pressures greater than 30 psig. During standard reactor building integrated leak rate tests, sealed equipment can experience differential pressures as high as 69 psig. Consequently, TMI-2 monitors had to be removed from the building during each integrated leak rate test. This only increased wear-and-tear on the instruments and the potential for installation mistakes.

Investigations have also shown that radiation monitors with connecting cables longer than 152 m can fail at radiation levels as low as 500 mR/h. The failure mode, called a foldover effect, causes control room ratemeters to show decreasing radiation levels, when in fact the levels in the plant may be increasing. The foldover is caused by a normally inconsequential impedance mismatch along the cable connecting the monitor to the ratemeter. As radiation levels increase, the instrument circuit operates at a higher frequency and the mismatch is amplified, causing a faulty drop in the ratemeter readout.

The foldover effect is amplified by another factor associated with failure of radiation monitors. Monitor components called metal oxide silicon transistors are known to degrade with accumulated radiation exposure. The TMI studies reveal that their degradation also compounds the foldover effect.

All TMI-2 studies indicate that these monitors may not be suitable for use in a reactor building. While the monitors perform satisfactorily in support building environments, they are subject to failure in the harsh environments which can be encountered in reactor buildings. □

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ADDRESS CORRECTION REQUESTED

UPDATE

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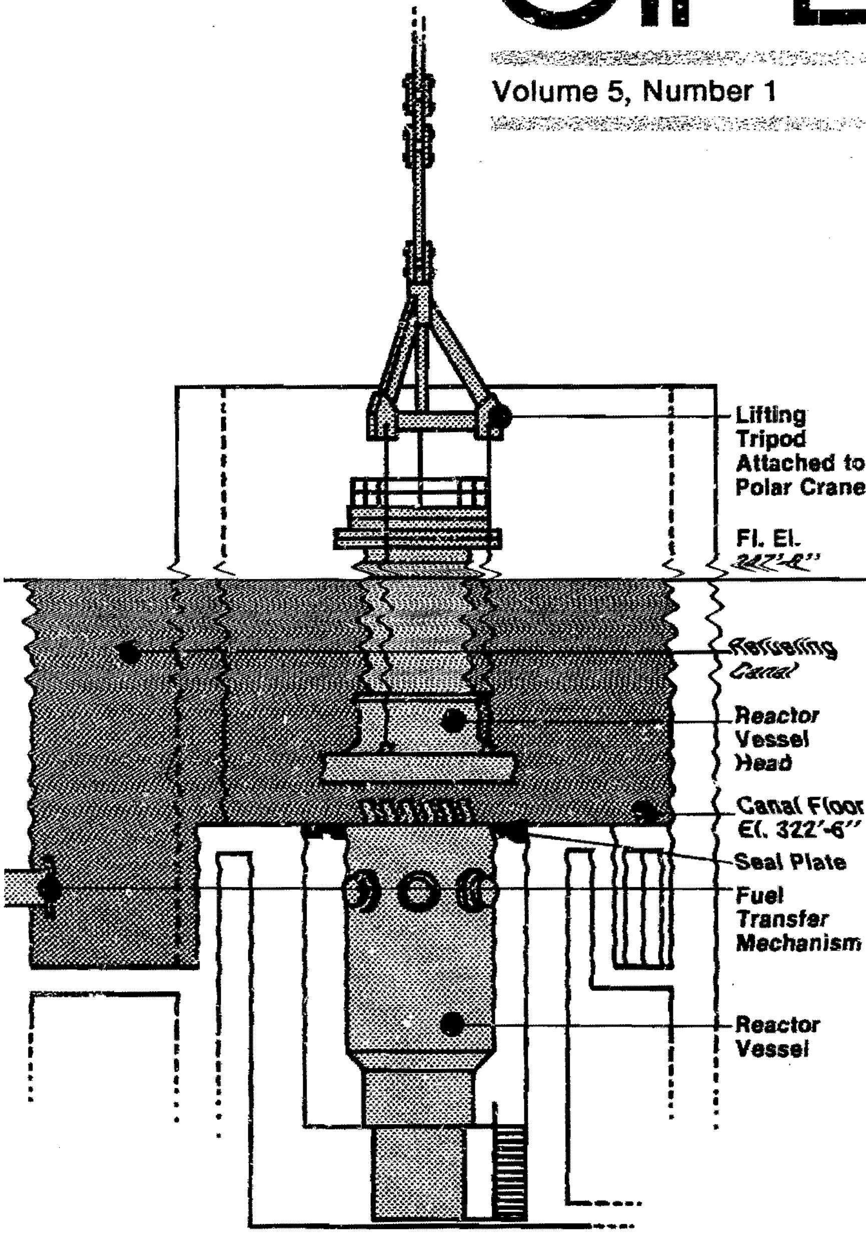
TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 5, Number 1 November 1984



TMI-2 Head Safely Lifted

Two teams of more than 40 workers labored around-the-clock to successfully lift the head from the damaged Three [illegible]

The completion of this stage in the cleanup effort provides ready access to the internal components and fuel of the Unit 2 reactor.

The head was lifted on the evening of *July 24 and was seated on its storage stand shortly after midnight the next day. The head, including the service structure, lead blanketed shielding, and lift rigging,* weighs approximately 159 tons. The head consists of two major components: the domed cap of the reactor vessel and the head service structure (see Figure 1).

***Figure 1*. Section view of reactor vessel head removal.**

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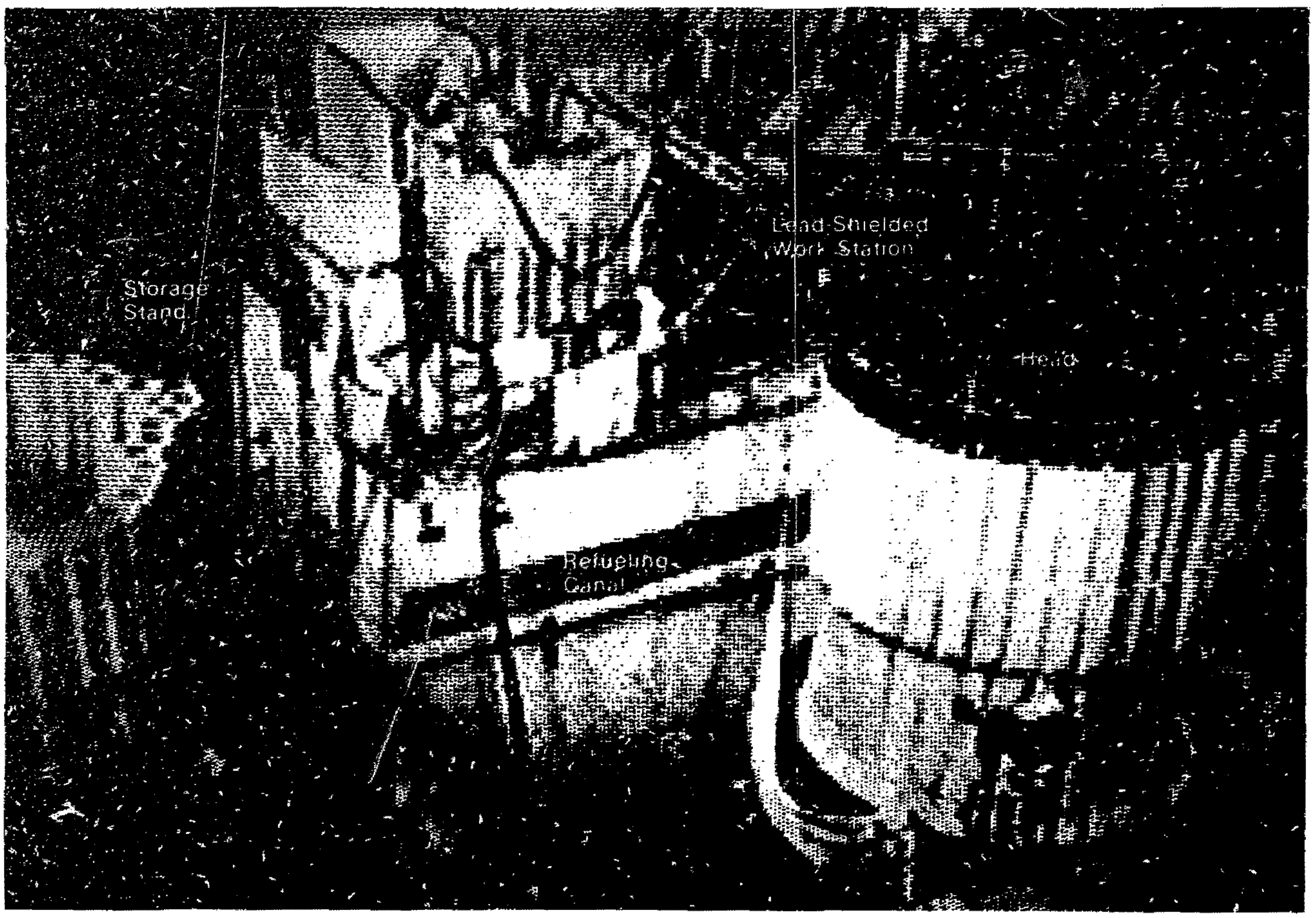


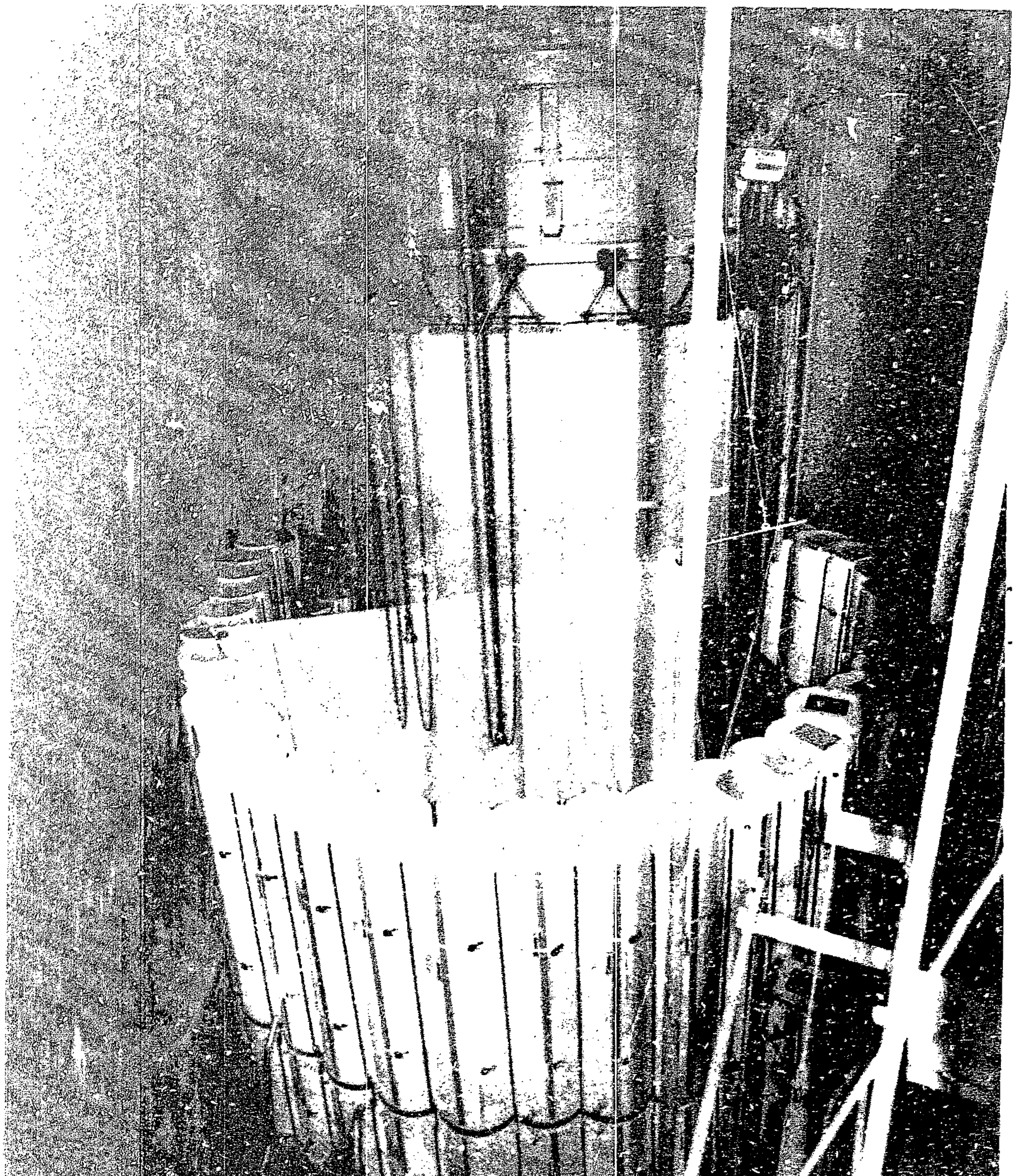
Figure 2. **Photo taken from television monitor of the head traveling toward the head storage stand.**

Attached to the polar crane with three cables and lifting tripod, and covered with 13 tons of lead blankets, the head was first lifted a fraction of an inch so that workers could ensure that the head was level. With the head then raised 3 feet above the reactor vessel, workers wrapped a plastic "diaper" underneath to prevent contaminants there from being spread during travel. Moving at a rate of 1 to 2 feet per minute, the head was raised 38 feet and then moved south and east towards its storage stand (see Figure 2).

During the entire process, engineers and technicians were located in a command center in the TMI-2 Turbine Building and monitored the head lift activities by closed-circuit television and mobile radio. The workers inside the Reactor Building worked most of the time inside a lead-shielded work station to minimize exposure. The Reactor Building was isolated during the lift, and radiation monitors placed inside the building showed no radiation releases during the entire operation.

The final lowering of the head was delayed while the guide holes in the head were aligned with the guide pins on the storage stand. Surrounding the head storage stand are 12-foot-high columns filled with sand that act as radiation shields. The columns were originally filled with water but were drained and refilled with sand because of leakage. Figure 3 shows the head seated on the storage stand.

Figure 3. **Reactor vessel head seated on storage stand.**



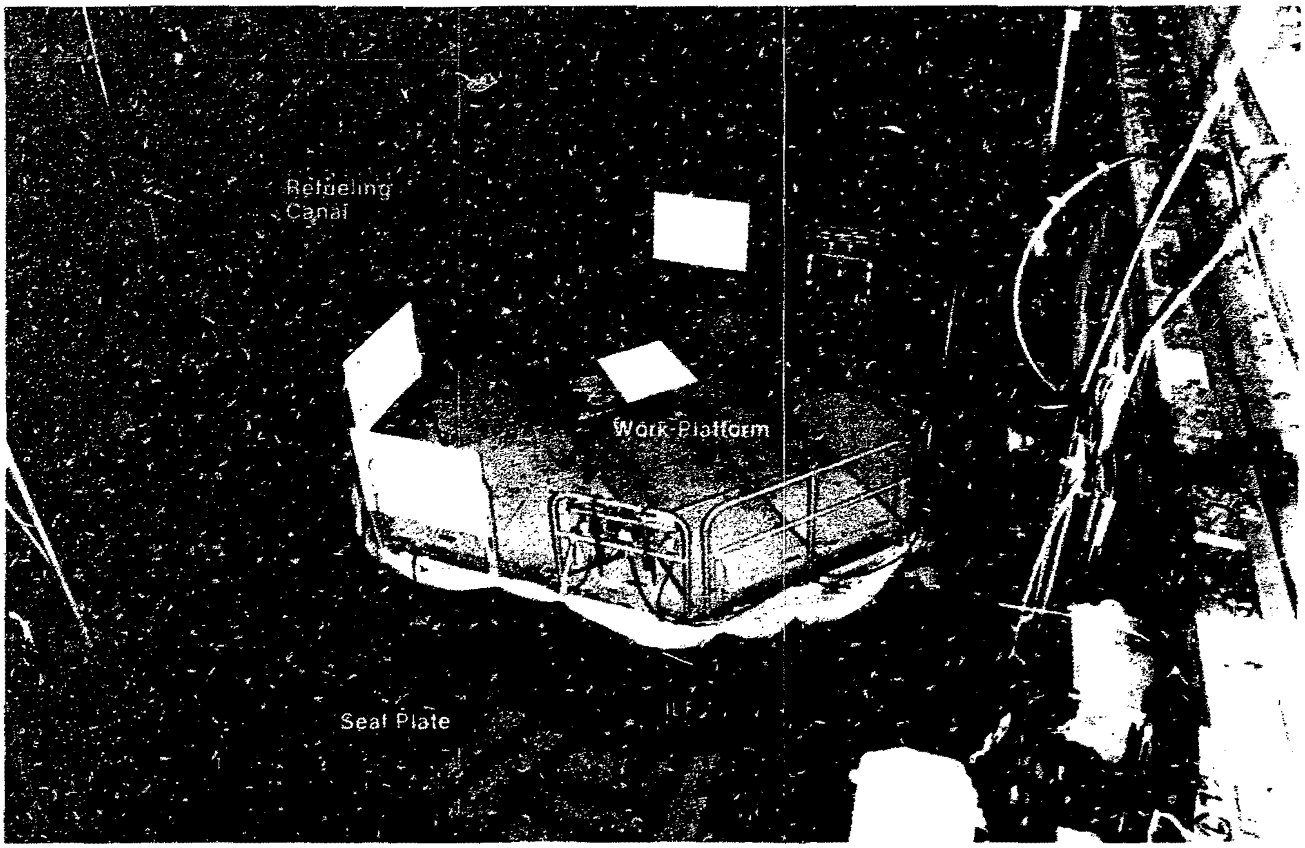


Figure 4. **IIF and work platform in place on the reactor vessel.**

After the head was successfully landed on the storage stand, workers released the lifting rig and attached it to the Internals Indexing Fixture (IIF) located on the operating floor of the Reactor Building. This 6-foot-high steel cylinder, used during normal refueling operations, was placed on top of the open vessel where the cylinder will remain throughout the entire defueling process. Once the IIF was attached, water was added to the Reactor Coolant System, filling the IIF to a depth of about 5 feet. This configuration provides shielding from radioactivity and will allow the plenum and fuel to be extracted through the IIF without flooding the refueling canal. Once the IIF was filled with water, workers installed a 1-3/4-inch-thick lead-lined steel work platform on top of the IIF, completing the head removal/IIF installation procedure. Figure 4 shows the IIF and platform in place on the reactor vessel.

During placement of the work platform, a minor malfunction of one of the switches on the polar crane caused it to stop when the work platform was within an inch of the IIF. Workers manually lowered the work platform the rest of the way by turning the turnbuckles on the crane's lift rigging.

Throughout the head removal process, the radiation levels were less than originally anticipated. Readings taken at the refueling canal were 3 R/hour, which was 10 to 15 times lower than projected. In the lead-curtain cubicle, workers experienced radiation levels of 30 mR/hour, lower than the 50 to 150 mR/hour anticipated. While removing their protective clothing, six workers experienced minor skin contamination, which was subsequently washed off with soap and water.

With the head lifted and the IIF in place, the first major phase toward successful cleanup has been accomplished. Currently, the schedule calls for initial plenum jacking in December 1984 and defueling to begin the following July.☐

Months of Preparation Lead to Safe Head Lift

The successful head lift in July 1984 climaxed months of preparatory work in and out of the TMI-2 Reactor Building. Safety played a key role throughout the operation, from underhead characterization to placement of the head on the storage stand.

One of the early objectives of the TI&EP's Reactor Evaluation Program was to determine the best approach to safely remove the reactor vessel head. The approach chosen was to remove the head dry, without flooding the refueling canal. This is essentially the same technique used in normal refueling operations and was considerably less time consuming than removing the head wet, which would have required subsequent decontamination of the refueling canal and processing of the canal water.

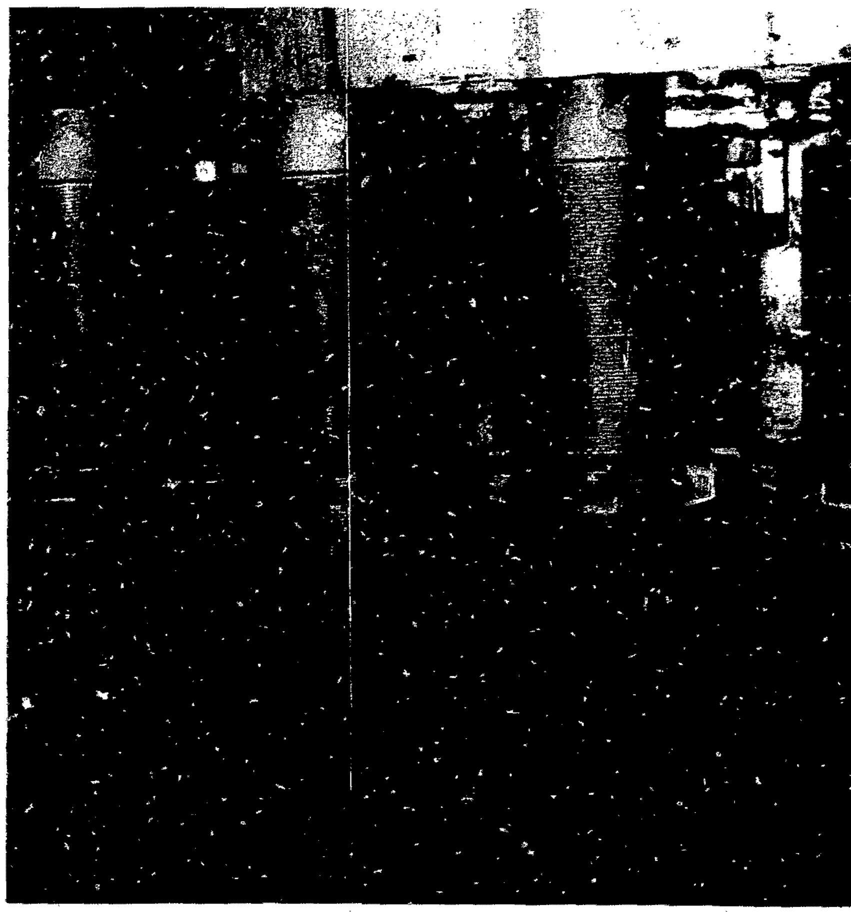
The Underhead Characterization Program confirmed the decision to remove the head dry. This program included the closed-circuit television examinations of surfaces under the head and on top of the plenum, radiation measurements inside the vessel, and analyses of debris samples from the plenum's upper surface.

While cameras saw much debris hanging from the underside of the plenum, its top surface—between the plenum and the head—showed no apparent damage or distortion and little debris. After obtaining gamma and beta radiation readings of this debris, technicians removed some samples which were tested for pyrophoric reaction. The test, conducted at Battelle Pacific Northwest Laboratory's TMI facility, demonstrated the debris posed no pyrophoric hazards.

The next major step in head removal preparations followed in February 1984, when the polar crane was successfully load tested and qualified to lift the reactor vessel head and service structure. The crane lifted and maneuvered a 214-ton load of missile shields, the lifting frame, and assorted rigging assemblies.

Major preparations were conducted in the five months preceding the actual lift. The 60 studs that fastened the head to the reactor vessel were partially detensioned to identify the studs that might have been stuck as a result of corrosion. Studs are detensioned by first stretching the studs and then loosening the nuts on them (see Figure 5). As expected, workers encountered some difficulty turning the stud nuts but succeeded using penetrating oil and a striking bar and hammer. Two of the studs were removed at that time, leaving holes in the head flange that later were lined up with the two guide pins on the storage stand on which the head was seated. In a later entry, the workers fully detensioned and removed the other 58 stud and nut assemblies, each weighing 670 lb, and placed them in storage racks. Finally, the stud holes were filled with a preservative and sealed, preventing them from corroding.

Figure 5. **This closeup shows some of the 60 studs that fastened the TMI-2 reactor head to the reactor vessel. The nut on the lower portion of each stud maintains tension on that stud.**



After the head studs were partially detensioned, the reactor vessel was refilled and pressurized. Processing of the Reactor Coolant System water could then resume. By sending the water through the Submerged Demineralizer System, its radioactivity was reduced. Also, the water's boron concentration was increased from 3700 to about 5000 ppm, thus increasing the safety margin for prevention of criticality (nuclear chain reaction) during later defueling operations. After the processing was complete, the reactor vessel was depressurized and water partially drained to below the reactor vessel flange before head lift.

Clearing a path for the head to be transported through the south end of the refueling canal, the auxiliary fuel handling bridge was dismantled and moved to the north end of the canal. The bridge is a crane that straddles and trolleys over the refueling canal.

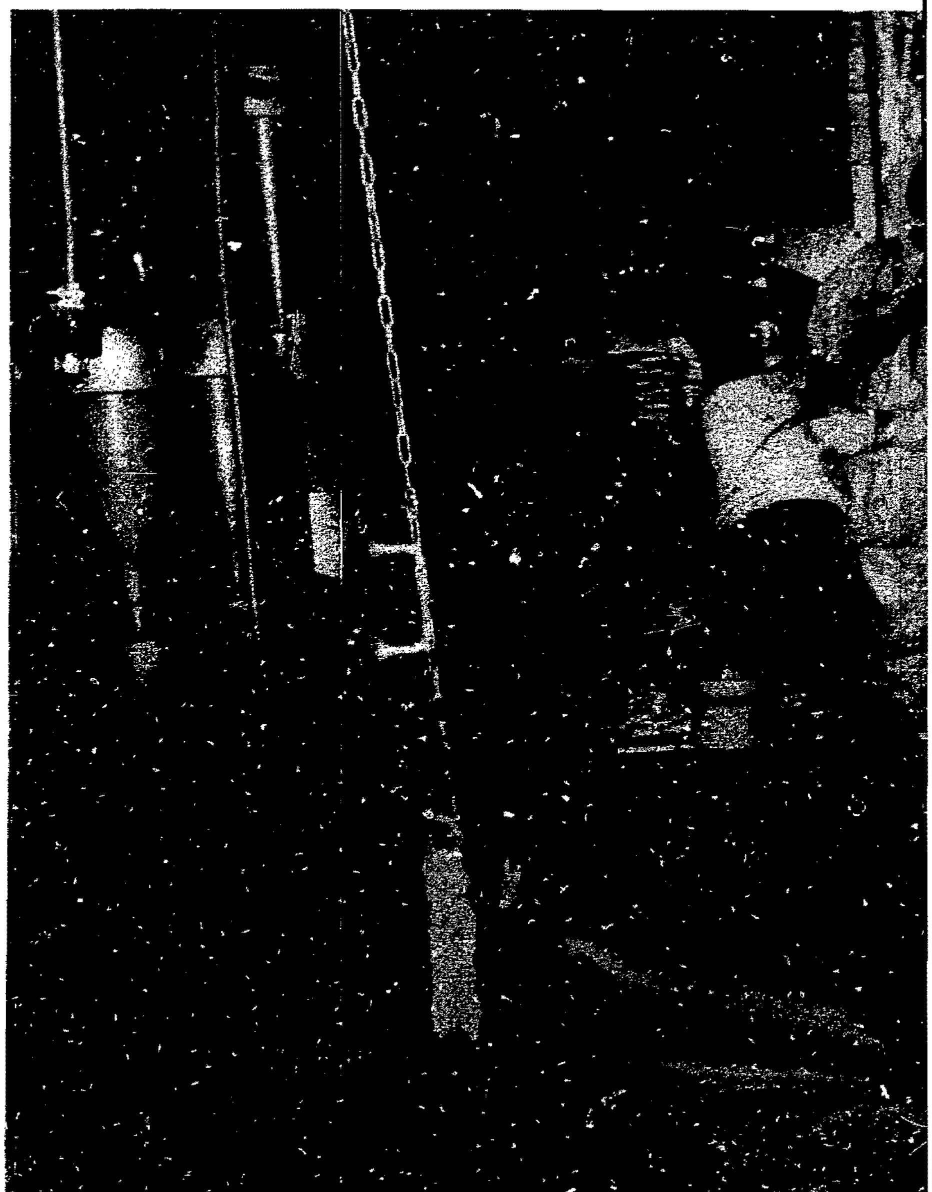


Figure 6. **A TMI-2 worker pours a sealant around the canal seal plate. The sealant and a metal seal plate were placed between the refueling canal and the reactor vessel so the canal could have been flooded, if necessary, for shielding.**

A few days before head lift, the remaining 66 lead screws were parked, or raised from inside the reactor vessel up into the reactor head service structure. Other important prelift jobs included installing cameras to monitor the head to maintain alignment as it was lifted; stripping the head of remaining insulation, wiring, piping, and equipment for adequate accessibility; preparing the IIF for placement on the reactor vessel after head removal; assembling in the Reactor Building the IIF work platform; and installing a system to process the Reactor Coolant System water within the reactor and IIF. The water is being pumped through this system to remove radioactivity from the reactor coolant system water, thereby keeping radiation levels low in work areas above the vessel.

Head lift planners were aware that head lift could have resulted in an air particulate radioactivity buildup or radiation intensity in the area around the top of the vessel, possibly requiring the refueling canal to be flooded. They therefore took precautions for such a contingency, fully inspecting the canal, sealing all penetrations in the canal walls and floors, and modifying the water systems so the canal could have been flooded with borated water—and subsequently drained.

A seal plate was installed, closing the gap between the reactor vessel and the refueling canal. On a partial mockup of the canal seal plate, workers practiced various techniques to apply the sealing compound that was to be used in the cavities and joints of the seal plate. Figure 6 shows a TMI-2 worker actually pouring the sealant around the reactor vessel.

Training was, in fact, a critical part of the head removal effort. By rehearsing in the Unit 2 Turbine Building on mockups of the reactor head, IIF, IIF work platform, and other components and apparatus, workers were prepared to enter the Reactor Building and carry out their functions safely and efficiently. Consequently, they were able to reduce their time in the building and minimize their exposure.

Training was one of a number of items established to make head lift a safe activity. Some other controls included the use of shielding, protective clothing and respirators, personal dosimeters, radiation monitors, and television cameras, the combination of which were designed to keep radiation exposures to a minimum.

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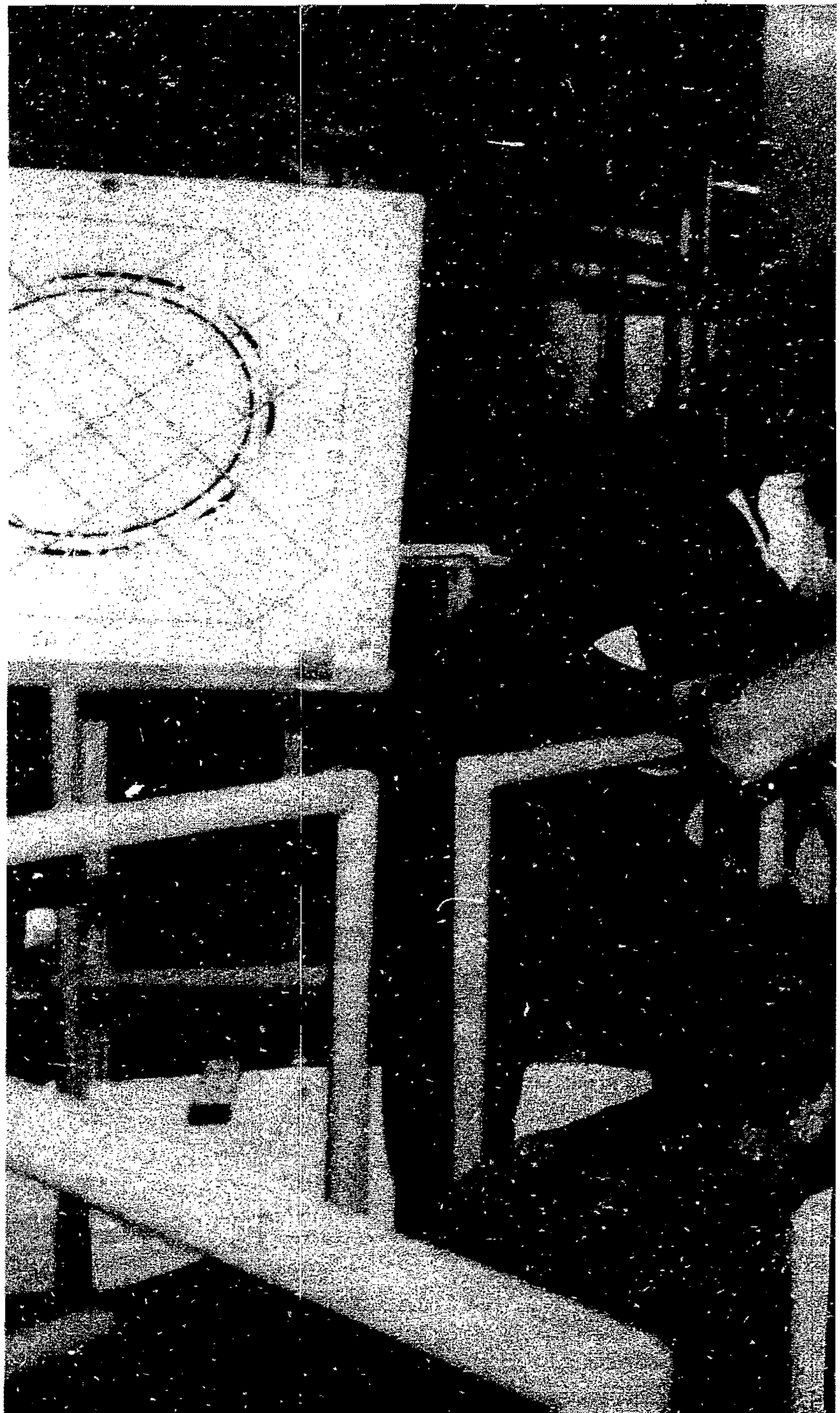
Next Step:

Plenum Jacking, Removal Planned



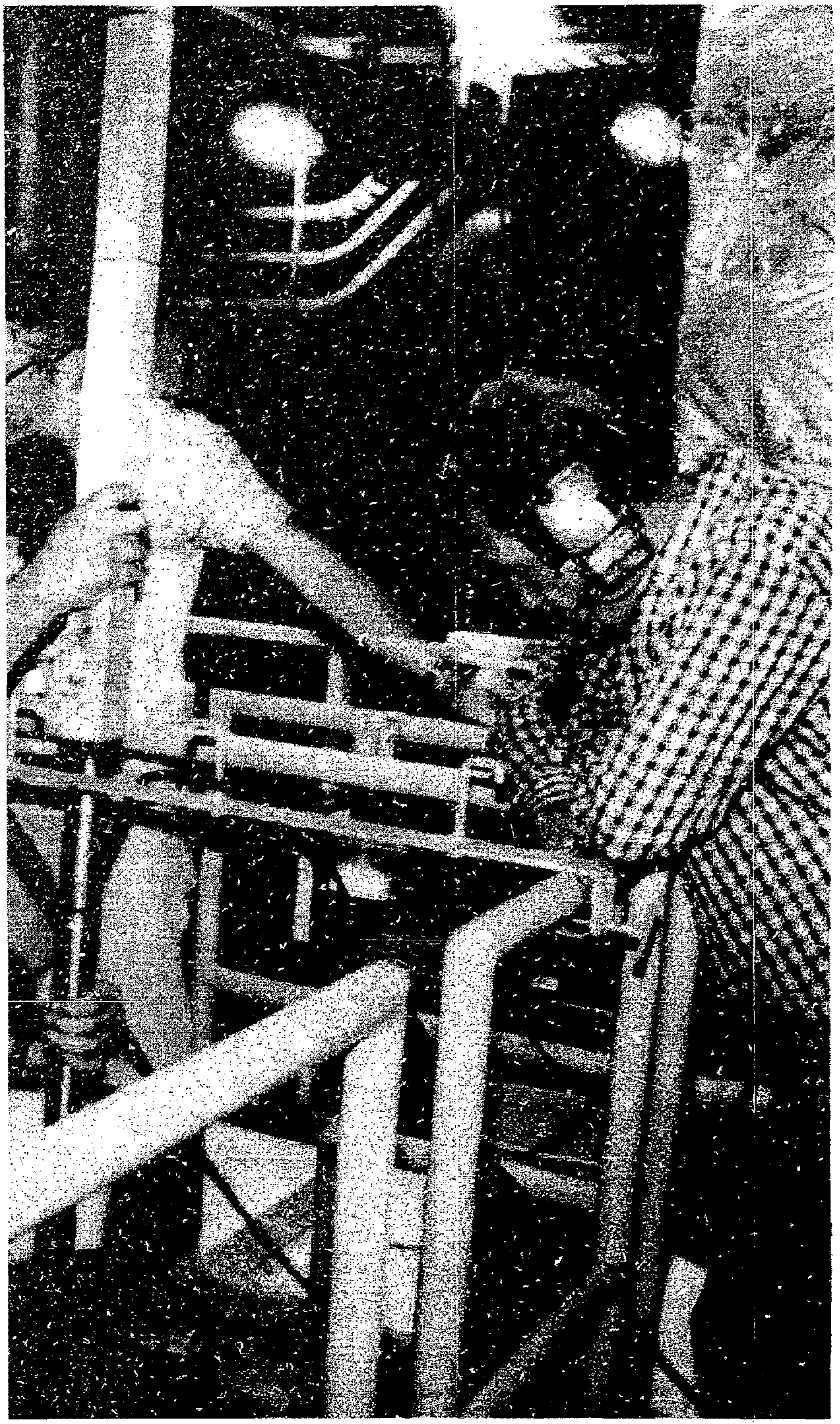
No sooner was the head removal project completed when the TMI-2 recovery program turned its primary focus to the next major stage in reactor disassembly: plenum removal.

The plenum assembly, a 55-ton cylindrical structure above the reactor core, houses the control rod guide tubes. It is scheduled for initial jacking in December and placement in the deep end of the refueling canal in May 1985. Over the past couple of months, the TI&EP and GPU Nuclear have been getting the plenum assembly ready for its initial jacking.

As a preparation, technicians are visually inspecting the plenum, using specially designed underwater cameras, recorders, lighting, and long-handled camera-positioning tools, to determine the amount of debris on the underside and peripheral surfaces of the plenum, as well as on top of the fuel assembly end fittings (see Figure 7). If a great amount of debris is found and considered to be a possible hindrance to the plenum lifting operation, the technicians may remove it by water lancing or vacuuming.

Also during the inspection, workers will attempt to separate unsupported end fittings using newly designed end fitting separation tools. If some end fittings remain attached, they may leave them since the plenum would still be able to sit evenly on a stand in the refueling canal.

End fitting separation is considered to be the first intentional movement of significant quantities of fuel in the damaged reactor core. This action can not cause



core criticality because the nearly 5000 ppm of boron in the Reactor Coolant System water in the vessel prevents criticality, regardless of fuel geometry.

The visual inspection of the plenum is not designed solely to check for debris and to test the knock-off tools, but also to see how much clearance remains in certain normally tight areas between the plenum and the core support shield that encircles the reactor vessel. Technicians want to establish whether the plenum has been damaged or distorted in these vital areas.

In December, workers operating four 60-ton hydraulic jacks will initially lift the plenum about 2-1/2 inches. The workers will then check for fuel separation, after which they will jack the plenum another 6-1/2 inches to be sure the plenum has a free path out of the reactor vessel.

The work will then be completed in early 1985, when a dam will be installed to hold water in the deep end of the refueling canal, a plenum storage stand will be put in place, the deep end will be flooded, and the plenum will be lifted, placed on its stand, and covered.

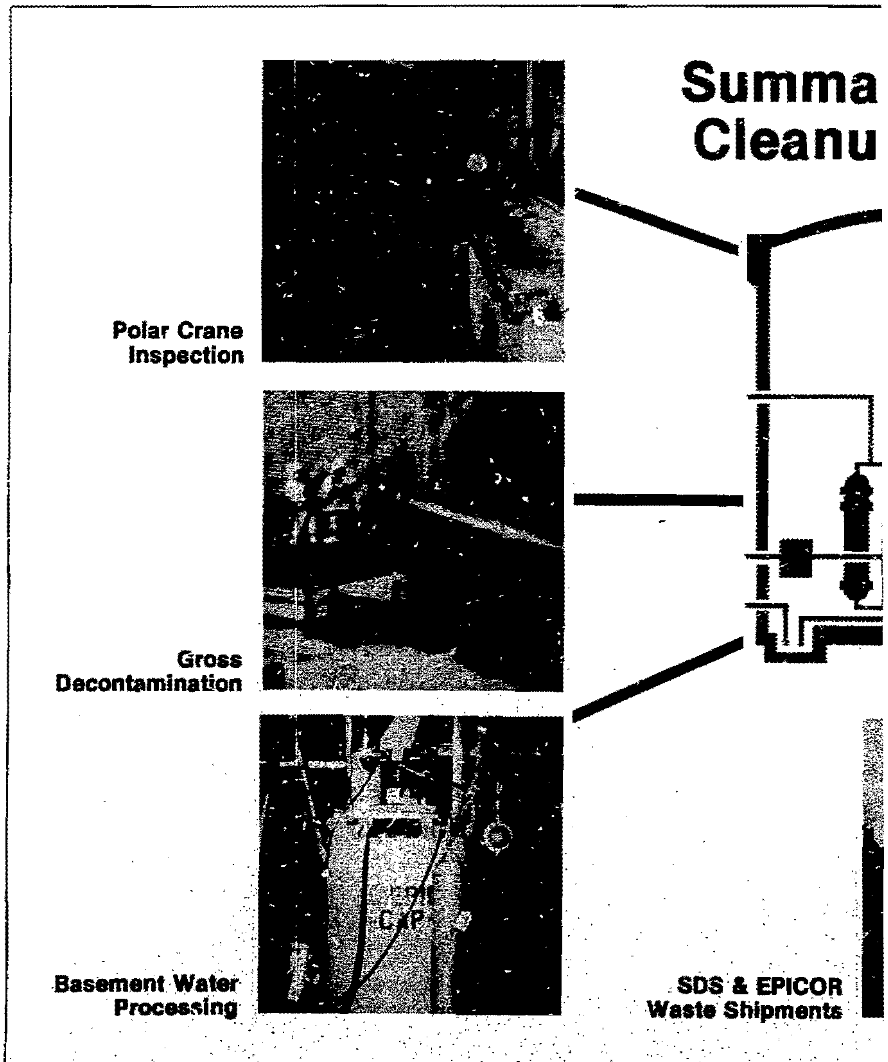
The outcome of this entire phase of the reactor disassembly project will be detailed in a future *Update* issue. □

***Figure 7.* Workers lower new plenum inspection equipment into a large model of the TMI-2 reactor as they receive training in the plant's Turbine Building. The grid on the map of the plenum (left) provides guidance.**

The TI&EP— What Has it Accomplished? What is in the Future?

The safe removal of the TMI-2 reactor vessel head marked the successful completion of Phase 1 of the defueling sequence. Many TI&EP sponsored activities, along with intensive head lift preparations, contributed to the achievement of this major milestone. These activities began in 1980 when the DOE Technical Integration Office (TIO) was established.

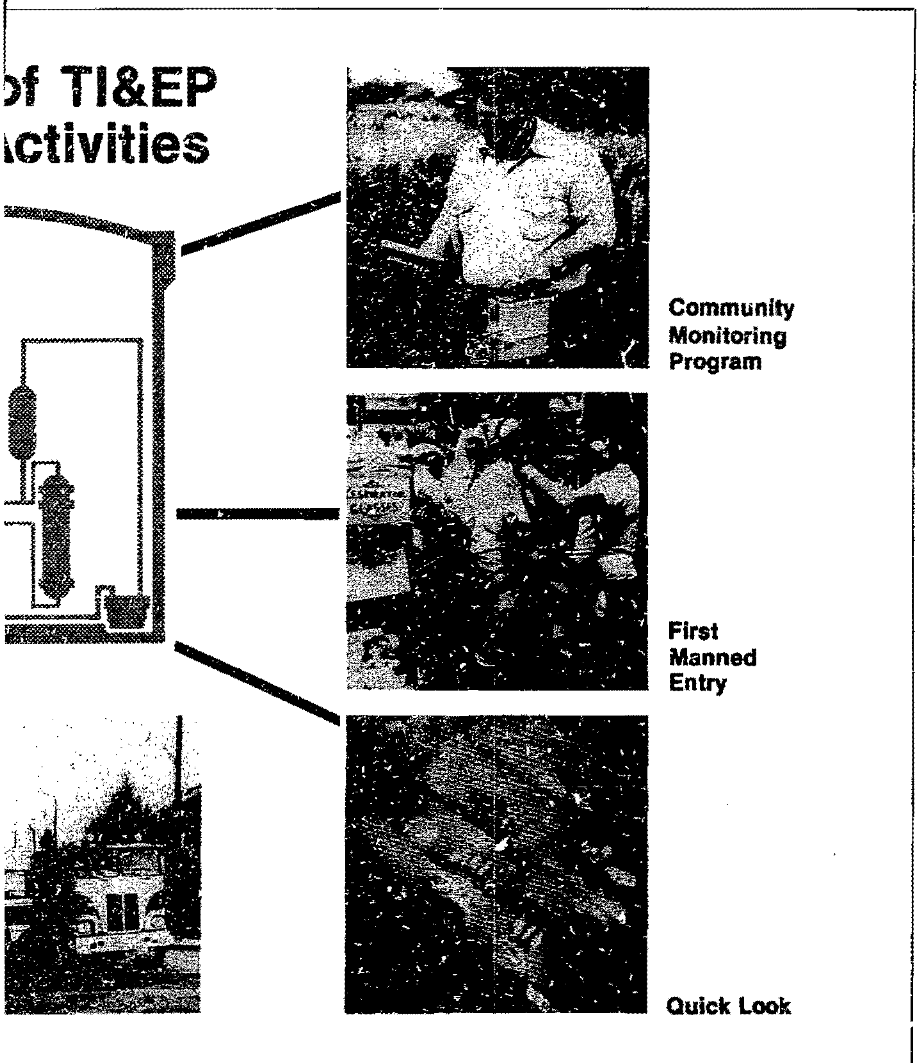
The first major step toward defueling the damaged reactor occurred in July 1980 when the first manned entry into the Reactor Building occurred. To support this activity, the TI&EP established a Citizens' Radiation Monitoring Program, which proved to be one influential factor in alleviating the fears of local residents regarding adequacy of monitoring during



the venting of ^{85}Kr from the Reactor Building—a prerequisite for manned entries. The program was designed to provide a credible source of information about radiation levels to the citizens in the communities adjacent to TMI during ^{85}Kr venting. The program represented a unique effort to build citizen confidence in public information and remains active in six communities today.

As manned entries into the Reactor Building increased, the TI&EP sponsored early inspections of the polar crane. These inspections provided recovery engineers with vital information on the extent of damage to the crane so that a safe, cost effective refurbishment of the necessary crane components could be conducted as expeditiously as possible. TI&EP engineers also provided technical electrical engineering evaluations to support the polar crane recovery—a critical path milestone for head removal that was completed in February 1984.

Probably the single TI&EP sponsored event that provided the greatest impact on the cleanup occurred in July and August of 1983 when the first inspections inside the reactor were conducted. Not only did this activity provide the first pictures of the actual conditions of the core, but it conclusively demonstrated that work in and around the reactor itself could be performed safely and efficiently. The activity, called "Quick Look," also proved that reactor internal components could be safely removed and handled and it paved the way for future underhead and in-vessel (core) characterization programs.



Other TI&EP activities also provided valuable contributions to the cleanup, but were not nearly as visible as Quick Look or head lift. Some of those activities included the gross decontamination experiment designed to determine the most effective means of reducing loose surface contamination, fission product deposition and mass balance, Reactor Building characterization, and shipping and disposal of accident generated wastes.

A major milestone in the Waste Immobilization Program was reached in the summer of 1983 when the last ion-exchange wastes used to decontaminate accident water were shipped from the Island for research and development projects and disposal. The two ion-exchange media systems, called EPICOR II and Submerged Demineralizer System (SDS), decontaminated more than a million gallons of accident generated water and captured approximately 95% of the radioactive elements released from the Reactor Coolant System as a result of the accident. (See articles published in previous editions of the *Update*.)

Another major cleanup milestone, elution of cesium from the plant's Makeup and Purification System demineralizer resin, is scheduled for completion in late 1984. Completion of this activity will essentially complete the Waste Immobilization Program's role in the TMI-2 cleanup.

Now that the head has been removed, the major thrust of the TI&EP is toward plenum removal. In addition, the defueling and core shipping phase is gaining momentum.

The plenum inspection equipment has already arrived on the Island, and training for the actual plenum inspections is well underway. The inspections, scheduled to begin in October, will be followed by removal of fuel rod stubs that are adhering to the plenum's underside. Once this step is complete, the plenum will be raised some 2-1/2 inches using hydraulic jacks to check for intact fuel assembly separation. This operation is currently scheduled to be completed by the end of 1984 and will be followed by transport of the plenum assembly to a storage stand located in the deep end of the refueling canal in the spring of 1985.

The defueling and core shipping activities have made significant progress. Westinghouse Electric Corporation, the defueling equipment contractor, has completed the preliminary design for the defueling tooling. The tooling final design and fabrication are expected to be completed before July 1985, when early defueling is currently scheduled to begin. Early defueling, which basically consists of a vacuuming technique, is projected to be completed by early fall of 1985 and will be followed by bulk defueling. Preliminary design of the fuel shipping/storage canisters is essentially complete. The first canisters are scheduled for delivery in early spring.

After completing many months of engineering evaluations and studies, TI&EP engineers selected the cask designed for rail shipping as the best method for transporting the TMI-2 core to the Idaho National Engineering Laboratory (INEL). Based on this concept, engineers have begun the preliminary cask design.

In addition to the plenum and core activities, the TI&EP is continuing to support the cleanup effort by analyzing samples of core materials and internal components. These efforts, as well as similar efforts in the past, will provide GPU Nuclear recovery engineers with the necessary data to formulate the best approach in solving complex recovery problems. □

Videotapes Detail Head Removal Operations and Successful Waste Disposal System

The TI&EP recently completed two videotapes, now available for loan without charge. One of the programs, titled "TMI-2 Head Removal—One Step Closer to Recovery," details the head removal operation carried out July 24 through July 27, 1984, in Unit 2 at TMI. With actual footage from inside the Reactor Building, this videotape takes the viewer step-by-step through the lift, transport, and storage of the reactor vessel head. The program also discusses the followup work of shielding and covering the opened reactor vessel and some of the preparatory work done in the months previous to the major event.

"EPICOR II: The Evolution of a Successful Waste Disposal System," also recently released, examines how this demineralizer system processed contaminated water through three stages of organic and inorganic ion-exchange media. The videotape specifically discusses EPICOR II system processing of the water, development of a prototype gas sampler that sampled and purged the EPICOR II liners of radiolytic gas, shipments of the liners to the INEL for major research and development studies, and preparations for liner burial in high integrity containers. □

These videotapes may be obtained by contacting 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

TWO DEBRIS CHARACTERIZATION TECHNIQUES FIELD-PROVEN

Engineers at TMI and the INEL have completed an extensive study of various nondestructive assay methods to locate and characterize debris distributed in the TMI-2 primary coolant and connecting piping systems. As a result of this work, they have found two techniques to be highly feasible.

The first technique, ultrasonic detection, uses ultrasound beamed through the pipe wall, then reflected back from the surface being examined. From the reflected ultrasound, or echo, a small computer constructs a contour map of the pipe's inner surface and the debris. Using this technique, engineers are capable of locating a full range of debris, including complete and fragmented fuel pellets, pieces of nonfuel material, and fine powder.

The second technique, passive gamma-ray detection, uses measurements of the gamma rays associated with the decay of certain fission products in fuel to detect and characterize fuel debris in the piping. A high-resolution gamma spectrometer was used in experiments to detect the fission product ^{144}Ce, chosen for its chemical inertness, low volatility, long half-life, and associated high-energy gamma rays.

Each technique has its disadvantages, but both are field-proven, and test engineers believe the advantages far outweigh the disadvantages. Among the disadvantages, the ultrasonic method cannot distinguish between fuel and nonfuel debris, and the shell walls of piping insulation degrade acoustic signals, so the insulation must be removed for the ultrasonic technique to be effective. The passive gamma-ray technique cannot detect nonfuel debris.

However, the advantages are: the gamma-ray technique is especially appropriate for assaying fuel debris in piping, whether or not the pipe is insulated. Also, the two techniques are complementary; used together, engineers can locate fuel and nonfuel debris, determine fuel and nonfuel fractions, and measure the amount, thickness, and physical distribution of all forms of debris in reactor plant piping systems. And both techniques use commercially available equipment.

BIAS VOLTAGE MEASUREMENTS OF LPM CHARGE CONVERTERS MORE RELIABLE

In all pressurized water reactors licensed since 1978, the proper operation of the loose parts monitoring (LPM) system of the reactor vessel and related reactor coolant components must be demonstrated on a regular basis. In some reactors, the system's performance is a limiting condition for continued operation. But the normal routine surveillance procedures, which rely on audio output, will not detect when the system is degrading. A more reliable method of monitoring the state-of-health of an LPM system is by taking regular dc bias voltage measurements of the converters. This is the conclusion of the DOE Instrumentation and Electrical (I&E) Program, which has been researching selected instrumentation and electrical components used in TMI-2 and other nuclear power facilities.

After studying LPM system charge converters removed from the TMI-2 and Sequoyah-1 nuclear power stations, I&E engineers found that the converters, which use field effect transistors of metal oxide silicon, degraded as a function of accumulated radiation dose. These converters, however, were not designed to be radiation tolerant, nor does the manufacturer, Endevco, claim them to be. The TMI-2 instruments had been mounted in low radiation dose areas but failed after being exposed to unusually high radiation after the accident. The Sequoyah-1 charge converters had been mounted under the reactor vessel where they failed as a result of high accumulated radiation dose after 156 effective full-power days.

Plants that use charge converters that are not radiation qualified are recommended to take regular measurements of converter dc bias voltage, which will shift upwards as radiation dose is accumulated until the limit of the power supply rail voltage, normally 30 volts, is reached. By measuring the charge converter dc bias voltage and looking for a higher-than-normal level, plant operators can effectively monitor radiation degradation. The normal bias voltages for the TMI-2 and Sequoyah-1 charge converters were 13.5 and 18 volts, respectively.

In monitoring the converter's audio output, the only normal indication of degradation is a decrease in the usual background vibration levels; this output is not a constant that would indicate to personnel that the converter degraded since its last test. Consequently, a plant could be operating in violation of technical specifications and U.S. NRC Regulatory Guide 1.133 R1, with control room personnel unaware of the condition.

In response to its failed charge converters, Sequoyah-1 replaced its units with temperature and radiation hardened converters. All nuclear power plants are recommended to consider installing charge converters with temperature and radiation tolerant components able to withstand normal plant conditions. Strategic location and shielding of converters can also significantly reduce radiation damage, prolong the service life, and increase the reliability of radiation sensitive equipment installed in a reactor building.

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TMI Unit 2 Technical Information & Examination Program

UPDATE

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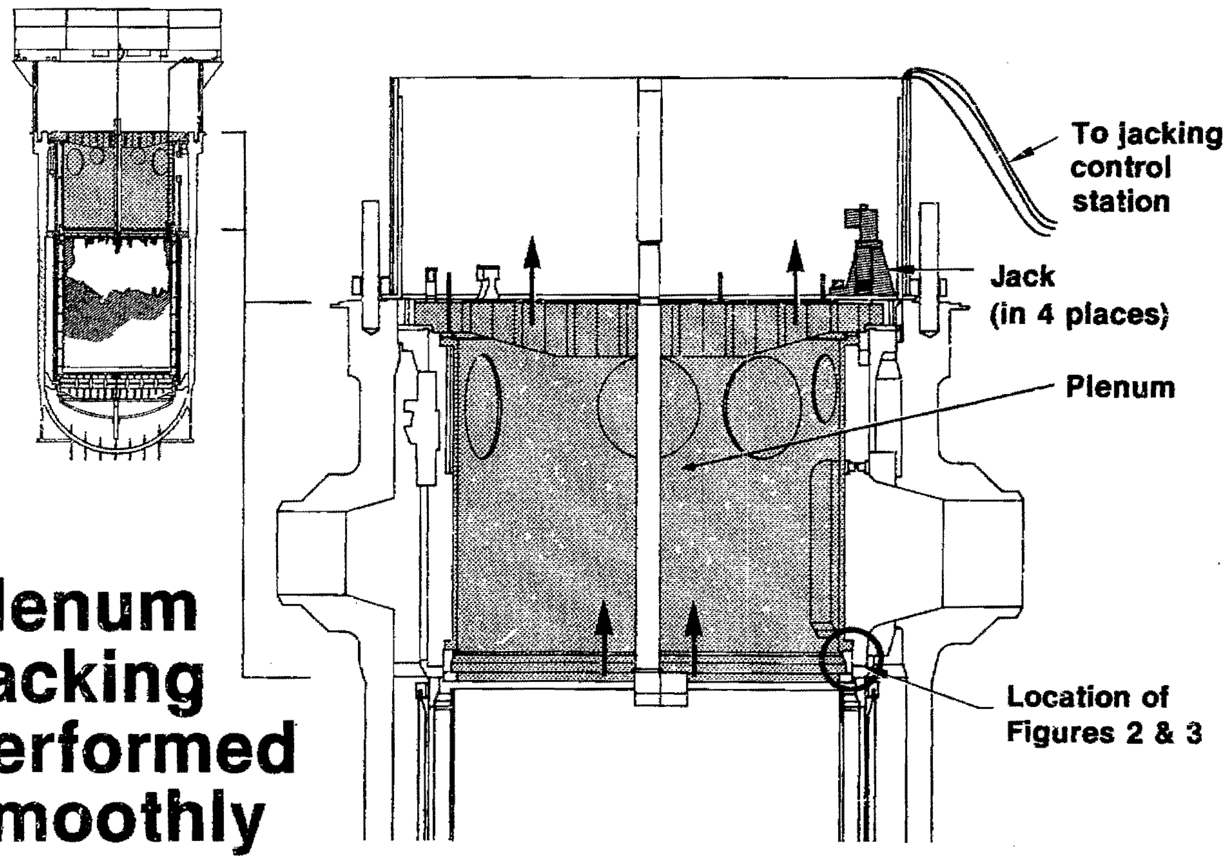
Plenum Jacking Performed Smoothly



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.

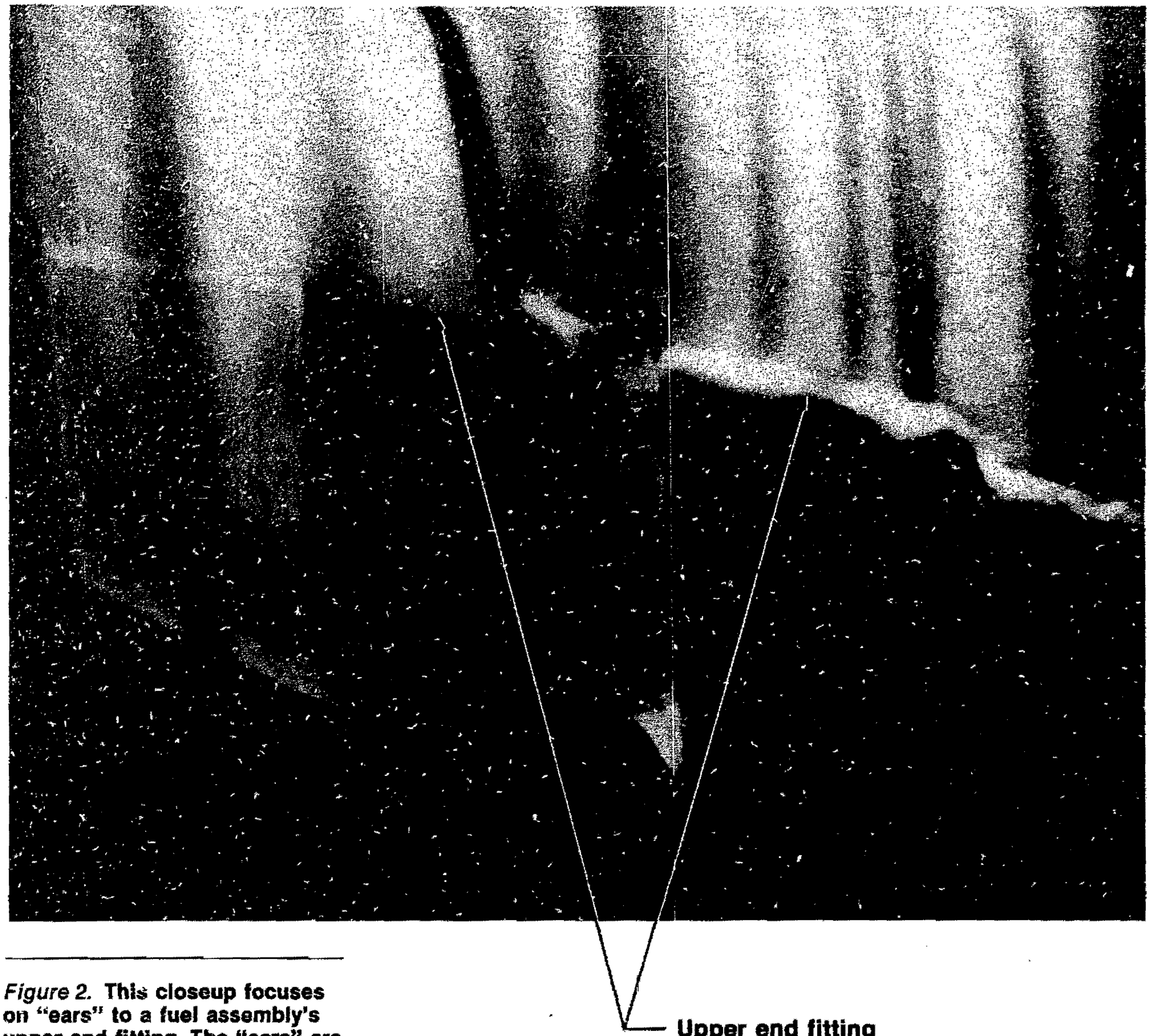


Figure 2. **This closeup focuses on "ears" to a fuel assembly's upper end fitting. The "ears" are beveled to guide the upper end fitting into the plenum assembly.**

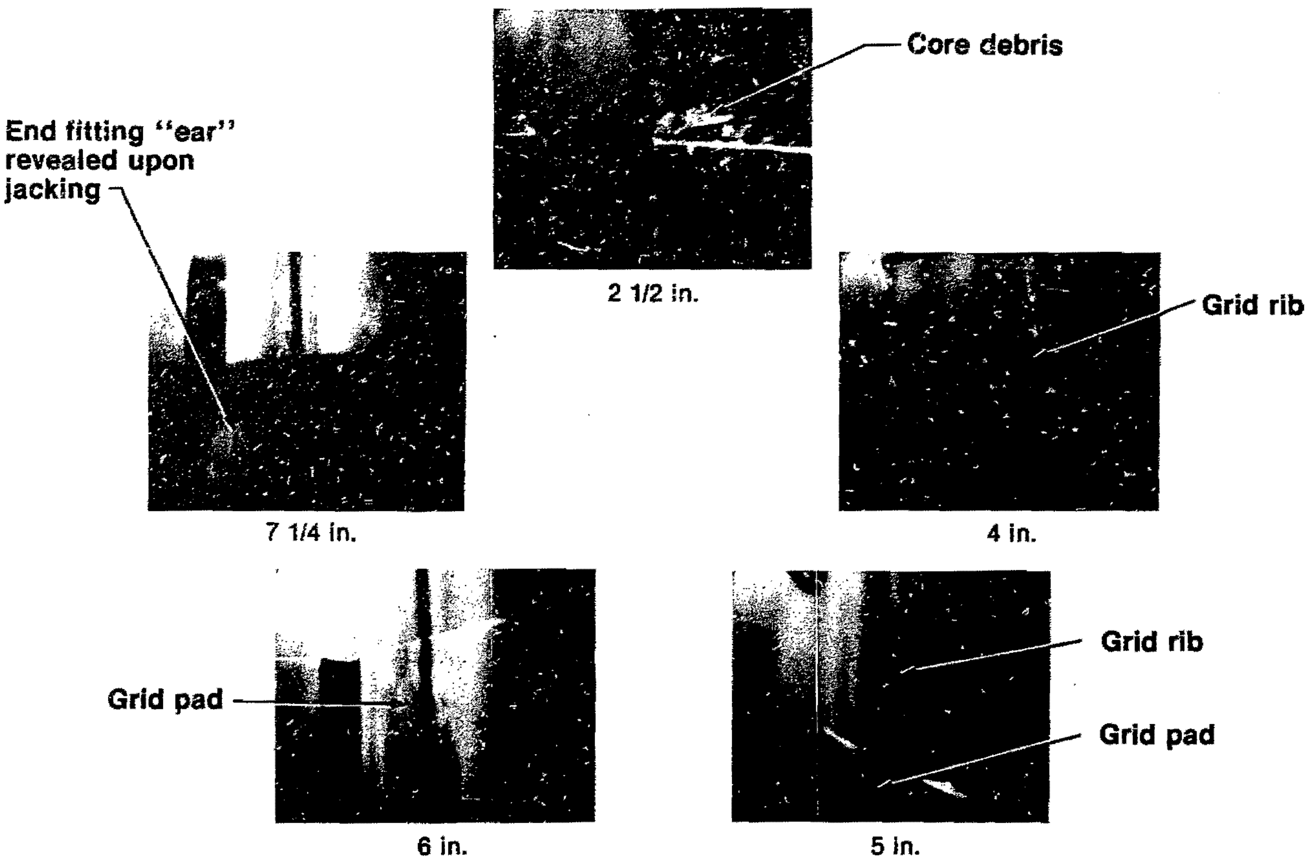


Figure 3 illustrates the jacking sequence from 2-1/2 in. to the final jacking position of 7-1/4 in., with the camera closely focused on an inside section of the plenum. Clockwise from top center, the figure shows the plenum at 2-1/2 in., 4 in., 5 in., 6 in., and 7-1/4 in. In the final picture, an end 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 and have to be processed.

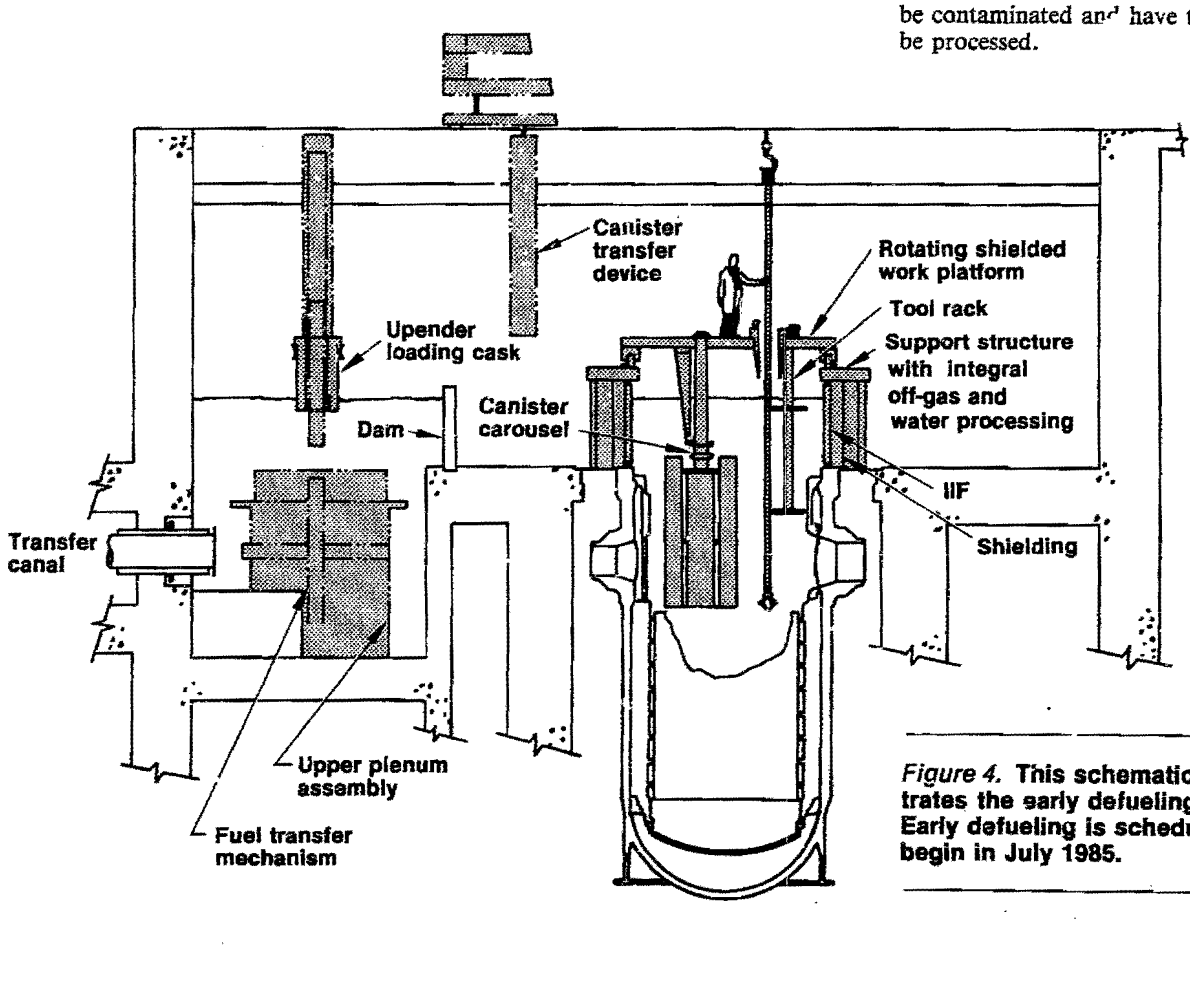


Figure 4. **This schematic illustrates the early defueling system. Early defueling is scheduled to begin in July 1985.**

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.

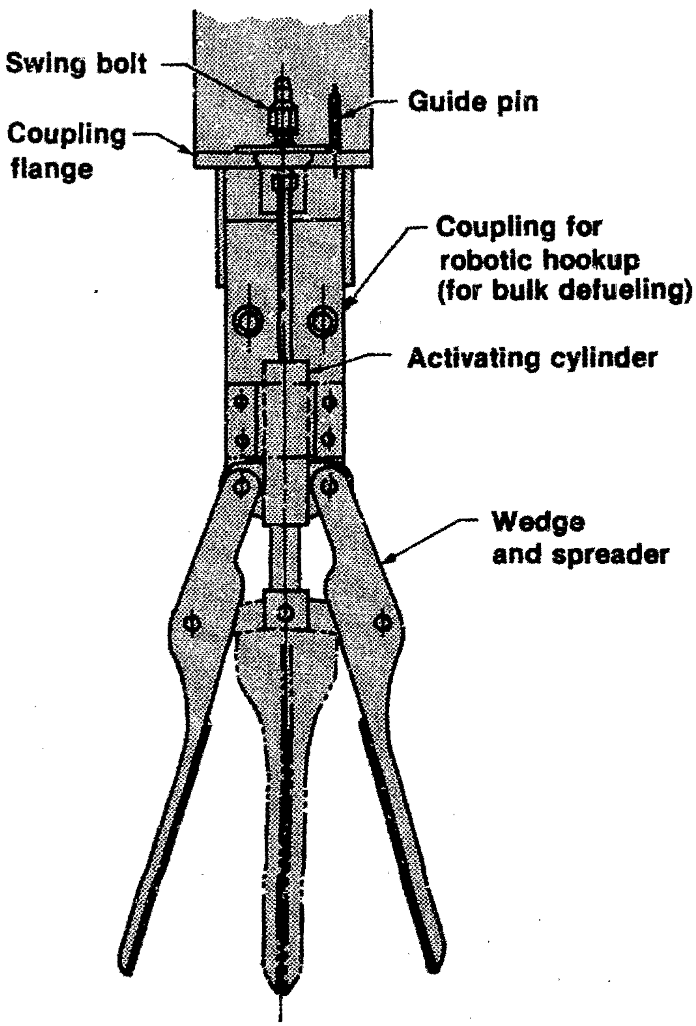
Among the tools are locking pliers to grip large pieces of debris or adjust hoses and cables; three- and four-point grippers to pick up objects from the debris pile; a grapple to lift irregular pieces, such as end fittings and spider assemblies; single rod shears, similar to scissors and capable of cutting 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.)

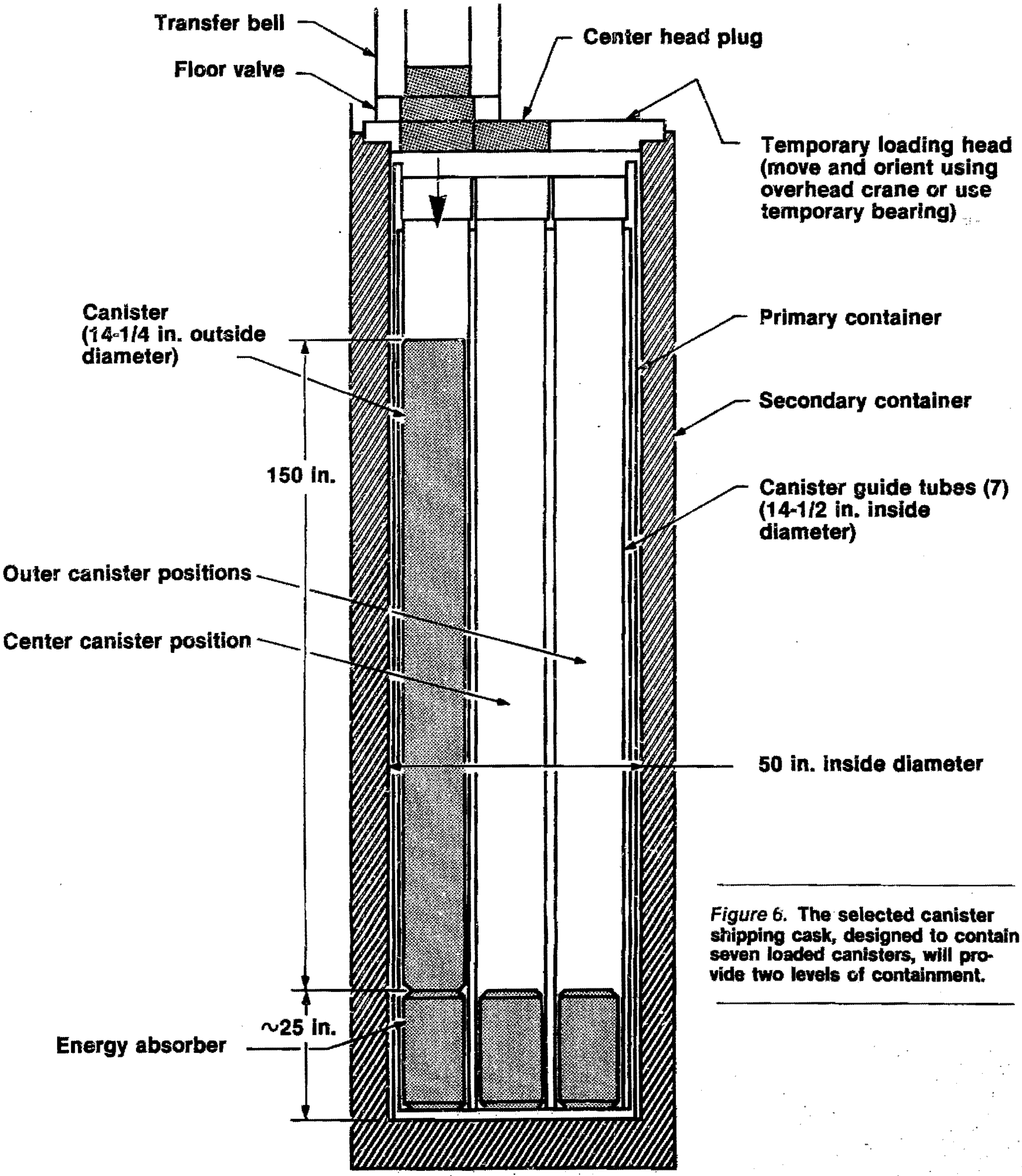


Figure 6. **The selected canister shipping cask, designed to contain seven loaded canisters, will provide two levels of containment.**

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. □

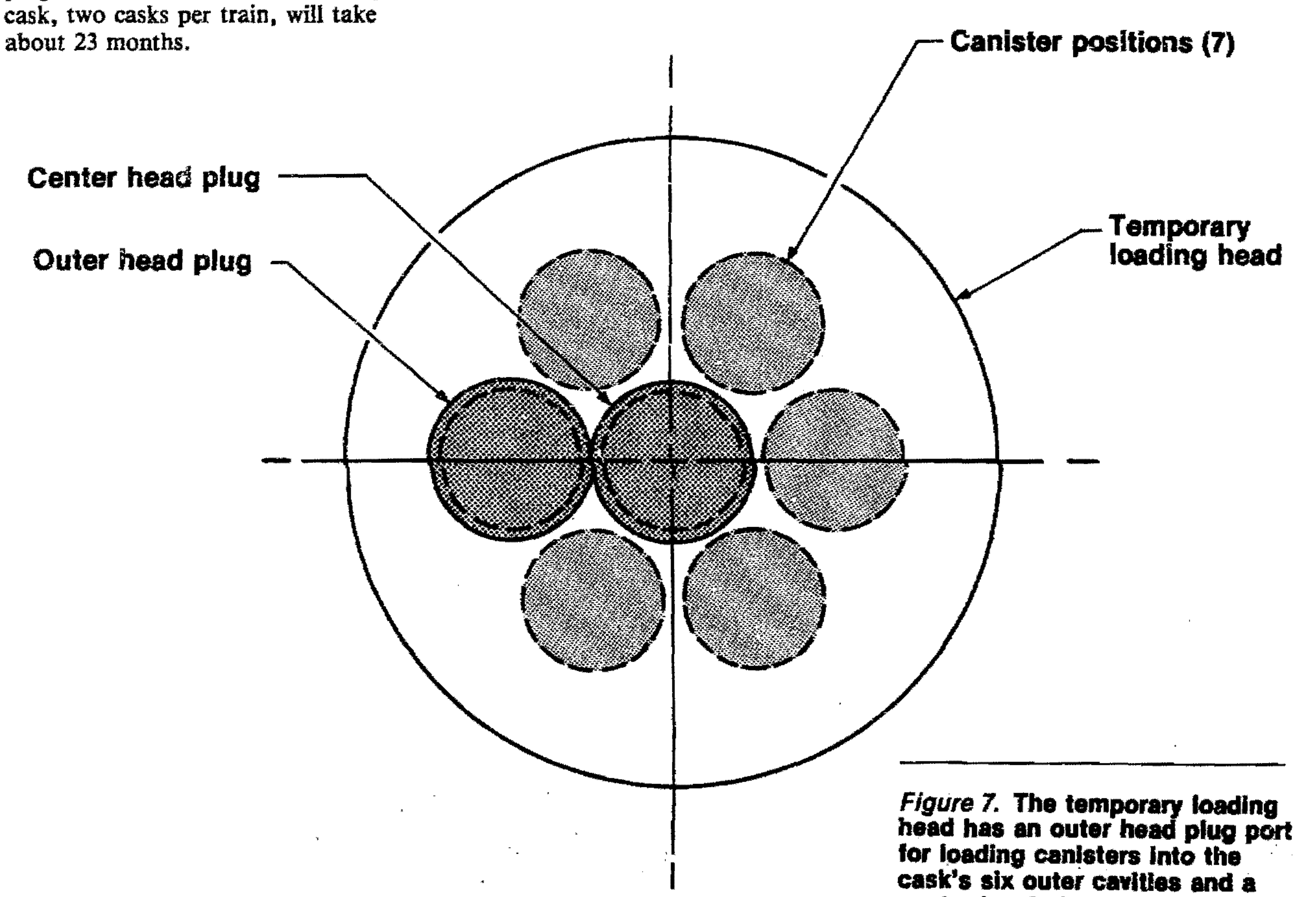


Figure 7. **The temporary loading head has an outer head plug port for loading canisters into the cask's six outer cavities and a center head plug port to load the seventh canister into the center cavity.**

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

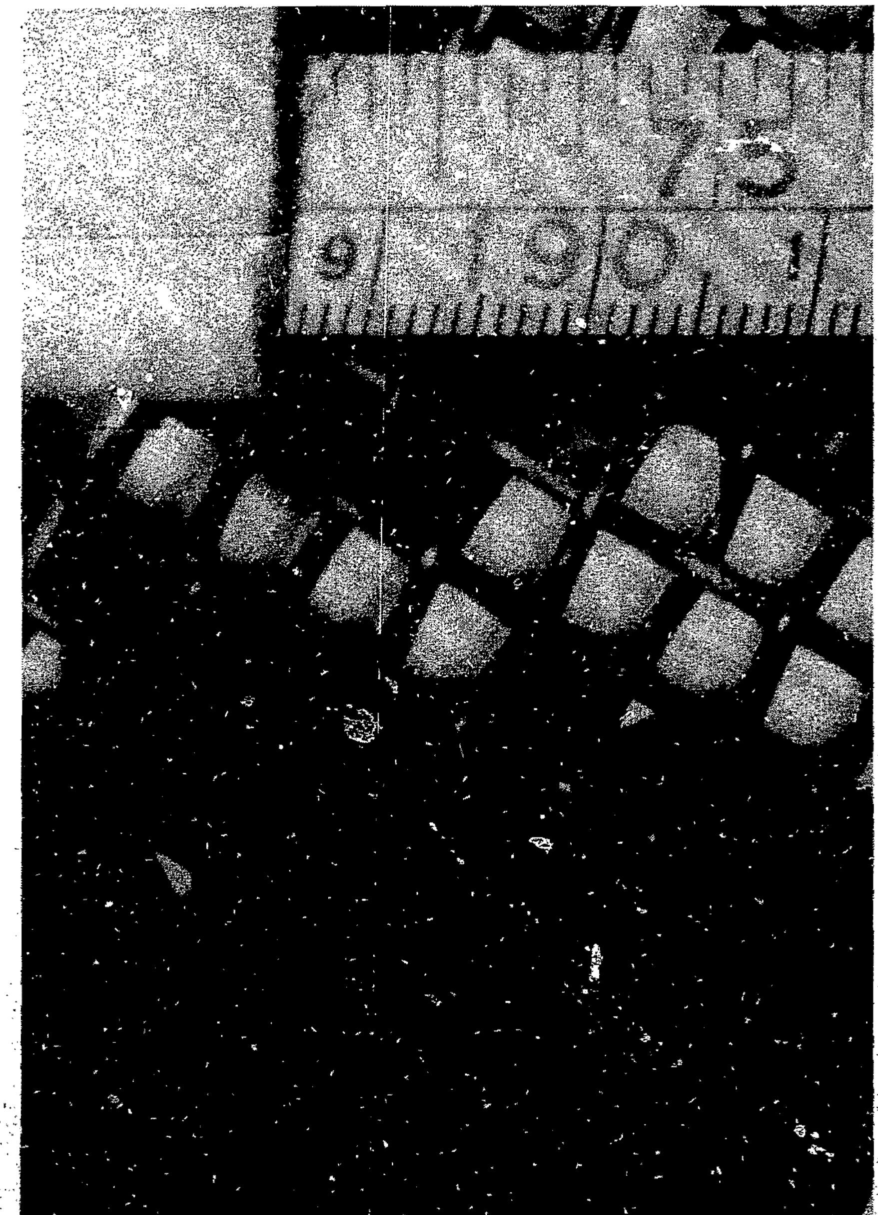


Figure 8. **These particles of greater than 0.157 in. are among the debris in the sample obtained 37 in. into the rubble bed at the core's mid-radius.**

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.

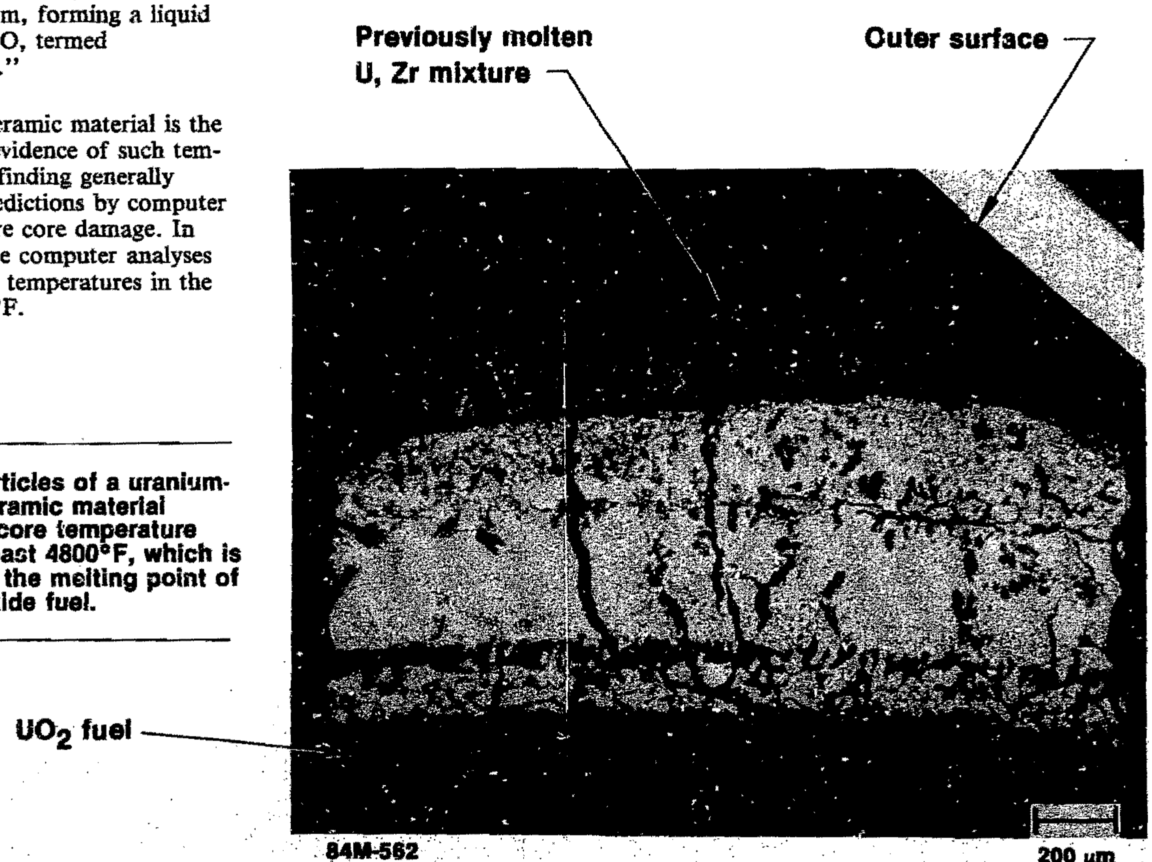


Figure 9. **Particles of a uranium-zirconium ceramic material indicate the core temperature reached at least 4800°F, which is 280°F below the melting point of uranium dioxide fuel.**

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.

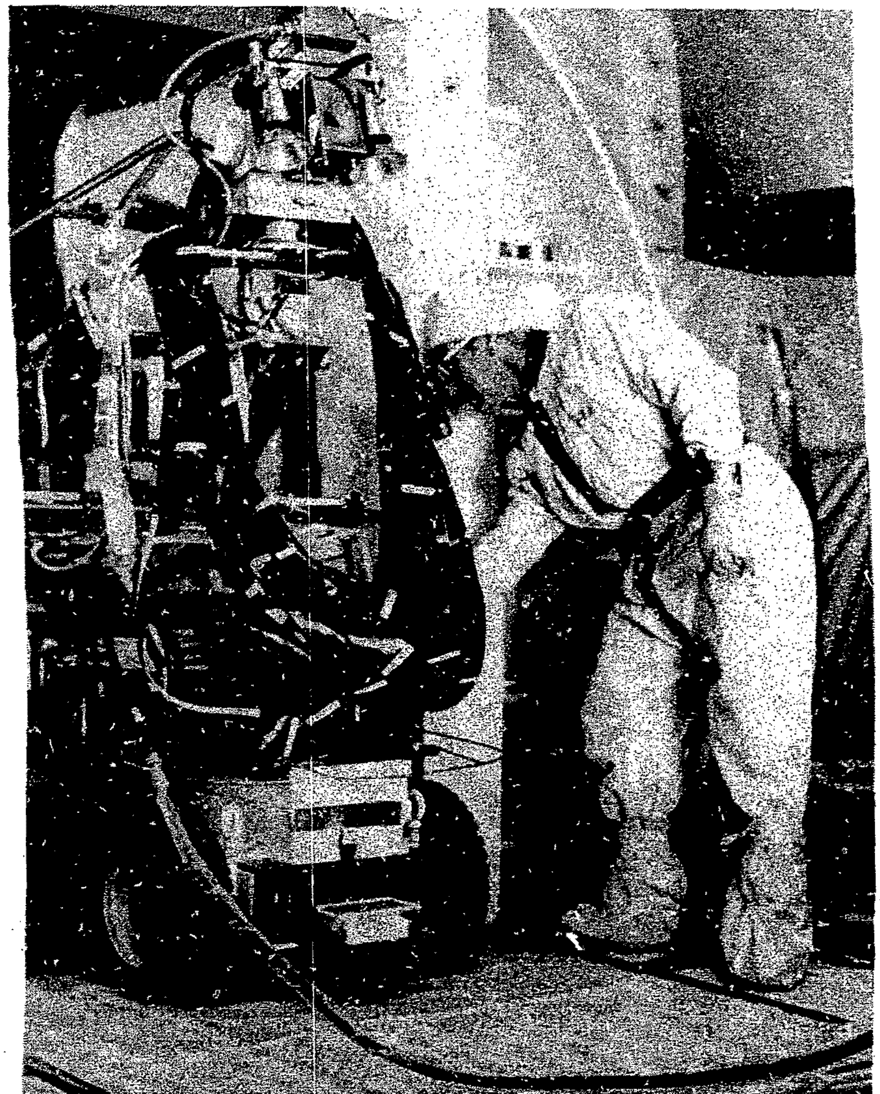


Figure 10. **This 6-ft-tall, remote-controlled vehicle used radiation instruments and three television cameras to inspect the TMI-2 basement.**

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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TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 5, Number 3 August 1985

New Electrical Diagnostic System Supports Maintenance Activities

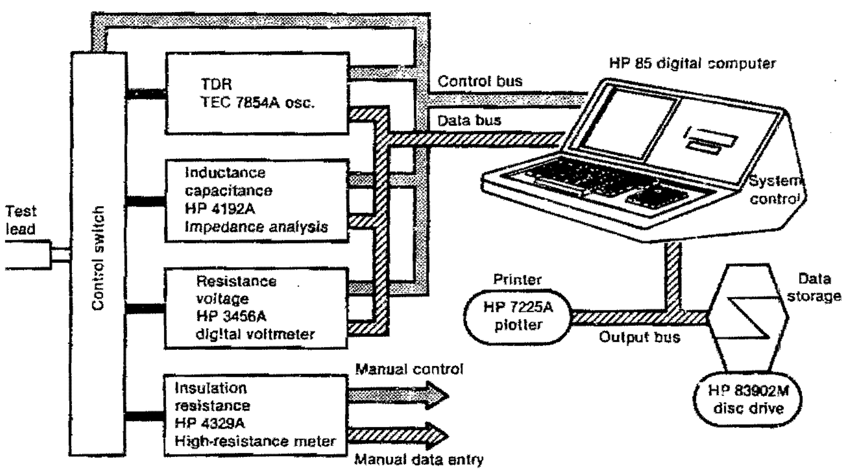


Figure 1. **Electrical circuit characterization and diagnostic system.**

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).

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.

Published by EG&G Idaho, Inc., for the U.S. Department of Energy

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.**

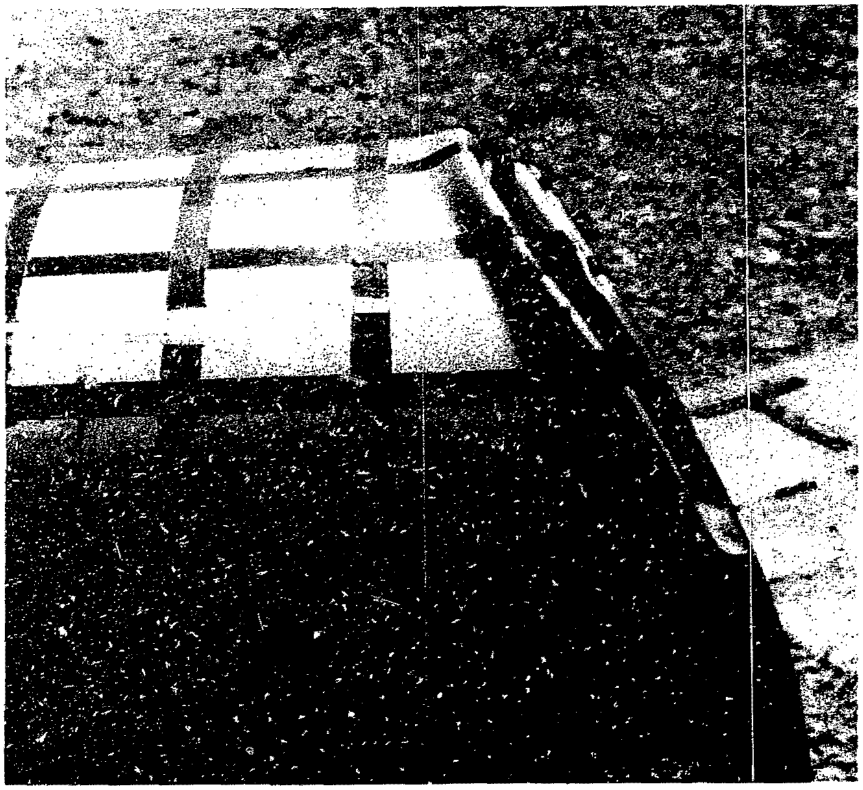
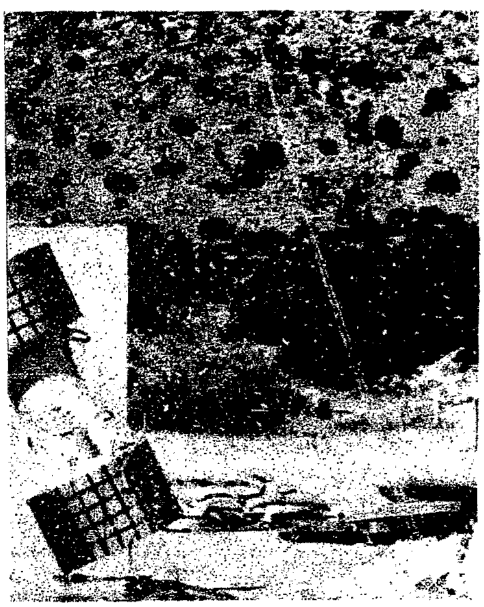


Figure 4. **Overpack damage from oblique drop.**



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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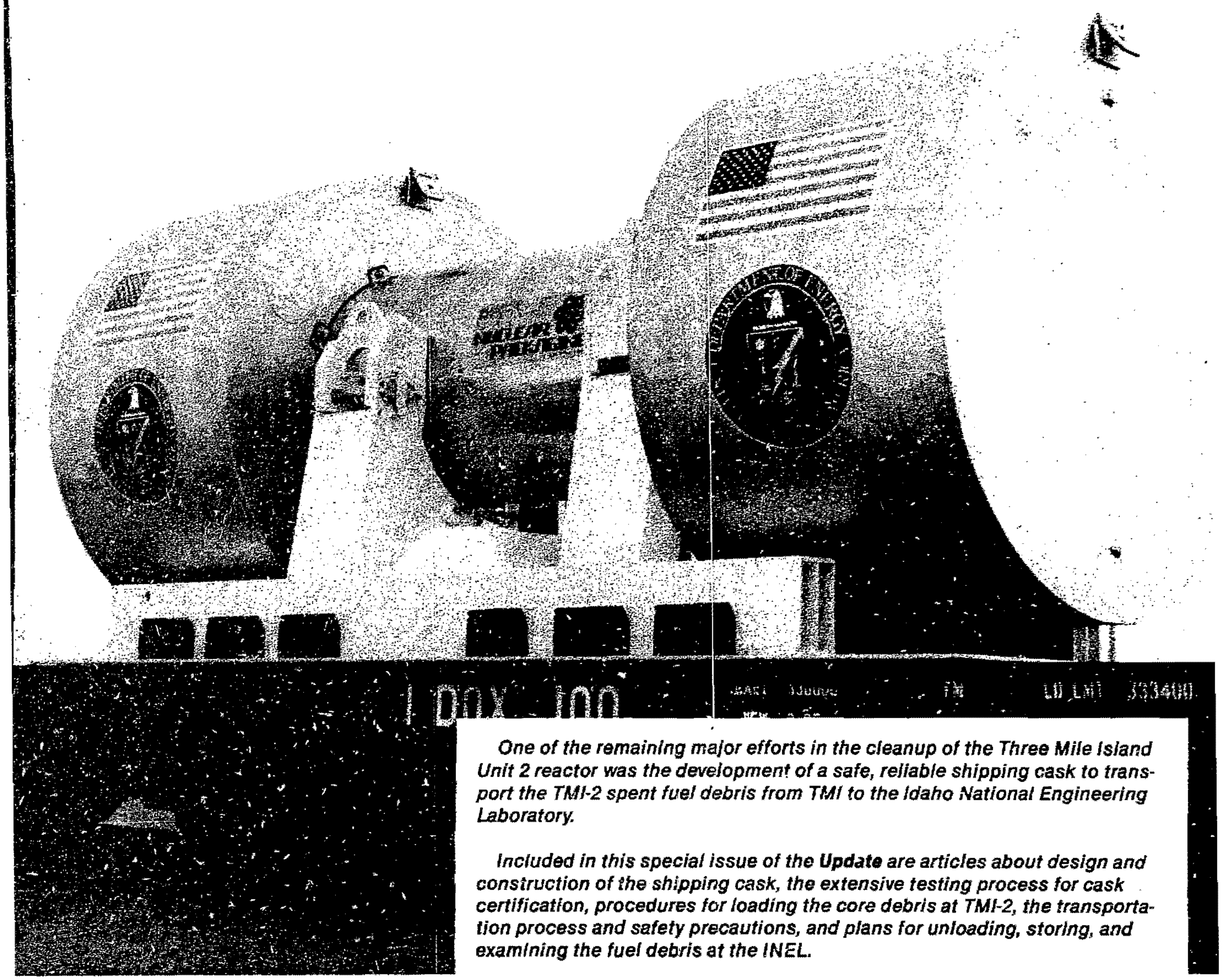
TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 6, Number 1 April 1986

Special cask in place on 150-ton railroad car.

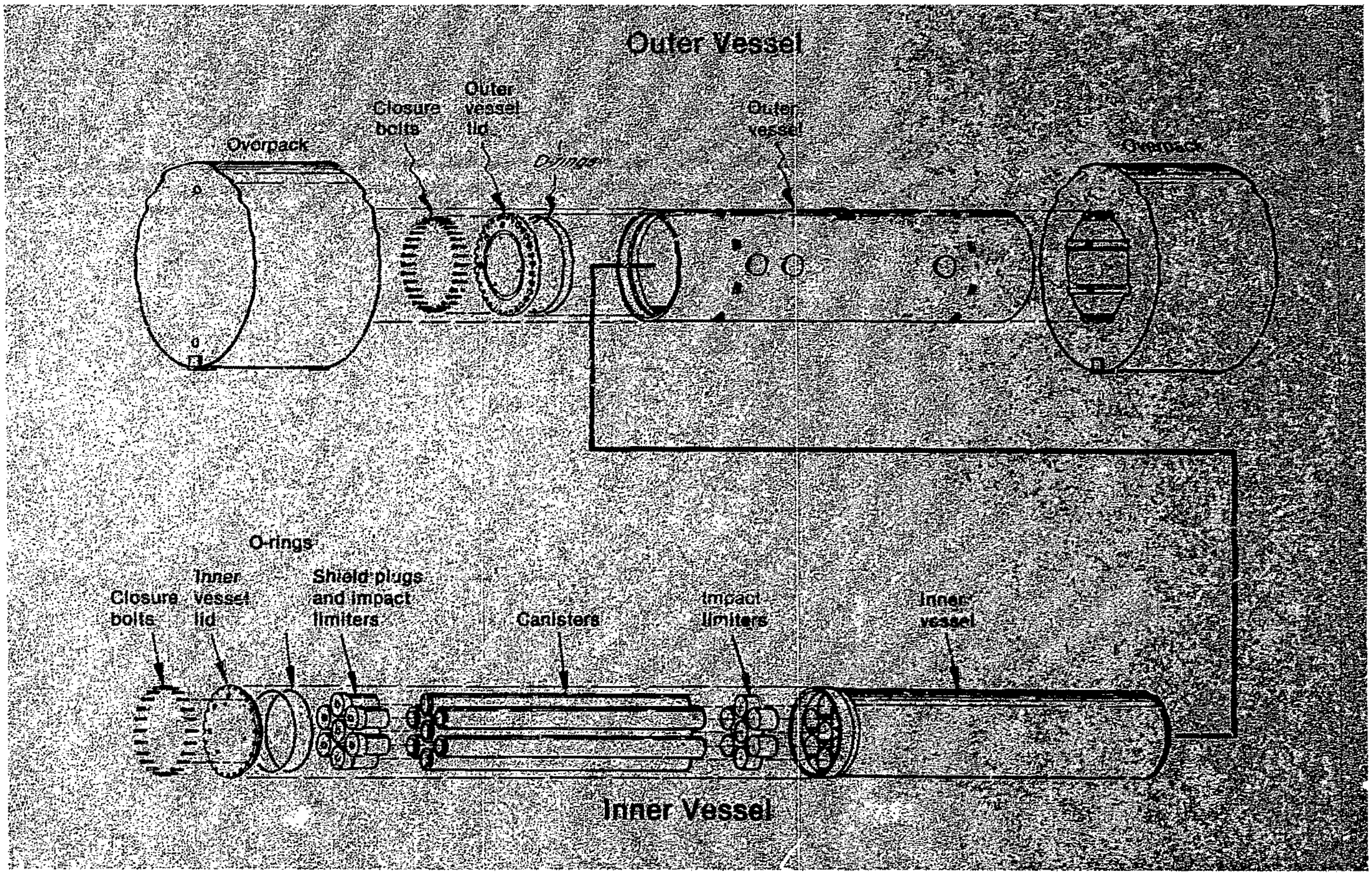


One of the remaining major efforts in the cleanup of the Three Mile Island Unit 2 reactor was the development of a safe, reliable shipping cask to transport the TMI-2 spent fuel debris from TMI to the Idaho National Engineering Laboratory.

*Included in this special issue of the **Update** are articles about design and construction of the shipping cask, the extensive testing process for cask certification, procedures for loading the core debris at TMI-2, the transportation process and safety precautions, and plans for unloading, storing, and examining the fuel debris at the INEL.*

Published by EG&G Idaho, Inc., for the U.S. Department of Energy

Special Cask Developed for Core Debris Shipments



Exploded view of the rail cask outer and inner vessels.

In 1984, the Department of Energy (DOE) signed a contract with GPU Nuclear Corporation to accept TMI-2 core debris for use in a research and development program aimed at understanding the accident sequence at TMI-2. DOE is taking the responsibility for transporting, storing, and ultimately disposing of the entire core. The first of more than 250 canisters filled with TMI-2 debris is expected to be delivered by GPU Nuclear to DOE in mid-1986; the shipping program is expected to last two to three years.

During the planning stages for handling core debris, EG&G Idaho (a DOE prime contractor at the Idaho National Engineering Laboratory) investigated spent fuel shipping cask options. The requirements for TMI-2 debris transport led to the decision that new casks be designed, certified, and fabricated for this unique project rather than modify and recertify existing casks. EG&G Idaho also evaluated whether canisters should be transported by truck or rail.

While truck-mounted casks could transport one to three fuel canisters each, the use of a rail cask that holds seven canisters has significant advantages. With more canisters in a rail cask than in a truck cask, fewer shipments will be needed. Only 35 to 40 rail shipments will be required, compared with the potential for more than 250 truck shipments.

Fewer shipments reduce the chance for an accident involving the cask during the transportation sequence and thereby reduce the total risk to the public. In addition, fewer shipments mean fewer loading and unloading operations and reduced radiation exposure to workers. For the overall TMI-2 shipping operation, the use of rail casks is projected to be more efficient and less costly than if truck casks were used.

The choice of rail to transport the TMI-2 core debris led to the development of the Nuclear Packaging, Incorporated (NuPac) 125B rail cask. This cask was designed, tested, and fabricated specifically for transporting the TMI-2 spent fuel debris to the INEL. The cask was certified by the Nuclear Regulatory Commission (NRC) in April 1986.

When the cask design was started in late 1984, several unique factors about the condition of the TMI-2 spent fuel had to be considered. Existing spent fuel shipping casks are certified only for transporting assemblies of undamaged spent nuclear fuel. The NuPac 125B rail cask had to be certified to transport spent fuel debris from the TMI-2 accident. Without the cladding that surrounds the spent fuel in an intact assembly, two barriers are needed during transport to comply with NRC regulations.

Under NRC regulations a cask with two barriers is required. Each barrier is a specified containment boundary that must meet stringent requirements for structural strength and demonstrate that an uncontrolled release of the contents will not occur, even after a sequence of accident conditions.

This double containment in the NuPac 125B rail cask is accomplished by use of two separate and strong vessels, one inside the other, each with a thick lid and seals that will be leak tested before each shipment. In addition to the cask inner and outer containment vessels, there are canisters into which the fuel debris will be loaded underwater at TMI. These canisters are another barrier that prevents a release of material during transport. A complete shipping package includes the double containment cask and its canisters, making three levels of protection to ensure the safety of the public.

Leaktight Design

Another unique feature of the NuPac 125B rail cask is the extremely small rate of leakage of radioactive materials that is allowed after a sequence of serious accidents. Each of the two cask containment vessels was designed, built, and tested to a leakrate low enough that the term "leaktight" is applicable, even during and after hypothetical accident conditions.

The leakrate for leaktight is defined as one-tenth of one-millionth of a cubic centimeter of gas per second at a pressure difference of one atmosphere across the containment boundary. This leakrate is equivalent to about three cubic centimeters in a year, or a bubble growing to about the size of a pingpong ball. Only gas could escape...not radioactive particles.

This low leakrate applies for leakage from the inner to the outer containment vessel, as well as from the outer vessel to the environment. The canisters and containment boundaries in the rail cask will ensure that an uncontrolled release of material to the environment will not occur.

Another important design consideration in developing a safe shipping package for the fuel debris was the control of gases that are generated when radioactive materials are in contact with water. The radiation that is emitted splits nearby water molecules into hydrogen and oxygen gases by a process called radiolysis.

These gases must be controlled during transport of wet radioactive materials or a flammable gas mixture could result. The method of control for TMI-2 fuel debris shipments is to use a catalyst that recombines the hydrogen and oxygen gases into water and allows safe transport of the fuel debris.

One other important consideration in the rail cask design was ensuring that the nuclear fuel contents would remain subcritical under all conditions. Subcritical means that the self-sustaining splitting of atoms that occurs in a nuclear reactor cannot occur in the cask.

The rail cask and the fuel debris canister designs ensure subcriticality of the nuclear fuel. This feature—an overriding design consideration—led to the incorporation of criticality control structures into each canister and the inner containment vessel of the cask.

The criticality control materials are positioned and supported to ensure subcriticality of the nuclear fuel by absorbing neutrons needed to achieve a chain reaction. With these neutron absorbers, subcriticality is maintained even after the sequence of accidents is considered.

Inner Containment Vessel

Each cask consists of an inner containment vessel that fits into an outer containment vessel. The inner vessel is fabricated starting with a hub-and-spoke structure made of stainless steel plates that are welded together. This structure is welded to two large forgings at each end. The structure prevents the seven canisters and their supports, which fit into each opening in the structure, from crushing each other in impact accidents.

Each canister fits into a stainless steel tube that forms part of the containment boundary of the inner vessel. Each tube is welded at the bottom to a thick plate that seals the tube closed at this end. The containment boundary is completed with a massive forging to which the tubes are welded and the thick, stainless steel lid that is bolted to the forging.

The 5-inch-thick lid is bolted down with 24 3/4-inch-diameter bolts. Around the edge of the lid are two O-rings that form the bore seals, which are inspected and leak tested before each shipment.

In addition to the stainless steel plates that separate the seven containment tubes, there are one-inch-thick plates welded around the outside that stiffen the inner vessel and form voids between the plates and the outer surface of the containment tubes.

A neutron absorbing material that solidifies like concrete is pumped like grout into these voids. The neutron absorber ensures that the canisters remain subcritical and the strength of the material, together with the plates, protects the containment tubes from damage should an accident occur.

For added safety, another design feature is incorporated inside the inner vessel. Located at the end of the containment tubes are removable energy absorbers that protect the canisters by crushing under accident conditions. Each energy absorber is an aluminum honeycomb material that limits the axial impact forces on the canisters.

The upper energy absorbers are attached to the bottom of shield plugs—short, solid cylinders of stainless steel added for worker radiation protection. After canisters are loaded into the cask, the shield plugs reduce the radiation from the fuel debris to levels that allow workers to replace the inner vessel lid and test the seals.

Outer Containment Vessel

Like the inner containment vessel, the outer containment vessel has many safety features included in the design. The outer vessel is called a composite wall cask because there are three thick layers of metal that form the wall of the cask. Two layers are stainless steel shells, one inside the other, that have a gap of nearly four inches between them. Molten lead is poured into the gap between the shells. The molten lead pour is accomplished after a brick oven is built around the outside of the cask. The entire cask is heated to a temperature hotter than the melting point of lead and the molten lead is added. When the lead cools and solidifies, it becomes an effective shield to reduce radiation levels outside the cask to below acceptable levels. After controlled cooling of the cask, the shielding effectiveness of the lead is checked with a radiation source to ensure there are no voids in the lead.

The larger stainless steel shell is two inches thick, while the shell that fits inside is one-inch-thick stainless steel. Both shells are welded at the bottom to a thick base plate that is carefully machined to the correct dimensions for welding.

Both shells are also welded to a large upper forging of stainless steel that is machined to very precise dimensions where the outer vessel containment seal is formed. The 7.5-inch-thick lid is bolted in place with 32 1.5-inch-diameter bolts. Around the edge of the lid are two O-rings that form the bore seals, which are inspected and leak tested before each shipment.

Attached to the outer shell are thick, short cylinders of stainless steel that are used to lift or hold down the cask during use. These attachments, also known as trunions, are designed and tested to show that they can support more than the weight of the loaded cask.

Another attachment to the outer shell is a structure called the shear block. This attachment absorbs forces during transport that would jolt the cask forward or backward, and protects the trunions from high inertial loads which may be encountered during transport.

Another safety feature of the rail cask is a thermal shield that would help protect the cask in an accident involving fire. The thermal shield consists of a wire wrapped around the outer shell every couple of inches, covered by a thin sheet of stainless steel welded over the wire, leaving an air gap between the thin sheet and the outer shell. This air gap reduces the amount of heat that can flow into the cask body in a fire because air is a poor conductor of heat energy. The thermal shield and the high heat capacity of the cask would keep temperatures low inside the cask if a fire occurred.

One other structural safety feature gives the cask a dumbbell-shape appearance. Large energy absorbers, called overpacks, are attached to each end of the outer shell. Each overpack is made of a thin plate of stainless steel and filled with foam that crushes on impact, absorbing energy and protecting the cask body. The effectiveness of the overpacks was demonstrated by a series of drop tests, done as part of the cask certification process, that showed the safety of this cask design feature. (An article about the drop tests appears in this Update issue.) □

Special Canisters Designed to Hold Spent Fuel Debris

Three different types of canisters are being used to defuel the TMI-2 reactor. Each has the same general external appearance—a stainless steel vessel 14 inches in diameter by 150 inches long. All have features that ensure safety during transport inside the rail cask.

The first type of canister is called a fuel canister and has a removable upper lid. With the lid removed, there is a square opening into which damaged fuel assemblies with a full cross-section can be lowered.

The second type is a knockout canister and is used in a hydraulic vacuum defueling operation. Water and pieces of debris are vacuumed up with a tool and pumped through the inlet of a knockout canister. The pieces of debris settle out of the water as the flow velocity decreases in the relatively larger diameter of the canister. The water, with residual fine pieces of debris, leaves the knockout canister and enters the third type of canister—a filter canister. This canister captures the fine debris on pleated, 0.5-micron stainless steel filters.

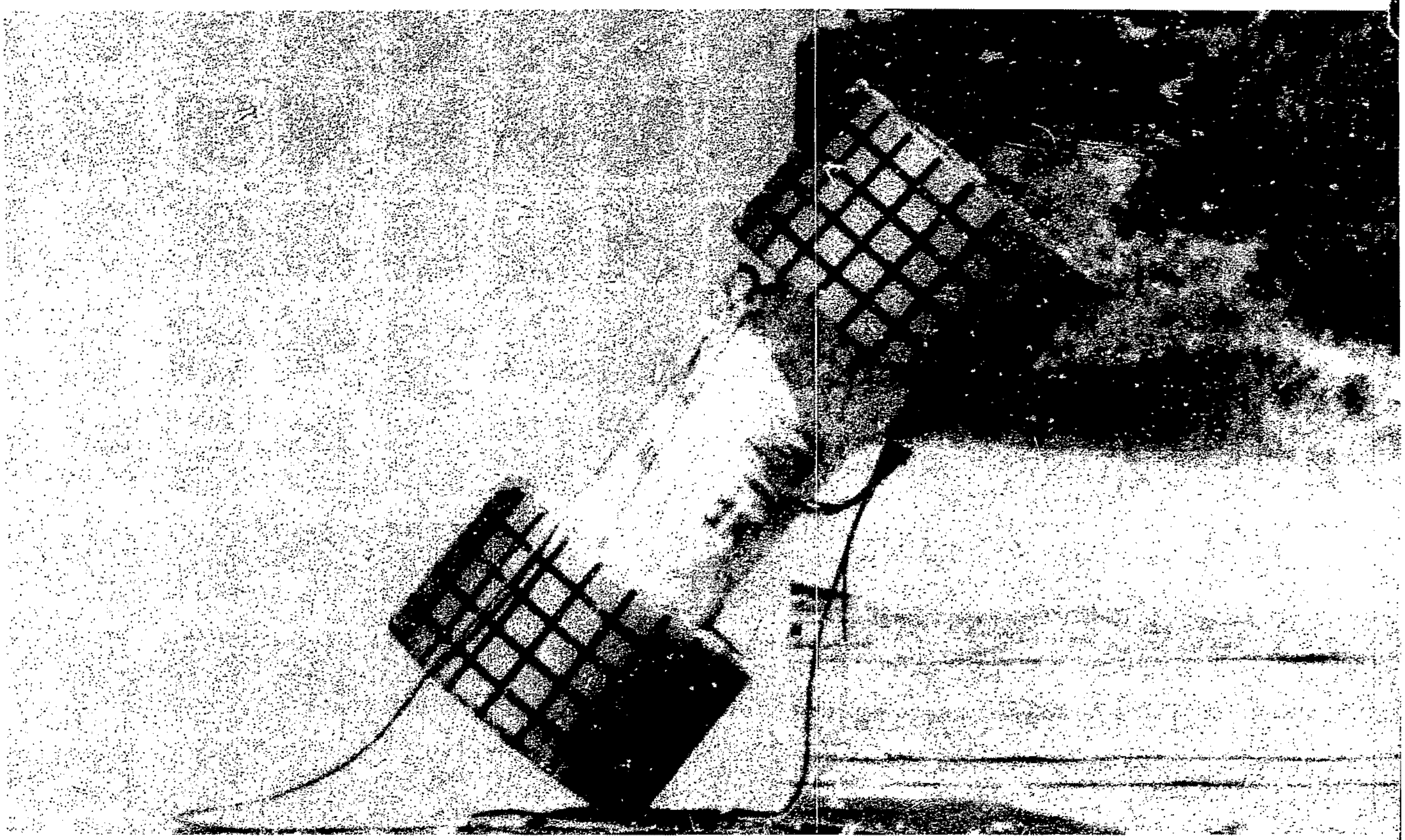
Neutron absorber materials are also built into all three canister types to ensure subcriticality of the nuclear fuel. In the fuel canisters, there is a square of borated aluminum sandwiched between two sheets of stainless steel. To ensure that the square does not move in an accident, lightweight concrete is added to fill the space between the outside of the square and the inside of the canister shell.

The neutron absorbers in the knockout canisters are located inside one large control tube and four small outer tubes. Each tube contains pellets of boron carbide that are seal welded inside. The tubes are supported along their length by thick plates that limit movement of the tubes.

In the filter canisters, the mass of the stainless steel filter media and a central tube of boron carbide pellets (as in the knockout canister) act as the neutron absorbers.

In all three types of canisters, both the upper and lower canister heads have beds of catalytic materials that recombine the radiolytically generated hydrogen and oxygen gases back into water and prevent the formation of combustible gas mixtures. □

Thorough Analyses and Tests Performed for NRC Cask License



Oblique drop at the instant before impact.

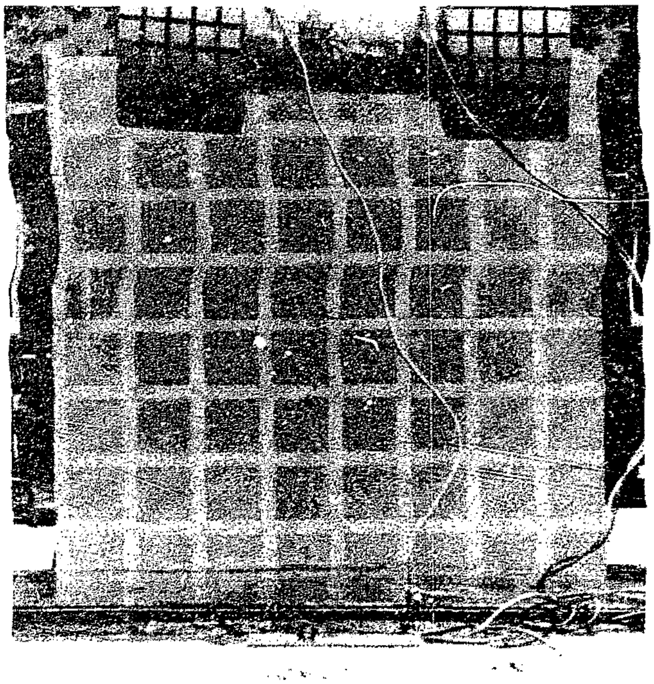
Obtaining certification from the NRC for the NuPac 125B rail cask required thorough analyses of the cask structures, thermal behavior, containment capability, shielding performance, and controls that ensure subcriticality.

The certification for the rail cask is based on an extensive three-volume safety analysis report. The report contains both the results of computer analyses and data from drop tests that were performed to demonstrate the structural integrity of the cask and canisters.

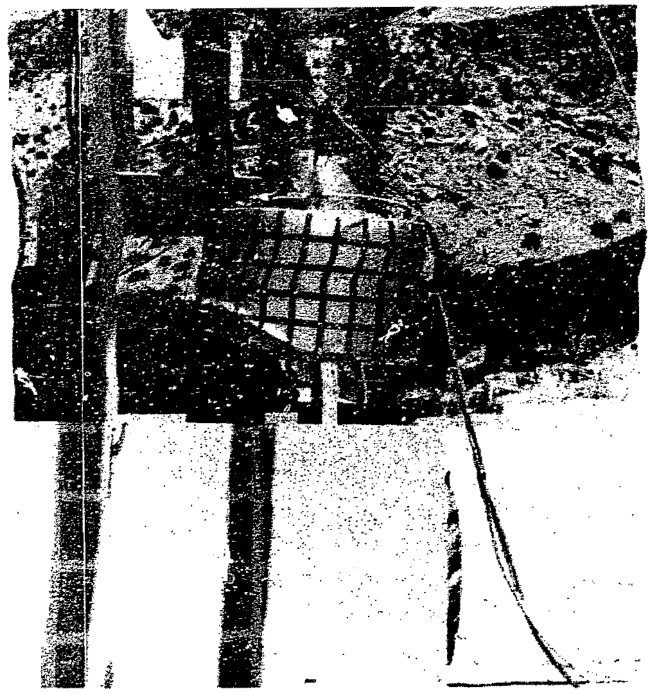
The results of the drop tests confirmed the predictions made in the structural analyses on the strength and behavior of the cask and canister structures during accident conditions. The drop tests provide conclusive evidence of the validity of the analytical models. The test results were given to the NRC to accelerate resolution of potential delays for questions about the amount of conservatism used in the structural analyses.

Cask Tests

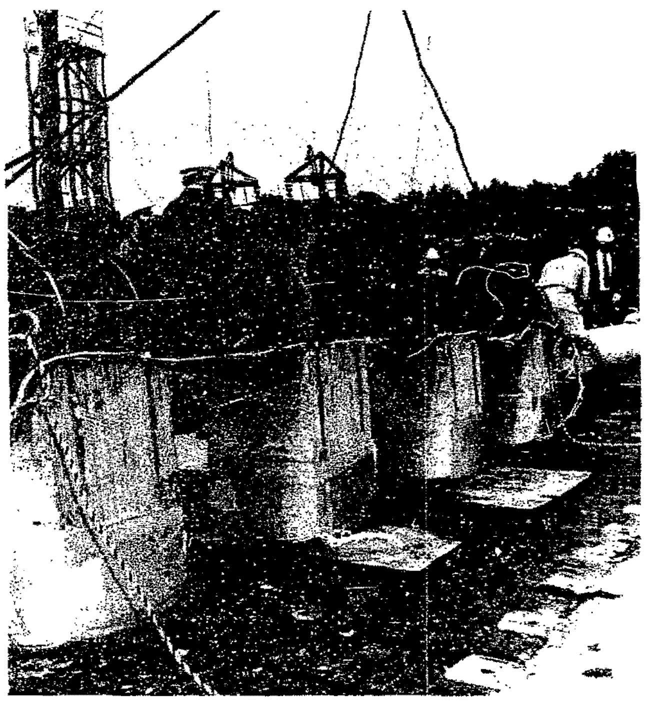
To ensure that only safe packages are used in transport, NRC regulations require that spent fuel shipping casks survive a series of severe accidents, including (in sequence) two drops of the package in an orientation to produce the maximum damage. The first drop is from 30 feet onto an



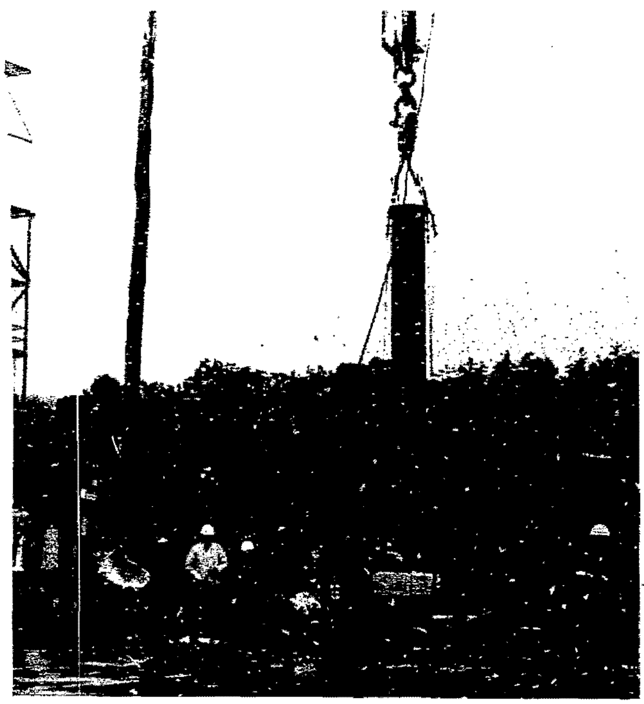
End puncture drop at the instant before impact.



Puncture drop height and orientation check.



Cask simulation vessel with simulation impact limiters for horizontal drops.



Cask simulation vessel and simulation impact limiter for vertical drops.

unyielding surface, followed by a drop from 40 inches onto a steel rod that is long enough to produce maximum damage to the package. The two drops are followed by a 30-minute fire at a temperature of 1475°F, after which the package is assumed to be flooded with water so that controls for subcriticality can be evaluated.

The damage from the 30-foot drop, for both cask and canisters, was first predicted analytically for every possible angle of impact and then demonstrated with a series of drop tests. For the cask drop test program performed at Sandia National Laboratories, a one-quarter-scale model was used. (Scale-model testing is an engineering practice that is used extensively in solving problems in aerospace, civil, mechanical, and nuclear engineering. The scaling laws are widely accepted and provide a cost-effective method of demonstrating design adequacy.) The scale-model tests confirmed the predicted behavior of the full-size cask.

Several drops were made with the quarter-scale model to show, for different cask orientations, the maximum damage to different parts of the cask. Three drops were from 30 feet onto an unyielding surface. Two of the three drops were conducted at a temperature of -20°F to simulate an accident at subfreezing temperatures that might cause brittle materials to fracture upon impact.

The first 30-foot drop was onto the bottom end of the cask to determine how well the cask walls, lids, and closure bolts performed. The test also demonstrated that the energy absorbers within the inner vessel adequately protected the canisters. The oblique angle drop from 30 feet was onto the lid, at an angle that would maximize the stress on the cask body The side drop from 30 feet was done to produce maximum loads on the inner vessel.

The first 40-inch drop onto a puncture rod demonstrated the integrity of the cask side wall in an accident where the outer foam overpacks are not effective in absorbing energy and the cask wall must absorb the impact of a protruding object. The second 40-inch drop onto the lid showed how the cask lid would remain undamaged in a puncture accident without reduction of the impact energy by the overpacks.

After the drop tests, the cask was disassembled, inspected, and damage to the overpacks was documented. The model cask was measured, leaktested, and x-rayed to ensure that any structural damage would be found. As expected, the test data confirmed the damage predicted by the analysis for the drop conditions.

The tests showed conclusively the safety of the cask, even in accidents involving severe impacts. For comparison, the impact in a drop from 30 feet onto an unyielding surface is about the same as an impact at 90 miles per hour into two feet of reinforced concrete.

Canister Tests

A series of drop tests with the fuel canisters showed that the square shroud did not move when surrounded by the lightweight concrete in the canister. A full-size knockout canister was subjected to four 30-foot drop tests at Oak Ridge National Laboratory.

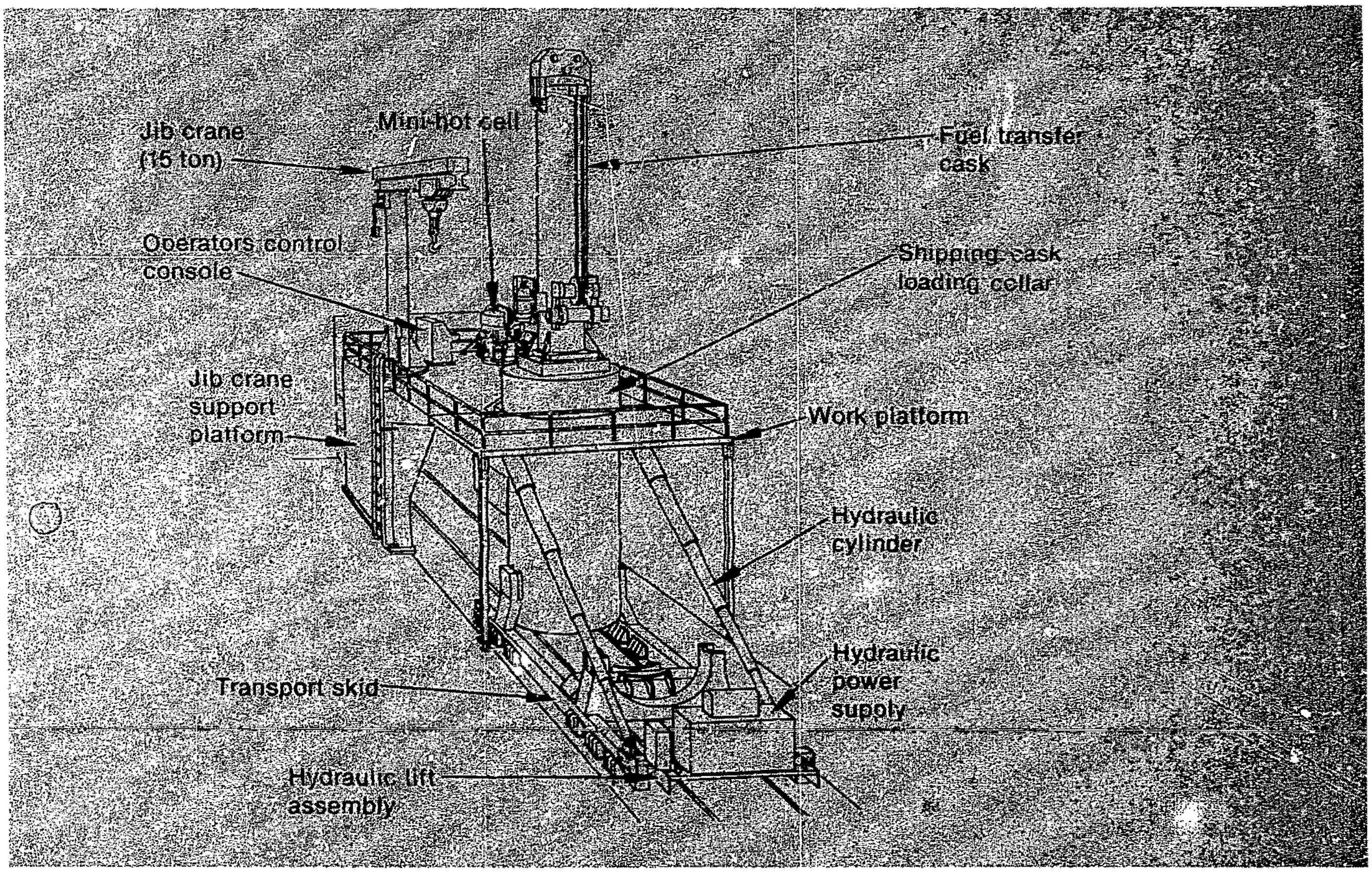
Two of the tests were with the canister in a vertical orientation. One drop test, onto the bottom of the canister, showed that the canister internal structures could safely withstand the force of the fuel debris coming down and compressing the tubes in the structure that contain the neutron absorbers. The second vertical drop was onto the upper end of the canister to show that the weight of the fuel debris could not apply forces that would pull the internal structure apart.

Two other drops were made with the canister horizontal to investigate bending and twisting of the internals. All four tests showed that the tubes containing neutron absorbers experienced no deformations beyond those determined by computer analyses of the structures.

Besides the drop test program, a thorough test program was performed on the catalyst beds installed in each canister to recombine the hydrogen and oxygen gases generated by radiolysis of water. In each test, the performance of the catalyst bed was measured while hydrogen and oxygen gases were added at a flowrate about three times what is expected to be generated in a TMI-2 debris canister.

The testing program helped determine the size and shape of the beds to be built into each canister. The effects of the environments to which the catalyst beds would be exposed, such as chemicals in the water in the TMI-2 reactor, were also investigated. The catalyst test program provided conclusive evidence of the satisfactory performance needed to ensure safe transport of the TMI-2 fuel debris. ☐

New Loading Procedure Developed for Debris Canisters



TMI fuel cask loading components.

Because the spent fuel storage pools at TMI-2 were being used for accident recovery operations, fuel debris canisters could not be loaded underwater into a shipping cask, which is a traditional industry practice. Instead, the NuPac 125B rail cask is loaded in the TMI-2 truck bay, with the canisters brought to the rail cask in lead-shielded transfer equipment.

The cask loading procedure begins after the overpacks are removed from the cask. The railcar and cask are positioned under a cask unloading station in the truck bay. Screw jacks on the cask unloading station are used to lift the cask and the transport skid from the railcar. The railcar is moved out of the truck bay, the cask and skid lowered to the floor, and the truck bay door closed. The cask unloading station is then moved and stored out of the way.

Two hydraulic cylinders are attached to the cask to raise it from a horizontal laydown position to a vertical position. The cask is locked in place by attachment to a support tower. A work platform is bolted around the cask and connected to the tower. The cask is opened by removing the lids of the outer and inner containment vessels, and a shielded loading collar is installed. A mini-hot cell is moved over the cask and collar to remove and hold a shield plug from one of the seven tubes in the cask.

A canister is transferred from the spent fuel storage pool by the fuel transfer cask and lowered into the shipping cask. The canister transfer process is repeated six more times. Radiation exposure to workers is controlled by the lead shielding that is built into the mini-hot cell, fuel transfer cask, and loading collar.

After canister loading is finished and the mini-hot cell and loading collar are removed, both the inner and outer vessel lids of the cask are replaced and independently leak-tested to ensure that the cask is assembled correctly. The cask is then lowered to a horizontal position, placed on the railcar, reassembled with overpacks, and inspected and surveyed for radiation levels before being moved to the TMI north gate for transport by the railroad carrier.☐

Rail Transportation Program Developed for Cask

In conjunction with the development of the NuPac 125B rail cask and railcar, a transportation program was formulated to ensure the safety of the public while the cask and railcar are in transit to Idaho. The Union Pacific Railroad is the only railroad which serves INEL and was requested by EG&G Idaho to publish a rate for TMI-2 fuel debris traffic from TMI-2 to INEL. The Union Pacific Railroad in turn contacted Conrail, (the railroad that serves the TMI site) as well as other potential connecting carriers serving the northeast United States. EG&G Idaho and DOE are reviewing the potential routes to ensure that they are appropriate in terms of track safety and service requirements.

The railroads being considered are hazardous-material carriers that consistently earn railroad industry recognition for safety of operations and maintenance of track. Evaluation of the routes proposed by the railroads will include various factors such as the highest quality track available, which results in the shortest possible schedule using regularly scheduled railroad service. The routes ultimately selected will be through relatively low populated areas where possible. These requirements will result in a route with connections and tracks that have a low accident frequency index and a minimum number of switching stations.

The casks will ride on new railcars, each with 8 axles and a load capacity of 150 tons. A special design consideration for the rail cars was a safety margin such that the rated capacity of the railcar comfortably exceeded the loaded weight of the cask.

Railroad personnel will maintain continuous contact and use surveillance controls during transport. The railroads have the responsibility for handling any incidents that may occur during shipping and have established emergency procedures and trained personnel to handle hazardous shipments.

In the unlikely event of an accident during shipment, the railroad would take the initial action of isolating the train. Based on the severity of the accident, a nationwide emergency response system could be mobilized if necessary. Because of the safety designs built into the TMI fuel shipping casks, it is highly unlikely that, even in a rail accident, a breach of container integrity would occur.

Should an emergency occur, the DOE has established eight regional offices to provide radiological assistance. Any of these offices can mobilize an emergency response team within two hours; the team can arrive at an accident scene within eight hours. Nationwide, 28 DOE radiological assistance teams are available. The number of personnel responding and type of equipment assigned would depend on the nature of the emergency.

The total shipment time from TMI to Idaho is expected to be less than two weeks. With more than 250 canisters expected to be used and 7 canisters per cask, 35 to 40 shipments are planned. While one cask is being loaded at TMI, another will be being unloaded at the INEL.

Shipments are expected to begin in mid-1986 and should be completed in two to three years. Before actual shipments begin, the designated governor's representative in each state through which the shipments pass will have received a notice of the pending shipping campaign. DOE, which is responsible for shipping the TMI-2 fuel debris, will continuously monitor all aspects of the fuel shipping program.☐

Core Debris to be Stored at INEL; Researchers to Have Access

On arrival at the INEL, the rail cask is removed from the railcar and transferred to a truck transporter for the 30-mile trip to the research and storage facility Hot Shop at Test Area North. Inside the Hot Shop, operations for unloading the canisters from the cask are done remotely.

Each canister is withdrawn from the cask, taken to a pool of water, and lowered into a storage module. Each module holds up to six canisters. When a storage module is full, each canister is vented with a specially designed venting and gas sampling system before being filled with demineralized water.

The modules are moved to storage locations in the pool and placed together, but not interconnected. After each module is in place, a gas venting line is connected to each canister. These fuel storage modules were designed to be stable and subcritical under all potential accident conditions.

Storage of the TMI-2 core debris is planned for up to 30 years at INEL, a DOE-owned facility located 50 miles west of Idaho Falls, Idaho. At the INEL, researchers will have access to core debris for the core examination research and development program. Until now, they have had only small samples of the damaged core to examine. While progress in understanding the accident sequence at TMI has been made, scientists at the INEL and at other nuclear research facilities can develop the fullest possible understanding only by studying debris from many core locations. This stored material will offer them that opportunity. □

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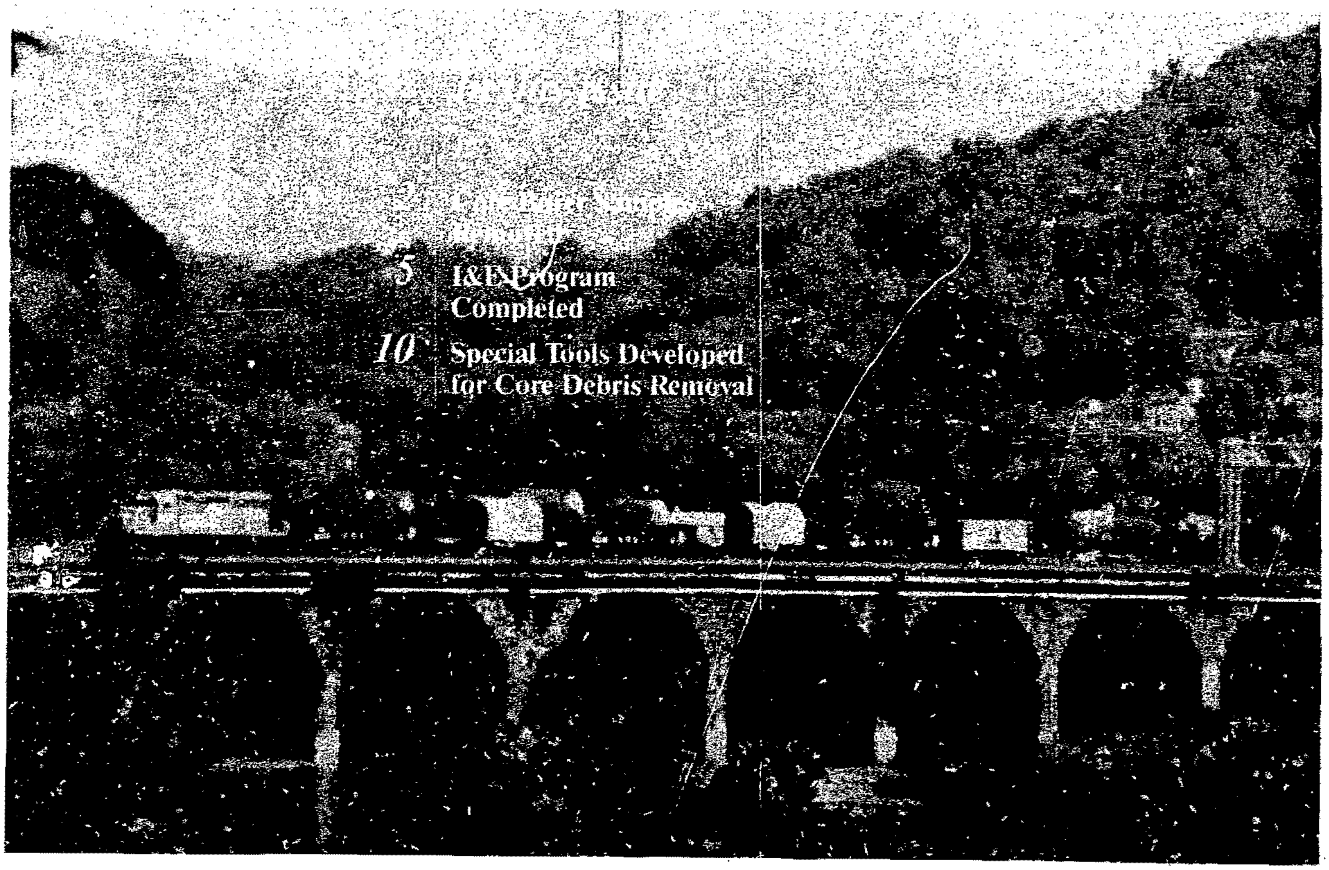
TMI Unit 2 Technical Information & Examination Program

UPDATE

Volume 7, Number 1 December 1986



Core Debris *Shipping* Program

Two shipments of TMI-2 core debris have been sent to INEL. The first shipment, with one shipping cask, *left TMI on July 20, 1986. The* second shipment, with two shipping casks, left TMI on *August 31, 1986* and included the *core bore samples. Both* shipments were uneventful. A third shipment is scheduled for mid-December.

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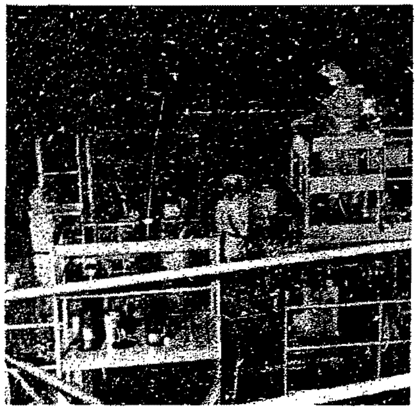
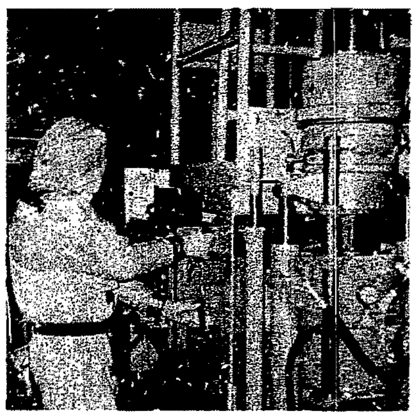
Core Borer Samples Removed



Three-man drilling crews . . . supervised by an EG&G technical advisor . . . operated 16 hours per day . . .



. . . operating the drill rig, monitoring its performance . . .

. . . repositioning equipment, and adding drill casing.

A significant milestone was reached at TMI-2 in July with the removal of stratified core debris samples from the reactor vessel. Once examinations and analyses are complete, the information gained from these core samples is expected to contribute to the resolution of several important research issues.

These issues include improving the definition of current core conditions, advancing the understanding of the accident scenario, and establishing location and distribution of retained fission products. In addition, information developed during core borer operations will significantly aid core debris removal (see box). This information was developed both from drilling data and from video inspections made through the bore holes.

The need for a thorough understanding of conditions inside the damaged reactor was recognized during the early stages of the TMI-2 program. More than simply identifying the end-state condition of the core, understanding the thermal, chemical, and mechanical processes that occurred during the accident was established as a priority concern. The release or retention of fission products by the core is at the center of severe accident predictions and related licensing issues, and was recognized as an important topic for investigation at TMI-2.

Similarly, the events and conditions contributing to the relocation of core materials, as well as the timing of those events, make up the major data points for reconstructing the accident sequence. Vital to all these considerations is the ability to acquire meaningful core samples.

The research community requires physical samples representing the spatial extent of damage or degradation. With analysis, the samples must be capable of providing data to characterize the variations in postaccident core materials present, as well as represent as-built variations in fuel assembly types and locations.

To be meaningful, the samples had to be traceable within the three-dimensional geometry of the core. Similarly, those responsible for defueling plans needed data on the type and distribution of altered core materials, both in the normal core space and in the regions within the lower core support structures. The latter information, to be useful, had to be available shortly after drilling. An overriding consideration was to minimize delays to the plant recovery and defueling operations.

The Core Stratification Sampling (CSS) Project, referred to as the "core borer" project, was developed as a coherent approach to the complex task of in-core sample acquisition. Starting with equipment and technology currently available in the mining/geology industry, the system was extensively modified to meet the special operating and environmental requirements of the TMI-2 Reactor Building.

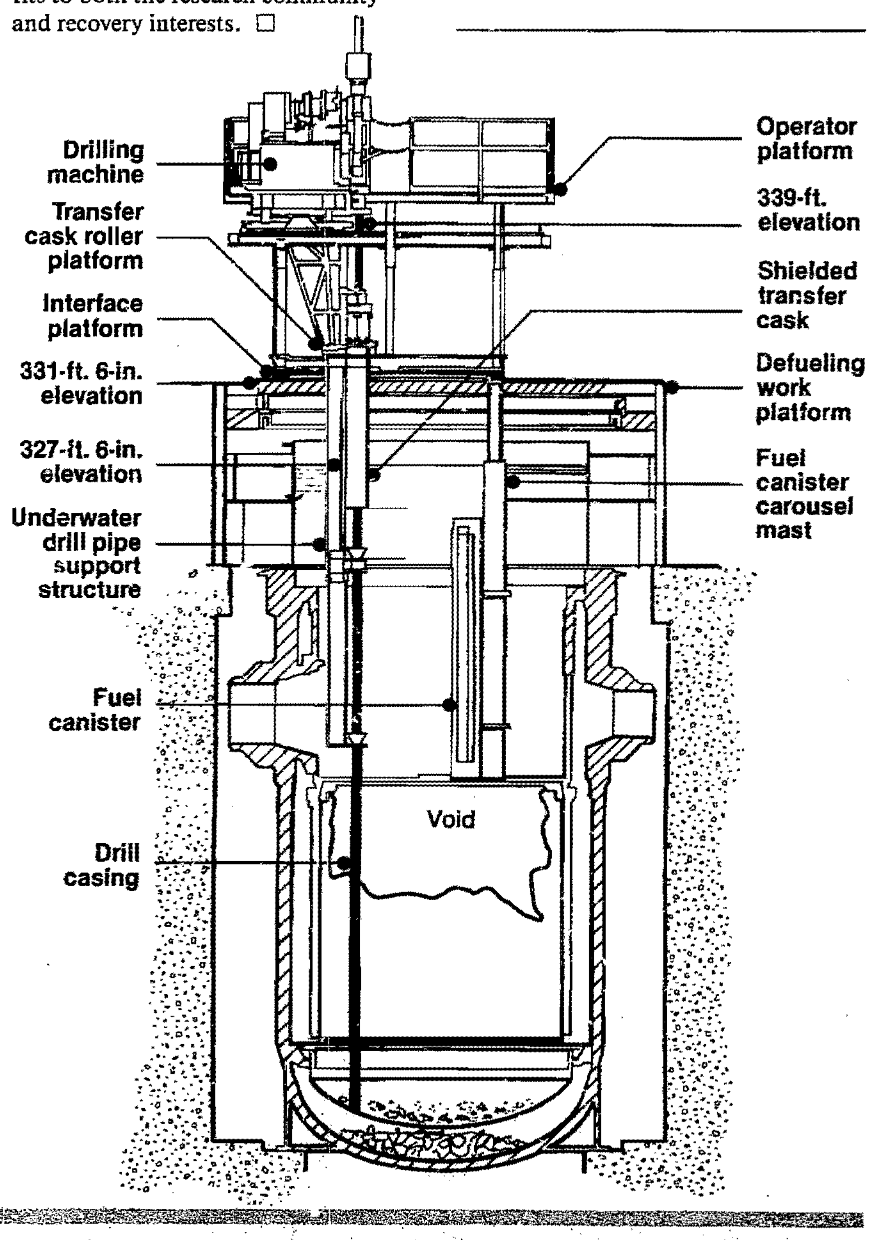
The drill unit was modified to provide precision positioning over the reactor vessel, to incorporate a microprocessor for operational control and safety interlocks, to record drilling parameters (torque, load, etc.), and to provide relevant plant protection functions. For the most part, the sample-cutting hardware was derived directly from the mining industry, with the drill bit the only major departure from standard, off-the-shelf equipment. The bit carried special teeth of diamond-faced tungsten carbide, the only configuration found to tolerate the combination of hard, ceramic-like materials as well as the ductile metallics encountered during the sampling operations.

Ten core samples were removed from the reactor vessel and loaded into five shipping canisters for shipment to the INEL. Once at the INEL, the sample materials were removed from the canisters and prepared for distribution to several laboratories where extensive examinations will begin.

Current examination plans include participation by both foreign and domestic laboratories, including facilities in Japan, Canada, and up to six European countries. The examination and analysis activities are expected to take more than two years to complete.

The unqualified success of the sample acquisition project is the direct result of a strong cooperative effort between GPU Nuclear (GPUN) and EG&G Idaho, Inc., with direct benefits to both the research community and recovery interests. □

Core boring machine for TMI-2 reactor.



Core Bore Findings Support Defueling

Drilling data and video inspections through the bore holes provided significant new information to support defueling operations. Among the findings were:

- The amount of force required to drill through the core indicates the core material, while containing a significant quantity of resolidified material, is not as hard as once thought.
- The normal core region contains loose debris, resolidified material, and apparently intact remnants of fuel assemblies, as expected.
- Damage to reactor components below the core region appears to be less than expected. Some minor damage was found on the eastern side.
- Less debris was found in reactor core-support components than expected.
- Most debris in the bottom of the reactor vessel appears loose enough to be removed with vacuuming equipment.
- During the 1979 accident, the bottom 2 to 3 1/2 feet of the core remained covered with water.

As a result of these findings, defueling planners are reviewing tooling requirements. To date, approximately 25 of the estimated 150 tons of core debris have been removed. □

Instrumentation and Electrical Program Completed

The TMI-2 Instrumentation and Electrical (I&E) Program, begun in 1980, was completed in June 1986. The program was designed to take advantage of the unique opportunity offered by the TMI-2 accident to evaluate a variety of instrumentation and electrical equipment for the effects of exposure to accident conditions including steam, spray, and radiation, as well as hydrogen burn and the resultant overpressure.

The examination of this equipment over a period of several years also provided information on long-term exposure to moisture. Findings of the TMI-2 I&E Program support the general conclusion that the plant instrumentation and electrical components performed well with respect to their required functions under accident conditions.

The TMI-2 I&E Program also identified and analyzed a number of installation problems and instrument response characteristics that led to misleading information and equipment failures. These problems included faulty seals and inadequate drains and vents to protect enclosed equipment against moisture, anomalous responses of radiation monitors, and substantial corrosion of electrical contacts over a period of a few years.

The equipment involved included the radiation monitors from which it has not been possible to determine the true radiation profile within the Reactor Building; pressure transmitters that failed because of moisture intrusion; the loose parts monitors that degraded and then failed due to the sensitivity of the electronics to radiation; various switches and contacts that are continuing to fail due to corrosion; solenoid operators for valves that trapped moisture within the assembly; and various other devices that suffered from moisture intrusion.

In addition to analysis of active equipment, cables and connectors have been carefully analyzed. Some 750 circuits were tested using the newly developed ECCAD system (see box). In addition, cables, or sections of cables, were removed from the Reactor Building for in-depth laboratory analyses.

Major Findings

Two major findings have emerged from the program: (1) more attention must be given to the prevention of moisture intrusion during the design, construction, operation, and maintenance of nuclear power plants, and (2) while basic engineering designs of electronic devices are generally adequate, applications engineering and specifications should be improved. These two findings are closely related.

Moisture Instrusion—The major cause of I&E equipment failure was moisture intrusion, generally caused by inadequate seals on housings, conduits, fittings, and connectors. Where seal integrity was maintained at the cable entry into the equipment housing, the internals were generally not corroded and the device was operable.

For example, seven pressure transmitters were removed from the Reactor Building for evaluation at the INEL. All had been located above flood level in the Reactor Building and were exposed to approximately the same environment. Three of the pressure transmitters were made by manufacturer A and four by manufacturer B. All of the A units survived the accident and postaccident; one of the B units survived the accident and postaccident, and another B unit survived the accident and one year of postaccident before failing.

ECCAD System Description

The Electrical Circuit Characterization and Diagnostic (ECCAD) system, developed under the TMI-2 I&E Program, can make a significant contribution to predictive maintenance for electrical circuits. The ECCAD system is a computer-controlled measurement system designed to characterize electrical circuit parameters that might impact the ability of a circuit to perform its function. For example, if the circuit energizes a motor for a motor-operated valve, the ECCAD system can determine if all connections or contacts are good, if proper voltage can be applied to operate the motor, and if the motor is electrically functional.

The system functions by measuring basic electrical parameters and by sending an electromagnetic pulse through a circuit. By analyzing the reflected pulse and related electrical data, the condition of the circuit can be determined and exact locations of circuit abnormalities can be established. Further, this information is stored in the computer and can be compared with data taken earlier or later to determine if circuit deterioration is taking place.

The ECCAD system is composed of standard electronic test equipment that is readily available on the commercial market. The system is controlled by a Compaq personal computer. The computer:

- Controls individual instruments, setting critical values
- Performs a self-test on the instruments
- Sequences the instruments
- Formats the data, ensuring a standard data set of high quality.

For additional information on ECCAD, refer to *Update*, Vol. 5, No. 3, August 1985, or contact:

DOE Site Office
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Middletown, PA 17057 □

Laboratory analysis showed that all of the failures resulted from moisture intrusion. Those units that survived had either an adequate internal seal (manufacturer A) or a properly installed conduit and junction box (one of manufacturer B's units). A proper installation specification, calling for sealing the unit (as was done by manufacturer A), or for a junction box with breather, drain, and correct conduit entry would have precluded moisture intrusion and extended the life of the equipment.

Other Findings—While moisture intrusion was the major cause of equipment failure, other significant findings were made.

- ***Dome Monitor***

The Reactor Building dome radiation monitor, with shielded ion chambers and electronics, was the only radiation monitor inside the Reactor Building with the capability to measure and indicate LOCA-level radiation. This monitor was the subject of extensive postaccident examination in efforts to understand the monitor response and to determine radiation levels inside the Reactor Building during the accident.

The dome monitor design shows that insufficient consideration had been given the fact that the energy content of the radiation changes with time during the course of an accident. By not requiring a flat gamma energy response under all radiation conditions, radiation measurements were inaccurate. Also, the electronics (specifically the MOS transistors) were significantly degraded by radiation exposure. Specifying and testing the dome monitor design for postaccident radiation dose levels could have led to improved performance of this equipment.

- ***Area Radiation Monitors***

Three radiation monitors were selected for early removal in an attempt to establish an improved knowledge of radiation levels during the accident. All three were located in the Reactor Building and were exposed to the accident and postaccident environment. All three monitors were of the Geiger-Mueller (GM) tube type, with an accompanying electronics package which fed square waves (one for each GM pulse) to an electronics package mounted outside the Reactor Building.

One ARM provided an erroneous (low) indication of the high radiation levels. It was discovered that the area radiation monitor gave on-scale readings when it should have given high, off-scale readings. The device did have a fail-safe circuit that was supposed to ensure high, off-scale readings for high input radiation levels. However, in the presence of the accident radiation (estimated to be between 2.5×10^5 Rads and 1×10^6 Rads), the circuit did not work. Failure to require proof of performance at high radiation levels resulted in misleading information that could have hampered accident mitigation activities.

- ***Loose Parts Monitor Charge Converters***

Charge converters associated with the loose parts monitoring system were found to have failed due to radiation sensitivity of semiconductors. This failure occurred in the first few days of the accident when the system was being monitored very closely to detect loose parts moving through the systems and to assess core damage.

This type of failure would mask or distort real loose parts signals. The studies at TMI-2 led to the determination that similar failures were occurring during normal operating conditions at another operating nuclear plant. This problem was subsequently corrected through redesign by the manufacturer.

The specification of a required radiation operating level and total radiation dose for this equipment could have led to the use of an alternate design or installation at a location with a lower radiation environment.

- ***Solenoid Valves***

  Two Class 1-E solenoid valves were removed from the Reactor Building air cooling and purge system. Both solenoids were operational except that one limit switch failed to respond to the valve position. One valve shell was rusted from moisture that had entered the solenoid housing, due to a flaw in the configuration of the conduit installation. The limit switch failure was moisture related and the lead wire insulation to both valves had embrittled. The long-term integrity of these valves could have been improved by ensuring protection against moisture intrusion as well as by specifying the use of materials that would not prematurely age and embrittle from heat or radiation.

These examples, typical of the equipment problems found during the TMI-2 I&E Program, led to the following general conclusions:

- Moisture intrusion is the major cause of equipment failure and, as such, must be considered in specifications, equipment designs, and installation and maintenance procedures.
- Applications engineering should be performed on a wider range of equipment, not just safety-related equipment. Analysis should include abnormal (e.g., LOCA) operating conditions and should address information needs for accident mitigation activities.
- Qualifications testing should include normal and abnormal radiation environments when it is vital that equipment continue to operate in such adverse environments.
- Predictive maintenance should be encouraged to avoid unnecessary interruption of electrical circuits for maintenance purposes. NRC studies show that 35% of electrical failures are maintenance-induced. The use of diagnostic or trending systems (such as an ECCAD system) would allow maintenance to be performed only where needed.

Further information on the TMI-2 I&E Program is available in the following reports. Copies of these reports are available from:

TMI-2 Technical Information and
Examination Program
P. O. Box 88
Middletown, PA 17057

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Special Tools Developed for Core Debris Removal

Table 1. **Defueling activity and related tooling.**

Defueling Activity	Tooling
Core debris sizing and reactor internals disassembly	Shears, shredder, impact chisel, cutting station, abrasive saw, brushes, abrasive water jet, cavitating water jet, plasma arc torch, incore instrument cutter, core boring machine, and cutoff saw.
"Pick and place"	Top access partial fuel assembly removal tool, scoops, hooks, tongs, grippers, tampers, sweepers, debris container handling tools, cranes, and handling bridges.
Fines/debris vacuuming	An integral fines/debris vacuuming system with specialized capturing canisters and an assortment of vacuum nozzles.
Tooling support equipment	Work platforms and support structures, control systems, cable management system, closed-circuit television viewing and lighting system, robotic arm manipulator, tooling positioners and stabilizers, debris canisters and buckets, and canister positioning system.

A number of special tools have been developed to meet the unique challenge of removing TMI-2 core debris. They are being used inside the reactor vessel, underwater, in a radioactive environment, and are operated from up to 35 feet away.

The current tooling inventory represents the culmination of several years of intensive technical planning. The overall philosophy calls for the simplest, least-developmental tools and techniques. Tooling is permitted to become more complex and developmental only as dictated by proof-of-principle testing, operational experience, and increasing knowledge of core conditions.

In late 1982, GPUN and their subcontractors, with funding support from DOE/EG&G Idaho, Inc., started the reactor vessel defueling tooling development effort. The thrust of this effort was to provide a tooling system capable of removing approximately 100 tons of uranium dioxide fuel and 50 tons of core components from the TMI-2 reactor vessel. The initial fuel and core debris removal tooling was delivered to TMI in time for the first phase of reactor vessel defueling, starting in October 1985. (Reactor vessel defueling operations are expected to be completed by December 1987). This tooling, and the defueling tooling that will follow, provides the means to prepare the reactor vessel core material and to place it in specially designed debris canisters. These canisters will be placed in temporary storage at the Idaho National Engineering Laboratory, with DOE having responsibility for their ultimate disposal. (See item on shipping program on page 1 of this issue of *Update.*)

Tooling requirements are based on four phases of reactor vessel defueling as follows:

- Initial defueling—removal of fuel element end fittings and other loose debris, including vacuumable fines, from the rubble bed.
- Core region defueling—removal of debris remaining after the completion of initial defueling in the core region. This phase is differentiated from initial defueling in that significant debris sizing operations will be performed. It is also intended that the removal of the once-molten, "hard crust" will be accomplished during this phase.
- Lower head defueling—removal of debris from the lower reactor vessel head. The lower head includes the volume directly below the flow distributor.
- Core support assembly (CSA) defueling—removal of debris from the core support assembly. The CSA consists of bolted, stainless steel subassemblies including the core support shield, core barrel, thermal shield, lower grid, incore instrument guide tubes, and flow distributor.

In addition to uranium dioxide fuel, the core material consists of fuel rods, end fittings, control rod material, spacer grids, fuel cladding, instrument strings, control rod spiders, and neutron poison materials.

All the defueling tooling is designed for remote operation, underwater in the reactor vessel, and is controlled at or near the main work platform located over the reactor vessel. While several tools are hoist mounted and manually operated, most of the tooling is hydraulically operated. The tool "end effectors," which represent the mechanical devices performing the work, are designed for mounting on poles and tool positioners up to 35 feet long. The accompanying table lists the tools.

The main work platform, on which most of the defueling tooling is staged and operated, is located above the reactor vessel flange. The work platform is shielded and can be rotated. It is equipped with ports and slots covered with removable hatches. These openings permit workers to use defueling tooling and support equipment inside the reactor vessel, while minimizing radiation exposure.

Before being placed into service, the tooling and support equipment are functionally tested to ensure that they will interface as designed and perform as intended. Functional testing is normally performed at the manufacturer's facility or on-island at the defueling test assembly reactor vessel mockup. GPUN is currently reviewing plans for the design, fabrication, and testing of CSA and lower head cutting tools and equipment. This tooling will complete the reactor vessel defueling tooling requirements. Recent reactor vessel core boring and associated video inspection results suggest that there is no reactor vessel core condition that the present and anticipated defueling tooling and support equipment inventory cannot accommodate.

During the past few years, robots have played an important role in the TMI-2 cleanup program, helping to reduce worker radiation exposure. To date, five different devices have been used to test or probe in high-radiation areas of the plant. Thus far, no remote-controlled device has been used inside the reactor vessel. As indicated in the table, a robotic arm has been purchased and is expected to be used in the vessel as a light-duty defueling operations manipulator.

The final development of this tooling will complete a major milestone leading to ultimate disposition of the TMI-2 plant. □

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