TMI Unit 2 Technical Information & Examination Program

UPDATE

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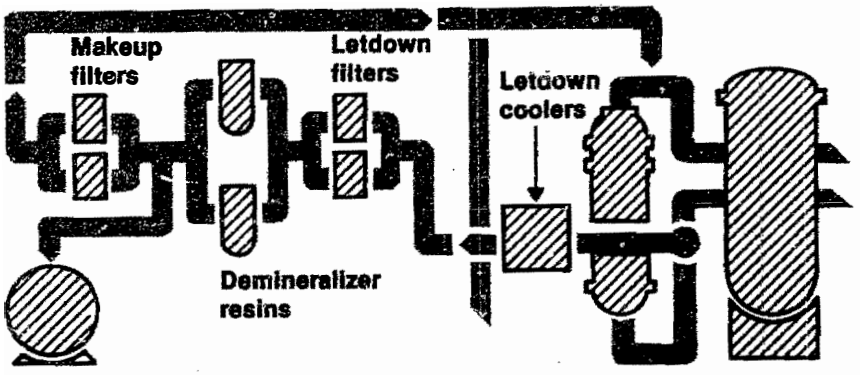
Resin Characterization Supports Waste Removal Efforts

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

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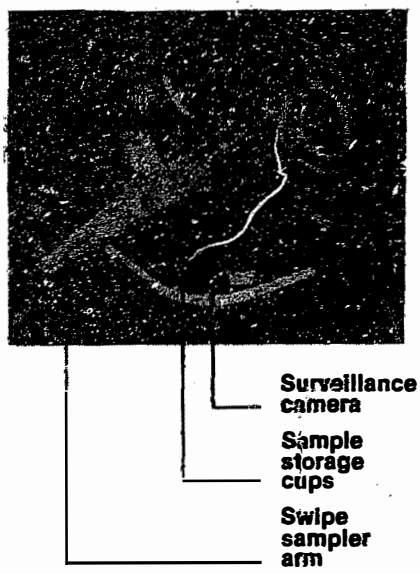


Figure 2. **Remotely operated transport vehicle developed for exploratory work in the demineralizer cubicles.**

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 ± 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 1.7×10^9 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.

	Demineralizer A		Demineralizer B	
Gas	**(8 psig)**	**(4 psig)**	**(8 psig)**	**(4 psig)**
^{85}Kr $\mu Ci/cm^3$	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**

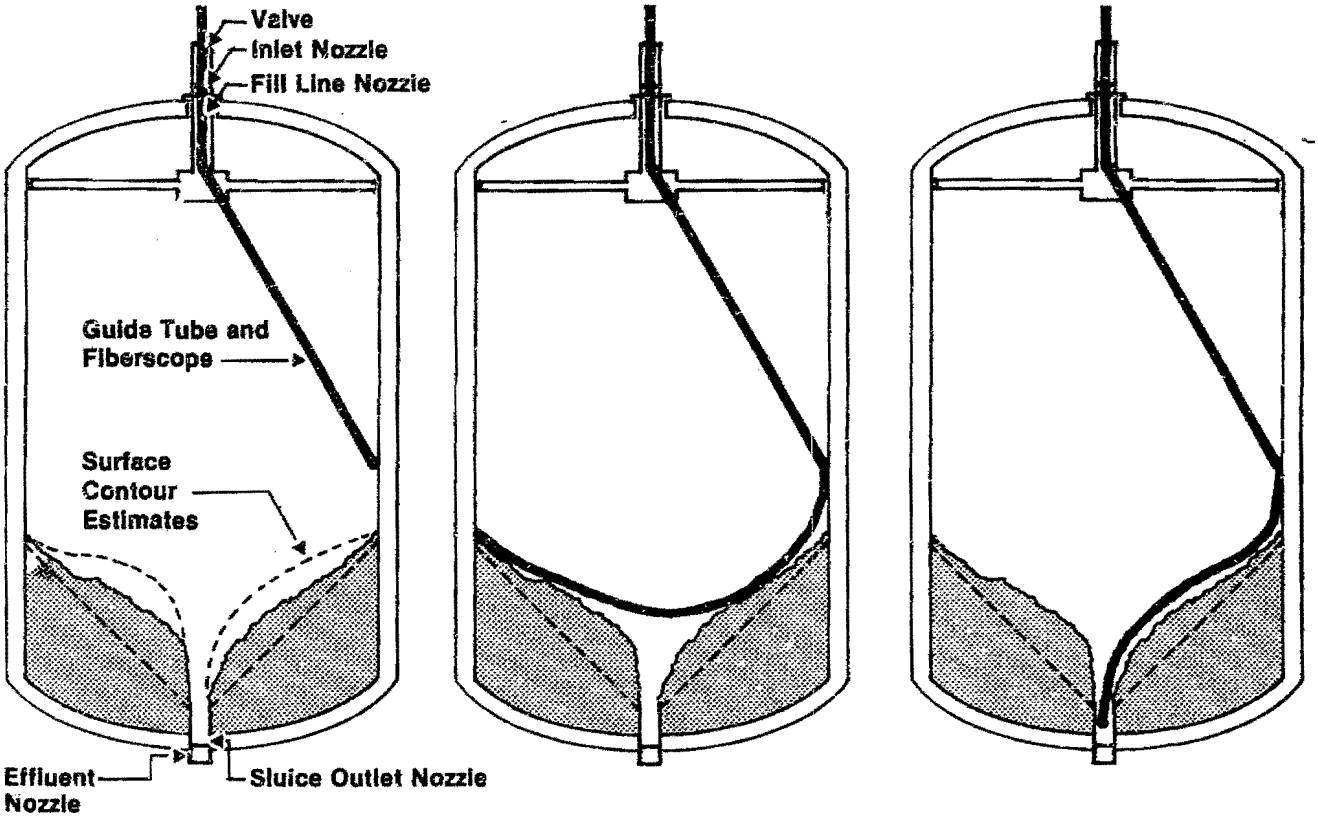


Figure 3. **Pathways of fiber optic horescope during examination of demineralizer A vessel internals.**

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.

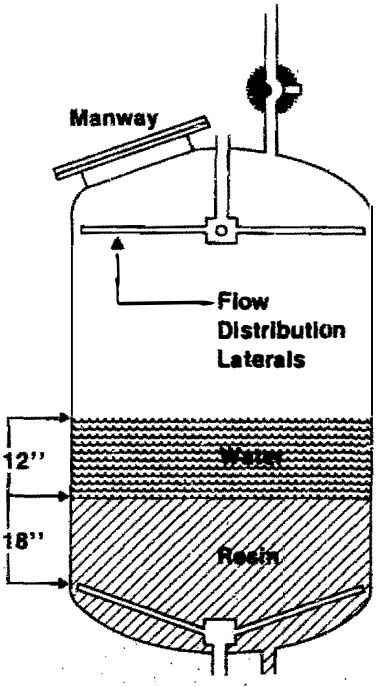


Figure 4. **Estimates of resin and water levels in B vessel.**

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

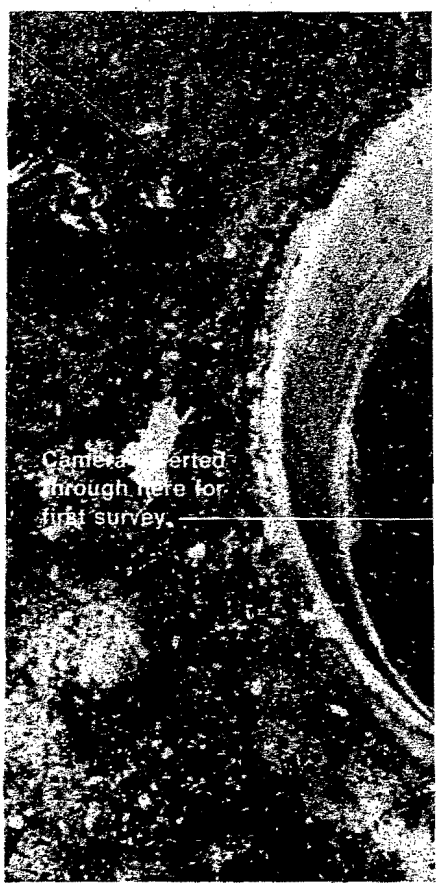
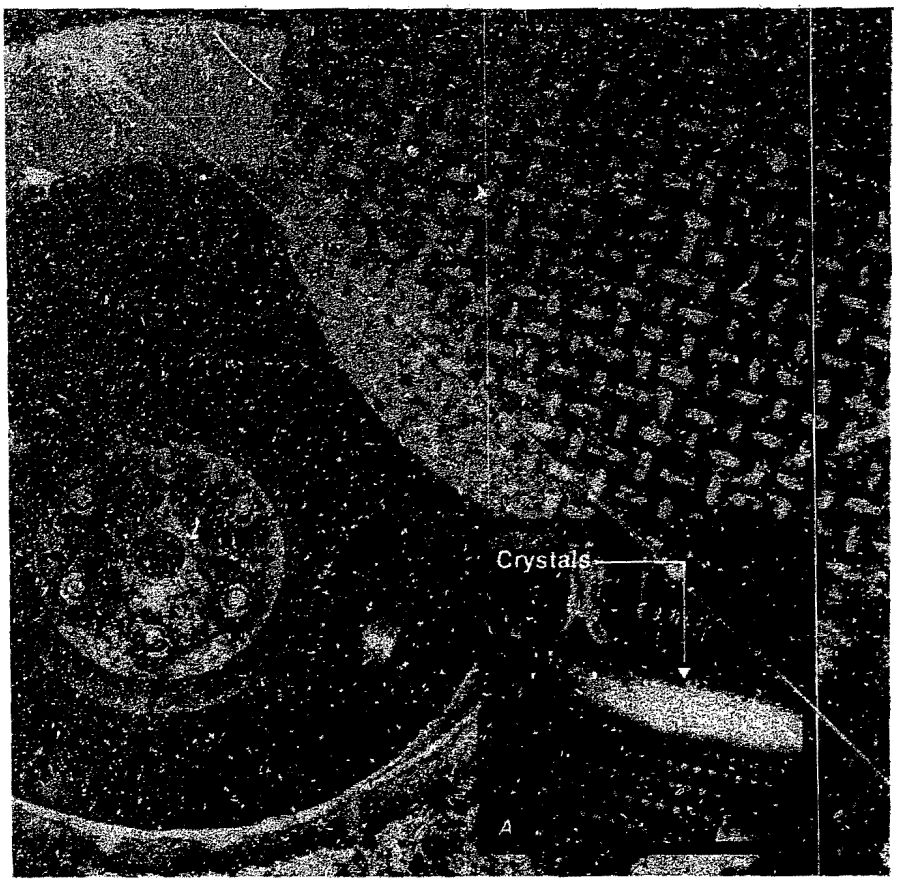
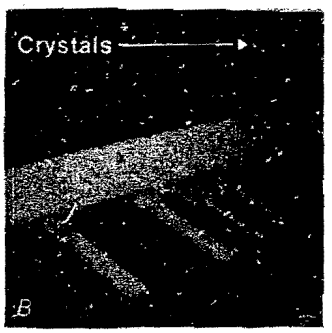


Figure 5. **The 30-in. manway in the 305-foot elevation that provided access to the reactor building basement for camera surveys.**





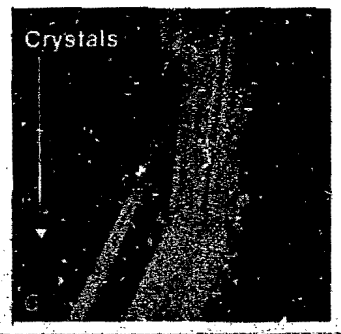


Figure 6.

A. The top of the sump inlet trash rack in the Reactor Building basement. Evidence 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 colite 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.

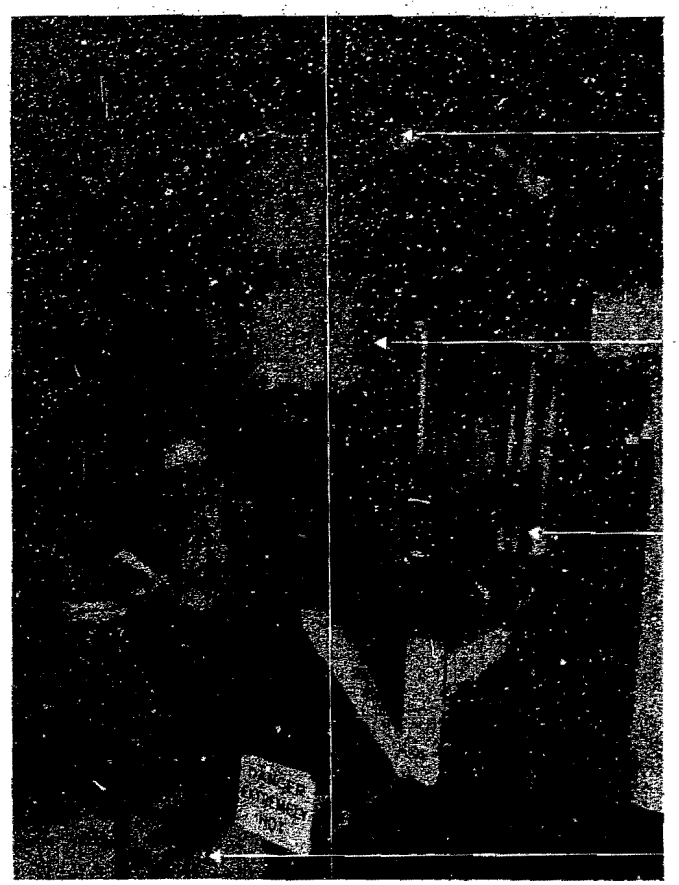


Figure 7. **In-can melter system in use at Pacific Northwest Laboratory to vitrify ZDS zeolites.**

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 nsed 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.

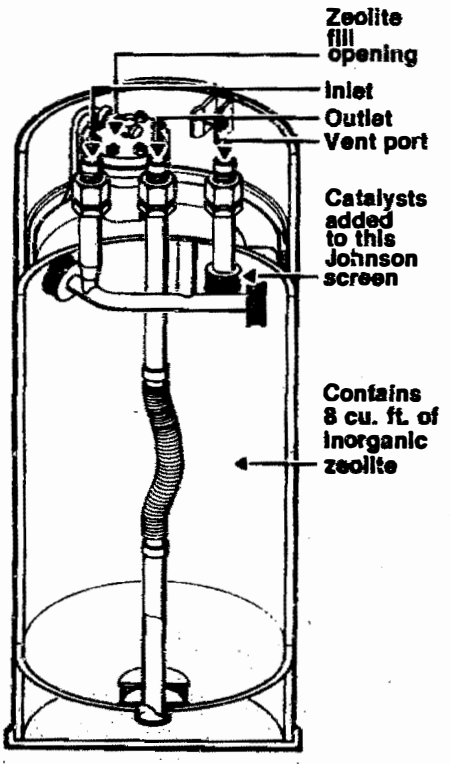


Figure 9. **Cutaway view of an SDS liner showing Johnson Screen to which catalysts are added.**

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.

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

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

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.

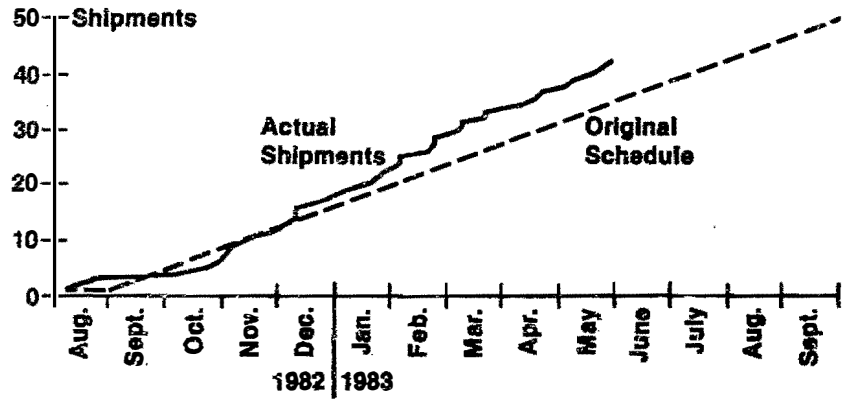


Figure 10. **Actual shipments of EPICOR liners are proceeding ahead of original projections, with completion in July, two months ahead of schedule.**

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 exampie, the I&IC Group has given presentations tc 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.

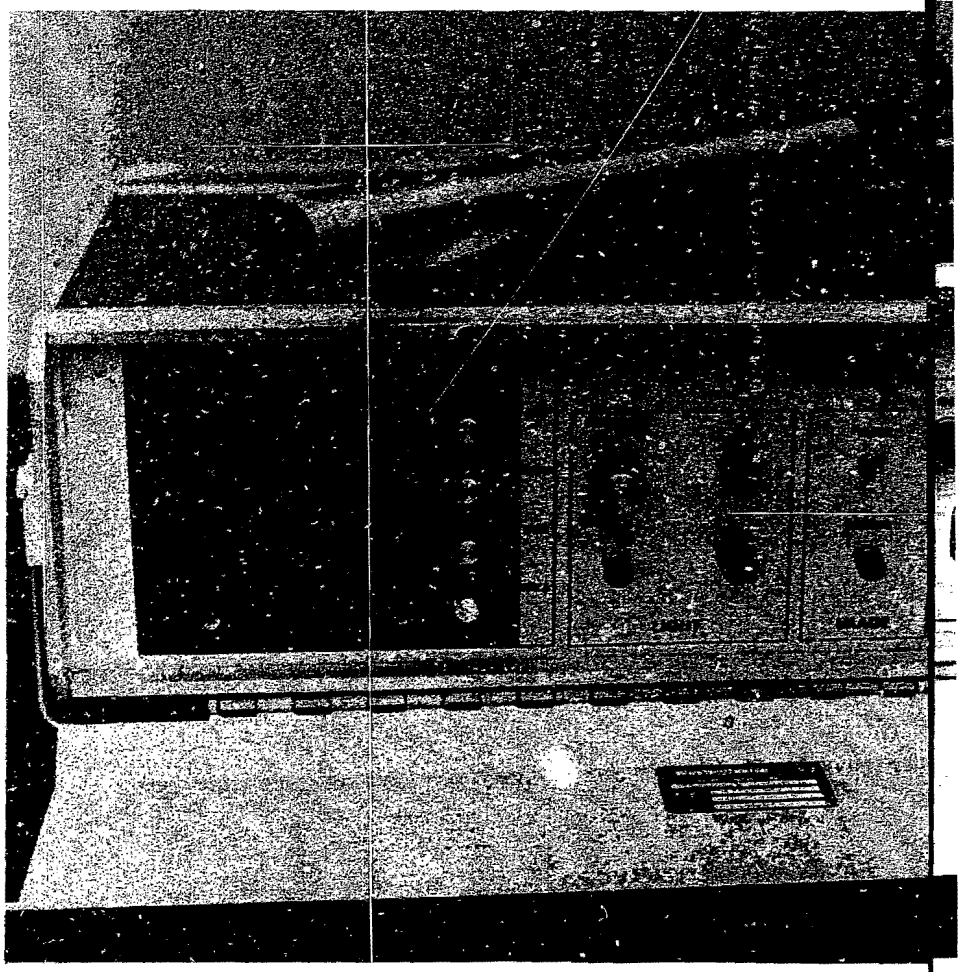
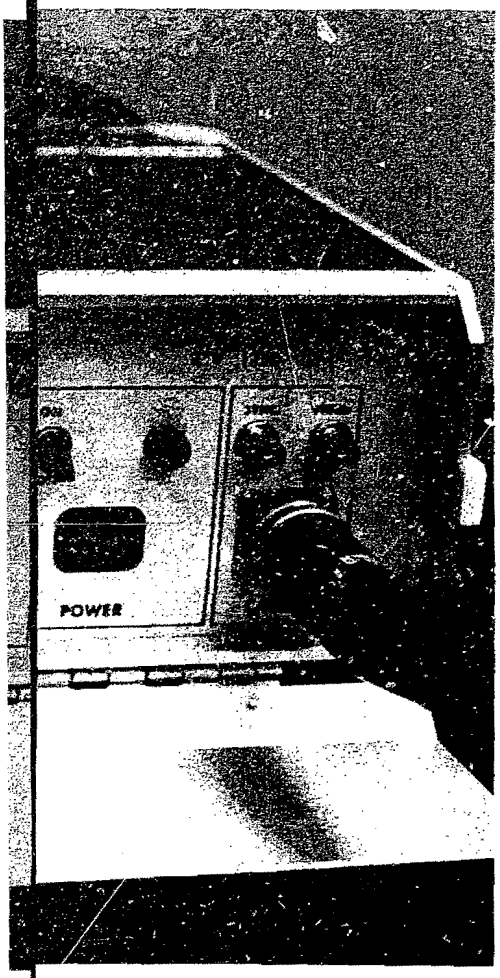


Figure 11. **Quick Look inspection camera and control unit.**



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.

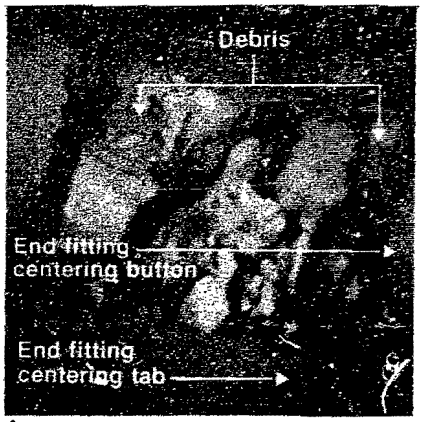
B. Damage to E-9 upper end fitting looks like metal after it has been cut by a torch.

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.



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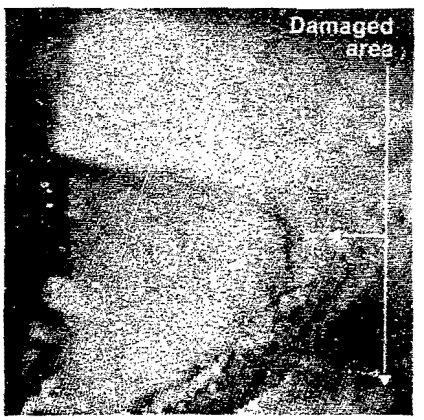

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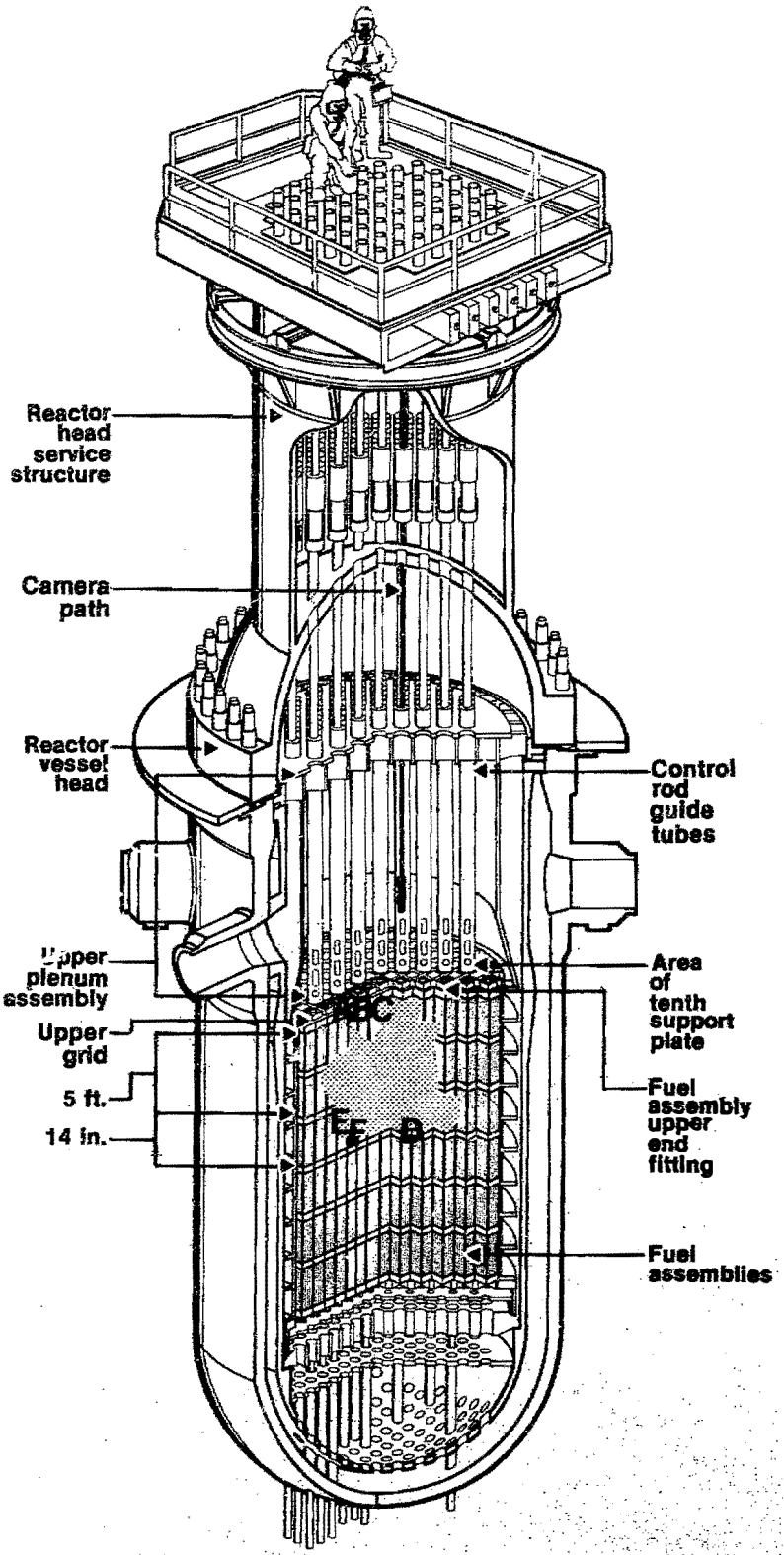

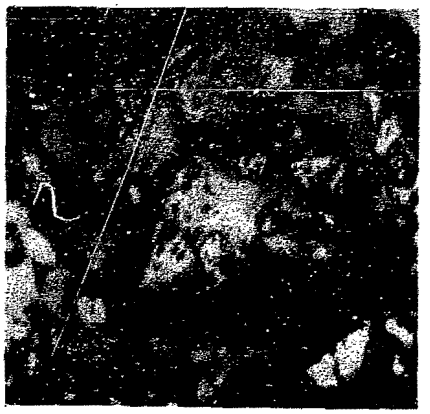

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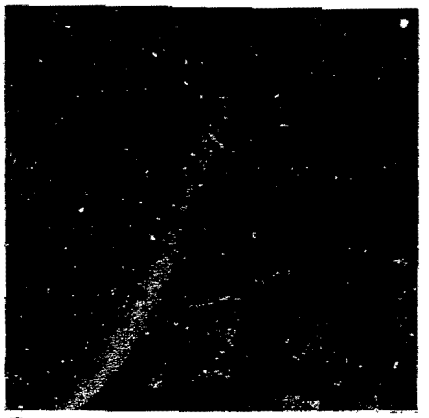

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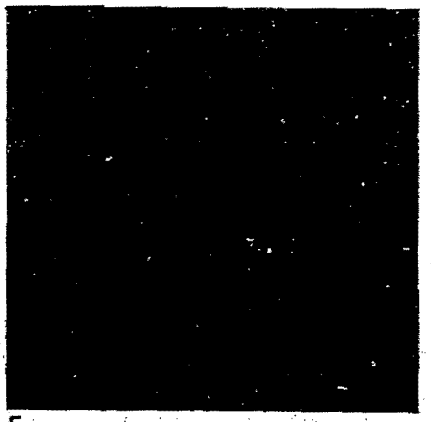

F

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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