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U.S. DEPARTMENT OF ENERGY THREE MILE ISLAND RESEARCH AND DEVELOPMENT PROGRAM 1986 ANNUAL REPORT

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ABSTRACT

Defueling of the Three Mile Island Unit 2 (TMI-2) reactor continued through 1986. This report summarizes this work and other TMI-2 related cleanup, research, and development activities. Other major topics include:

- Core stratification sampling and other data acquisition tasks,
- The fuel shipping program,
- Waste immobilization and management,
- Decontamination and dose reduction, and
- Future uses and applications of TMI-2 data.

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U.S. DEPARTMENT OF ENERGY THREE MILE ISLAND RESEARCH AND DEVELOPMENT PROGRAM 1986 ANNUAL REPORT

PROGRAM PURPOSE AND HISTORY

During 1986, the U.S. Department of Energy (DOE) Three Mile Island Research and Development Program continued. With the cooperation of the plant operator GPU Nuclear Corporation, the Electric Power Research Institute (EPRI), and the U.S. Nuclear Regulatory Commission (NRC), the U.S. Department of Energy (DOE) established the Technical Integration Office at TMI in 1980 to carry out DOE's research and development objectives. These objectives included obtaining and analyzing data on the March 1979 accident and its aftermath, developing new techniques for responding to the unique challenges at TMI, and transferring these findings and technologies to the commercial nuclear power industry.

The Technical Integration Office is DOE's primary data gathering and distribution arm at Three Mile Island Unit 2 (TMI-2), with primary interests in:

- Providing support to GPU Nuclear for recovery operations,
- Supporting the core debris shipping program through on-site preparations and monitoring, and
- Providing samples and other data in support of the TMI-2 Accident Evaluation Program objectives.

Current objectives of the TMI-2 Accident Evaluation Program are:

- To understand the physical and chemical state of the TMI-2 core and related structures and the external influences which affected the accident,
- To understand what happened during the accident and to provide a qualified data base and standard problem of the TMI-2 accident to provide a benchmark for severe-accident analysis codes and methodologies,
- To understand the relationship between the phenomena and process controlling the accident and the important severe-accident and source-term technical issues, and
- To ensure that the results of the program are effectively transferred to the nuclear industry.

Those specific "in-vessel" technical issues for which our understanding can be significantly improved through additional TMI-2 research include questions related to reactor system thermal-hydraulics, core degradation, and fission product behavior.

The Fuel and Waste Handling and Disposition Program is adapting and developing advanced waste processing technologies to manage the special wastes generated as a result of the accident.

Through its evaluations, the DOE TMI-2 R&D program expects to (a) identify possible changes in equipment standards and regulations; (b) improve the understanding of fission product release pathways and retention mechanisms; (c) provide the industry with new techniques for decontamination and recovery; and (d) have an understanding of the accident scenario and sequence. In conjunction with the program, an Information and Industry Coordination staff was established to communicate program findings directly to the industry through various information networks.

Since the accident, DOE, GPU Nuclear, and their contractors have reached a number of milestones in the recovery operation. In 1979, the EPICOR II system was deployed to clean the contaminated water in the basement of the Auxiliary Building. Fifty of the heavily loaded prefilters from that system were subsequently transported to the Idaho National Engineering Laboratory (INEL) for interim storage and research. Following development of a concrete-reinforced High Integrity Container, 46 of the prefilters were transported to the U.S. Ecology commercial disposal facility in the State of Washington. Research continues with the remaining four filters. Television cameras and radiation instruments were used in the first inspection of the Reactor Building. In 1980, 43,000 Ci of radioactive krypton gas were safely vented from the Reactor Building, allowing workers to make entries on a routine basis. In 1981, the submerged demineralizer system (SDS) began to decontaminate the radioactive water from the basement of the Reactor Building. Transport of the SDS vessels containing the waste to the Hanford Engineering Development Laboratory (HEDL) for research and disposition was started in May 1982. Also in 1982, workers lowered cameras into the damaged reactor and conducted the first inspection of the core.

In the summer of 1983, the last solid waste from the processing of original accident-related water was shipped from TMI-2. Later that year, further explorations inside the reactor vessel produced the first samples of the damaged core and the clearest videotapes of the damaged core to that date. In addition, using sonar technology, a topographical map of the core void was prepared.

A number of major steps toward reactor disassembly dominated activities during 1984. In February, the TMI-2 polar crane was load tested for qualification to lift the reactor vessel head. Five months later, the head of the reactor was successfully moved to its storage stand in the Reactor Building, and shielding was installed over the vessel, giving workers safe access to the reactor's internal components to prepare for defueling. Finally in December, the plenum assembly was inspected, cleaned of hanging debris, and jacked 18.4 cm above its seated position. Throughout 1984, preparations for removal of the core were ongoing with the development and procurement of defueling canisters by GPU Nuclear and the placing of a contract for fabrication of two rail shipping casks by DOE.

As a prelude to defueling, the plenum assembly was lifted from the reactor vessel after a major engineering effort. In October 1985, actual reactor defueling operations began with core debris being loaded into the specially designed canisters.

In April 1986, a Certificate of Compliance was issued by the NRC for the NuPac 125-B Rail Cask. The first shipment of core debris was made in July, and by year-end five shipments totaling 35 debris canisters had been made.

In July, computer-controlled drilling equipment was mounted atop the reactor vessel and ten core-bore samples were extracted. Video examinations were performed through the bore holes. Drilling parameter data, enhanced video pictures, and the removed samples are providing valuable information for further development of the accident scenario and guidance for reactor defueling.

The accomplishments to this point have been significant not only in moving the entire recovery effort closer to completion, but in demonstrating that every new challenge this unique situation presents can be met; and the DOE program has been instrumental in keeping the industry well informed about the progress at TMI-2.

SUMMARY OF SIGNIFICANT ACCOMPLISHMENTS OF 1986

Fuel and Waste Handling and Disposition Program

Important efforts under the topic of Fuel and Waste Handling and Disposition include waste immobilization, disposal of accident-generated water, core transportation, and water clarity. Monitoring of an SDS vessel buried in an instrumented concrete overpack continued in 1986. Temperature, moisture, pressure, and particulates are being recorded as part of the Monitored Retrievable Burial Demonstration Program.

An estimated 2.1 million gal of processed TMI-2 water will have accumulated by the end of the cleanup program. This water will require disposition via a Nuclear Regulatory Commission (NRC) approved method. GPU Nuclear undertook a study of possible disposal methods and in July 1986 proposed evaporation as the preferred method.

Transportation of the core debris from TMI-2 to the INEL commenced in July 1986. Preparations for core transportation were an extensive effort in 1986 as discussed in detail in following sections. The debris is shipped in the NuPac 125-B Rail Cask, which holds seven defueling canisters. The core debris will be stored at the INEL for up to 30 years, during which time it will be available for examination and research.

In early 1986, a biological growth in the reactor vessel water reduced visibility to the point that defueling operations were severely hampered. After testing a number of biocides, hydrogen peroxide was used as a shock treatment to kill the microbes. The Defueling Water Cleanup System (DWCS) was then found to be ineffective because a large number of very small metallic particles that were generated during the accident clogged the sintered metal DWCS filters. Addition of a coagulant to the water and modifications to the DWCS appear to have solved the water clarity problem.

Accident Evaluation Program

Video examinations of and samples from the lower head debris were analyzed in 1986. The accident scenario was updated based on new inspections and the core stratification sampling project. Based on the core boring data, it is postulated that the molten core material flowed into the lower plenum from the core near the core periphery in the southeast quadrant of the core. The primary flow path appears to have been through relatively undamaged fuel assemblies and the core support assembly.

Collection and analysis of data continued in an effort to determine the fission product inventory in the TMI-2 buildings and equipment. The fission product inventory in the core and reactor coolant system is also being determined via detailed chemical examination of samples of various primary system components and core materials.

A major effort was the core stratification sampling program or "core bore." A drill was mounted on the reactor vessel and material samples were extracted from 10 locations in the core. Video inspections in these core locations revealed the first clear picture of the condition of the lower core and core support assembly. The core bore samples were transported to the INEL for research purposes.

Reactor Evaluation Program

Defueling, though hampered by water clarity, removed over 20% of the core debris in 1986. Defueling began by loading loose debris into defueling canisters with hand-operated tools. After a pause for water clarity work and the core bore project, an attempt was made to break up the fused mass found at the center of the core. This was partially successful and normal defueling operations resumed in December 1986.

Planning for lower CSA and lower head defueling continued in 1986. A plasma arc cutting system and a remote manipulator system were chosen for further development. Systems for defueling the pressurizer, the pressurizer spray line, and the plenum were designed in 1986.

Ex-vessel fuel location continued with efforts in the major RCS component, the reactor building basement, and the makeup and purification system. In all cases, less fuel was found than expected.

GPU Nuclear presented a plan for disposition of the TMI-2 plant after cleanup to the NRC. The plant will be in a condition known as Post-Defueling Monitored Storage which will be safe, stable, secure, and pose no threat to the public.

FUEL AND WASTE HANDLING AND DISPOSITION PROGRAM

Waste Immobilization

The Submerged Demineralizer System (SDS) was used during the Three Mile Island Unit 2 (TMI-2) nuclear reactor cleanup to remove cesium and strontium from contaminated water. Nineteen SDS vessels, packaged one at a time in a shielded and licensed shipping cask, were transported to Rockwell Hanford Operations. Of the nineteen vessels, the contents of three were vitrified, fifteen were placed in overpacks, and one was overpacked for long-term monitoring. Long-term monitoring of SDS vessel No. D10011 in its concrete overpack was undertaken to determine actual burial conditions during the proposed test period of 20 to 50 yr as part of the Monitored Retrievable Burial Demonstration Program. Monitoring of the vessel pressure began in June 1984; the vessel was placed in its concrete overpack in September 1984. Temperature, moisture, and particulate monitoring began in November 1984, after burial of the vessel and its overpack had been completed. The results of the monitoring are as follows:

- An evaluation of the moisture, particulate, pressure, and temperature data obtained to date from SDS vessel No. D10011 shows no abnormalities. The humidity of the air between the vessel and its overpack is relatively low, indicating dry, ideal storage conditions. Suspended particulate materials in the air surrounding the vessel are very low and no radioactivity has been detected.
- The rate of pressure rise in the vessel decreased from approximately 9.4 lbf/in^2-yr to approximately 3.5 lb/in^2-yr as its pressure rose from a few psi (absolute) to one atmosphere. Therefore, it appears that the initial pressure rise was caused partly by a slow leak of air into the vessel or that the net gas generation rate has decreased by a factor of approximately three.
- The net gas generation rate plus the air inleakage rate initially totaled approximately 70 l/yr. The gas generation rate when the gas pressure was at one atmosphere (leakage essentially zero) was approximately 25 l/yr. It is expected that the pressure will reach its peak during the next year when the generation rate equals the leak rate.
- An analysis of the temperature data shows that temperatures are at or near their maximums. The calculated current peak temperature of the zeolite near its vertical centerline and below its upper surface is 265°F. The maximum recorded vessel temperature is 136.3°F.

Core Transportation

Major accomplishments were made in preparing and transporting the damaged core from TMI-2 to the INEL. At the INEL, the core will be stored and portions used for research by the Core Examination Research and Development Program pending final disposition.

Nuclear Packaging, Inc. (NuPac) completed fabrication of the second NuPac 125-B Rail Cask (with skid and railcar) and miscellaneous cask handling and loading equipment (i.e., vertical lift fixture, horizontal lift equipment, fuel transfer cask, mini hot cell, jib crane) in January 1986. In April 1986, the Transportation Certification Branch of the Nuclear Regulatory Commission issued the Certificate of Compliance for the NuPac 125-B Rail Cask.

The second cask and all loading equipment were transported to Hanford Engineering Development Laboratory (HEDL) in Richland, Washington, for use in an integrated system test. The equipment was assembled, operationally tested, and used for training of personnel from GPU Nuclear. The test demonstrated that all components functioned as designed, and that operating procedures developed by GPU Nuclear were usable. Whenever difficulties were identified in equipment or software, corrections were made at the facility. Once the testing was completed, the equipment was disassembled, transported to TMI-2, and reassembled.

In parallel with the Integrated System Test at HEDL, EG&G Idaho used the first cask and some specialized hardware custom designed for use at the INEL in training exercises. The equipment functioned as intended, and the detailed operating procedures written for using the cask and equipment proved to be adequate.

Several months before the first shipment of core materials from TMI-2 to the INEL, EG&G Idaho (representing DOE) notified each state along the rail route between TMI-2 and the INEL of the planned action by DOE to transport core debris through their political jurisdictions. Representatives of each state also were invited to TMI-2 for a public viewing of the cask and equipment and a detailed briefing of the route, safety considerations, and emergency preparedness. Several representatives accepted the invitation and voiced approval that the briefing was useful in satisfying specific concerns.

Preparation of canisters for shipment from TMI-2 includes: (a) killing micro-organisms in the canister with a hydrogen peroxide solution (if required) (b) dewatering canisters (a minimum of 2 times each), (c) sampling of gas in each canister to ensure catalytic control of radiolytic gases, (d) pressurizing each canister to 2 atmospheres with argon gas, (e) checking for leaks, (f) removing the relief valves, (g) weighing each canister, and (h) decontaminating each canister as it is moved to the cask.

Loading of the cask begins by removing overpacks from the cask at TMI-2. The railcar and cask are positioned under a cask-unloading station in the Truck Bay of the Fuel Handling Building. Screw jacks on the cask-unloading station lift the cask and transport skid from the railcar . The railcar is moved out of the Truck Bay; the cask and skid are lowered to the floor, and the door of the Truck Bay is closed. The cask-unloading station is moved and stored out of the way. Two hydraulic cylinders are attached to lift the cask from horizontal to vertical position, where the cask is locked in place by attachment to a support tower (see Figure 1). A work platform is bolted around the cask, connecting it to the tower. The cask is opened by removing the lids of the outer and inner vessels, and a shielded loading collar is installed. A mini-hot cell is moved over the cask and collar to remove and hold a shield plug from one of seven tubes in the cask. A canister is transferred from the spent fuel storage pool by the fuel transfer cask. The fuel transfer cask with canister is placed on the rail cask. The canister is lowered into the cask and the loading process repeated six more times, using the shielded equipment to reduce radiation exposure to personnel.

After loading is finished, lids of the inner and outer vessels are replaced and individually leak-tested to ensure that the rail cask is assembled correctly. The cask is rotated to the horizontal, placed on the railcar, reassembled with overpacks, inspected, and surveyed for radiation levels. The environmental cover is secured over the cask and the train is assembled. The train consists of an engine, a buffer car, a cask, an additional buffer car (if 2 casks are transported, the second cask and another buffer car are added), and a caboose. Instructions are given to the engineer of the train, including normal operations and what to do in the unlikely event of an accident.

Conrail Railroad Company accepts the loaded cask at TMI-2 and transports it through Pennsylvania, Ohio, Indiana, and Illinois. At East St. Louis, the cask is transferred to Union Pacific Railroad which transports it through Missouri, Kansas, Nebraska, Colorado, Wyoming, and Idaho to the INEL.

At the INEL, the cask is received at the Central Facilities Area, where the environmental cover and overpacks are removed. The cask is lifted from the railcar by a gantry crane and transferred to a truck transporter for travel to the Hot Shop of Test Area North. In the Hot Shop, the cask is rotated to vertical, moved to a cask storage stand, tested for internal airborne contamination, and opened. All operations involving manipulation of canisters are conducted remotely. Each canister is withdrawn from the cask, transferred to the vestibule of the water pit, and lowered into an underwater module situated atop the pool cart. Each module holds a maximum of six canisters. When a module is full, each canister is vented and filled with demineralized water. Then the module is conveyed to the water pit, where modules are rowed together (but not interconnected) forming the storage rack. Once the module is in place, a vent line is connected to each canister. The cask is reassembled, transferred to CFA, loaded onto the railcar, and returned to TMI-2, where the operation starts again.

Storage of TMI-2 core debris at the INEL is planned for a maximum of 30 years. During that period, the scientific community will have core debris material available for examination and research. Samples of material will be collected after removing a canister from the water pit, transferring it to a hot cell in the hot shop, and opening it remotely.

Rockwell Hanford Operation assisted the transportation effort by performing a series of tests of the catalytic recombiner built into each canister. The tests evaluated effects of various contaminants on the catalyst. That is, Rockwell was asked to determine whether or not certain contaminants impeded or accelerated efficiency of the catalyst to recombine gases generated radiolytically. The contaminants tested included: (a) concentrated boric acid/water solution with dye penetrant and developer solutions, (b) diluted BIOSPERSE-250 and 14% nitric acid solutions, and (c) a "worst case" test needed to

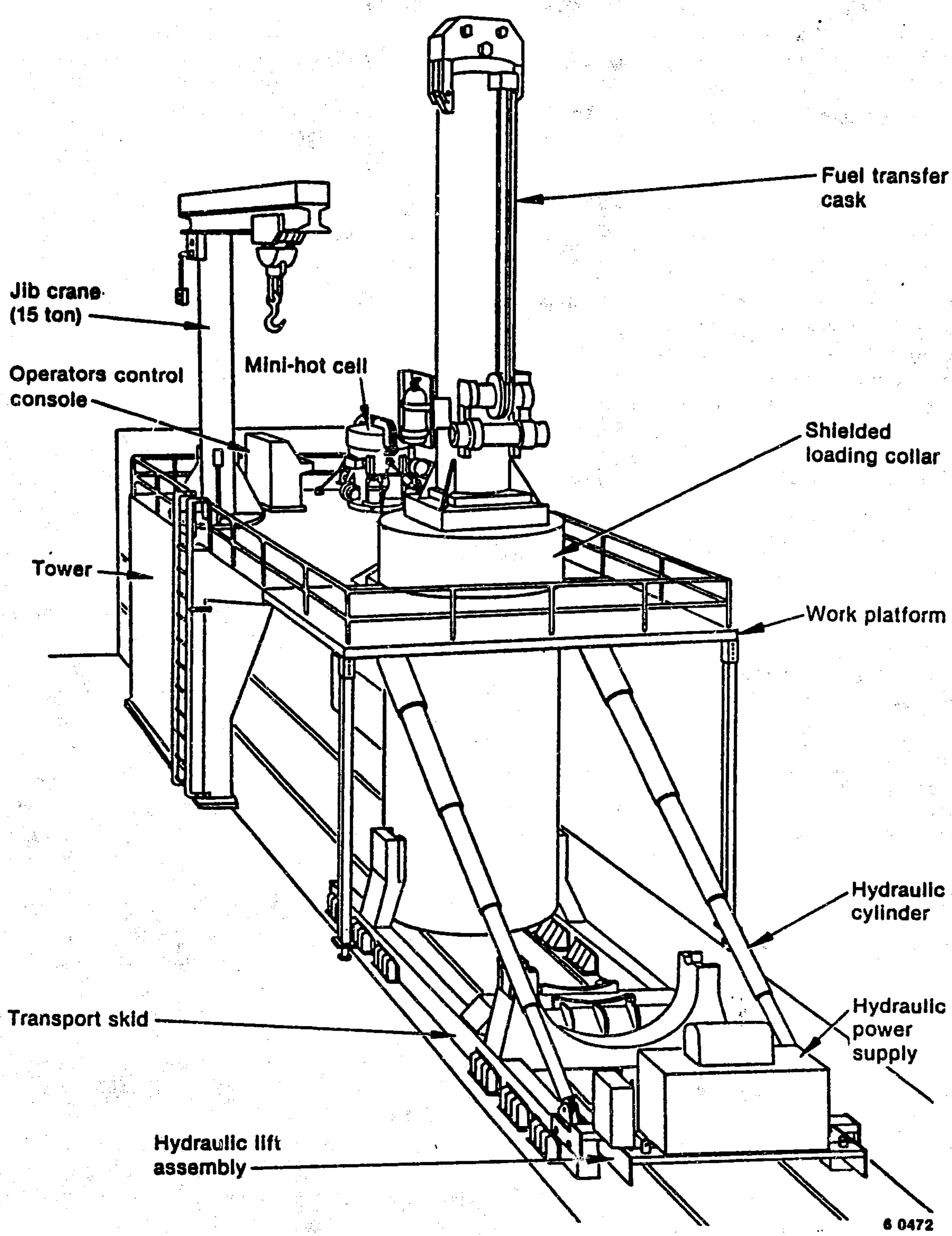

Figure 1. TMI-2 fuel cask loading components.

establish bounds for the combination of contamination conditions the canister catalyst could encounter before shipment. The test exposed the catalyst to Okite cutting fluid, Magnaflux dye penetrant chemicals, hydraulic fluid, reactor coolant microorganisms, and 400 ppm hydrogen peroxide solution. In addition, recombiner catalysts tests were performed on a hydraulic fluid called Quintolubric 807-SN and a coagulant called BETZ #1182. In all tests except the 14% hot nitric acid, the catalyst continued to perform adequately. The 14% hot nitric acid poisoned the catalyst.

ACCIDENT EVALUATION PROGRAM

Examination Requirements and Systems Evaluation

Fission Product Inventory. During 1986, a detailed plan was formulated for the analysis and reporting of the fission product information from TMI-2. This plan which was intended only for internal use received extensive review by EG&G Idaho fission product experts and will form the basis for fission product analysis through the duration of the TMI-2 Accident Evaluation Program (AEP).

The samples extracted from the lower plenum rubble bed during 1985 were examined and the fission product contents were documented. These samples were extracted from the top of the debris bed in the lower plenum near the reactor vessel wall and at two different azimuthal locations, the south and the southwest quadrants.

Samples from the lower debris bed indicate that this mass, which experienced temperatures as high as 3100 K, retained approximately 16% of the fission product Cs, but less than 10% of the fission products Sb and Ru and less than 5% of the fission product I. A summary of the retained fission product fractions in the lower plenum debris is given in Table 1 and is compared with retentions measured on grab samples from the upper debris bed.

The video examinations of the lower plenum debris bed carried out during 1985 and 1986 have been analyzed and a report of the analysis results is being prepared. On the basis of that analysis, it is estimated that approximately 16% of the original core mass now resides in the lower plenum and that the packing fraction of the debris is approximately 0.50.

Retention of the nonvolatile rare earth fission products, ^{144}Ce and ^{154}Eu, are roughly equal in both the upper and lower debris beds. The retention of ^{137}Cs is only slightly lower in the lower debris bed where a far lower retention would have been expected on the basis of its volatility. The expected lower retention is exhibited by iodine, an average of only 2% being retained in the samples examined. Antimony and ruthenium retentions are also lower in the lower plenum material; however, larger fractions of these fission products may be found in metallic phases which have yet to be examined. Because these lower plenum results showed a wide variation and are based on the analysis of only eleven particles, they should be regarded as preliminary data.

Standard Problem. The TMI-2 accident will be used as a benchmark for severe accident analysis techniques. Such techniques are being applied by industry and regulatory agencies to estimate the source term from low-probability severe accidents. The standard problem is a formal exercise in which several participants will apply their analytical methods to the TMI-2 accident using common initial and boundary conditions. The results of the analyses will be compared among the participants and with the measured or determined actual conditions during the accident. Organizations interested in participating include the NRC and foreign countries through the Organization of Economic Cooperation and Development's Committee on the Safety of Nuclear Installations.

A standard problem package was distributed at the end of 1986 to participating organizations. The package contains the necessary information to perform an analysis: initial plant conditions; boundary conditions, such as operator actions; and plant configuration (i.e., a complete geometric description). Additionally, to assist the analyses, the balance of the package contains a best-estimate accident scenario and selected results of a demonstration analysis performed with the state-of-the-art severe accident analysis code—RELAP5/SCDAP.

Accident Scenario. Development of the TMI-2 accident scenario is the focal point of the AEP and involves integration of information from (a) TMI-2 reactor system measurements recorded during the accident; (b) the end-state characterization of the TMI-2 core, core support structures, and reactor vessel; and (c) independent experiments simulating fuel and fission product behavior during severe accidents.

Previous scenario work was based on the data available through mid-1985. Recent work in the following areas provided the basis for an improved understanding of the accident:

- Determination of damage to the lower core, core support assembly (CSA), and lower plenum region,

Table 1. Average radionuclide retention in the upper and lower vessel debris beds

Radionuclide	Percent of Inventory Retained[a]			
	Lower Plenum		Upper Plenum	
	Average	Range	Average	Range
^{129}I[b]	2	0-10	22	10-28
^{137}Cs	16	9-22	21	6-32
^{125}Sv	5	3-10	28	18-38
^{106}Ru	7	4-9	55	35-86
^{144}Ce	114	106-124	114	90-130
^{154}Eu	85	75-94	90	60-108

a. Compared with whole core average ORIGEN-2 analysis (μ Ci/gU) and average uranium content.

b. Average radionuclide content based on the average of a number of small (5-40 mg) particle analyses.

- Examination results from lower plenum debris samples,
- Defueling of the upper debris bed and intact fuel assemblies from the core periphery,
- Analysis of potential damage to the lower vessel head and the instrument penetration nozzles,
- Analysis to improve understanding of the degraded core heatup and the mechanisms which controlled core relocation,
- Analysis of the reactor system thermal-hydraulic response that includes RCS coolant inventory and core heat transfer rates, and
- Improved interpretation of the source range monitor and in-core self-powered neutron detector responses.

The accident was initiated by cessation of secondary feedwater flow. The steam generator boiled dry, and the resultant reduction of primary-to-secondary heat exchange caused the primary coolant to heat up, surge into the pressurizer, and increase the primary system pressure. The pilot-operated relief valve (PORV) opened to relieve pressure, but failed to close when the pressure decreased. The first 100 min of the accident can be characterized as a small break loss-of-coolant accident (LOCA) with resultant loss of primary coolant and decreasing pressure. It differed from the scenario expected during such a LOCA in that the pressurizer liquid level remained high. This was interpreted by the reactor operator as an indication that the reactor coolant system (RCS) was full of water when, in fact, the RCS was continually voiding. Up to 100 min, the core was covered with sufficient water to be cooled.

The reactor coolant pumps were turned off at 100 min, and core heatup was initiated as the water level stratified and decreased below the core top. By 150 min, an exothermic zircaloy-steam reaction was initiated, dramatically increasing the core heatup rate. As a result, zircaloy melting temperatures were exceeded, resulting in relocation of the molten zircaloy and some liquefied fuel to the lower core regions, solidifying near the coolant interface. These conditions persisted until 174 min. By this time a region of consolidated, degraded core material existed in the lower, central regions of the core as shown in Figure 2. Coolant flow through this consolidated material was probably negligible. The intact fuel rod stubs in the lower core region indicate that the lower 0.5 m of the core remained cool.

A reactor coolant pump was turned on briefly at 174 min, and coolant was pumped into the reactor vessel. The resultant thermal-mechanical forces,

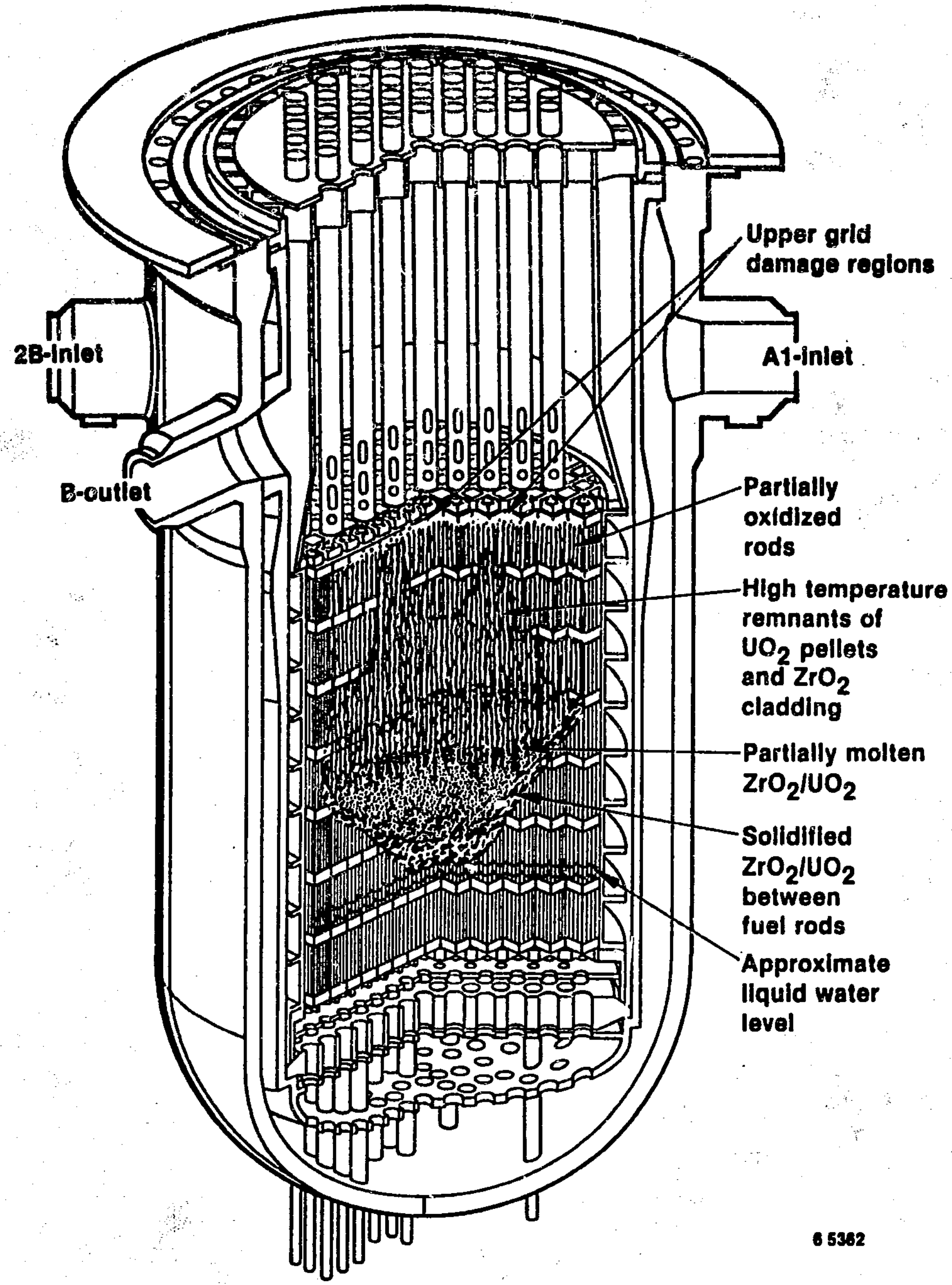

Figure 2. Hypothesized core damage configuration (173 min).

generated from the rapid steam formation, are believed to have shattered the oxidized fuel rod remnants in the upper regions of the core, forming a rubble bed on top of the consolidated core materials as shown in Figure 3. The consolidated core materials continued to heat up during the next 50 min (174 to 224 min), even though coolant delivery to the reactor vessel from the pump transient and emergency core cooling injection is estimated to have covered the core by approximately 210 min. By 224 min, much of the consolidated region had reached temperatures sufficient to melt the U-Zr-O ternary mixture.

On-line TMI-2 data recorded during the accident indicate that the crust surrounding the consolidated core failed and, between 224 and 226 min, some of the molten core material relocated to the lower plenum. Based on the end-state core and core support assembly (CSA) configuration and supporting analysis of the degraded core heatup, it is

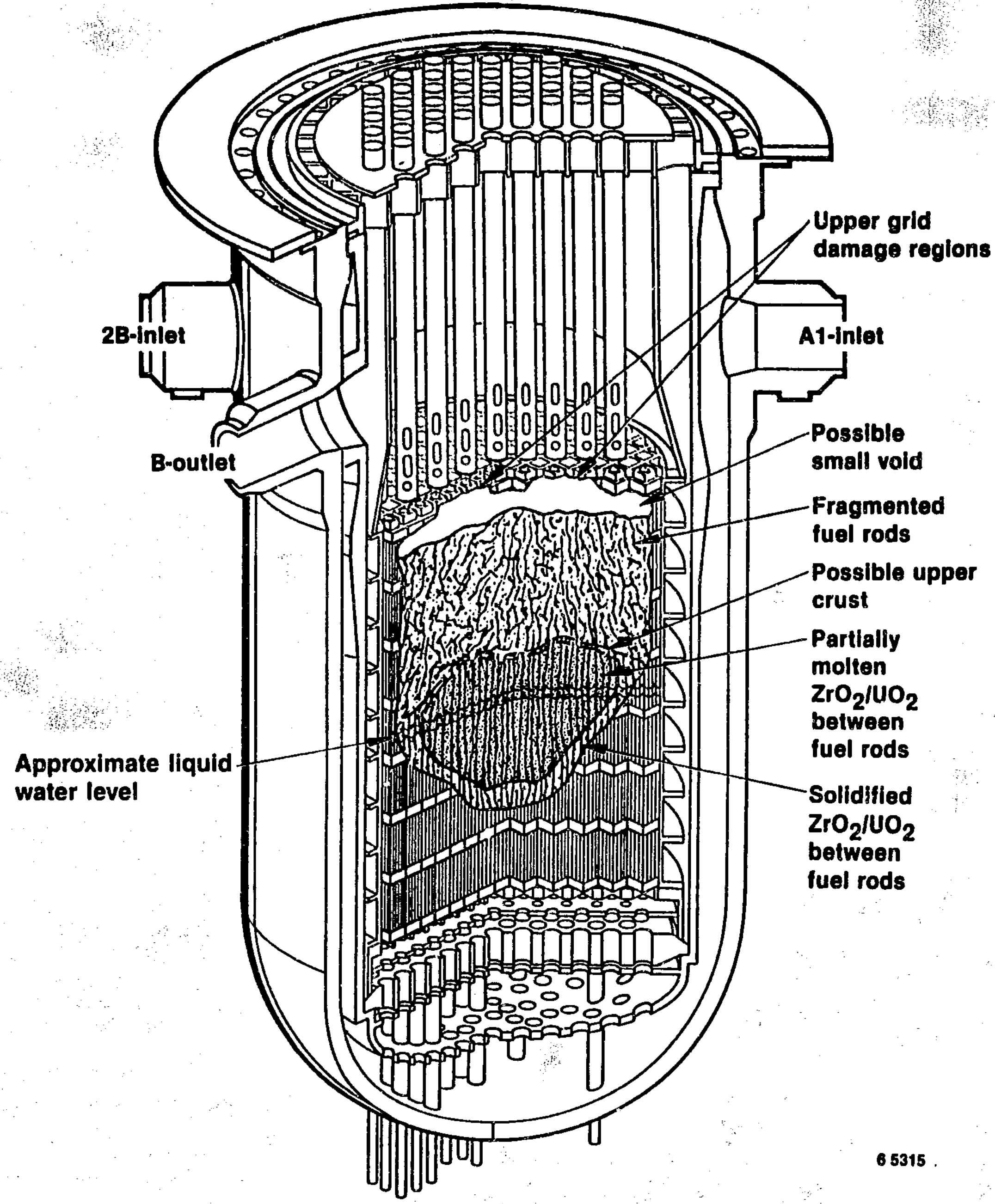

Figure 3. Hypothesized core damage configuration (175-180 min).

believed that the crust failure occurred near the top of the molten core region in the southeast quadrant of the reactor vessel. Crust failure allowed the molten core material to flow from the mid-core region downward through the core to the lower plenum region of the reactor vessel. The configuration of the core during this relocation process is shown in Figure 4. Limited damage to the CSA may have occurred as the core material flowed to the lower plenum. Estimates of the maximum pressure vessel wall temperatures indicate that the melting point of stainless steel was not exceeded, even at the inside surface of the pressure vessel liner. The Inconel instrument penetration tubes, however, may have melted in the lower plenum above the vessel penetration weld. If this occurred, freezing of molten material is predicted to have plugged any holes in these tubes.

Important questions relative to core failure and relocation of the molten core material remain to be

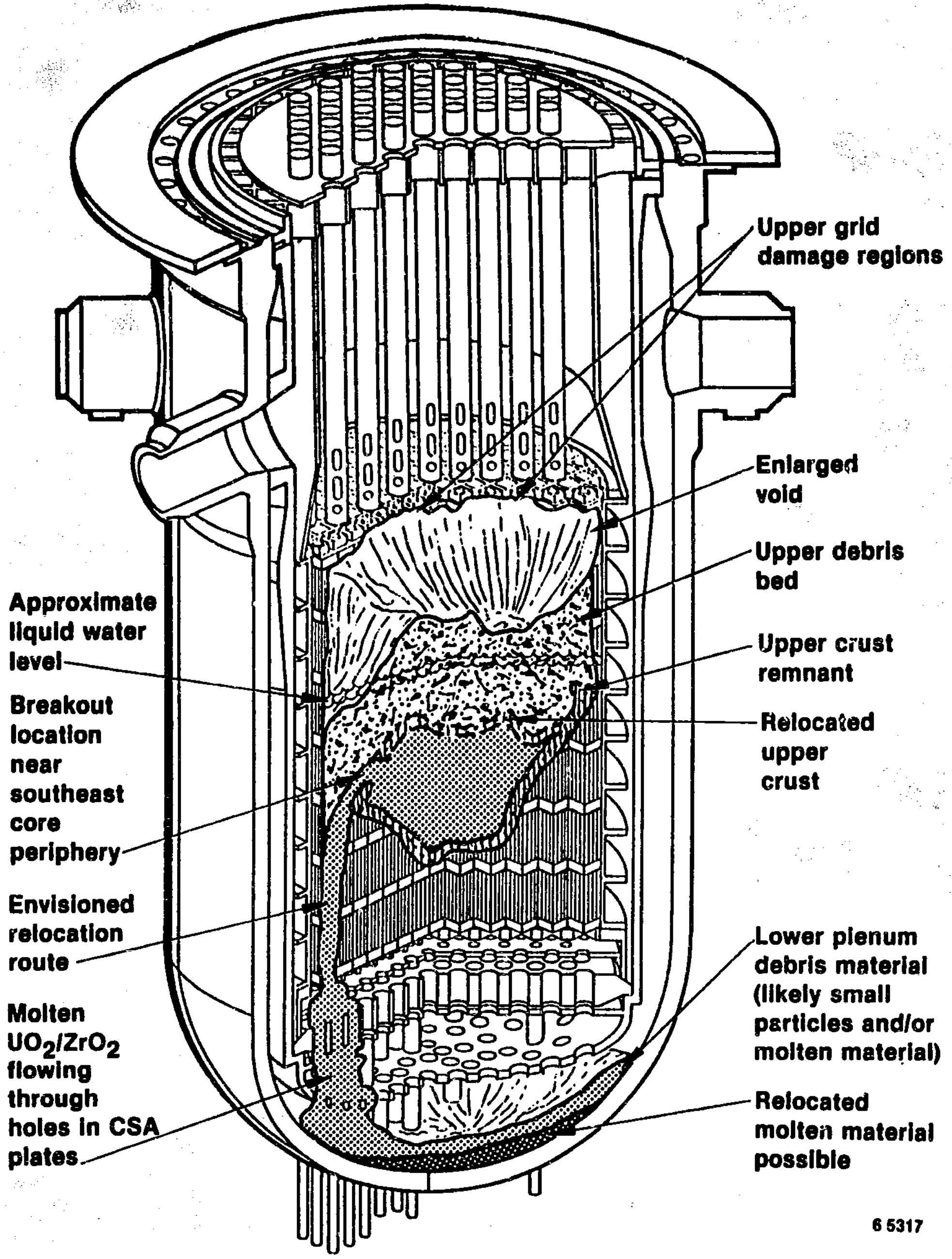

Figure 4. Hypothesized core damage configuration (226 min).

answered. Perhaps the most important of these is: What was the mechanism or mechanisms leading to failure of the crust surrounding the molten consolidated core? There appear to be several plausible failure mechanisms; however, an unequivocal answer to this question will require additional inspection of the core crust in the east quadrant and sample acquisition and examination of the crust material to determine its composition, material interactions, and physical and chemical properties. Additional inspections of the core region in the east quadrant of the vessel are necessary to determine if the crust failure was localized or global in nature.

Another important question concerns the extent of damage to the reactor vessel. Additional inspection and sample examination data are necessary to adequately characterize the lower plenum debris and possible damage to the pressure vessel lower head and instrument tube penetrations.

Even though the accident scenario development is not complete, several important conclusions are emerging relative to core damage progression and the consequences of severe core failure during severe accidents, based on the TMI-2 research. These include:

- When the high-pressure injection was initiated at 200 min, core cooling resulted. However, the molten, consolidated core material continued to heat up, despite the presence of water surrounding the crust.
- There is no evidence that a steam explosion occurred when the crust surrounding the consolidated molten core material failed, allowing approximately 16% of the total core mass to fall into the water-filled lower plenum.
- The presence of water in the lower plenum terminated the accident progression and prevented failure of the reactor vessel lower head.
- Fission product release from the fuel and the RCS is very sensitive to the volatility of the individual elements or chemical species. Release of the high-volatile fission products (iodine and cesium) was less than expected, especially cesium, up to 20% of which was retained in the previously molten core material that relocated to the lower plenum. Retention of the medium- and low-volatile fission products in the reactor pressure vessel (RPV) was nearly complete, although significant amounts of antimony and ruthenium were released from the fuel and are believed to be bound to metallic structures in the reactor vessel.

Analytical and Experiment Support

The Data Evaluation and Data Base Development staff of the Accident Evaluation Program has two principal areas of responsibility: (a) the evaluation of on-line measurements that pertain to the standard problem, and (b) the development of data bases which eventually will contain the information required to run a standard problem.

Most of the on-line data has been digitized and entered into a computer file. Data sufficient to run the standard problem have been evaluated. These data include the initial conditions such as mass flows, core power, and initial temperatures and pressures. Also, boundary conditions such as secondary pressures and levels and primary system makeup and letdown flows are included. Three data bases have been made available to the standard problem participants: (a) plant configuration, (b) sequence of events, and (c) initial and boundary conditions. During 1987, two additional data bases will be developed: (d) core bore, and (e) fission product.

The core bore data base will contain data relating to the core bores completed in 1986 and the end-state of other core conditions. The fission product data base will contain data regarding the temporal, spatial fission product inventory.

Information and Industry Coordination

The *Proceedings of the First International Information Meeting on the TMI-2 Accident* was published and distributed to attendees of the conference and other interested parties. About 700 copies of the Proceedings have been distributed.

Five papers on recent progress in the TMI-2 Accident Evaluation Program were presented at the NRC-sponsored Water Reactor Safety Research Meeting in Washington, DC in October 1986. Full papers covering the presentations have been prepared and will be published in the Proceedings of the meeting.

At the meeting of the American Chemical Society in Anaheim in September 1986, Accident Evaluation Program personnel presented two papers on the chemical processes important to severe accidents. These papers will be published in the *Advances in Chemistry Series.*

The revised Accident Evaluation Program Description document, which was prepared in draft form in 1985, was published in early 1986. This document establishes the objectives and goals of the program and outlines the basic research methodology necessary to attain those goals and objectives. An update of the TMI-2 accident scenario, based on recently obtained information from the core boring operation, analyses of on-line instrumentation data, and engineering analysis of core failure mechanisms and thermal response of the structures in the lower plenum, was also published during 1986. Before publication, the revised scenario was subjected to peer review by an *ad hoc* committee composed of members of industry and academia.

A large number of presentations were made during the year, highlighted by a presentation to the U.S. NRC Commissioners by the DOE Program Manager and a presentation to the DOE Assistant Secretary for Nuclear Energy.

Sample Acquisition and Examination

Program Planning. In 1986, the TMI-2 accident information needs of the TMI-2 Accident Evaluation Program were converted into specific sample acquisition and examination plans. The initial sample examination plans established an integrated sample examination program to be sponsored by DOE, at both the INEL and private laboratories, and by the NRC at ANL-E. In December, the plans were issued in a report entitled *TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan—Executive Summary*, EGG-TMI-2-7121. In 1987, an updated unabridged version of the Examination Plan will be published. This version will include the CSNI Examination Plan and a listing of over 400 TMI-2 accident reference documents, both developed in 1986. The reference documents contain information about the TMI-2 accident evaluation - program planning, core damage, and fission product inventory release.

Fission Product Inventory Characterization

EX-RCS Fission Product Inventory Characterization. In 1986, efforts continued to locate and characterize the core fission products and materials, which had been captured in the TMI-2 buildings and equipment after release from the reactor coolant system. The characterization effort concentrated in the reactor building basement because other TMI-2 areas are decontaminated or believed to be sufficiently explored and characterized.

Figure 5 identifies locations of *in situ* measurements and sample collections made by GPUN in the reactor building basement since the TMI-2 accident. In 1986, these locations included: (a) the concrete wall and floor core borings (using both a robot and remotely operated tools), (b) gamma detector surveys near the basement walls and through the core instrument string calibration tubes to the space below the reactor vessel, (c) gamma spectrometer surveys of the letdown cooler room, and (d) thermoluminescent detector (TLD) string surveys of about 15 locations.

Other characterization work included (a) a TMI-2 AEP-sponsored analysis by a private laboratory (SAI) of two reactor building basement sediment samples collected by robot in 1985, and (b) testing at the INEL of the concrete bores to determine the amount of fission products that could be leached from the concrete by soaking in a reactor coolant (a borated water) solution.

Preliminary findings of the *in situ* measurements and sample examinations conducted during 1986 are:

1. The concrete is the principal source of radiation in the basement with radioactive contamination (cesium and strontium) penetrating throughout the porous concrete block and about 1/4-in. (90% of radioactivity) into the high-density concrete walls and floor, with less contamination in coated (painted) areas.
2. The radioactive contamination of the basement walls is concentrated near the water level (5.5 to 8.5 ft above the floor) in the flooded basement as shown in Figure 6. (A 24.2% correction factor is applied to provide equivalence.)
3. It may be possible to remove most of the radioactivity fission products from the concrete by leaching using a borated water solution.
4. The estimated 5 metric tons of sediment on the basement floor contains only small quantities of core materials (1.7 to 3.2 kg of UO_2).

Characterization of the reactor building basement may continue in 1987 with possible detailed

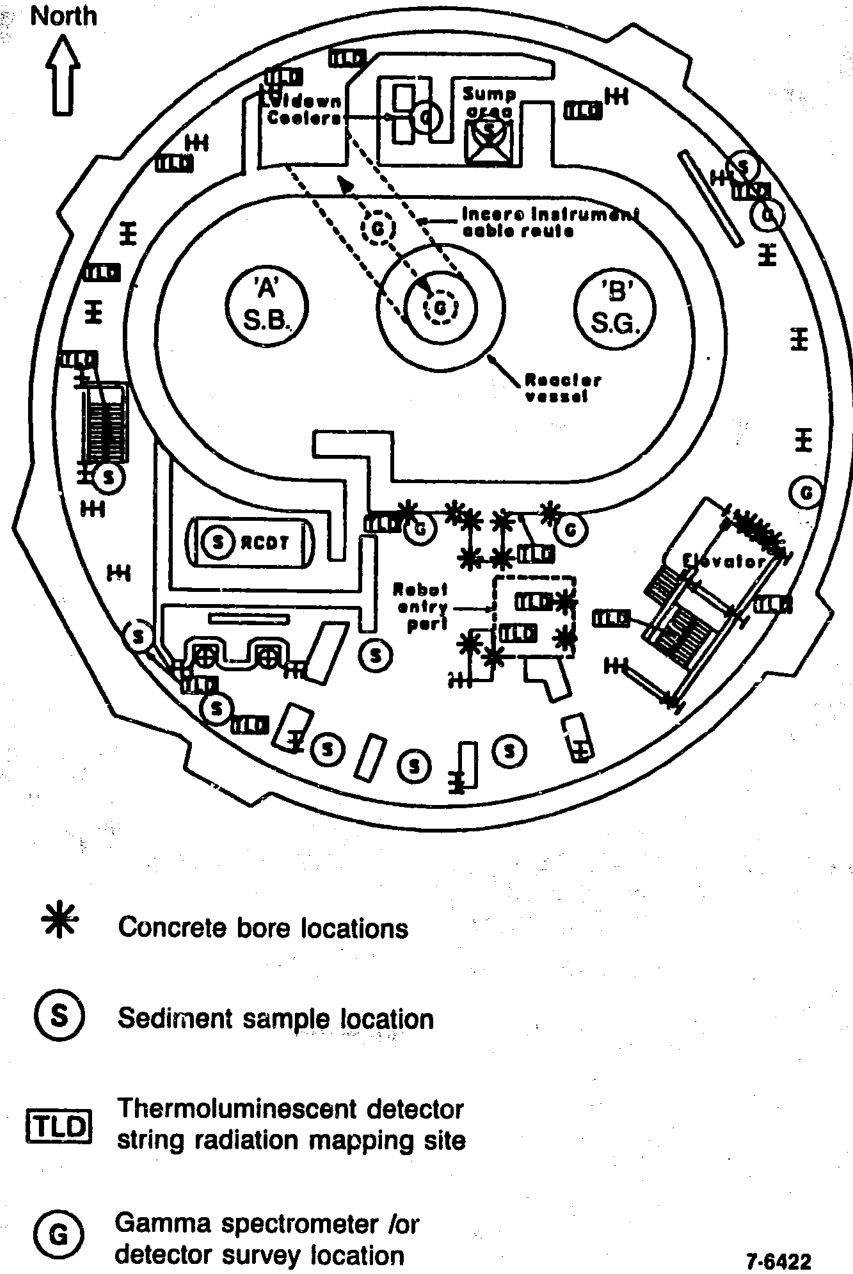

Figure 5. TMI-2 reactor building basement FPI sample locations.

analysis of concrete core samples for fission product quantity and samples from the sludge batches removed from the reactor building basement during the desludging program.

RCS Fission Product Inventory Characterization. In 1986, activities continued in an effort to locate and characterize the core fission products and materials which had been captured in the RCS vessels (except the reactor vessel) and piping. The 1986 activities included: (a) exploration, with video systems, of the pressurizer and steam generator upper head regions to locate loose debris, (b) exploration, with a fiber-optic borescope, of the steam generator

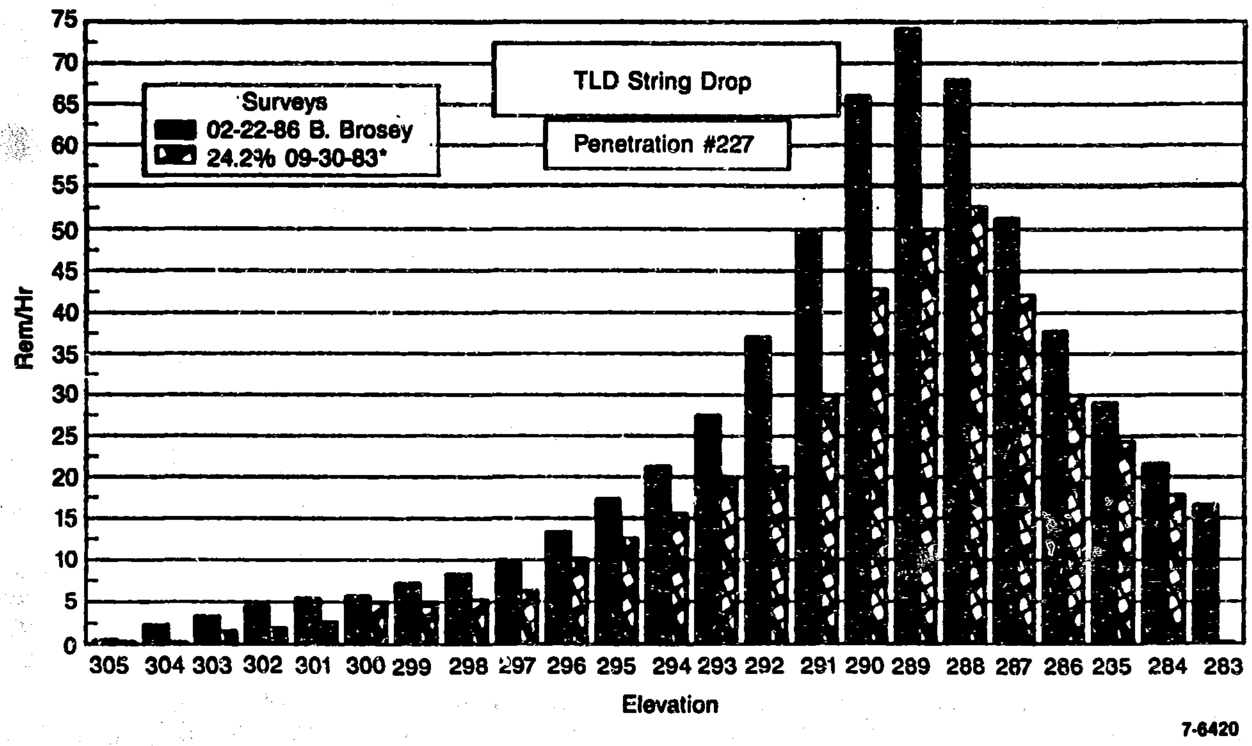

Figure 6. Radioactive contamination near reactor building basement wall.

(SG) tubes and SG lower head and adjacent cold leg piping to locate loose debris, (c) collection of loose debris samples from the pressurizer lower head and the tops of both steam generator tube sheets, (d) acquisition of the manway cover backing plates from the pressurizer and both steam generator upper heads for characterization of adherent surface deposits, (e) analysis of a pressurizer lower head sediment sample by a private laboratory (Westinghouse), and (f) preliminary analysis of the deposits on the manway cover backing plates by a private laboratory (Battelle Columbus Division).

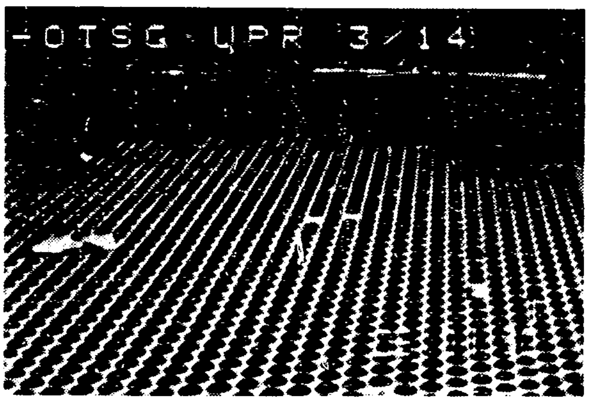

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Figure 7. A-loop steam generator tubesheet top.

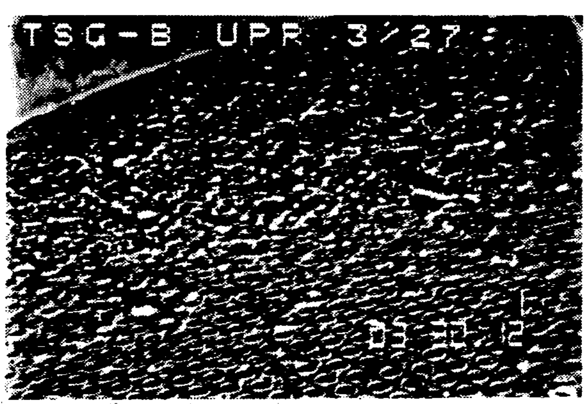

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Figure 8. B-loop steam generator tubesheet top.

Preliminary findings from surface deposit examinations, sample collections and acquisitions, and debris sample examinations conducted in 1986 are:

1. The distribution of loose deposits in the RCS is estimated as follows:

Area	Loose Deposit Quantity and Characteristics
Pressurizer	15 liters of slurry and possible flakes from heater rod surface deposits.
	Particle sizes are 40° greater than 50 microns; 40° in the 50 to 20 micron range; and 10° each in the 20 to 10 and 10 to 5 micron ranges.
A-loop steam generator	0.5 to 1.0 liters of solids on tube sheet top (see Figure 7); some pieces longer than 1-in.; uranium content is low.
	10 to 15 liters of slurry in the lower head.
A-loop cold leg.	20 to 30 liters of slurry.
B-loop steam generator	1 to 4 liters of solids on tube sheet top (see Figure 8) with some particles about 1/2-in. diameter.
	Possible mixture of fuel pellet fragments, prior molten material, and oxidized cladding.
	Uranium content may be less than 70°.
	15 to 30 liters of slurry in the lower head.
B-loop cold leg	40 to 60 liters of slurry.

2. The adherent surface deposits in the RCS have the following features:

   a. Thickness of 1 to 15 microns in the pressurizer and steam generator upper regions.
   b. Fission product retention by the surface deposits is greatest in the B-loop steam generator and least in the pressurizer. The difference is one to two orders of magnitude. Figure 9 shows results of gamma scans across the face of the manway cover backing plates.
   c. The retained fission products are primarily ^{137}Cs and ^{60}Co.

Characterization of the core fission products and materials retained in the RCS will continue in 1987 with completion of the characterization of surface deposits on the manway cover backing plate, private laboratory analysis of the loose debris on top of the steam generator tube sheet, and private laboratory analysis of loose debris collected from other RCS regions.

Reactor Vessel Internals Characterization and Sampling. During 1986, activities included fabrication of special tooling for sample acquisition and examination, video surveys of core material conditions during defueling, and acquisition and examination of samples. The video surveys and sample acquisitions were made more difficult by the biological growth community which developed in the reactor vessel and prevented (a) identification of much of the material being loaded into the fuel canisters after January, and (b) clear video surveys of surfaces and objects exposed by the loose debris removal.

In 1986, reactor vessel exploration was nearly completed, with video surveys and/or sample collections in the core lower region, the lower core support assembly, and the reactor vessel lower head central regions. The reactor vessel internals areas not yet adequately explored are:

- The actual relocation pathways of the molten core material in the northeast, east, and southeast sections of lower core region,
- The northeast, east, and southeast regions of the lower core support assembly where previously molten core material has solidified,
- The central region adjacent to the reactor vessel head where a concentration of nonfuel core material is believed to be deposited.

Special Tooling. Special tooling completed in 1986 included: equipment for unloading TMI-2 fuel canisters at the INEL, a boring machine to collect samples of the fused lower core region material, and a gamma spectrometer scanning instrument for characterizing the gamma-emitter distributions of the TMI-2 core bores and rod and tube sections from unmelted distinct core components.

The fuel canister unloading equipment included the following tool assemblies:

- A machine-tool type device (Figure 10) to remotely translate and rotate the fuel canister and remove and install the canister cap,
- Special tools for extracting distinct core components (fuel and control rod assembly upper ends), small debris, and core bore drill barrels from the fuel canister,
- A video system for observing conditions and extraction tool manipulations inside the fuel canister,
- Special tools for potting and section-cutting rod bundles and core bores,
- Tools for disassembling core bore barrels and fuel assembly upper end fittings, and
- Tools for handling, venting, and draining the fuel canisters.

In 1986, the tooling fabrication and operation demonstrations were completed. In June, the tooling was installed in the INEL TAN Hot Cell, and successfully used to recover core bores and the upper ends of fuel and control rod assemblies from five fuel canisters. Extraction of assembly upper ends was difficult for two reasons: (a) inverted upper end fittings had been forced into the fuel canisters, and (b) small objects had became wedged between the canister walls and the fuel assembly upper end boxes during extraction.

The core boring equipment, completed in 1985, was installed in June on the TMI-2 defueling platform as shown in Figure 11. It was successfully used to extract core material from the lower core region at nine core positions. Attempts to recover core material at three reactor vessel lower plenum region positions using the core boring equipment were unsuccessful due to the absence of core

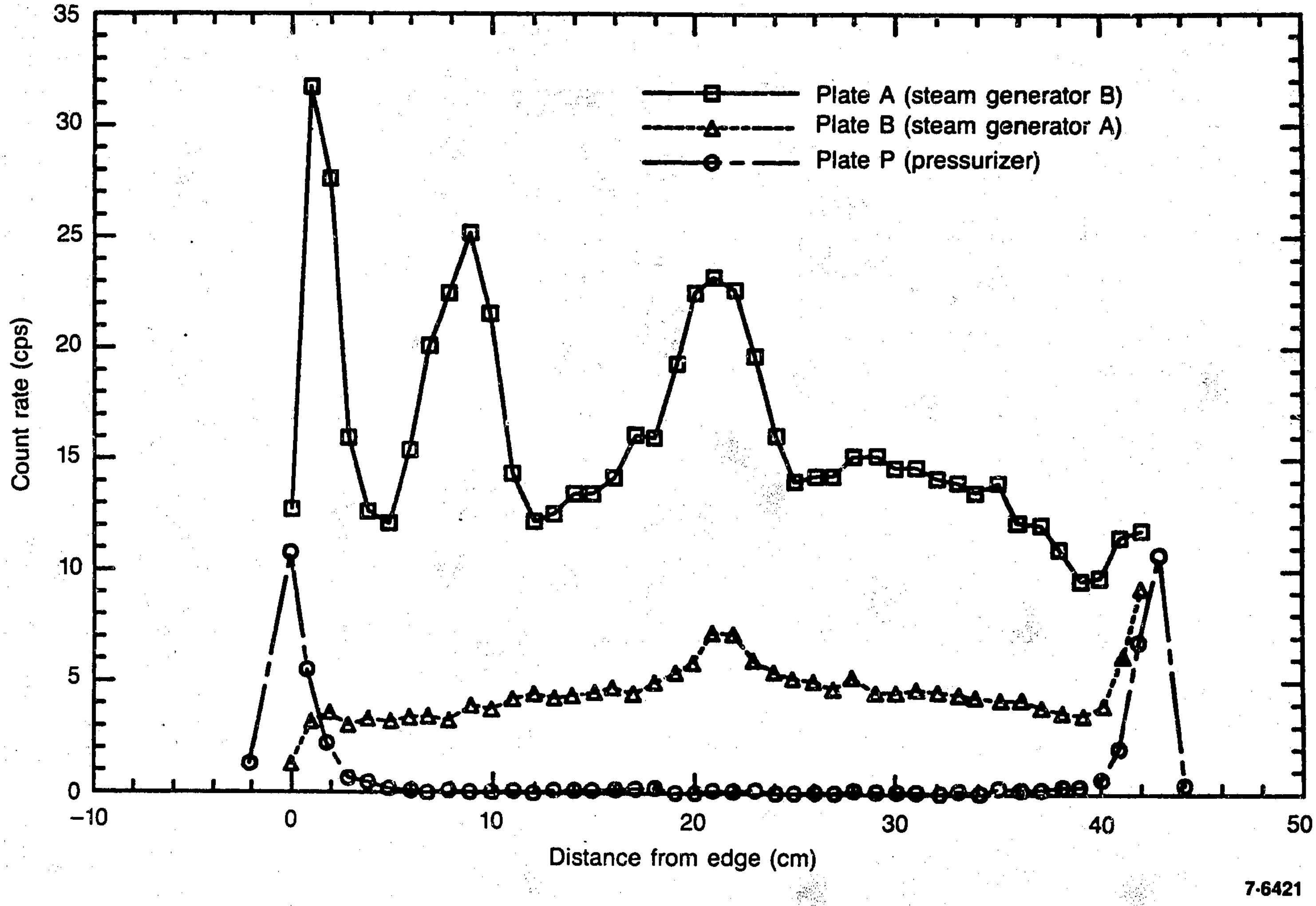

Figure 9. RCS manway cover backing plate surface Cs activity profiles.

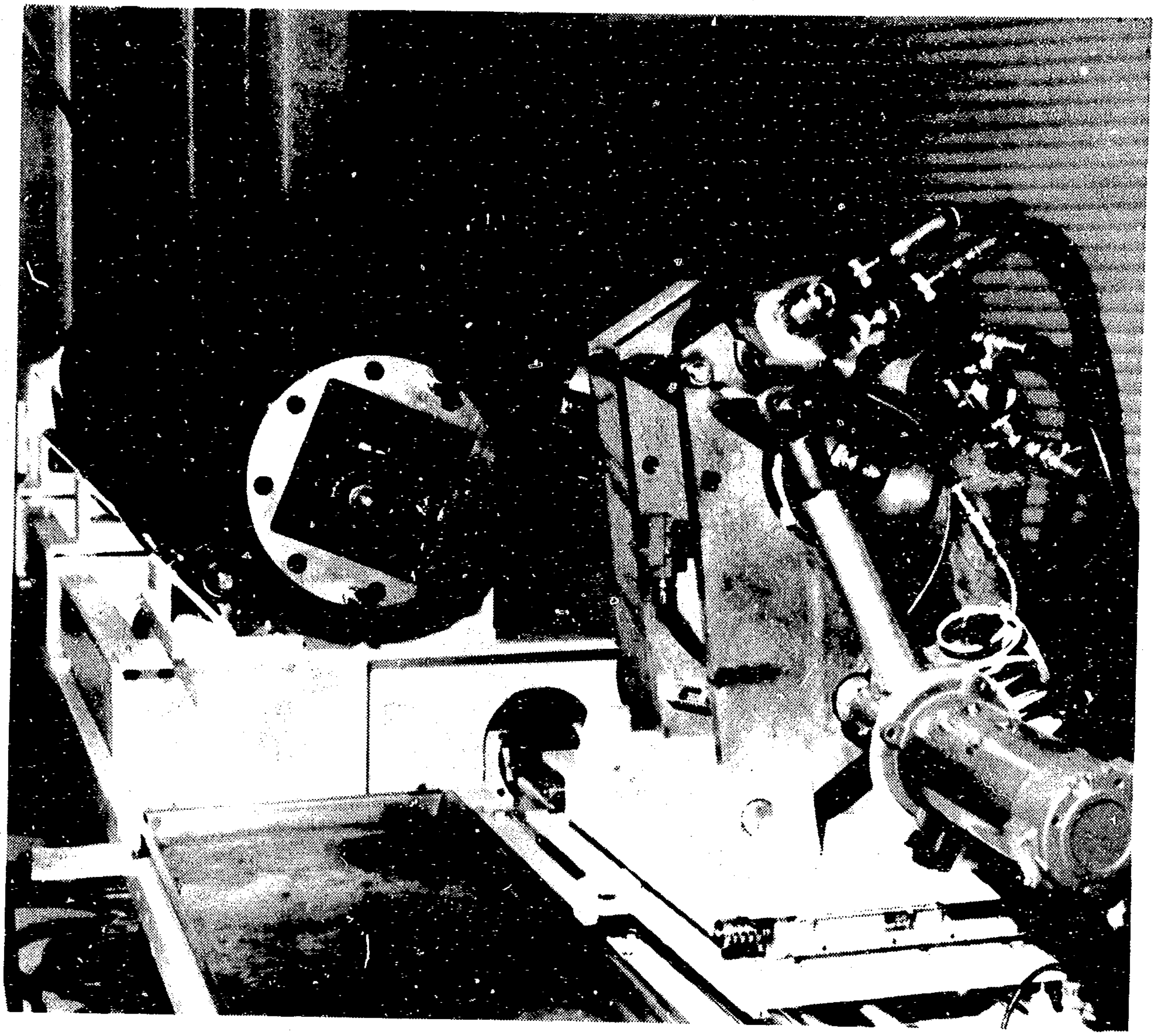

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Figure 10. Fuel canister unloading device.

material in the lower core support assembly and probable loose grain conditions in the core material deposited on the reactor vessel lower head. For unknown reasons, only a small amount of core material was recovered from a tenth lower core region position. Figure 12 shows the 10 core positions that were drilled with the core boring machine. After successful completion of its mission, GPUN used the core boring equipment as a milling device to loosen and/or pulverize the fused material in the core central region. The milling operations consisted of: (a) initially (August) drilling into the 6-ft diameter central core region at 48 locations with a 2-in. diameter solid-faced bit and (b) an expanded (November) drilling of the 8.5-ft diameter central core region at 409 overlapping locations (Figure 13) with a 4.5-in. diameter solid-faced bit. Future use of the core boring equipment is possible to assist in loosening the fused core material in the outer core and the less-well-characterized material in the reactor vessel lower plenum regions.

The gamma scanner equipment (Figure 14) fabrication, operation, and calibration was completed in early 1986. The gamma scanner includes: (a) a computer-controlled, collimated, germanium-crystal gamma spectrometer, and a trolley-and-scanner-bed for positioning the object being

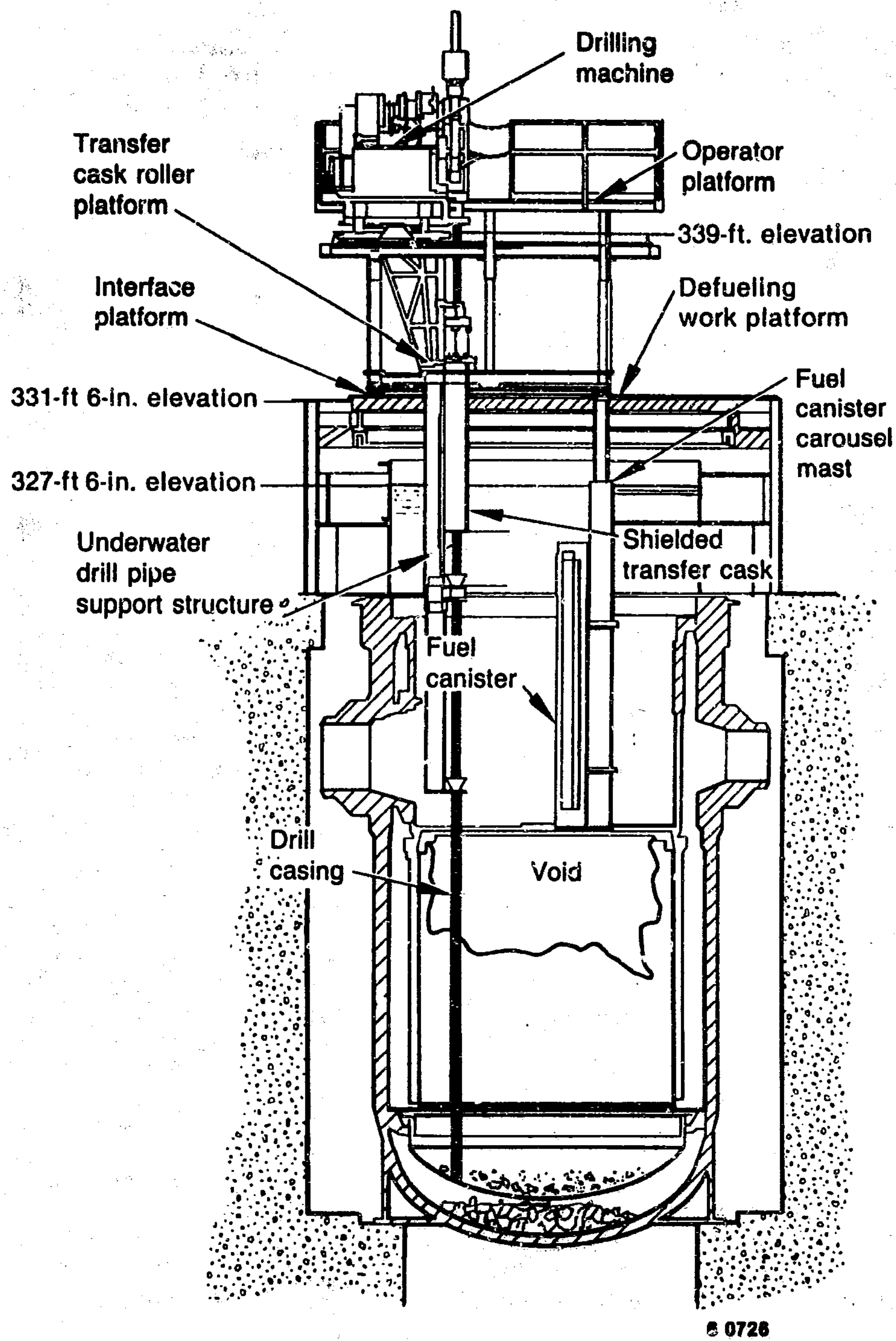
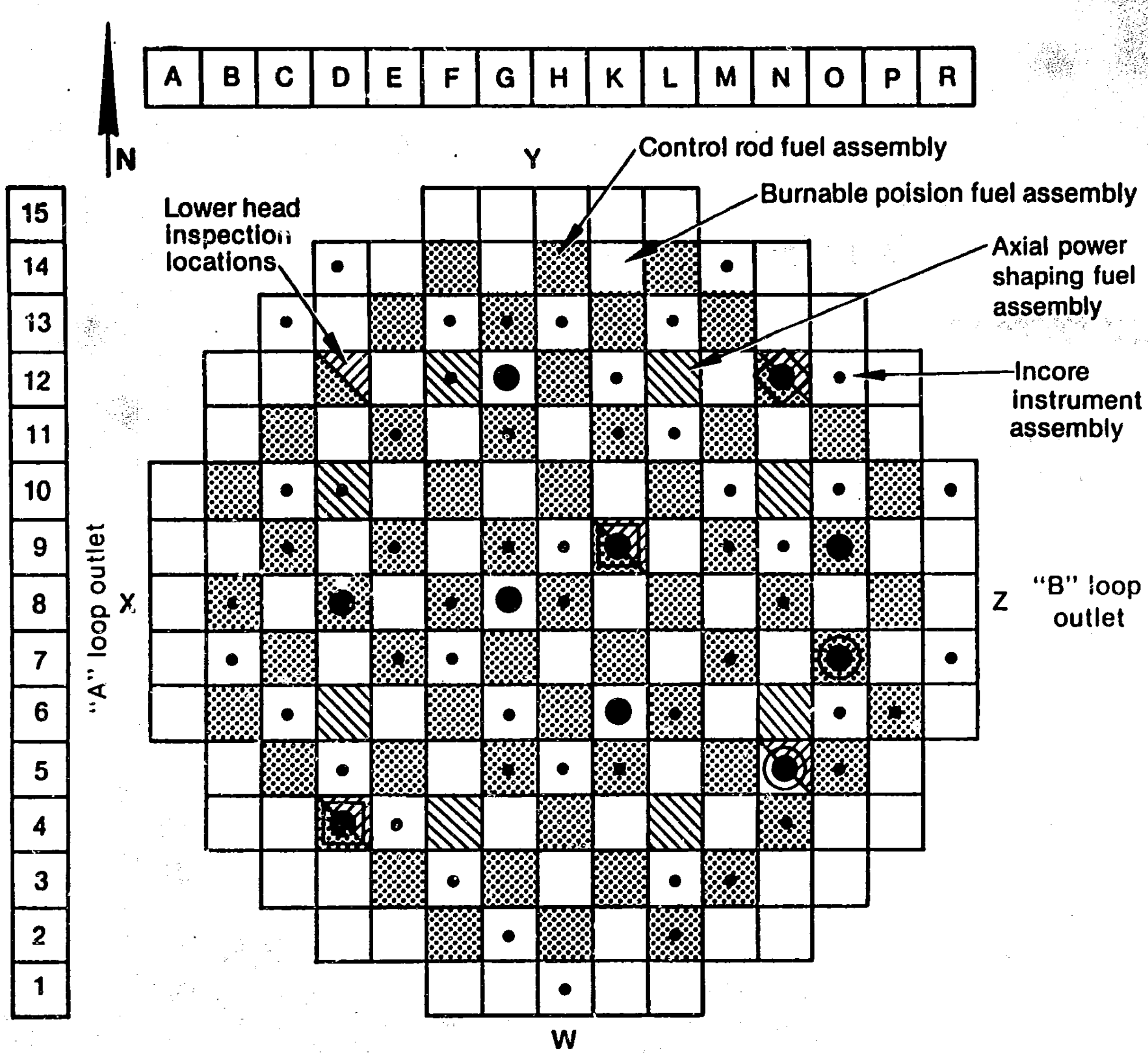


Figure 11. Core boring machine.



Samples from core region; visual inspection of core and core support assembly

Samples of core and lower plenum debris, visual inspection of core, CSA and lower plenum

Samples of core; visual inspection of core, CSA and lower plenum debris

Core samples, attempted CSA samples; visual inspection of core and CSA

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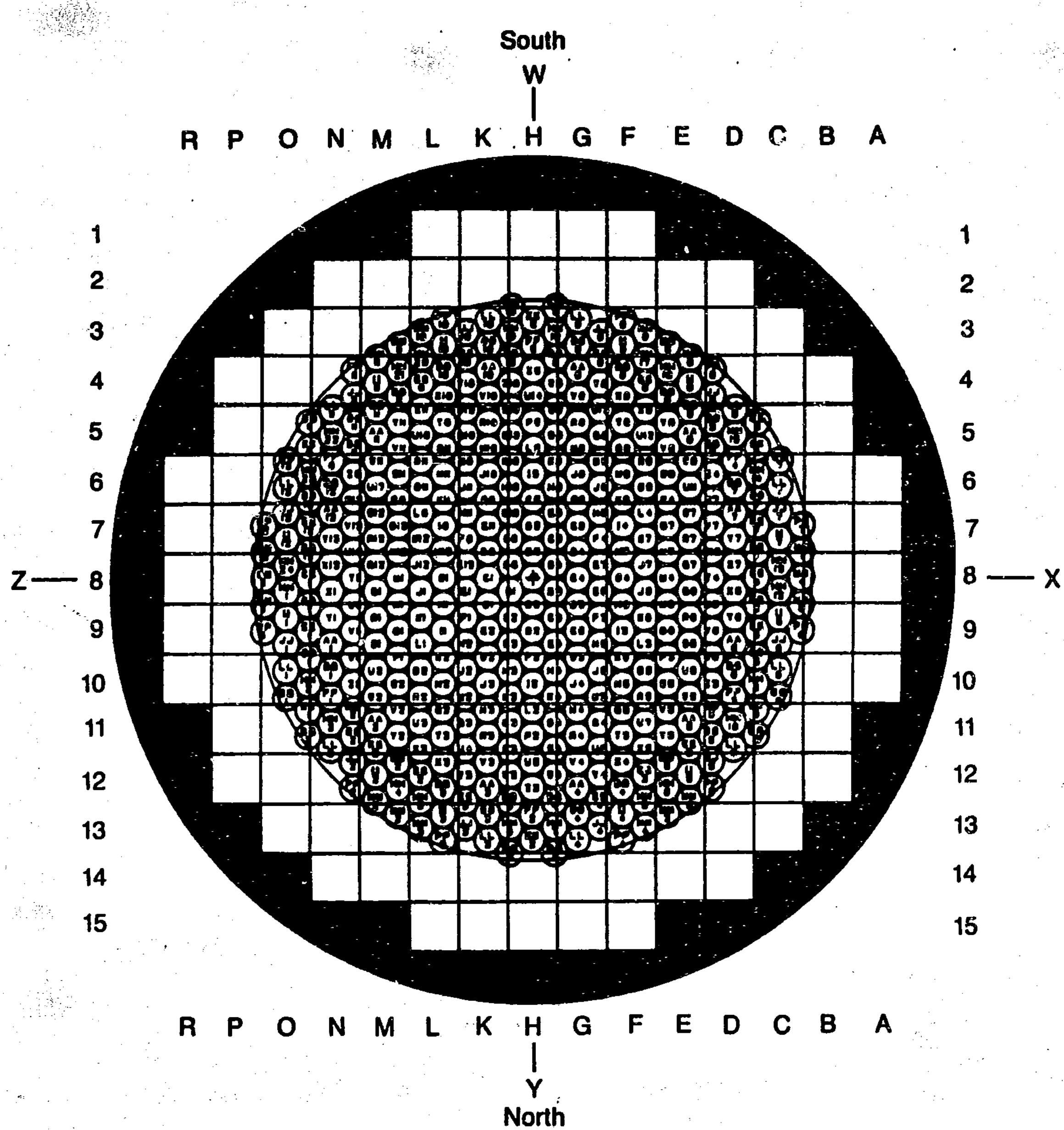
Figure 12. Ten core bore drilling locations.



No ligament 421-holes

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Figure 13. Drilling locations.

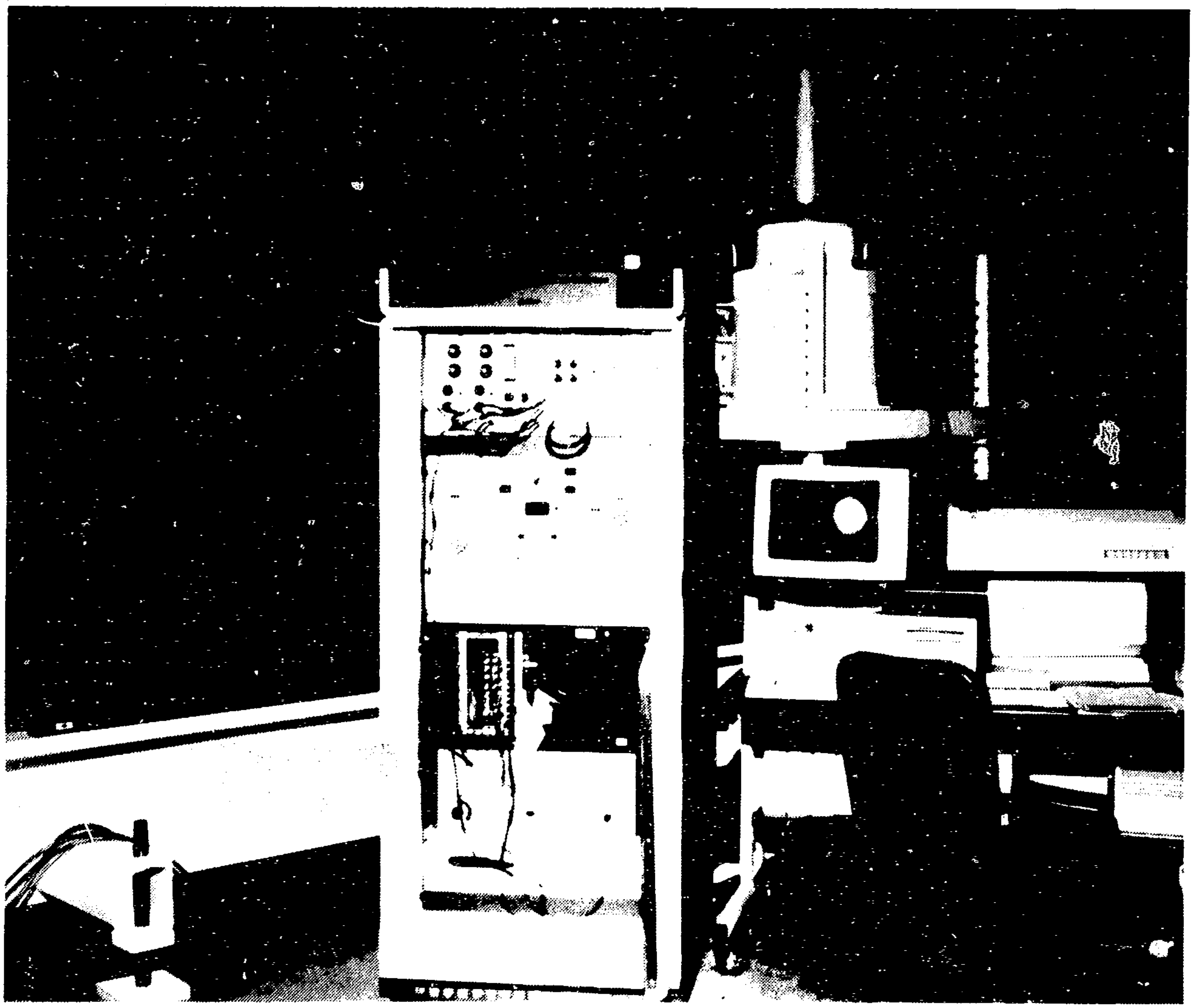

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Figure 14. Gamma A scanner assembly photograph.

examined in front of the collimator, and (b) acomputerized data acquisition system with a programmable multichannel analyzer and spectral analysis computer program software. The equipment was installed in the INEL TAN Area Hot Shop and successfully used to gamma scan fuel rods and control rod/guide tube upper ends recovered from fuel canister D-141 and the nine core bores.

Video Surveys. Video surveys of reactor vessel internals were conducted as follows:

Month	Reactor Vessel Region	Purpose
January	Core cavity floor (east side)	Examine core position P4 (peripheral) fuel assembly which had toppled onto the core cavity floor.
June	Core cavity floor at core bore drill sites	Locate core debris which might interfere with the drilling operation after removal of most of the loose debris between the original core cavity floor and the hard crust.
July	Ten core bore hole sides in lower core and region and exposed regions of the lower support assembly and loose debris on the reactor vessel lower head below the core bore locations	Determine condition of core material in the lower core region and within underneath the lower core support assembly. Locate damage to the lower core core support assembly and in-core instrument string guide structures.
October	Core cavity walls and floor	Determine condition and location of standing fuel assemblies and loose debris before resuming defueling
December	Core cavity walls and floor	Determine condition and location of standing fuel assemblies and loose debris before resuming defueling after the loosening of fused core material and pulverizing using the core boring equipment.

The video survey tape recordings were used at the INEL to produce still-image photographs using the special electronic image enhancement and production equipment described in the 1984 Annual Report. The still-image photographs were used as illustrations in the core-bore-drilling quick-look report published in September and will be used as illustrations in a core cavity wall and floor video survey still-image album and the fuel rod segment Examination Results Report to be published in 1987.

Findings from the 1986 video surveys include:

- Peripheral fuel assembly damage appears to increase significantly below the original core cavity and especially near the observed level of the hard crust encountered below the loose debris.
- A horseshoe-shaped ring of agglomerated core material was located in June between the 100-in. elevation above the fuel bottom and the hard crust level. The ring projected inward from the standing fuel rods as shown in Figure 15 and was uncovered by removal of loose debris underneath the original core cavity floor.
- Sufficient information was obtained in August from the core bore hole video surveys to estimate the overall damage and reconfiguration of the core. Those end state conditions are depicted in Figure 16 and summarized as follows:

Core Region	Percent of Core Material	
Still standing rod bundle geometry	42	
Loose debris (unmelted and previously molten core material mixture) below the cavity in the upper core region (the cavity was 26° of the original core volume)	23	
Previously-molten core material:	35:	
(retained in core boundary)		19
(escaped from core boundary)		16

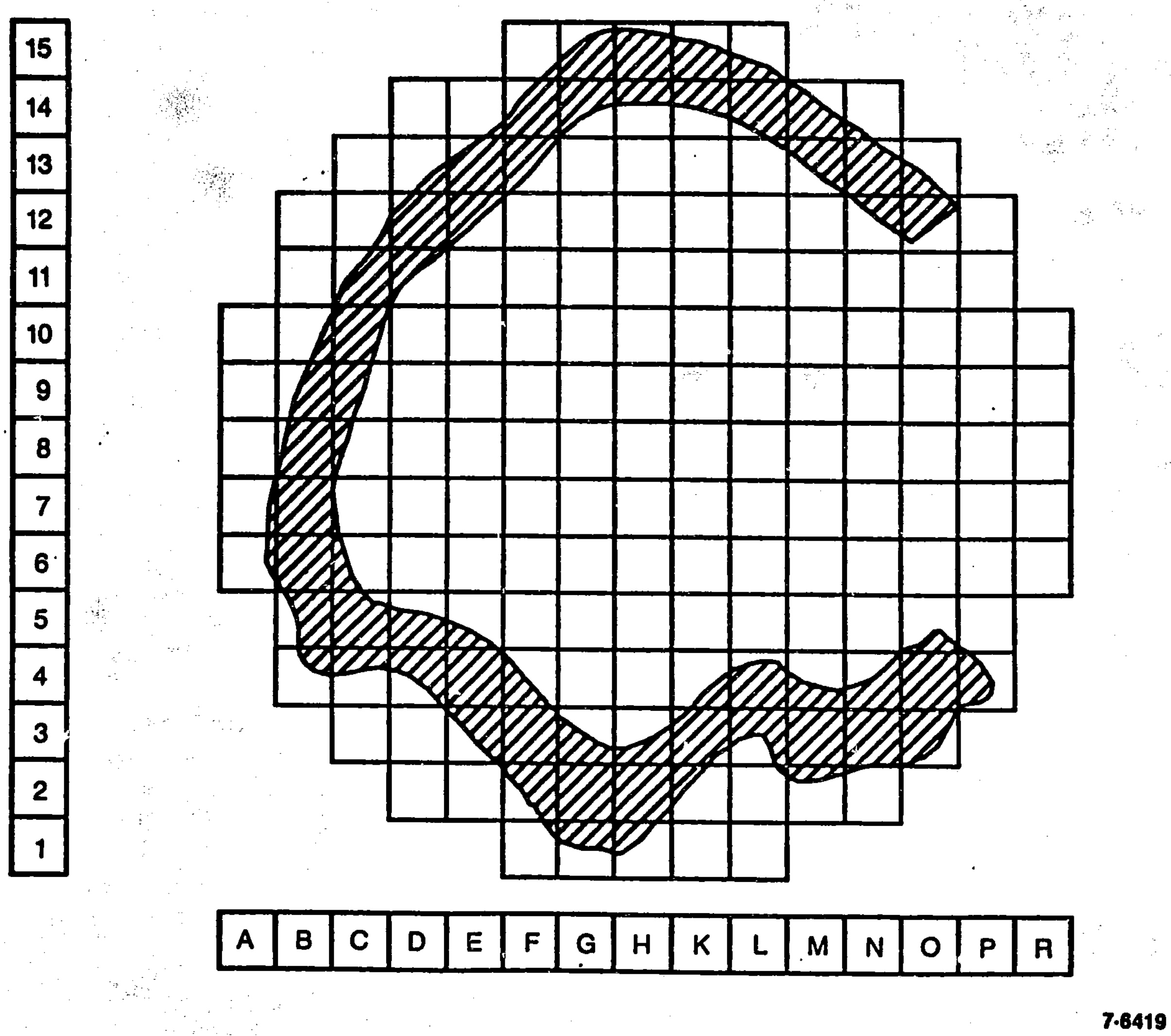

Figure 15. Estimated radial configuration of the upper ridge of agglomerated core material.

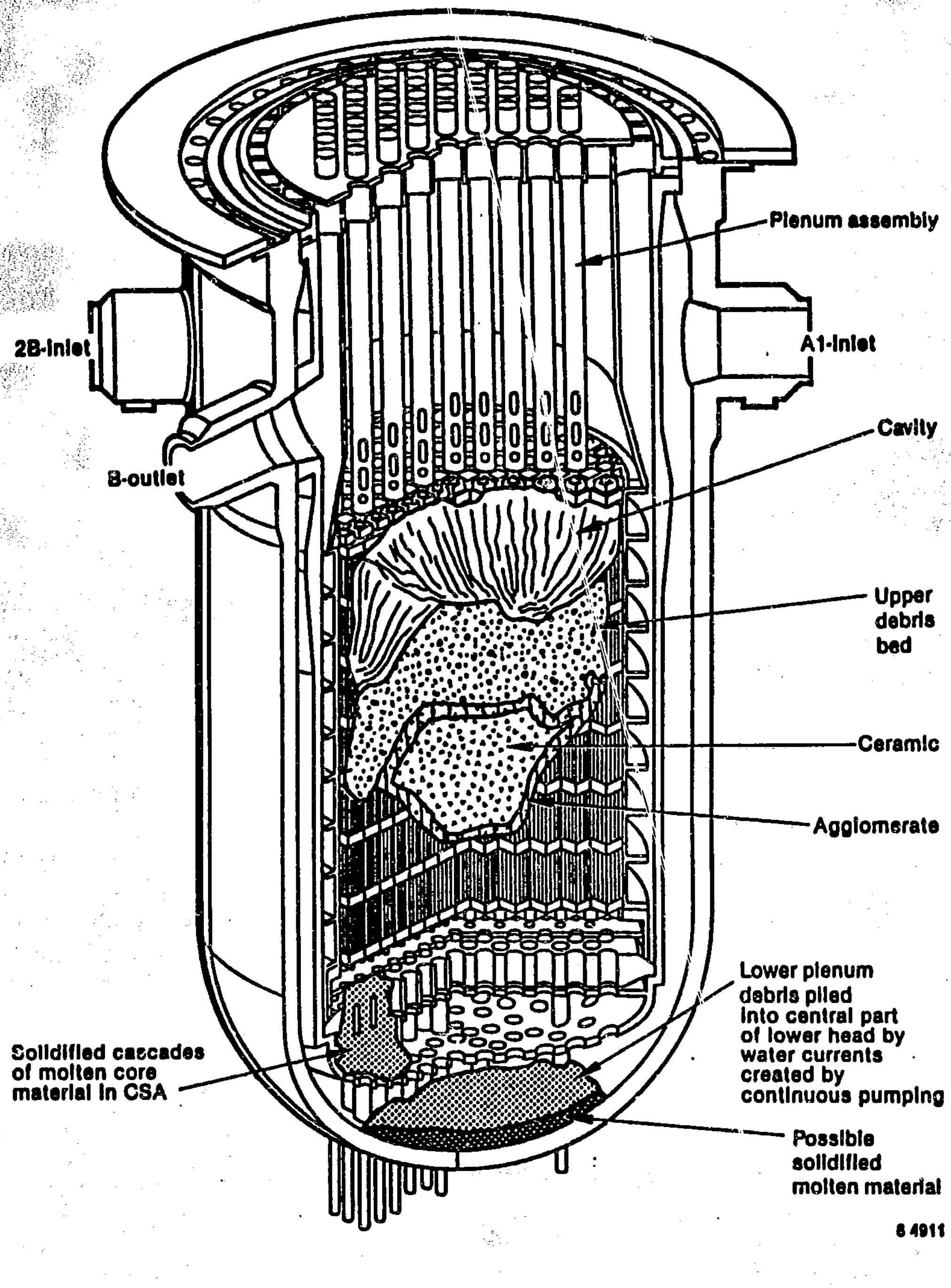

Figure 16. TMI-2 accident end-state core conditions.

- Figure 17 shows the estimated condition of core materials along Row 6 at the time of core boring in July.
- Previously molten core material had solidified only in the east side of the lower core support assembly as shown in Figures 18 and 19. This is the likely location of the relocation path of the previously molten core material.
- The December video surveys revealed large rock and boulder-size objects on the core cavity floor which might be dislodged portions of the horseshoe-shaped ring of agglomerated core material. Also, the core cavity floor was about 60-in. above the fuel rod bottom compared with the original 105-in. level, and the agglomerated ring appeared discontinuous with missing portions in the northeast and west regions.
- There is possible warping of the core's former wall at the corner between core positions P5 and R6.

Continued use of the video recording and image enhancement equipment is planned for the duration of the TMI-2 defueling program. The purpose is to document the condition of the reactor vessel internals which may provide TMI-2 accident sequence information. Assembly of still-image albums showing views of the horseshoe-shaped ring of agglomerated core material and the warping of the core former wall is planned for 1987.

Core Sample Acquisition and Examination Project.

Sample Acquisition. The 1986 sample acquisitions from the reactor vessel internals included: (a) completing the acquisition (shipment to the INEL) of the fuel rod segments from the core cavity wall, (b) retrieving additional samples of the loose debris underneath the original core cavity floor, (c) extracting 32 fuel and control rod assembly upper ends from fuel canisters D-141 and D-153 at the INEL, (d) acquisition of approximately 290 pounds of previously molten core material and rod tube segments from nine positions in the lower core region during core boring, (e) collection of a probable 4-ft long section of the core instrument string from the upper core region, and (f) retrieving at TMI-2 300 lb of previously molten core material after completion of the overlapping hole-drilling in the 8.5-ft diameter central core region. Included in the 300 lb is a possible 200 lb chunk from the horseshoe-shaped agglomerated material ring on the core cavity floor.

Plans for sample acquisition in 1987 include shipment to the INEL of the 300-lb sample of previously molten core material, retrieval of additional core material loose debris from the reactor vessel lower head region, and acquisition of fuel assemblies containing the molten core material from the probable relocation path on the core east side.

Sample Examination. During 1986, reactor vessel sample examination activities included:

- Reports on the H8 leadscrew support tube and the results on the loose debris in the reactor vessel lower plenum region,
- Examinations were also completed of:
  - The RV lower head debris,
  - Non destructive examination of the six fuel rod segments from the sides of the core cavity, and
  - 32 fuel and control rod assembly upper end fittings retrieved from containers D-141 and D-153,
- Preliminary examination was completed of:
  - Fuel and control rod guide tube segments from the upper end of core position C7,
  - 6 additional loose debris samples from the core cavity floor, and
  - The material retrieved from the lower core regions by the core bores.

The report by Battelle Columbus Laboratories entitled *Examination of the Leadscrew Support Tube from Three Mile Island Reactor Unit 2*, GEND-INF-067, was issued in March. The leadscrew support tube (LST) came from the center of the dome-shaped region between the plenum assembly and the reactor vessel head. The report presents results of removal, separation, and dissolution of the surface deposits, chemical analysis by secondary ion mass spectrometry, electron spectroscopy, inductively coupled argon plasma spectrometry, and x-ray diffraction techniques, and additional deposit studies using metallographic and scanning electron microscopy techniques. The results indicate the following:

- Some of the surface deposits were found to be loosely adherent and easily transferred

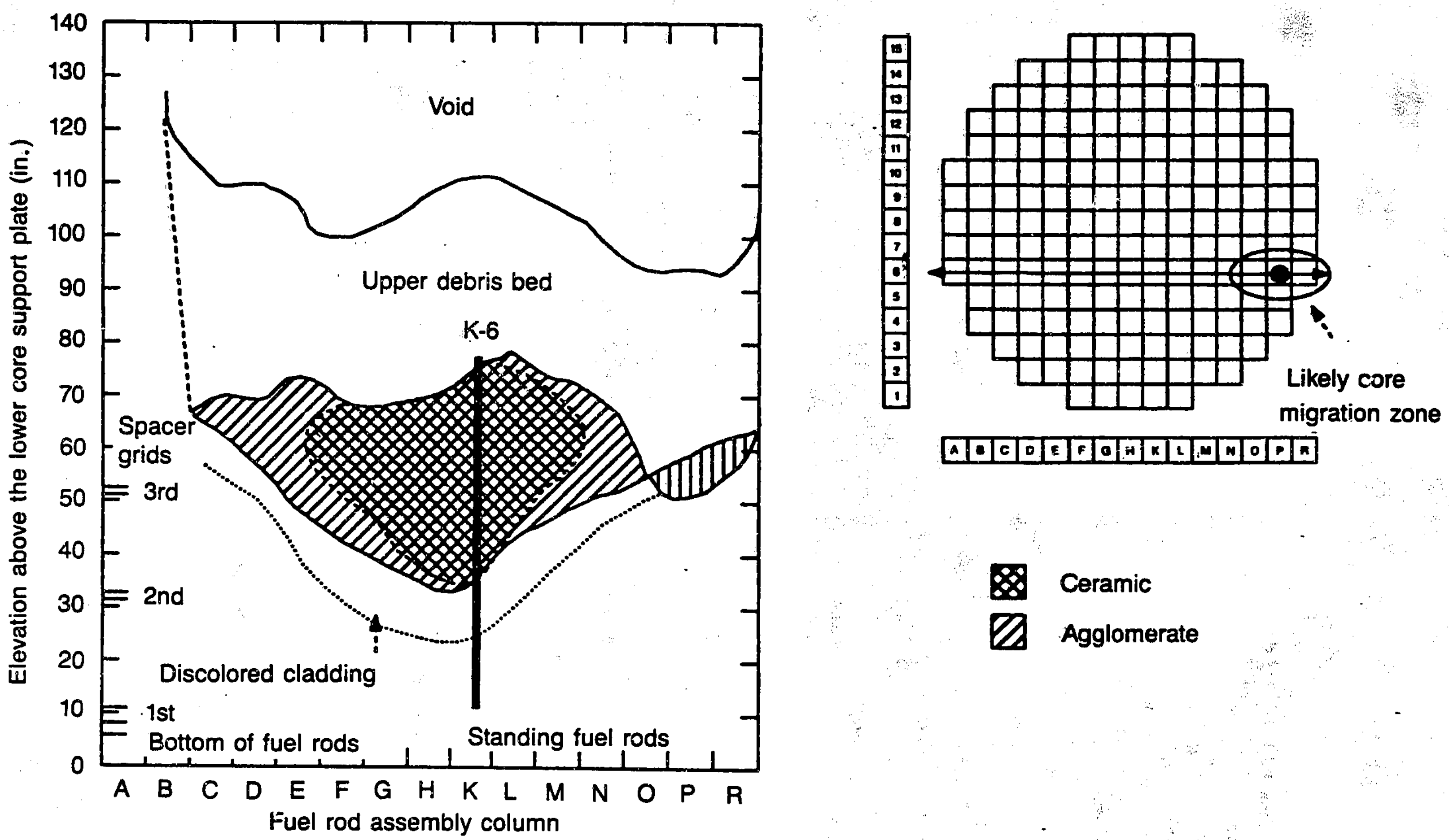

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Figure 17. Core cross-section—Row 6.

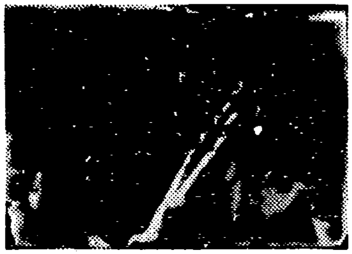

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Figure 18. Looking north from core position N12 at previously molten core material underneath the lower grid flow distributor below core position N13 with core-boring-produced fuel rod shard in foreground.

to materials coming in contact with the LST section. Metallographic examination of specimens from the section showed the presence of two separate and identifiable layers; loosely adherent and tightly adherent. The loosely adherent deposit on the inner surface appeared to be more porous and less uniform in thickness than that on the outer surface. The outer surface deposits exhibited numerous bright, metallic particles, while the inner surface deposits had few or none.

- The metallic particles observed at the bottom end of the section and in the outer surface deposits contained varying amounts of Ag, In, and Cd.
- Most of the inner and outer surface deposits appeared to contain iron and chromium in varying amounts. Very few isolated particles were observed containing U, Zr, or Sn. Chemical analysis by Electron spectroscopy showed: (a) the presence of Fe, Ag, In, Cd, and B in the outer surface deposits, (b) iodine was undetectable, (c) traces of cesium and tellurium were observed, and (d) the inner surface deposits showed only Fe, Cr, Ni, and O. Results from secondary ion mass spectroscopy showed the presence of Cs, I, U, and Zr. There appeared to be no systematic variation in concentrations of these elements with respect to the depth profile.
- The microstructural appearance of the base metal of the LST section showed carbide precipitates in the grain boundary. It is believed that the LST may have experienced temperatures in the range of from 510 to 732°C (950 to 1350°F) during the accident.
- Results from the examination of the LST section are, in general, consistent with those from the leadscrew examinations. The total mass of fission products deposited on the LST surface is a very small fraction of the mass of fission products released from the core. The extent of deposition may have been higher during the accident; if so, it has been reduced by subsequent washing.

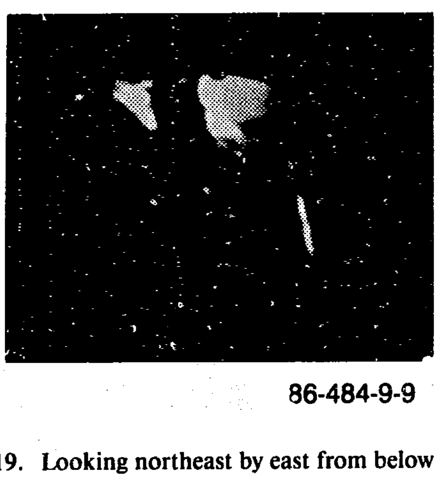

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Figure 19. Looking northeast by east from below core position O7 at previously molten core material flow underneath the lower grid flow distributor.

The upper core region loose debris examination report, entitled *TMI-2 Core Debris Grab Samples—Examination and Analysis, Parts 1 and 2*, GEND-INF-075, was issued in September. The report presents results of the examination of the core debris grab samples, including physical, metallurgical, chemical, and radiochemical analyses. The results indicate the following:

- Some regions in the core may have reached at least 3100 K during the TMI-2 accident,
- Fuel melting and significant mixing of core structural material occurred in the molten region, and
- Large fractions of some radionuclides (e.g., ^{90}Sr and ^{144}Ce) were retained in the core.

A significant amount of material showed no restructuring, indicating temperatures below about 2200 K.

The INEL/ANL-E investigations of the reactor vessel lower head region loose debris samples included a series of physical, metallurgical, and radiochemical examinations. The physical characterization measurements included porosity, dry weight, immersion density, and the associated radiation field. Mechanical properties measurements (e.g., drilling properties and crushability) were performed to support the reactor defueling effort now being performed by GPU Nuclear. The metallurgical examinations (e.g., grain size, composition, oxygen content) were performed to characterize the debris using optical metallography, scanning electron microscopy with energy dispersive x-ray analysis, Auger electron spectroscopy, microprobe analysis, and x-ray diffraction analysis. The radiochemical measurements included gamma spectroscopy, neutron activation analysis, liquid scintillation analysis, and delayed neutron production. Preliminary observations and results of the examinations are:

- Material is inhomogeneous, porous and cracked, with average density of 7.2 g/cm^3.
- Elemental composition includes uranium (70°), zirconium (20°), iron, aluminum, chromium, and nickel.
- Oxidized chromium and iron are found in the grain boundaries as a fine dendritic structure.
- Metallic inclusions in the fuel are mixtures of nickel and tin, nickel and silver, and nickel and ruthenium. Some isolated particles of silver are also found.
- Debris in the lower head is a mixture of stoichiometric uranium and zirconium oxides.
- Substantial amounts of ^{137}Cs are retained in the lower reactor vessel debris.
- ^{125}Sb and ^{106}Ru are retained to some extent in the debris, probably as metallic nodules.
- Lanthanides are retained almost entirely in the fuel material and are not released to the reactor coolant system.

The six fuel rod segments obtained from partially intact fuel assemblies still standing in the core were subjected to neutron radiography, gamma spectroscopy, and visual and photographic examinations, to evaluate the necessity of performing additional metallurgical and radiochemical analyses. Preliminary observations and results of the examinations are:

- Neutron radiographs of the segments show (a) some evidence of possible pellet cladding interaction, and (b) TMI-2 fuel rods used zircaloy sleeve spacers between the pellet stack hold-down springs and fuel pellets instead of ZrO_2 spacers.
- The gamma spectroscopy analyses indicate that retention of the radionuclides agrees with predicted values within the uncertainty of the analysis.
- The fuel rod segments were only slightly damaged during the accident sequence and are not suitable for further destructive examination except for use as possible comparison samples in other TMI-2 regions where fuel rod regions were not extensively damaged.

The examination of the fuel and control rod assembly upper end fittings in 1986 consisted of obtaining side and end-view photographs of each piece retrieved from fuel canisters D-141 and D-153 through an INEL hot cell periscope. Figure 20 shows the damage to the core center (core position H8) fuel assembly upper end fitting, Figure 21 shows the damage to a typical core periphery (core position H1) fuel assembly upper end fitting rod, and Figure 22 shows the damage to a typical core mid-radius fuel assembly upper end fitting (core position N9). The TMI-2 accident information that can be derived from photographs is limited. However, the photographs show that the upper end fitting of the core component represents the upper boundary of core material melting during the TMI-2 accident and that local jetting of hot gasses may have occurred at the core exit to cause the localized region of missing upper end fitting material.

Preliminary examinations were conducted of fuel and control rod segments from the upper end of the core position C7 fuel assembly which was recovered from fuel canister D-141 in August. The examinations included neutron radiographs, gamma spectrometer scanning, and metallographic analysis. The preliminary findings from the examinations conducted were that the conditions of the core materials was as expected for exposure to severe fuel damage event; except, the silver-indium-cadmium poison material had relocated upwards while molten into the hold-down spring region, indicating possible control rod cladding collapse during the core heatup phase. Figure 23 shows the neutron radiograph of the silver-indium-cadmium material solidified in and around the coil spring in the control rod upper region. Metallographic analysis confirmed the material was previously molten silver-indium-cadmium.

The examination of the six samples of loose debris retrieved from the core cavity floor

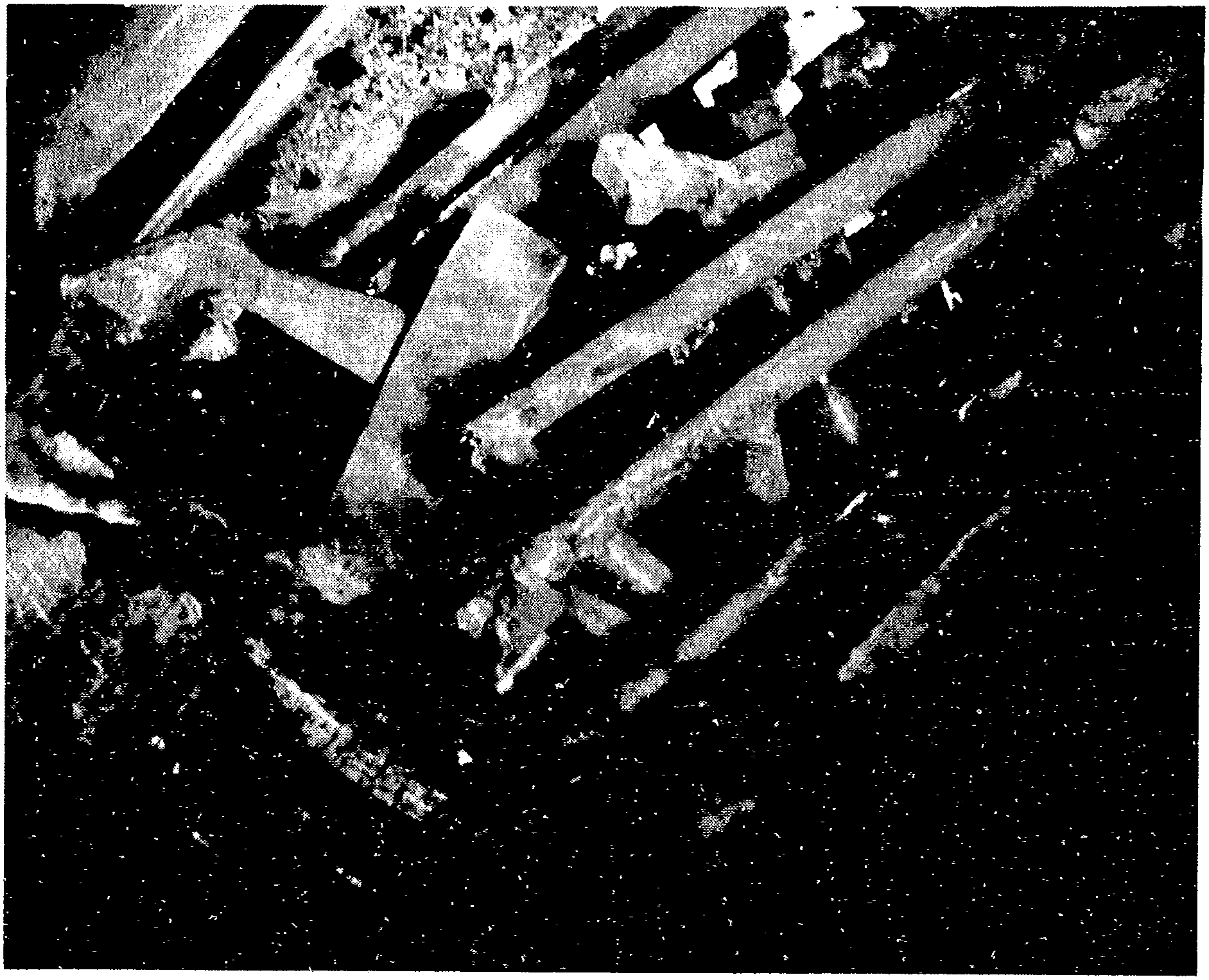

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Figure 20. Southwest corner of core position H8 control rod fuel assembly upper end fittings.

commenced in May. Radiochemical analysis, with emphasis on determining the ^{129}I abundance, was underway at year-end at the INEL.

The preliminary examination of the lower core region material retrieved by the core bores commenced in September. The preliminary examinations involved photography of the core material collected in core bore drill tubes, gamma spectrometer scanning of the core barrels, weighing the contents, and density measurements of the pieces of previously molten core material. The photographs and gamma spectrometer scanning indicated that the core boring partially recovered the material encountered in the lower core region and confirmed that no material from the reactor vessel lower plenum region was recovered. The core boring recovered nearly 300 lbs of material from the lower core region including: (a) eight 2.4-in. diameter cores and 100 rock-size pieces from the region of previously molten core material, and (b) 120 lower ends of fuel rods, control rods, burnable poison rods, guide tubes, and instrument tubes from below the previously molten core material region. The loss of core material from the core bores provided an indication that the ceramic and agglomerated core material regions would be frangible by milling-type tools. Density measurements were made on all eight 2.4-in. diameter cores and 34 of the approximately 100 rock-size (greater than 1-in. in some direction) pieces of previously molten core material. The average density for the rocks is 7.7 g/cm^3 with density extremes of 5.4 and 9.4 g/cm^3, compared with the eight 2.4-in. diameter cores which have an average density of 7.9 g/cm^3 with density extremes of 6.8 and 9.7 g/cm^3. The higher density, previously molten

86-384-5-1

Figure 21. Bottom and northside view of core position H1 fuel assembly upper end fitting.

core material appears to be from regions where undissolved fuel pellets are cemented together by previously molten core material which is frequently metallic-appearing.

Examination plans for samples from the reactor vessel internals in 1987 include the following:

- Completing the report on the six fuel rod segment nondestructive examinations,
- Reporting on the examinations of the contents (upper core region distinct components) of fuel canisters D-141 and D-153 and the six additional loose debris samples from the core cavity floor, and
- Completing the examination of the lower core region material retrieved by both the core bores and by the defueling tools after completion of the overlapping hole drilling.

86-384-3-6

Figure 22. Bottom and northside view of core position N9 burnable poison rod fuel assembly upper end fitting.

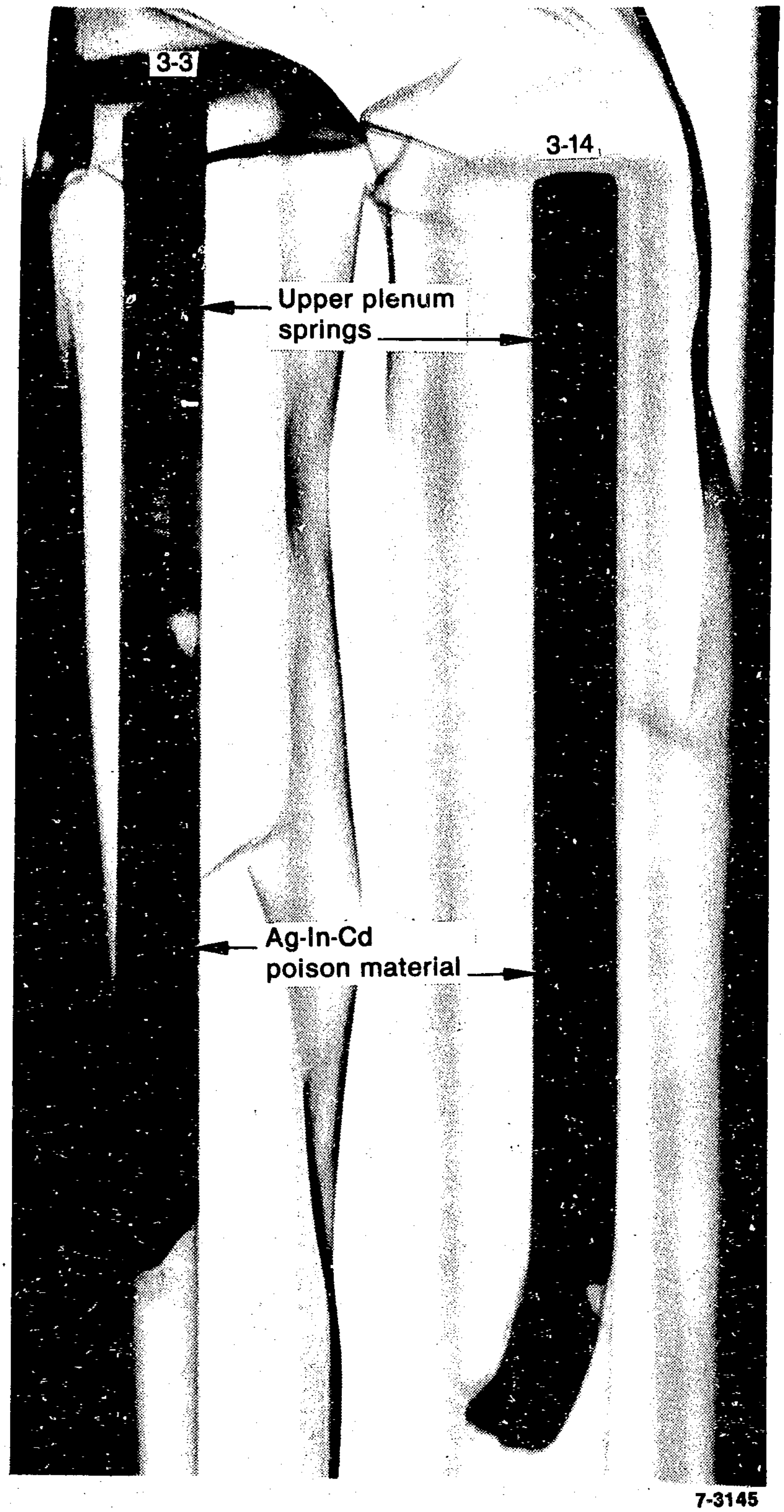


Figure 23. Neutron radiographs of core position C7 control rod samples 3-3 and 3-14 upper ends.

Defueling Operations

The first canisters of fuel were transferred from the Reactor Building to the "A" Spent Fuel Pool in January 1986. During the year, 60 canisters containing 60,150 lb of core debris were transferred. This is 20.5% of the total postaccident core materials to be removed.

In January 1986, the vacuum system was used for the first time. Difficulty was experienced with hydraulically lifting the material from the debris bed due to its compacted condition. In addition, debris too large to vacuum (e.g., end fittings) covered most of the debris bed, making access to vacuumable debris difficult. A water jet system was added to the nozzle to agitate the debris bed during vacuuming. This and other minor modifications led to a loading rate of approximately 3 lb/min. The vacuum system has not been used since it was tested in April.

Water clarity in the reactor vessel continued to degrade from that available at the start of the year. By February, visibility in the vessel was essentially zero; however, pick-and-place operations continued at a production rate which permitted the schedule to be maintained through mid-April. At that time, most of the loose debris on top of a hard layer in the core region had been removed and pick-and-place operations became nonproductive. Defueling operations were suspended to permit water processing operations, including destroying the microbial growth, to proceed to gain visibility so that evaluations of the debris bed could be conducted. By late May, limited visibility was available in the reactor vessel. Pick-and-place defueling operations resumed with little progress. During June, defueling shifted from pick-and-place operations to preparing the debris bed for core drilling operations.

The core stratification operation went smoothly after resolution of initial difficulties with the laser system that indexes the drill to the target location. Based on the visual examinations and computer records of feedback from the drill, the following preliminary conclusions were drawn: (a) the degree of force required in drilling through the core indicates the remaining core material, while containing a significant quantity of resolidified material, apparently is not as exceedingly hard as was once thought possible; (b) the core region consists of loose debris, resolidified material, and apparently intact remains of fuel assemblies; (c) damage to lower CSA components appears to be less than projected; but some damage to components was identified in the eastern side of the lower CSA, where a wall of resolidified material extending from the incore tube support plate to the flow distributor head could be seen - that area had not been observed in previous lower head video inspections and the degree of structural damage cannot yet be determined; (d) less debris was found in the lower CSA than was projected; (e) most debris in the bottom of the reactor vessel appears loose enough to be removed with vacuuming equipment; and (f) during the 1979 accident, the bottom 2 to 3-1/2 ft of the core remained covered with water.

With the completion of core sampling, the hollow drill bit was replaced with a solid-faced bit and the machine was used to drill holes into the core crust region to aid in breaking it up. This latter operation was not as successful as anticipated. After the core boring machine had been removed, the core region proved to be much harder than anticipated. Efforts to break up and remove the debris with long-handled tools such as the heavy duty spade bucket, heavy duty tong tool, hydraulic impact chisel, and the core conditioning tool were unsuccessful.

In September, a decision was made to reinstall the core boring machine and rubblize the core debris using a solid-faced drill bit. Although hampered by poor visibility, loose upper end fittings which would interfere with drilling operations were removed from the surface of the debris bed. The end fittings were fused together and were too large to be inserted into fuel canisters, so they were placed in shielded drums filled with borated water and stored on the 347 ft 6 in. elevation in the Reactor Building. Final disposition of these end fittings will be addressed later. Having cleared the debris bed of drilling interferences, the core boring machine was reinstalled on the shielded work platform in October.

Drilling operations began in late October using a 4-1/2 in. solid-faced drill bit containing an array of synthetic-diamond, tungsten carbide inserts. A 19-hole overlapping test pattern was first drilled, successfully demonstrating that the resolidified material in the debris bed could be broken up by this method. Examinations of the test pattern revealed that the fragmented area apparently subsided after drilling. Tests with long-handled tools demonstrated that the fragments of resolidified material could be easily loaded into defueling canisters. After the tests, the drilling machine was centered over the debris bed and production drilling began.

Core drilling operations were completed in mid-November. Using the core stratification drilling rig and solid-faced drill bits, a total of 409 holes were drilled in the resolidified material at the center of the core debris bed. Eleven of the planned drill holes were not made in order to leave potentially solid samples for later extraction. The drill penetrated to a depth of between 1-1/2 and 4 ft, and across a 50 in. radius from the core center. Approximately 14 in. of undrilled material, including peripheral fuel assemblies, remain surrounding the circumference of the drilled area.

At the completion of drilling operations, the drilling rig was removed and the defueling equipment was reinstalled on the shielded work platform. Defueling operations resumed with core topography and video surveys. Because visibility was limited to a few inches, the video survey was inconclusive. As a result of the topographical survey, some of the previous mass was unaccounted for. It is thought that this "missing" material fell between the stubs of fuel rods under the damaged area and, possibly, into the lower head.

Several large samples weighing 100- to 200-lb each were extracted with long-handled tools and placed in special baskets within a fuel canister. Additional sampling and fuel removal operations were hampered by a lack of visibility and by hard stops encountered when the spade bucket was used on the debris bed.

At the end of November, defueling operations were suspended to permit a concentrated effort to attain water clarity in the reactor vessel. Visibility obtained as a result of this effort permitted improved video surveys and topographical maps to be made in early December. The results indicated that the core drilling operation performed in October and November was not completely successful in breaking up the resolidified material into easily removable pieces. A number of rocks exceeding one ft in diameter were also identified; most of these are believed to have fallen in from the higher, undrilled peripheral region that surrounds the central, drilled area. This peripheral region consists of undrilled resolidified material and standing fuel assembly elements. In addition, several broken drill strings were located lying on or imbedded in the drilled surface of the debris bed.

The majority of the drill strings were loaded into canisters and attempts were made to load the loose debris. Given the limited visibility, the crust impact tool could not be used efficiently enough to break up the rocks; other methods for breaking them into smaller pieces are being developed. Although much of the smaller, loose debris proved difficult to pick up with long-handled tools, some areas of more accessible rubble at the edge of the drilled region were located and several canisters were loaded using the small spade bucket.

CSA and Lower Head Defueling

In September 1985, a Technical Plan for defueling the CSA was issued as the primary guidance document for CSA defueling tooling development. Based on this plan, EG&G and GPUN established a Scope of Work for development, design, and purchasing of tools to defuel the CSA.

As a result of core region defueling experience and additional inspections in the lower head, the plan was modified to integrate CSA and lower head defueling in April 1986. This approach anticipated that large size deposits of very hard material would exist in the CSA and lower head, requiring removal of a large portion of the lower CSA. Also, a CSA Defueling Working Team was formed to evaluate tooling concepts and EG&G performed tests using plasma arc, thermic rod, and abrasive water jet cutting tools.

A set of tools was selected for detail design and procurement based on the evaluations and tests. The primary cutting method selected is plasma arc. The abrasive water jet (already provided for core region defueling) would serve as a backup method. A cavitating water jet tool was selected for potential use in cleaning stainless steel surfaces and breaking up large pieces of core debris. Other mechanical tools, including abrasive wheels, flushing, and vacuum tools were also selected for development.

A Request for Proposal was issued in May 1986 for a cutting system, based primarily on plasma arc. Bids were received in June 1986. Two proposals stood out, one from Power Cutting, Inc. (PCI) and one from Ocean Systems Engineering (OSE). PCI was selected to develop the tooling to deploy the plasma arc torch, and to develop the plasma arc technology for use at the water depths at TMI-2. OSE was selected to develop a dual, remote manipulator system with mechanical tool end effectors. The manipulators would maneuver tools and lift and relocate debris.

A parallel development effort was started for lower head defueling based on an anticipation that the debris in the CSA and lower head may be easily flushed and broken. EG&G has begun the tool design and manufacturing effort. This approach utilizes the core boring machine to cut holes through two stainless steel plates in the CSA to provide access for a vacuum/flush tool to the lower head.

Evaluations of methods and development of tools to defuel the upper CSA (the core former plate region) will be performed in 1987.

Ex-Vessel Fuel Removal

Engineering work for ex-vessel fuel removal started in the third quarter of 1986. Three system designs were initiated and completed in 1986 for ex-vessel fuel removal: Pressurizer Spray Line Defueling System (PSLDS), Pressurizer Defueling System (PDS), and Plenum Defueling System (P/DS).

The PSLDS uses the existing Defueling Water Cleanup System (DWCS) as a source of water. The DWCS is connected to a temporary tie-in to the Pressurizer Spray Line, and the fuel in the spray line is flushed to the pressurizer and to the RCS cold leg 2A. The hose for the PSLDS has been routed in the reactor building and, in 1986, preparations were started for an early 1987 flushing evolution.

The PDS uses new hose runs and portions of the existing DWCS as a flow path for water from the pressurizer. Water is pumped from the pressurizer using a submersible pump, through a knockout canister and the DWCS filter canisters, and is returned to the pressurizer through a spray nozzle. The spray through the nozzle is used to agitate the fuel in the pressurizer and suspend it in the water so that it may be pumped out of the pressurizer. In the event that the DWCS filter canisters are not operable, water in the pressurizer containing fuel fines is pumped to the reactor vessel. Equipment is presently being prepared for a planned pressurizer defueling evolution in 1987.

The P/DS may be used to flush fuel fines/debris off of the plenum and onto the fuel transfer canal (FTC) floor. Water in the FTC is filtered and pumped through a high pressure pump, and is returned to the FTC through a spray nozzle. This high pressure spray is used to flush any fuel fines/debris off of the plenum. After the flushing is completed, the fuel fines/debris is vacuumed from the FTC floor. The fuel fines/debris are filtered from the water before returning to the FTC. Design of this system is complete for possible use near the final stages of defueling.

Radioactive Waste Technology

Processing Through EPICOR and SDS. One of the more important contributions of the TMI-2 R&D program has been the high-level radioactive waste technology developed at the national laboratories. From the standpoint of volume reduction, the use of the EPICOR system reduced the volume of radioactive waste by a factor of 10 and the submerged demineralizer system (SDS) reduced volume by a factor of 500 over conventional waste processing systems. Additionally, through DOE-sponsored research on the storage and final disposition of these wastes, techniques are now available to process, store, and dispose of large amounts of radioactive waste, especially radioactive cesium. The disposal of heavily loaded and highly radioactive zeolites from TMI-2 will assist the industry in addressing similar problems at operating nuclear power plants. With a strong attempt by low-level-waste disposal sites to reduce the acceptable volume to those sites, the utilities are looking at resins and other absorbent materials that have increased capacity for radioactive waste. The result would be to decrease the volume of low-level waste delivered to the waste sites by significantly increasing the radioactive loading on these materials. Consequently, the techniques established at TMI-2 for processing and disposing of these highly concentrated materials have assisted both the utilities in developing appropriate techniques for such processing and the waste disposal sites in establishing suitable regulations for their disposal.

Another application for the technology developed through the use of zeolites at TMI-2 which holds the promise of significant economic impact is in the area of selective radioactive isotope removal. To appreciate the potential, one must be aware of current industry practices and regulatory requirements.

Utilities typically ship their spent resin in containers loaded to a third or less of the NRC limits identified in 10 CFR 61. There are two primary reasons for this:

First, organic resins, the most widely used ion-exchange media, can become depleted with nonradioactive isotopes which would ordinarily be disposed of as chemical waste.

Second, certain isotopes, such as ^{137}Cs and ^{90}Sr, are restricted by 10 CFR 61 to very low concentrations in Class A Waste (1 Ci/m^3 and 0.04 Ci/m^3, respectively). These isotopes do not have to appear in even these low quantities to cause a waste container to exceed the Class A limits. The sum-of-fractions rule states:

> "For determining classification for waste that contains a mixture of radionuclides, it is necessary to determine the sum of fractions by dividing each nuclide's concentration by the appropriate limit and adding the resulting values. The appropriate limits must all be taken from the same column of the same table (Table 2, 10 CFR 61.55). The sum of the fractions for the column must be less than 1.0 if the waste class is to be determined by that column."

Thus, a small amount of ^{137}Cs and/or ^{90}Sr in a waste stream can cause the entire waste package to exceed the Class A limits and add the additional expense associated with Class B disposal. This is significant when one considers that Class B limits for ^{137}Cs and ^{90}Sr are 44 Ci/m^3 and 150 Ci/m^3, respectively. Couple these two together and one realizes why utilities are generating large volumes of lightly loaded Class A and B Waste. Clearly, a selective radioactive isotope removal method aimed at these troublesome radioisotope products can be a cost-effective means of reducing volumes by increasing curie loadings in Class A Waste and concentrating troublesome radioisotopes in Class B Waste.

The activity levels of the waste being processed at TMI-2 are, for the most part, much higher than any seen at normal operation plants. Organic resins are known to degrade after prolonged exposure to high levels of radiation; therefore, inorganic resins (zeolites) were used extensively in the water processing systems. The zeolite selected proved particularly effective for the removal of ^{134}Cs and ^{137}Cs. Laboratory examinations of EPICOR prefilters containing both zeolites and organic resins confirmed this. The zeolite captured most of the cesium, while other isotopes, such as ^{90}Sr, were uniformly distributed in the organic resin. On the other hand, examination of EPICOR prefilters containing only organic resin revealed that cesium and strontium were captured with equal effectiveness.

Synthetic zeolites are manufactured in a variety of types, each of which are precisely uniform in size and of molecular dimension such that they function as selective molecular sieves. Each type is effective in the capture of molecules of specific diameters. They also have a selective preference for polar or polarizable molecules. Thus, they can be manufactured (sized) to selectively remove radioisotopes such as ^{90}Sr.

The in-plant demineralizers became so heavily loaded with fission products released during the accident that the resin could not be removed through the normal pathway. A chemical process was devised and tested at Oak Ridge National Laboratory which demonstrated that 90% of the cesium absorbed on the resin could be removed by elution and that virtually 100% of the cesium removed by elution could be recaptured by zeolite exchange material in the submerged demineralizer system. This method was employed successfully to reduce the activity in the TMI-2 demineralizers. Clearly, a combination of the two technologies used so effectively at TMI-2 could be used by other utilities to reduce waste volumes. For example, as spent resins are sluiced to resin storage tanks, they could be sampled to determine curie content and ultimate waste disposal classification. If the samples revealed the cesium contents were too high for Class A Waste, then an elution process could be used to remove the troublesome isotope. Because of the large capacity of zeolites, the cesium waste package could be used to collect the cesium removed from many organic loadings. The result would be one package of Class B waste and the rest Class A waste.

Radioactive Waste and Combustible Gas Handling. Looking to the future, the handling of large radioactive components at TMI-2 and their subsequent disposal should assist operating nuclear utility plants in formulating and carrying out plans for the decommissioning of their power plants. During normal plant life, core components and pressure vessels will become radioactive and probably will be outside the NRC Class C designation for low-level waste, as are components being disposed of at TMI-2.

One area of radioactive waste handling where the TMI-2 program has made a significant contribution which is already being employed in the industry is in combustible gas generation. In September 1984, the NRC issued new requirements (IE Notice 84-72) dealing with the issue of combustible gas generation in radioactive waste containers. Specifically, the rule requires radioactive waste generators to demonstrate that waste containers do not contain combustible mixtures of gases by means of tests and measurements or to vent or inert the containers before transport. The available means to comply with this new rulemaking involved both increased cost and man-Rem exposure. Based on experience with gas generation in EPICOR II liners, the TMI-2 Program was able to develop a computer-assisted method of calculating combustible gas generation rates and, in turn, predicting safe storage and transport periods before combustible limits were reached. The method uses existing data (such as isotope content and curie loading) which are already being collected to comply with transport manifest and other regulatory requirements.

The NRC has reviewed the technical basis provided by the TMI-2 Program and has approved the use of calculations as an acceptable alternative for demonstrating compliance with IE Notice 84-72.

The Electric Power Research Institute (EPRI), with DOE support, developed a training program to teach radioactive waste shippers how to use the computer program, the technical basis, and how to implement and obtain NRC approval for a plant-specific program.

The first workshop was held in June at the EPRI Mechanical Equipment Application Center (MEAC) in Charlotte, North Carolina. Response to this workshop was so positive that a second workshop was held in July in Washington State. EPRI is planning a third workshop, again at MEAC, for the first quarter of 1987.

Gas generation technology has not helped only utilities. Battelle Pacific Northwest Laboratories (BPNL) raised concerns about hydrogen gas generation in hot cell waste. BPNL developed a set of conservative assumptions for various unknowns, based on observed results at TMI-2, which allowed BPNL to calculate the explosive hazard the waste represented. The state of New Mexico Environmental Evaluation Group was concerned with hydrogen gas generation in radioactive waste containing transuranic waste during transportation for the Waste Immobilization Pilot Plant (WIPP) program. They requested information on gas generation rates due to alpha radiation. BPNL suggested a simple test program which would produce correlation values that could be used with the gas calculation equations to accurately predict gas generation in alpha waste. The same information was provided to the Safety Analysis Review Group at Argonne National Laboratory after they contacted BPNL about a similar problem with transuranic waste being shipped in Trupact containers. This waste was destined for the WIPP program as well.

High Integrity Containers. A first-of-a-kind, high integrity container (HIC) was developed and tested by the U.S. Department of Energy, and approved by the State of Washington for use in disposing of 46 EPICOR-II prefilters from TMI-2. Fifty prefilters were transported to the INEL for storage, research, and disposition. The disposal demonstration of one EPICOR-II prefilter contained in an HIC paved the way for GPU Nuclear to dispose of 46 prefilters individually contained in HICs at the waste disposal facility operated by U.S. Ecology, Inc. in the State of Washington. [Four prefilters are being held in storage by the DOE for research by the NRC.]

EPICOR-II prefilters contained as much as 2200 Ci of radioactive isotopes. That amount of radioactivity exceeded criteria outlined in 10 CFR 61 for disposal as Class C low-level radioactive waste, unless the ion exchange media in each prefilter was either (a) solidified, or (b) the prefilter was totally enclosed in an HIC before disposal. A reinforced concrete HIC seemed the more attractive alternative for disposal of EPICOR-II prefilters, after economic and engineering evaluations projected the HIC to be cost-competitive with solidification. That projection was affirmed after the disposal demonstration.

One important outcome of the EPICOR disposal demonstration was the achievement of regulatory approval for use of the concrete HIC. Approvals were required from regulatory authorities at a time when the regulatory position on HICs was not well established, criteria for HICs were still being developed, and the HIC concept was relatively untested. The regulation 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," established a waste classification system based on radionuclide concentrations in the wastes. Class C wastes, like the EPICOR prefilters, must be stabilized. The "Technical Position on Waste Form" by NRC gave guidance for stability (e.g., solidification or use of high integrity containers), along with criteria for high integrity containers. However, that technical position was formative at the start of the DOE effort to dispose of EPICOR-II prefilters.

The approval process for use of the HIC took approximately four years and involved the cooperation of federal and state agencies, a public utility, and private industry. The process was a pathfinder; therefore, it is believed that much of the regulatory deliberation will be shortened for similar approval actions for other types of HICs. Figure 24 presents a network for achievement of the goal to develop and use the EPICOR/HIC. Shown are paths of (a) requirements identification, (b) development/testing, (c) licensing, (d) fabrication, (e) research, and (f) transportation. Figure 24 illustrates the complexity involved in that approval process and connotes the importance of the disposal demonstration.

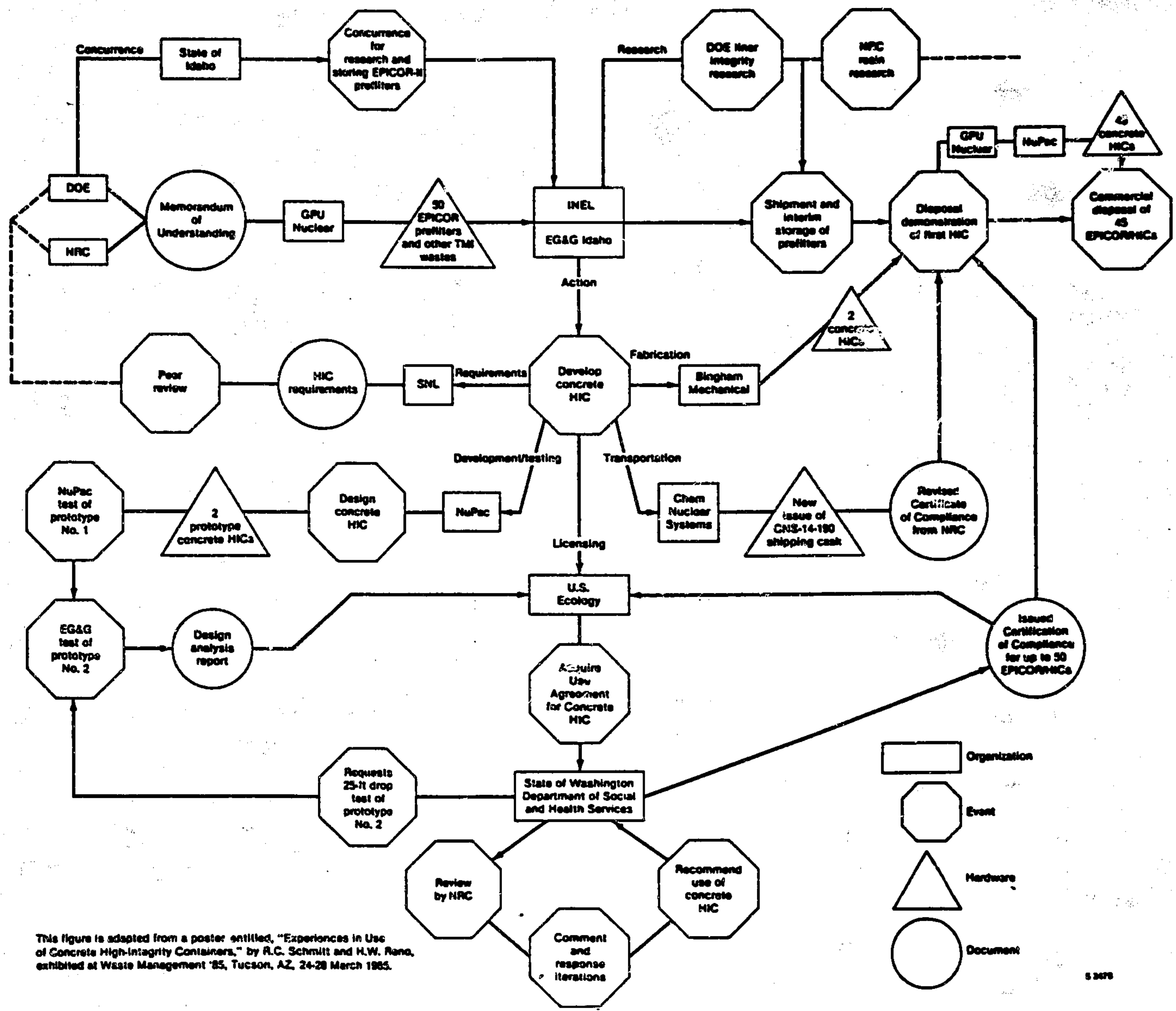
Since enactment of the Low-Level Radioactive Waste Policy Act of 1980 (PL 96-573), groups of states have entered into compacts for the purpose of mutually resolving disposal of low-level radioactive wastes. A disposal facility would be established in and operated by a "host" state for the benefit of the participating compact members and at the exclusion of other states. Although compacts are formed, no low-level waste disposal facilities have been established to date.

Part of the difficulty in establishing low-level waste disposal facilities is the debate over whether a facility should be below ground (e.g., shallow-land burial or deep below surface) or above ground (e.g., on a concrete or asphalt pad). The solution to this difficult problem may be the HIC, or a variation of it.

The concrete HIC, developed for disposal of EPICOR-II prefilters from TMI-2, is durable, licensed, and tested, and is equipped with a one-way vent system for the exhaust of gases produced inside. It would be relatively simple to adapt its design and scale its dimensions up or down according to need. In this way, the container could be used for both above-ground storage and below-ground disposal of low-level radioactive wastes, for it is durable, capable of withstanding mechanical deformation, resistant to internal and external corrosion, reinforced internally to withstand high pressures (should the vent system in the lid fail), inspectable externally, and designed to provide some radiation shielding; it is a cost-competitive alternative to solidification of Class C or greater low-level radioactive wastes.

Instrumentation and Electrical Equipment

Performance. The accident at TMI-2 provided an opportunity to evaluate a variety of instrumentation and electrical equipment for the effects of exposure to moderately severe accident conditions including steam, spray, and radiation, as well as



This figure is adapted from a poster entitled, "Experiences in Use of Concrete High-Integrity Containers," by R.C. Schmitt and H.W. Reno, exhibited at Waste Management '85, Tucson, AZ, 24-28 March 1985.

Figure 24. Achievement network for development and use of EPICOR/HIC.

hydrogen burn and the resultant overpressure. The examination of this equipment over a period of several years also provided information on long-term exposure to moisture. The TMI-2 Instrumentation and Electrical Program was established by the Department of Energy to recover as much of this information as possible. Although the primary thrust was the evaluation of the survivability and performance of safety equipment, the program revealed many weaknesses in the way we design, install, maintain, and test both safety and balance-of-plant equipment. Safety-related equipment performed quite well, with the only failures being pressure transmitters and motor-operated valves located in the basement that were eventually flooded. These findings reported here apply to all nuclear power plants, since equipment at TMI-2 was not unique and the observed problems can happen in a normal operating plant.

Design. During the course of the evaluation of the effects of the accident on the instrumentation, it became evident that many of the observed failures resulted from improper specifications of equipment.

The reactor building of a normal operating nuclear power plant creates a harsh environment for electrical equipment and instrumentation. Temperatures reach 130°F, some areas experience intense radiation fields, humidity levels are high, and building integrated leak-rate tests create differential pressures in excess of 60 psig. Nevertheless, there is little that distinguishes the balance of plant equipment installed in the TMI-2 reactor building from equipment serving a similar function in the auxiliary or turbine buildings. Examination of electrical equipment which failed as a direct result of the accident revealed that the failures were also likely to occur as a result of long-term exposure to the normal operating environment.

Installation. The dominant failure mode of the TMI-2 instrument and electrical equipment was due to corrosion. Water and vapor intrusion into the equipment housings caused erratic readings and ultimate failure. Where a reliable seal existed at the cable entry into the I&E equipment housing, the internals were generally not corroded and the instrument or electrical equipment was operable.

The installed configuration of some equipment clearly played a major role in their failure under accident conditions. In some cases, these problems resulted in failures during the first 24 hr after the accident. The findings demonstrate that even simple installation requirements, if incorrectly performed, can degrade the ability of the equipment to function as designed. Particular care should be taken with conduit and junction box seals, drains and vents, and the sealing of connector backshells to protect against moisture intrusion. Design practices should take into account those field activities that are critical so that installation problems are minimized. Installation practices should be controlled to assure they do not degrade the equipment design and application engineering requirements.

Maintenance and Testing. Analysis of failed instrumentation and electrical equipment at TMI-2 indicated that in some cases, inadequate maintenance and testing activities may have been contributing factors. For example, it was necessary to remove the dome monitor during containment leak-rate testing. This removal required opening the sealed container with the risk of damage to the monitor seal and connectors. This instrument later failed due to moisture intrusion. Many other components are similarly affected during the containment leak-rate tests. Considering the NRC statistical data relating that 35% of the abnormal plant occurrences are maintenance-related, it is an obviously prudent measure to minimize maintenance-related activities which may lead to equipment failure. In a case such as this, specifying and procuring equipment able to withstand other plant tests without modification or removal should improve equipment reliability. Likewise, designing tests to maximize data for maintenance decisions while minimizing the disturbance of equipment seals will improve equipment reliability.

In-Situ Test Methods. There has been a great deal of interest in the Electrical Circuit Characterization and Diagnostic (ECCAD) System. The ECCAD is a computer controlled measurement system designed to characterize electrical circuits in nuclear plants. EG&G Idaho, Inc. developed the system at the INEL for the U.S. Department of Energy in order to assist in the selection of electrical equipment for removal from the TMI-2 containment for extensive examination in the laboratory. The ECCAD system was designed to distinguish small differences in the electrical characteristics of different circuits and to identify anomalies. Follow-up laboratory analysis has proven just how successful the ECCAD system can be at distinguishing between normal and degraded circuits. The

ECCAD system can perform a similar function at normally operating nuclear plants or in any industry which relies on a large number of critical electrical circuits. The ability to detect an impending failure and correct it before it becomes a problem will pay dividends in increased plant reliability and capacity.

This was recognized by Southern California Edison who contracted with the Department of Energy to test circuits at their San Onofre Site. They used it to establish a technical basis for eventual plant-life extension. The Department of Energy also worked with Duke Power Company to conduct a pilot program at their Oconee nuclear facility. The work is presently in progress and DOE has provided equipment and two system experts. In June 1986, a company in the private sector, Pentek Corporation, started marketing ECCAD equipment and services. The EPRI Maintenance Equipment Application Center has demonstrated the ECCAD system to representatives of the nuclear power industry and a number of utilities are considering adding this system to their maintenance programs.

Interest in the ECCAD system has not been restricted to the private sector. United Nuclear Corporation has requested one of the TMI-2 ECCAD systems and training services in order to support a plant-life-extension program they are developing at the DOE Hanford site.

The ECCAD technology is also being used to support other activities. At present, a research program is being supported by the NRC to develop an *in situ* method of evaluating the functional condition of solenoid valves by analyzing inrush current. ECCAD equipment is being used for data acquisition. Southern California Edison is supplying the valves and Pentek Corporation is performing the work.

It is highly likely that ECCAD-type diagnostic systems will eventually become standard maintenance equipment at most electric utilities, both nuclear and nonnuclear.

Industry Standards Development

IAEA. The TMI-2 Program is involved with IAEA in the development of a technical report on "Handling and Treatment of Radioactive Waste from Unplanned Events at Nuclear Power Plants." The report will provide information on state-of-the-art technology for waste management systems and processes regarding the handling and treatment of wastes that have arisen from unplanned events at nuclear power plants. More importantly, the report will discuss which processes are the most effective for different waste forms and curie content. A user will be able to determine the best system available for his particular problem and the one which will generate the smallest total waste volume.

ANS. The TMI-2 Program is involved with a number of standards groups in the development or revision of industry standards. TMI-2 has much information and experience on the performance of in-containment radiation monitors. Working with an American Nuclear Society Standards Group, TMI-2 Program results are being applied to a new standard for the design and use of both portable and permanent in-containment radiation monitors. ANS Standard 5.7.2 deals with the radiological design criteria for postaccident radiation monitoring.

This standard will benefit the nuclear power industry by incorporating important information on postaccident radiation monitoring not available in current standards on the subject. Issues to be addressed include requirements to assure that postaccident instrumentation will be functional when needed and precautions required to avoid misinterpretation of postaccident data. This standard will reflect the lessons learned from detailed analyses of instrument failure modes at TMI-2. It will also reflect the experience of utilities in seeking to meet post-TMI-2 regulatory requirements.

ANSI. The TMI-2 Program is also working with the American National Standard Institute (ANSI) on the development of a new standard, ANSI N42.17B-D3.

The objective of this standard is to provide basic design and performance criteria for instruments and instrument systems used to continuously sample and measure airborne radioactivity concentrations, i.e., continuous air monitors (CAMs). The standard includes performance testing criteria for use in generic (type) tests of new instrument models. Included in this standard are both gaseous and particulate gross radioactivity monitors, as well as monitors for specific radionuclides or groups of radionuclides, used for radiation protection purposes. Instruments that provide monitoring of ambient airborne radioactivity by measuring the external radiation field associated with the airborne radioactivity (i.e., direct radiation monitors) are also covered, as are instruments designed to divide the airborne radioactivity into two or more specific components or fractions and monitor each independently.

IEEE. An extensive instrumentation and electrical equipment examination program provided a number of important results for industry. Although some design flaws were detected in electrical equipment and instruments, these were not nearly as prevalent as deficiencies in installation, maintenance, and surveillance testing as causes of failure. Consequently, the TMI-2 Program has been involved with "Subcommittee 3, Operation, Surveillance and Testing," of the Nuclear Power Engineering Committee (NPEC). NPEC is a subdivision of the Institute of Electrical and Electronic Engineers (IEEE). The standards under development deal with improved maintenance and surveillance practices.

Industry Actions from Lessons Learned

The effect of the 1979 TMI-2 accident on utilities has not only resulted in implementing improvements in operational practices, personnel training, staffing, and plant design, but it has also permanently changed the way utilities view their role within the nuclear industry. Utility dedication to self-improvement and self-regulation is evident through their participation in the various utility-supported organizations and by greater involvement on the part of management in all aspects of nuclear operations. Utilities are now working together more effectively than ever before. Improved communications, solutions to technical issues, data availability, and design reviews are examples of areas of unification that contribute to increasing the reliability and safety of plants.

The following sections summarize the actions taken by nuclear utilities resulting from the "lessons learned" at TMI-2.

Improved Standards. The nuclear utilities created the Institute of Nuclear Power Operations (INPO), an organization whose purpose is to promote greater levels of safety and reliability on the part of every nuclear utility. INPO developed a set of criteria and an evaluation program for measuring the performance of nuclear power plants. The evaluation process has provided the industry with a standard approach in establishing benchmarks for excellence. To date, 78 "Good Practices" have been established, identifying sound industry practices. INPO has completed over 200 on-site plant evaluations as a means of assisting utilities in meeting improved industry standards. The plant evaluations are performed in the following major areas:

- Organization and administration,
- Operations,
- Maintenance,
- Technical support,
- Training and qualification,
- Radiological protection,
- Chemistry, and
- Emergency preparedness.

Improved Communications. Each member utility has been required to establish a program to provide feedback of operating experience from their own nuclear plants and from experience throughout the industry. The response of industry, in establishing INPO to carry out this function, along with numerous others, has been outstanding. Several programs have been established to improve communications including: Significant Event Evaluation and Information Network (SEE-IN), the Nuclear Plant Reliability Data System (NPRDS), and NUCLEAR NETWORK.

SEE-IN provides for the systematic collection, analysis, and sharing of information on operating events. SEE-IN acts as an early warning system for potentially serious events.

More than 9,000 events a year are analyzed at INPO. Each event is analyzed by two different people to ensure a thorough review. Lessons learned from these events are transmitted as Significant Event Reports (SERs) to INPO members.

Significant events are studied further and, in some cases, INPO finds that an event has generic significance to a large segment of the industry. In these cases, INPO provides Significant Operating Experience Reports (SOERs) to members and participants. These reports provide recommendations on actions the utilities can take to prevent a further recurrence of the incident.

The Nuclear Plant Reliability Data System, or NPRDS, is another program that falls under INPO's events analysis activities. NPRDS is a voluntary program in which utilities track the performance of key nuclear plant systems components. Utilities do this by reporting two kinds of data: (a) baseline engineering data and design characteristics for these components, and (b) failure reports that are submitted whenever components fail to perform their intended function. Currently, the NPRDS engineering data base is more than 90% complete. When complete, INPO expects to have approximately 410,000 records. This is more than double the number of reports on the system at the end of 1983.

The number of failure reports also has increased dramatically. At the end of 1984, there were 21,700 failure reports on the system. That number stood at 30,000 in mid-1985.

NUCLEAR NETWORK is a worldwide, computer-based communications system. INPO's 55 member utilities, 13 international participants, and 13 supplier participants routinely use the 25 separate channels in this system. Each channel is dedicated to a particular subject (e.g., instrumentation). Today, an average of 50 entries is posted each day.

Another example of improved utility cooperation and better communications is the proliferation of utility owners' groups. Some of the owners' groups are oriented to address a specific technical or regulatory issue, while others, such as the vendor-oriented owners' groups, address a broad spectrum of generic issues. The vendor-oriented owners' groups were originally established after the TMI-2 accident to respond in a timely manner to the avalanche of "lessons-learned" requirements emanating from the NRC. This approach has been effective in making available technical resources in a manner that is timely, cost-effective, and uniform for all plants of similar design. Today, these owners' groups provide a forum for joint discussions and resolutions of generic issues through effective communications between members and other organizations such as EPRI, INPO, and AIF. The owners' groups review commitments made by industry management on generic issues and volunteer to address those issues where the group can make an effective contribution towards satisfying the industry commitment.

Training. The TMI-2 accident has been attributed in large measure to the lack of adequate operator training and staffing, along with inadequate control room design. To remedy these shortcomings, NRC increased operator training and examination requirements to include studies in heat transfer, fluid flow, and thermodynamics, with applications for understanding plant transients and mitigating core damage.

Nuclear Plant Training is dramatically different, better, and more comprehensive now than it was in 1979. At the time of the TMI-2 accident, nuclear plant training programs by the utilities varied widely. Then, some were very modest. Today, utilities are applying the latest techniques of job analysis and training program development to ensure that this training is performance-based and that the required knowledge and skills are developed and demonstrated by all operators. This expanded and systematic approach has brought about tremendous improvement in training in recent years.

Through INPO, the industry has established a system to accredit utility training programs for operations, maintenance, and technical support personnel. Accreditation requires utilities to meet established standards which are used to measure the quality of their training program content, instructor qualifications, and training facilities. The final decision on whether a utility meets these standards is made by an independent accrediting board.

Simulators have been recognized as one of the most effective training devices for operators. In 1979, there were 10 simulators in operation in the United States. Now there are 70 of these multimillion dollar machines operating, under construction, or planned.

As a result of the increased emphasis on training, utility staffs now include more than 2,100 full-time training personnel, almost 1,700 of whom are instructors. That's an average of 25 instructors for each plant operating or under construction, more than four times as many as in 1979.

In September 1985, U.S. nuclear utilities affirmed their commitment to training excellence by forming the National Academy for Nuclear Training. The academy provides training programs aimed at providing properly trained and qualified personnel to operate nuclear power plants.

NRC has reduced their involvement in establishing rules on the training of nuclear plant personnel and vested prime responsibility for training and accrediting these personnel to INPO.

Every operating nuclear plant is now required to have a degreed Shift Technical Advisor and two Senior Operators on each shift, with one Senior Operator always in the control room. A Senior Operator with the appropriate degree can also meet the requirement for a Shift Technical Advisor, if the utility so designates an individual. Each licensee has evaluated the administrative, non-safety-related duties of his shift supervisors and made reassignments to ensure their attention is focused on safe plant operations. Limits have been established on the amount of overtime work that can be performed by operating shift crews and key maintenance personnel.

INPO also provides human performance criteria on the development of writing guides for plant operations and maintenance procedures. INPO's writing guidelines provide guidance in the area of technical content and human factors considerations such as procedure format, use of illustrations, and clear writing techniques.

Human factors engineering has been applied to certain required design changes, which include:

- Control room safety-parameter display console,
- Valve position indicators,
- Auxiliary feedwater system initiation and flow instrumentation,
- Instrumentation for detecting inadequate core cooling, and
- Accident radiation monitoring.

Plant Modifications. There have been several major equipment, instrumentation, and technical design improvements incorporated into nuclear power plants as a result of the TMI-2 accident enforcement actions. These include:

- Installation of safety parameter display systems in control rooms,
- Control-room-operated venting capability of reactor coolant system and reactor vessel,
- Shielding for access to vital areas,
- Safety and relief valve positive position indication,
- Redundancy of H_2 recombiners or purge systems,
- Noble gas effluent monitors and postaccident sampling of iodines and particulates,
- Continuous indication of containment pressure, water level hydrogen concentration, and inadequate core cooling,
- Emergency power to the pressurizer heaters, and
- Emergency power supplies for pressurizer relief valves, block valves, and level indicators.

In May 1980, the NRC published the TMI-2 Action Plan, a comprehensive list of 176 action items involving 364 detailed tasks judged necessary to improve utility operations and NRC's regulation of nuclear power plants. Identified as the responsibility of the utilities were 39 items covering 132 tasks. Of these tasks, 39 involved equipment backfit items. Approximately 90% of the utility action items on operating reactors are now complete.

Emergency Response Planning and Preparedness. One of the most significant "lessons-learned" from the TMI-2 involved emergency programs. It was determined that the nuclear utilities would have to improve their emergency programs, which led to requirements in the following areas:

- Emergency preparedness organization and administration,
- Emergency plan,
- Emergency response training,
- Emergency facilities, equipment, and resources,
- Emergency assessment and notification,
- Personnel protection,
- Emergency public information,
- Inplant radiation monitoring, and
- Accident monitoring (noble gas, iodine, containment radiation, containment pressure, containment water level, and hydrogen gas concentrations).

These requirements led to numerous plant modifications, retrofits, and INPO support efforts.

INPO supports industry efforts in the emergency preparedness area with special assistance visits, evaluations, and other programs. They have conducted many assistance visits to help utilities establish effective programs and made observations of emergency drills to achieve performance-based evaluations of emergency preparedness programs.

INPO also acts as a clearinghouse for requests for assistance in case of an emergency at a nuclear plant. INPO has published and distributed an emergency resources manual that contains comprehensive information on available resources industrywide. Additionally, they maintain an emergency response center, through which they can coordinate assistance to a member utility.

To date, fifty-two utilities have agreed to and signed two voluntary assistance agreements concerning emergency preparedness. The agreements pave the way for uninhibited assistance by any and all utilities in case of a plant or transportation emergency by addressing potential legal or other barriers in advance.

In addition, Utility Owners' Groups have developed generic operating guidelines for each type of reactor that simplify the operator approach to analyzing and responding to plant transients. These efforts include: (a) develop new procedures using the guidelines; (b) validate the procedures on plant simulators; and (c) train and requalify the licensed personnel in the use of the procedures. This resulted in the development of "symptomatic"- rather than "event"- oriented procedures. In effect, the human factors concept was integrated into the operating procedures. The use of symptomatic procedures takes the guesswork out of operational transient response and directs the operator attention to correcting the symptoms. Symptomatic-oriented procedures, unlike event-oriented procedures, provide the operator with the capability to respond to all possible accident scenarios.

Results of Implementation of "Lessons Learned." While it is impossible to quantify precisely the impact of "lessons learned" on the nuclear power industry, it is clear that a number of improvements have been made since the TMI-2 accident. Areas of improvements include:

- Unplanned Reactor Shutdowns

  In the area of unplanned automatic reactor shutdowns, the U.S. utility industry has shown progress since 1980. During 1980, the industry experienced an average of 6 scrams/commercial unit while synchronized to the grid. By 1984, that number had dropped to an average of 3.5 scrams/commercial unit.

- Plant Performance

  Plant performance has improved as indicated by the number of significant events occurring at U.S. nuclear plants. During 1981, records indicate that an average of 1.64 events occurred for each of the 69 U.S. units operating. This number has been declining each year. Since 1984, it has stood at 0.7 events for each of the operating units.

- Plant Reliability

  Two indicators—forced outage rates and equivalent availability— provide examples of plant reliability. Trends in these areas remained nearly steady for the years 1980 through 1984.

- Thermal Performance

  The U.S. industry measures thermal performance by heat rate, or BTUs per kilowatt hour. A low heat rate reflects emphasis on thermal efficiency and attention to detail in maintenance. The U.S. industry's overall thermal performance shows a slight steady improvement from 1980 to 1984.

- Radiation Control and Exposure

  In the area of radiation control and exposure, the U.S. industry has shown improvement. At both boiling water and pressurized water plants, the U.S. industry has reduced the average collective exposure per unit from 1980 to 1984.

- Radioactive Waste

  Both PWRs and BWRs have reduced the amount of radioactive waste shipped from 1980 to 1984.

- Lost-Time Accidents

  The industry's lost-time accident rate showed significant improvement from 1980 through 1982. Since that year, the rate has been steady.

There is clear evidence that the U.S. nuclear power industry is experiencing the benefits of their efforts, and that nuclear power plants are more reliable and availability has been enhanced.

Accident Evaluation Program

The TMI-2 accident was unique in two important features relative to severe accident and source-term research. First, the accident occurred in a commercial LWR under thermal-hydraulic conditions typical of a large family of hypothesized severe accidents. Second, the damage to the core has been confirmed to be more severe than the existing severe fuel damage experimental data base. Because of these unique features, the accident offers the potential of increasing our understanding of many currently unresolved severe accident and source-term technical issues.

The major unresolved technical issues have been identified by extensive review studies by the technical community. These include U.S. Nuclear Regulatory Commission (NRC) review, based on the results of the Severe Accident Research Program; the Industry-sponsored Degraded Core Research Program (IDCOR); and extensive technical reviews by the American Nuclear Society, the American Physical Society, and DOE. Twenty-nine major technical issues were identified in these reviews. Because damage during the TMI-2 accident was primarily limited to the core and reactor vessel, the utility of the TMI-2 data is related to those technical issues associated with in-vessel core degradation and fission product behavior before vessel failure.

Based on technical issues which TMI-2 can impact, the objectives of the TMI-2 Accident Evaluation Program are:

- To understand the physical and chemical state of the TMI-2 core and related structures and the external influences which affected the accident,
- To understand what happened during the accident and to provide a qualified data base and standard problem of the TMI-2 accident to provide a benchmark for severe-accident analysis codes and methodologies,
- To understand the relationship between the phenomena and process controlling the accident and the important severe-accident and source-term technical issues, and
- To ensure that the results of the program are effectively transferred to the nuclear industry.

Those specific "in-vessel" technical issues for which our understanding can be significantly improved through additional TMI-2 research are identified in Table 2 and include questions related to reactor system thermal-hydraulics, core degradation, and fission product behavior.

Recent TMI-2 characterization and analyses have provided the basis for an improved understanding of the accident scenario, including the core damage progression during the accident. Important research findings to date include:

- When the high-pressure injection was initiated at 200 min, core cooling resulted. However, the molten, consolidated core material continued to heat up, despite the presence of water surrounding the crust.
- There is no evidence indicating that a steam explosion occurred when the crust surrounding the consolidated molten core material failed, allowing approximately 10% of the total core mass to fall into the water-filled lower plenum.
- The presence of water in the lower plenum terminated the accident progression and prevented failure of the reactor vessel lower head.
- Fission product release from the fuel and the RCS is very sensitive to the volatilitiy of the individual elements of chemical species. Release of the high-volatile fission products (iodine and cesium) was less than expected (especially of cesium, up to 20% of which was retained in the previously molten core material that relocated to the lower plenum). Retention of the medium- and low-volatile fission products in the reactor pressure vessel was nearly complete, although significant amounts of antimony and ruthenium were released from the fuel and are believed to be bound to metallic surfaces in the reactor vessel.

Important questions relative to core failure and relocation of the molten core material remain to be answered. Perhaps the most important of these is, "What was the mechanism or mechanisms leading to failure of the crust surrounding the molten consolidated core?". There appear to be several plausible failure mechanisms; however, an unequivocal answer to this question will require additional inspection of the core crust in the east quadrant and sample acquisition and examination of the crust material to determine its composition, material interactions, and physical and chemical properties. Additional inspections of the core region in the east quadrant of the vessel are necessary to determine if the crust failure was localized or global in nature.

Another important question yet to be answered concerns the extent of damage to the reactor vessel. Additional inspection and sample examination data are necessary to adequately characterize the lower plenum debris and possible damage to the pressure vessel lower head and instrument tube penetrations.

Supporting analysis is necessary to interpret the data and improve our understanding of the formation of the degraded core and eventual failure of the supporting crust. Analysis is also needed to better estimate the interaction of the molten core material with the vessel coolant, formation of core debris,

Table 2. TMI-2 related severe accident and source-term technical issues

RCS Thermal-Hydraulics

- Coupling along core degradation, RV thermal-hydraulics, fission product behavior, and hydrogen generation
- RV natural circulation

Core Damage Progression

- Damage progression in core
- Core slump and collapse
- Reactor vessel failure modes
- Hydrogen generation

Fission Product Behavior

- Release of lower-volatility fission products
- Chemical reactions affecting fission product transport (includes chemical form)
- Tellurium behavior
- Vaporization and relocation of control rod materials

and long-term cooling of the degraded core materials, both within the original core boundaries and in the lower plenum.

Application of future TMI-2 research towards resolution of the relevant technical issues will be very much dependent upon demonstrating a consistent and comprehensive understanding of the accident with respect to core damage progression and fission product release and transport; developing this understanding is our most important program objective. Considerable progress in developing the accident scenario has been made but additional work is necessary. The following information will be developed: more realistic temperature bounds during the initial core heatup; the extent of fuel relocation and resulting configuration of the noncoolable core regions; core relocation into the lower plenum; the physical and chemical interactions between the molten core materials and the lower plenum structures (including the reactor vessel); and the formation of a coolable configuration within the lower plenum. These questions will be resolved through additional sample acquisition and examination of the core and reactor vessel materials and supporting analysis to integrate the examination results, the plant thermal-hydraulic response as characterized by online instrumentation during the accident, and other independent severe fuel damage research. This will provide the desired understanding of the core damage progression and resulting fission product release (i.e., the accident scenario). The unresolved questions regarding the TMI-2 accident progression discussed above have almost a one-to-one correspondence to the outstanding severe accident and source-term technical issues (summarized in Table 2) which have impacted the nuclear industry.

Continuation of the TMI-2 Accident Evaluation Program will achieve the following objectives: (a) complete our understanding of the TMI-2 accident, including the end-state distribution of fission products; (b) provide the opportunity to apply the TMI-2 research results towards resolving the more general severe accident and source-term technical issues; and (c) ensure industry coordination and participation in defining and carrying out the program.