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

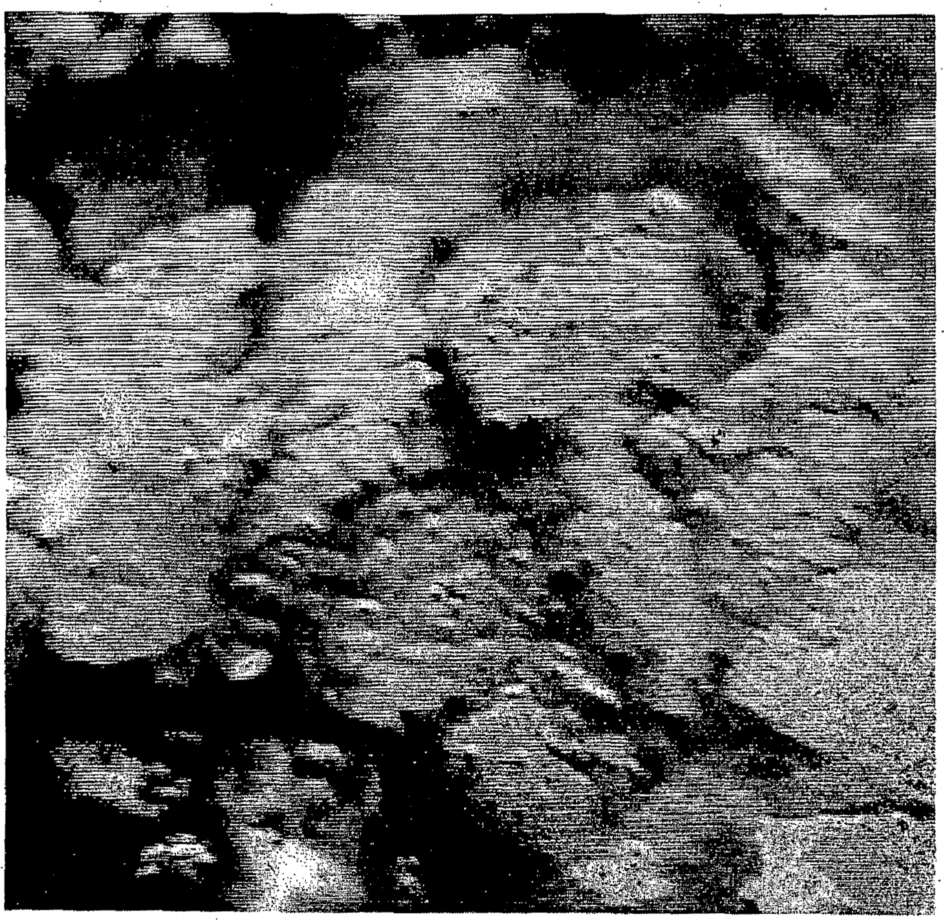

Figure 1 **Closed-circuit television closeup view of the rubble bed.**

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.

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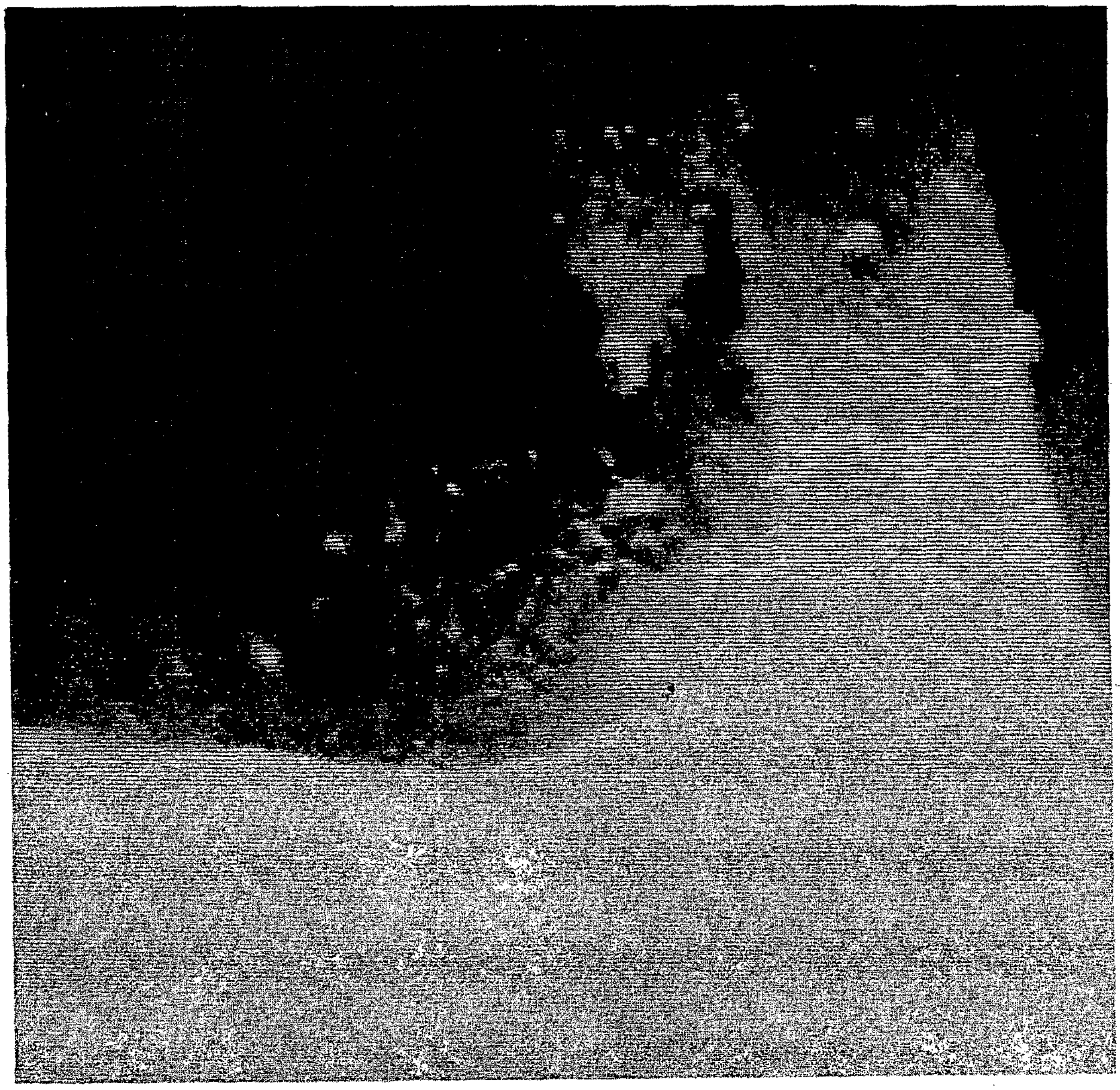


During the quick look, data samples of vented gas and reactor coolant liquid were obtained and are being analyzed. Altogether, the observations and samplings are providing early data on:

- 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.

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

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.

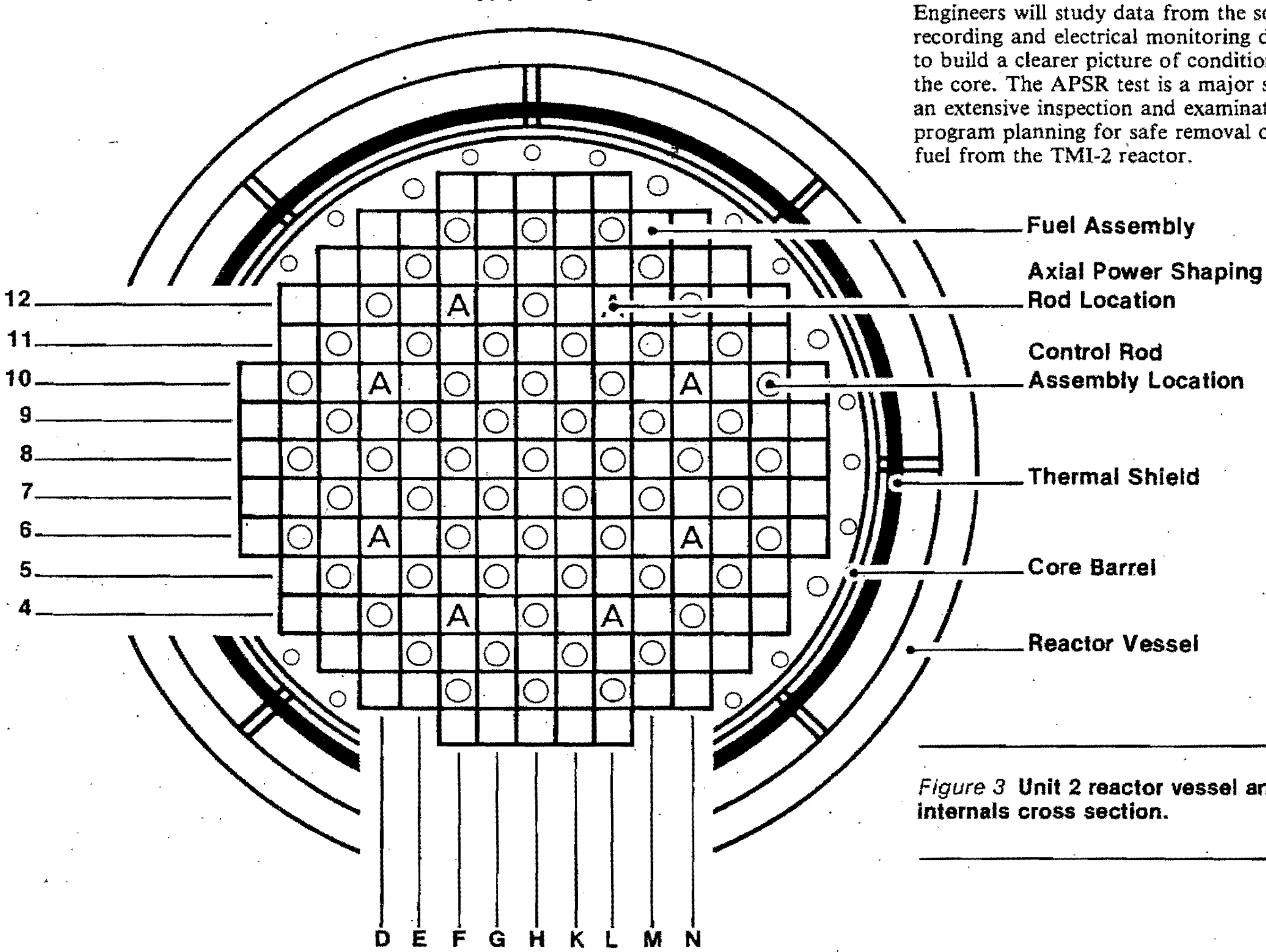
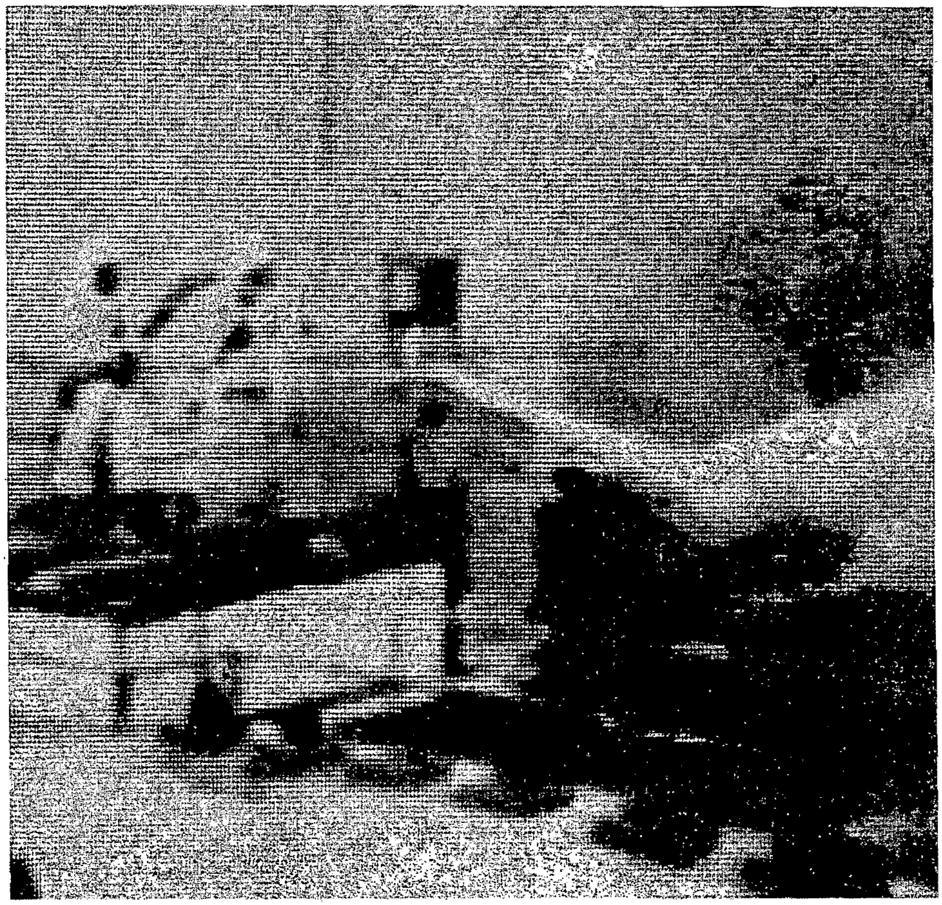


Figure 3 **Unit 2 reactor vessel and internals cross section.**

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*).



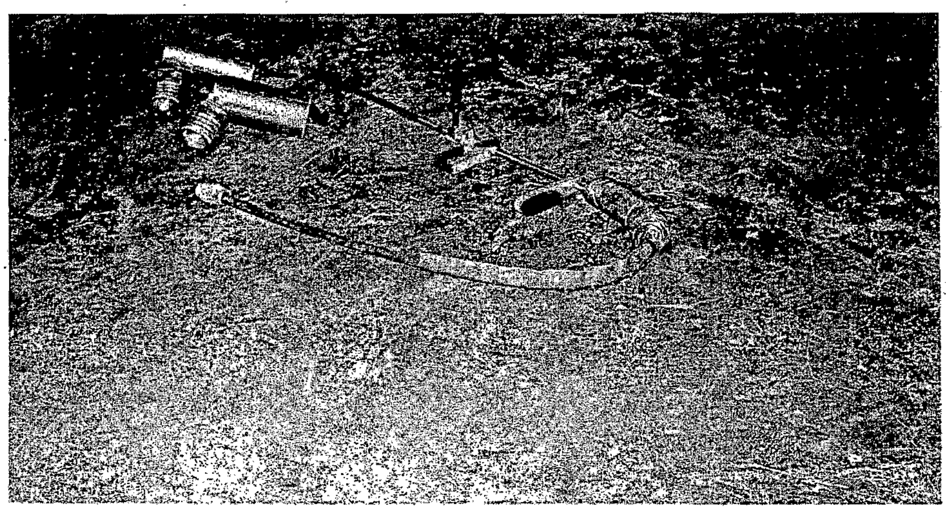


Figure 4 **Technicians hydrolasing the top of the D-rings.**

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

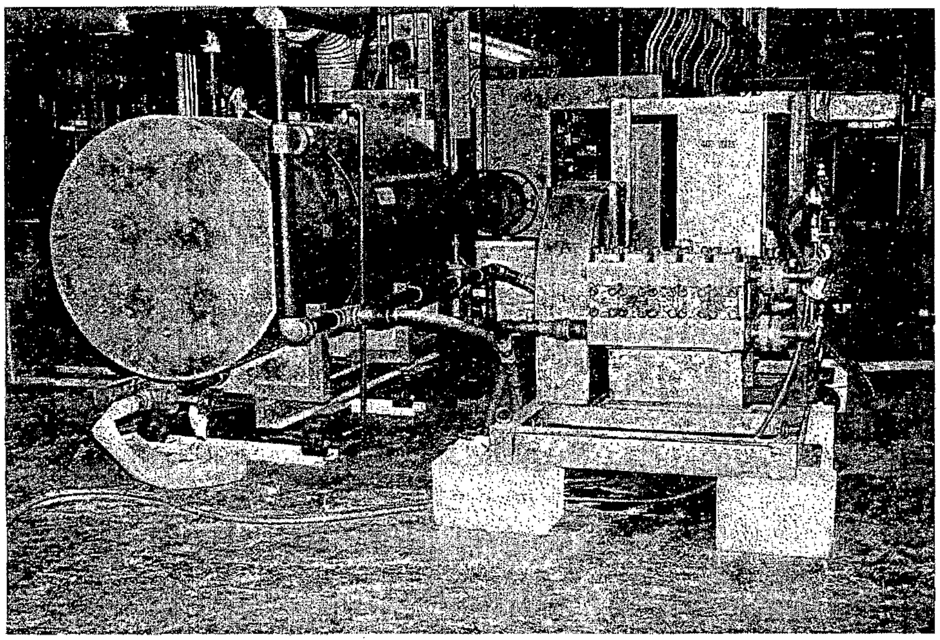


Figure 6 **Gross decontamination experiment hot water heater and high pressure pump.**

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.

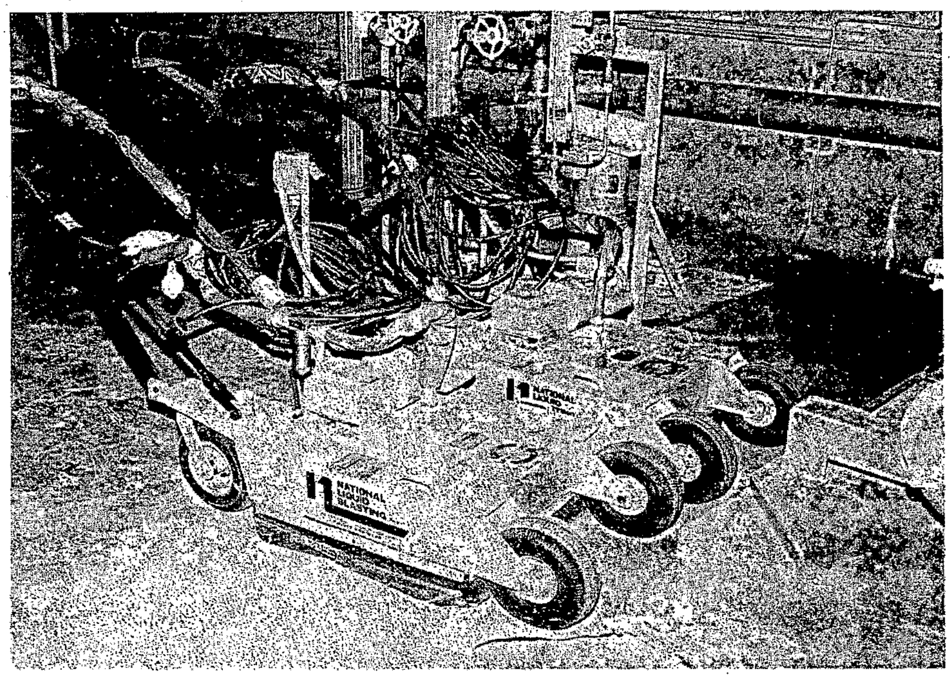


Figure 7 **Gross decontamination experiment spinjet sprayers.**

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.

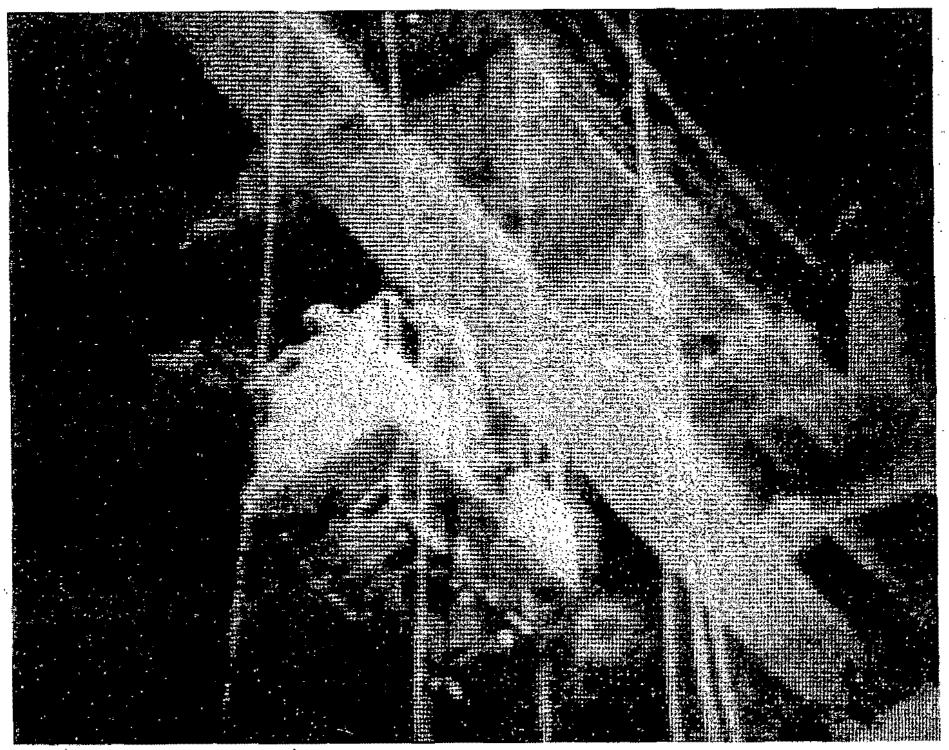


Figure 9 **Technicians ascending to the polar crane on the spider lift.**

Dose Levels Reduced in TMI-2 Reactor Building

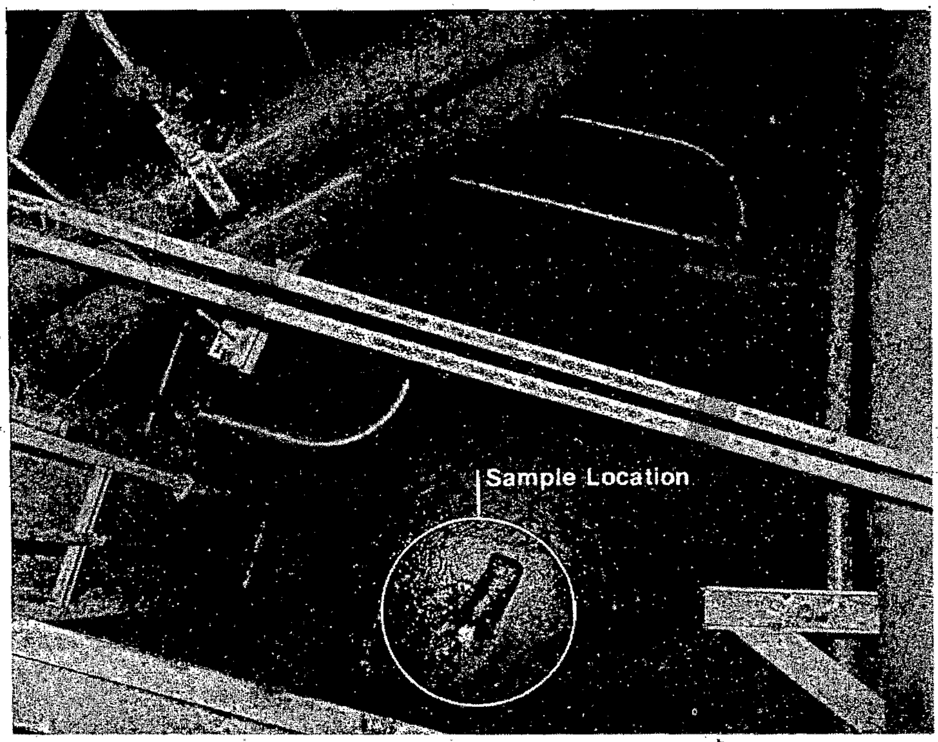


Figure 10 **Basement floor one level below 305-foot elevation after sample was taken.**

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**

	Ion-Exchange System	
Element	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.

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.

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

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.

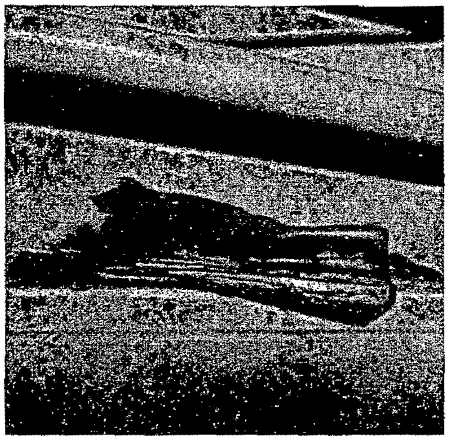


Figure 12 **Burned and crumpled maintenance manual on the fuel handling bridge.**

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.

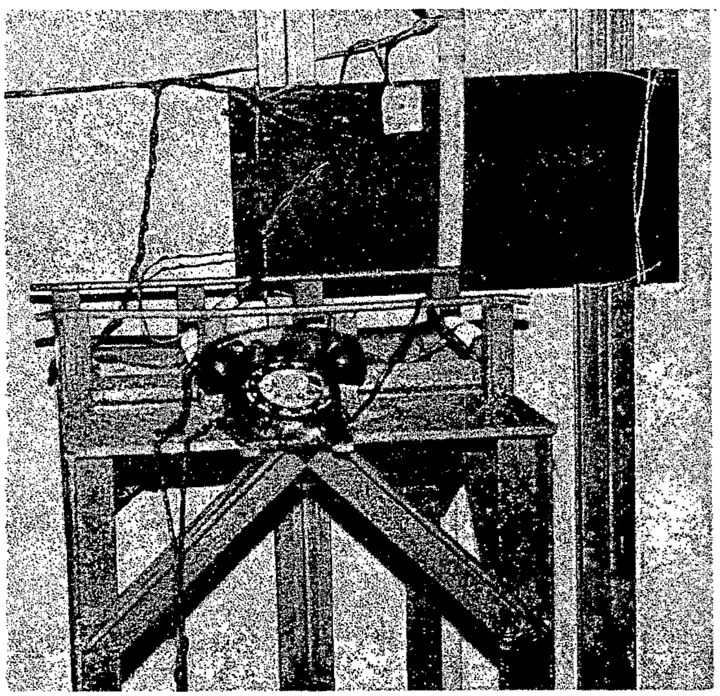


Figure 13 **Deformed telephone on the 347-foot elevation.**

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*.

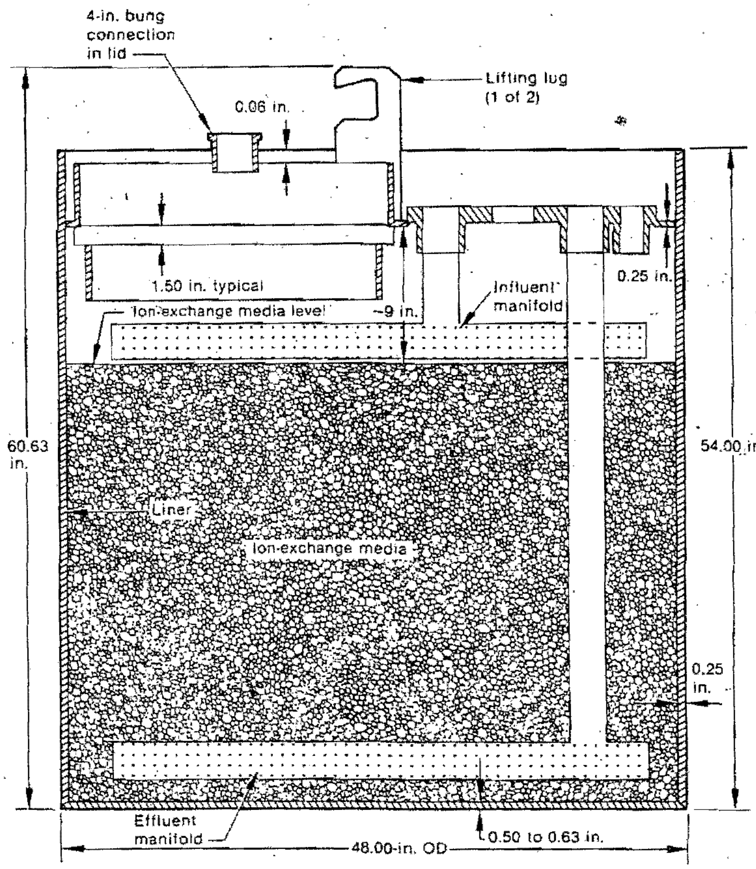


Figure 14 **Cutaway sketch of EPICOR II prefilter liner.**

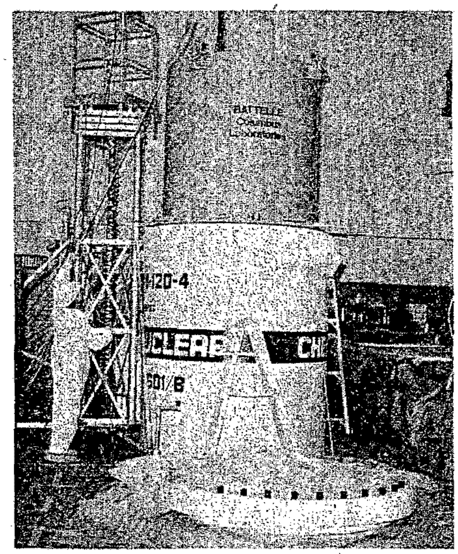


Figure 15 **BCL transfer and storage device atop the PF-16 shipping cask.**

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.

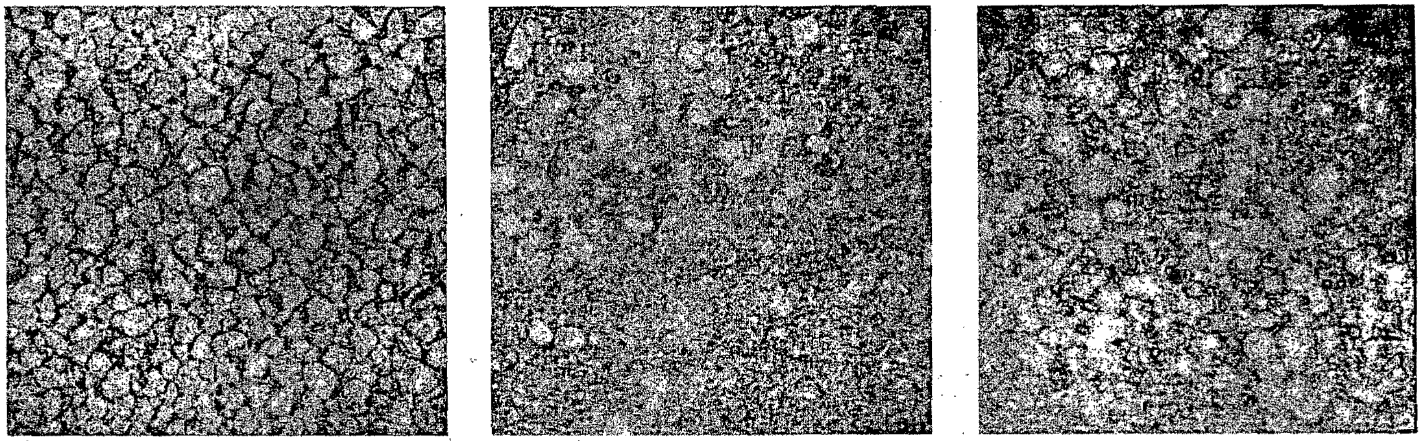


Figure 16 **Top, middle, and bottom regions of ion-exchange media core sample (30x).**

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 x 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 ± 0.01 x 10^{-2}
Gross alpha	5.9 ± 0.01 x 10^{-4}
Strontium 89/90	5.2 ± 0.1 x 10^{-4}
Antimony 125	7.94 ± 0.42 x 10^{-4}
Cesium 134	1.32 ± 0.02 x 10^{-3}
Cesium 137	1.308 ± 0.005 x 10^{-2}
Plutonium 238, 239, 240	< 1.0 x 10^{-4}
Uranium 238	< 1.0 x 10^{-4}

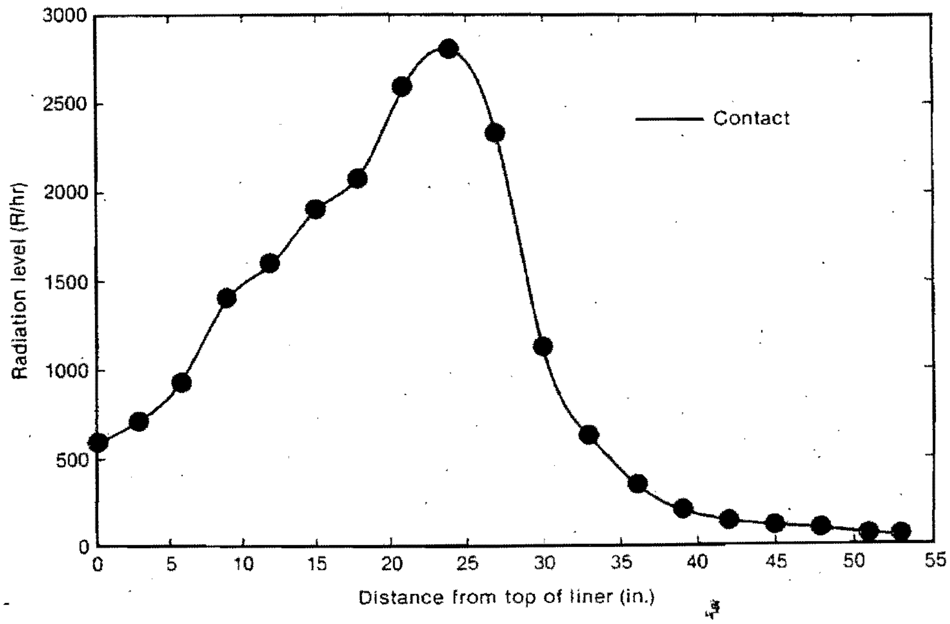
Scientists performed preliminary examination of the core sample using the hot cell stereo microscope at approximately 30x magnification while the sample remained encased in the Lucite tube. The three well-defined regions that were observed are shown in *Figure 16*. The top region, presumed to be an inorganic 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 x 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.*

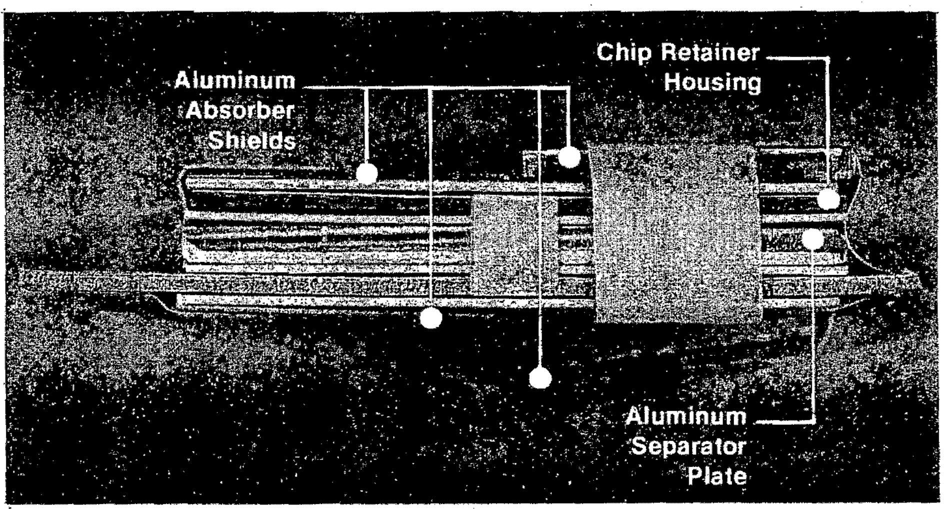
The characterization of PF-16 provided reseachers with valuable information about the behavior of highly loaded ion-exchange media and yielded baseline data for the development of safe handling, storage, and disposal techniques. One of the most important results of this characterization work is the fact 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.**



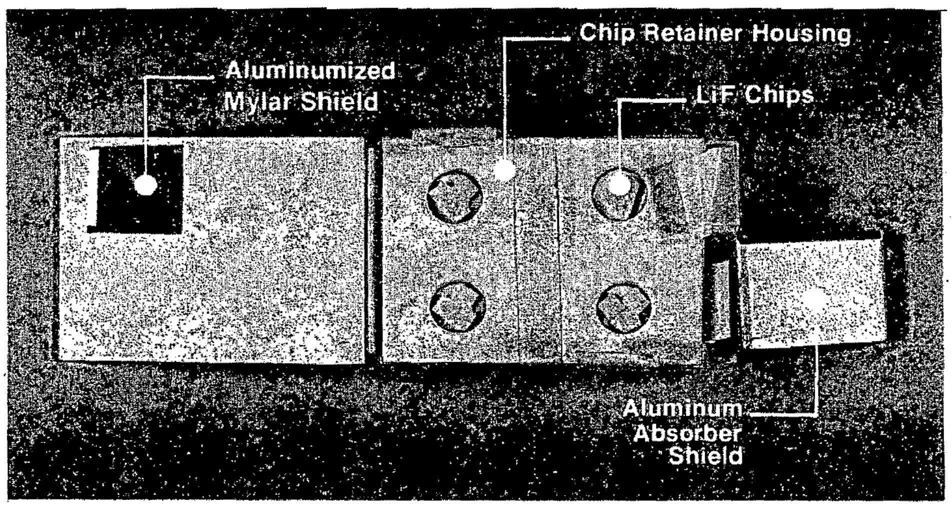


Figure 19 **TLD center section**

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.

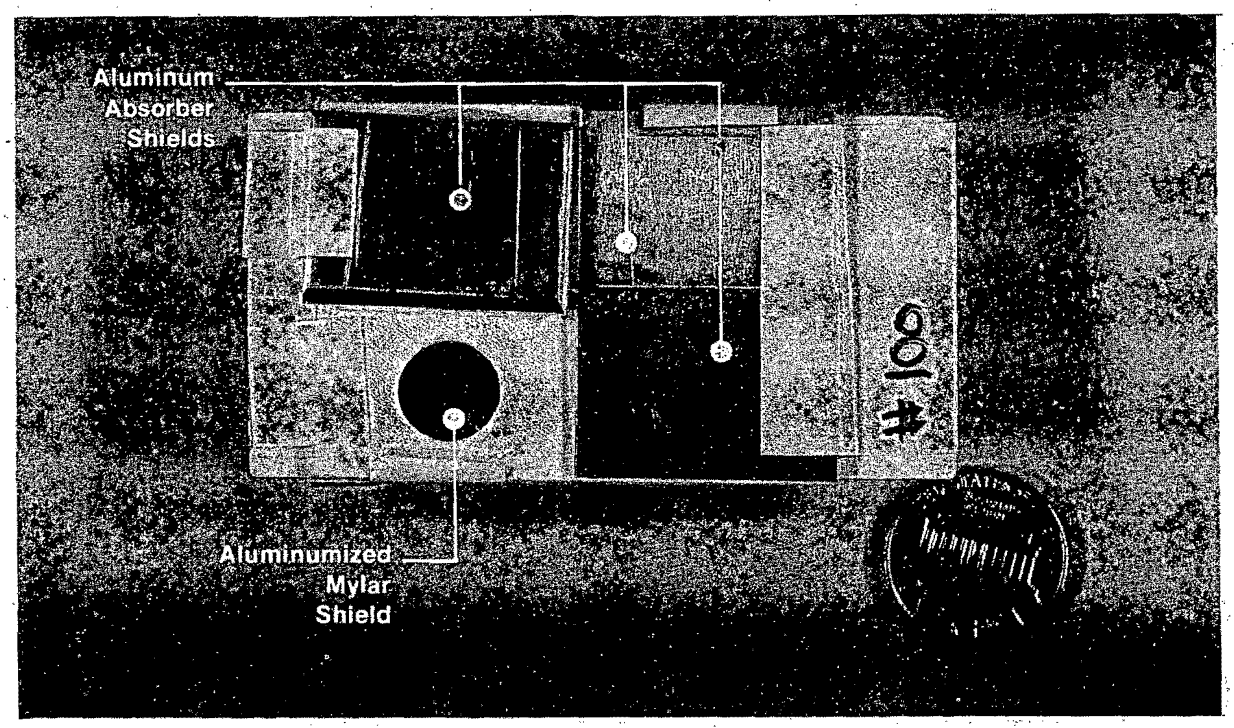


Figure 20 **TLD top view.**

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.

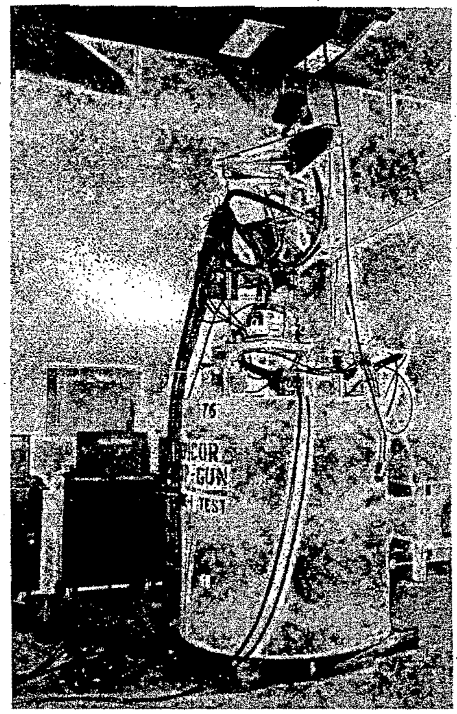
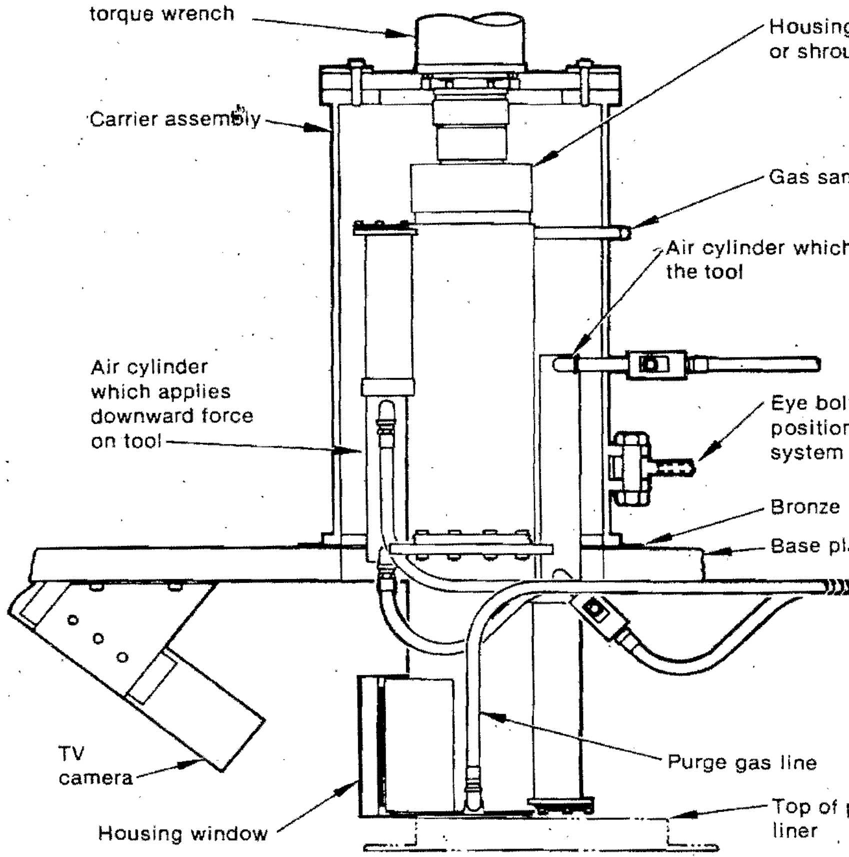


Figure 21 **Prototype gas sampler installed on EPICOR II liner.**

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.

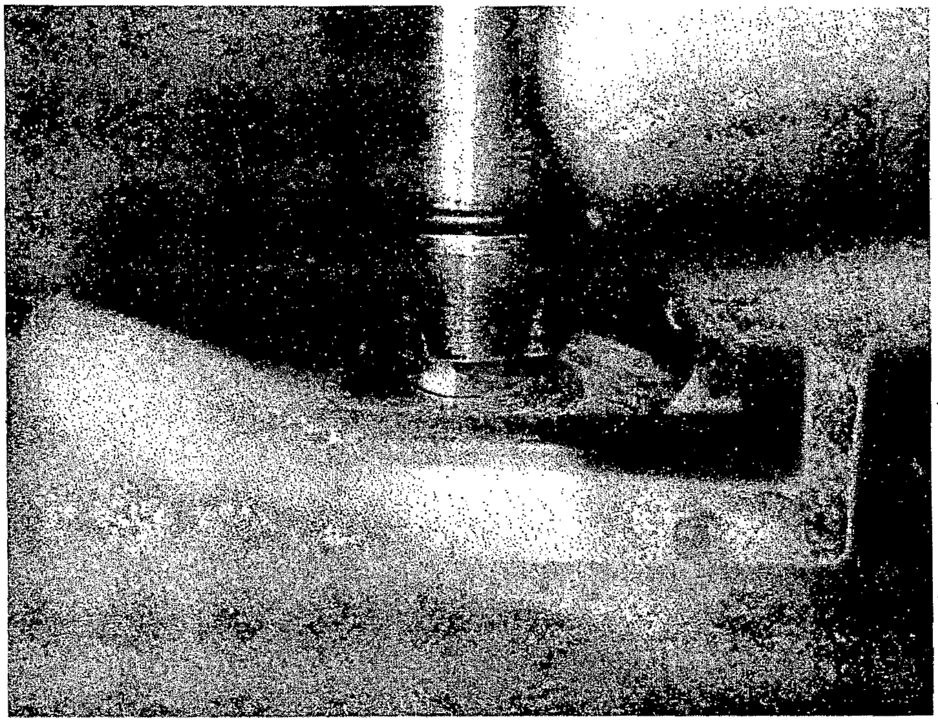


Figure 23 **View through shroud window of tool tip engaged in liner vent plug.**

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.

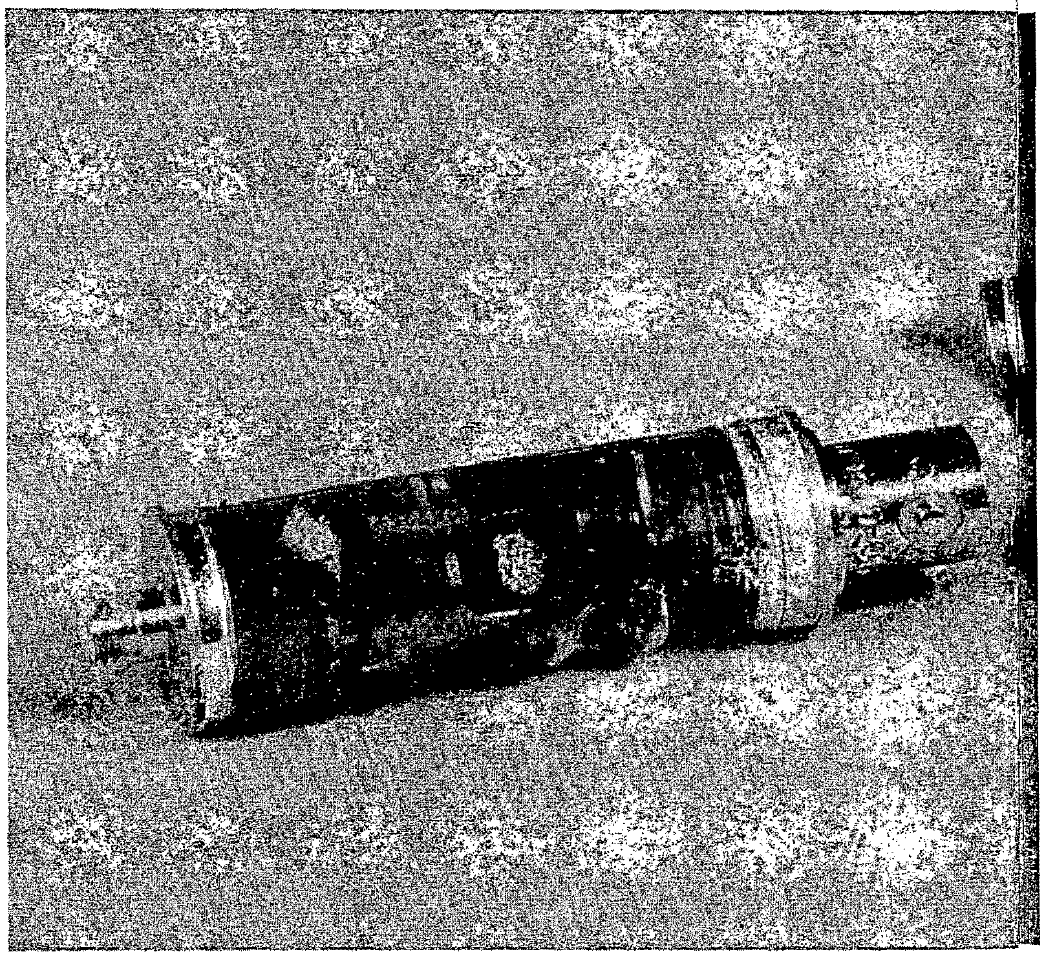
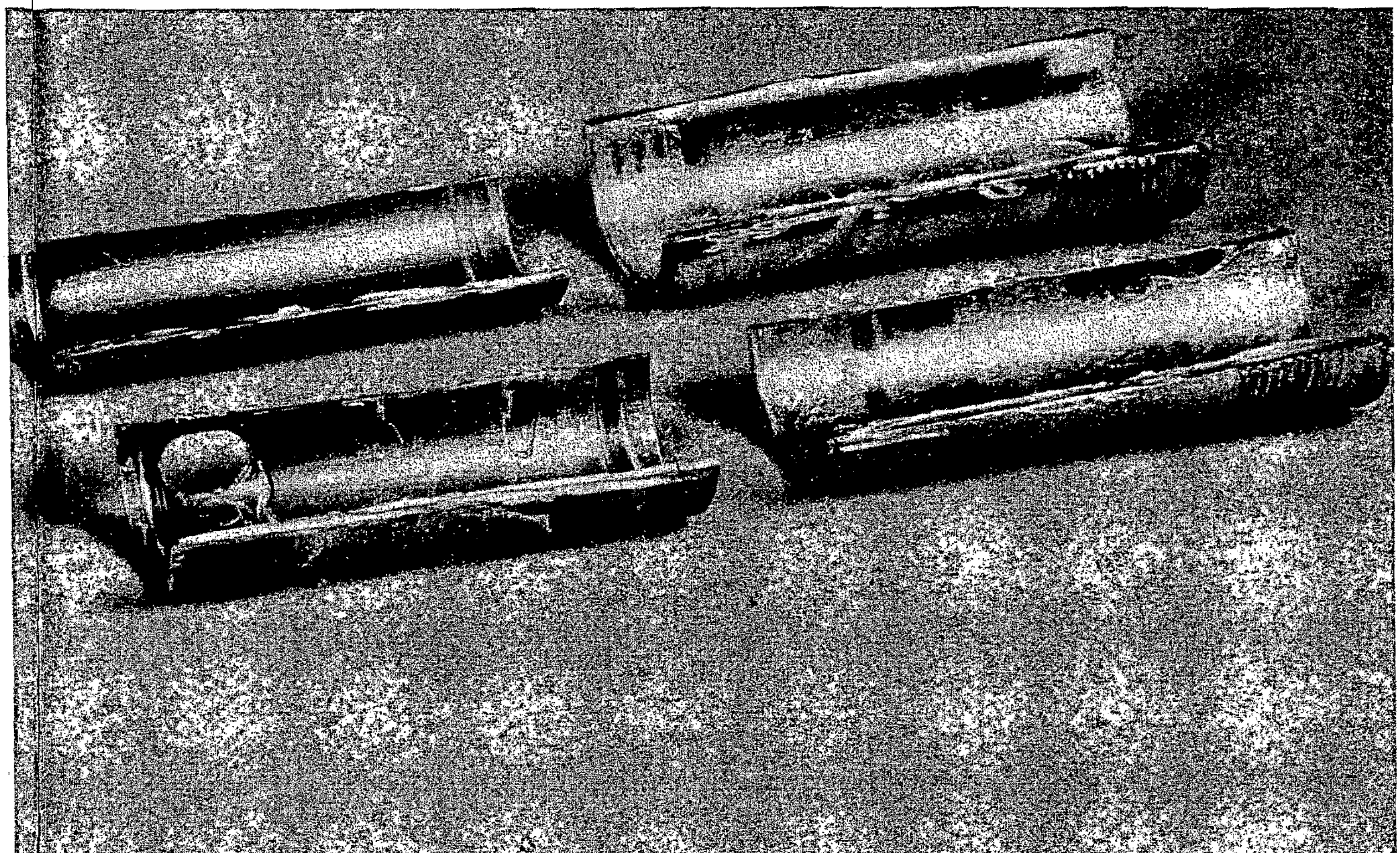


Figure 24 **Failed charge converter after disassembly.**



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

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.

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UPDATE

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