Case Study
Remediation of Three Mile Island Unit 2 in Preparation for Decommissioning

Prepared by:

John P. Clements
U.S. Nuclear Regulatory Commission

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1. INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) reactor, near Middletown, Pa., partially melted down on March 28, 1979. A combination of equipment malfunctions, design-related problems and worker errors led to TMI-2's partial meltdown and very small off-site releases of radioactivity. This was the most serious accident in U.S. commercial nuclear power plant operating history, although its small radioactive releases had no detectable health effects on plant workers or the public. Its aftermath brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the Nuclear Regulatory Commission (NRC) to tighten and heighten its regulatory oversight. All of these changes significantly enhanced U.S. reactor safety. [1]

1.1 Technical Characteristics

The Three Mile Island (TMI) Nuclear Station is located on Three Mile Island in the Susquehanna River, approximately 10 miles southeast of Harrisburg, Pennsylvania. It is in Londonderry Township of Dauphin County. Goldsboro is located 1.93 km [1.2 miles] west of Three Mile Island, and Middletown is located 4.83 km [3 miles] north of the facility. An aerial view of the TMI site is shown in Figure 1.

On November 4, 1969, following a public hearing, a construction permit for TMI-2 was issued by the Atomic Energy Commission. An operating license for TMI Unit 2 was issued on February 8, 1978. Between issuance of its operating license and March 28, 1979, TMI-2 had operated for about 95 effective full-power days (or the equivalent of 3,165 MW-days per metric ton of low-enriched uranium oxide). Prior to the accident, the unit had been operating without interruption since March 7, 1979. At full power, the TMI-2 reactor plant would supply 890 MW (2,770 MW thermal) of electric power to the utility's transmission system; together with TMI Unit 1, the two units had a 1,700-MW electric generating capacity. TMI-2 was operating at 97% power when the accident occurred.

TMI-2 was a pressurized-water reactor with the nuclear steam supply system (NSSS) provided by Babcock &Wilcox Company. Under normal reactor operation, the primary coolant water inside the reactor vessel was maintained at around 302 degrees Celsius [575 degrees Fahrenheit]

and 1.52 x 10^7 Pa [2,200 psi]. Heat generated by the fission process within the reactor core was removed by means of the primary coolant to two steam generators where steam was produced to operate a turbine.

The reactor vessel and components that were exposed to radioactivity were located within the reactor building (also referred to as the containment building). The turbine generator, feedwater system, and electrical generation equipment were housed inside the turbine building. The remaining support systems, e.g. the high-pressure injection pumps and the makeup and let-down systems, were located in the auxiliary building adjacent to the reactor building. A general plant diagram for TMI-2 is shown in Figure 2. [2]

Figure 1 – Aerial View of TMI

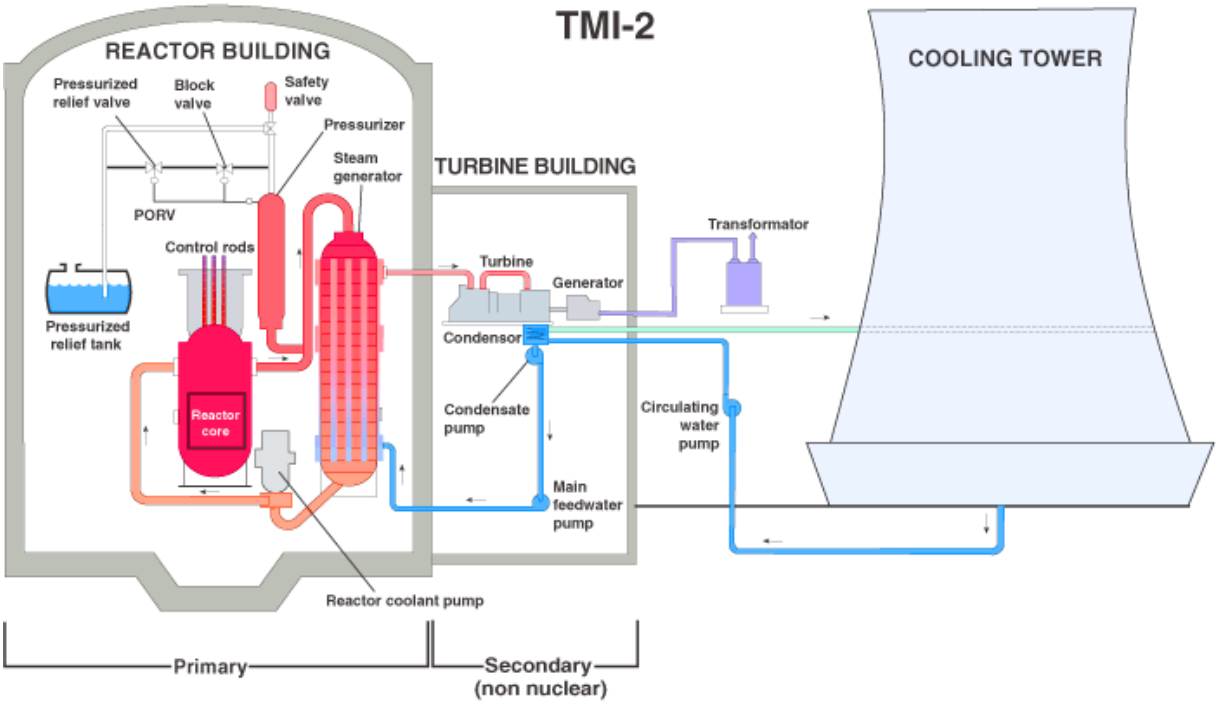

Figure 2 – General Plant Diagram for TMI-2

1.2 The Accident

Potential Factors Leading to the Accident

It was noted in the Report of the President's Commission on The Accident at Three Mile Island titled "The Need for Change: The Legacy of TMI" that TMI-2 had repeated problems with the condensate polishers (the filtration systems that remove dissolved minerals from the feedwater system). During the 18-month period before the accident, no effective steps were taken to correct these problems. These polishers probably initiated the March 28, 1979 sequence of events.

Preceding the accident, the TMI-2 shift foreman had been overseeing maintenance on the plant's Number 7 polisher. His crew was using a mixture of air and water to break up resin that had clogged a resin transfer line. Later investigation would reveal that a faulty valve in one of the polishers allowed some water to leak into the air-controlled system that opens and closes the polishers' valves and may have been a factor in their sudden closure just before the accident began. This malfunction probably triggered the initial pump trip that led to the accident. The same problem of water leaking into the polishers' valve control system had occurred at least twice before at TMI-2. Had the earlier polisher problem been corrected, the March 28, 1979 sequence of events may never have begun. [3]

Summary of Accident Events

The accident began about 4 a.m. on Wednesday, March 28, 1979, when the plant experienced a failure in the secondary, non-nuclear section of the TMI-2 plant. Either a mechanical or electrical failure prevented the main feedwater pumps from sending water to the steam generators that remove heat from the reactor core. This caused the plant's turbine-generator and then the reactor itself to automatically shut down. Immediately, the pressure in the primary system (the nuclear portion of the plant) began to increase. In order to control that pressure, the pilot-operated relief valve (a valve located at the top of the pressurizer) opened. The valve should have closed when the pressure fell to proper levels, but it became stuck open. Instruments in the control room, however, indicated to the plant staff that the valve was closed. As a result, the plant staff was unaware that cooling water was pouring out of the stuck-open valve for about 142 minutes.

As coolant flowed from the primary system through the valve, other instruments available to reactor operators provided inadequate information. There was no instrument that showed how much water covered the core. As a result, plant staff assumed that as long as the pressurizer water level was high, the core was properly covered with water. As alarms rang and warning lights flashed, the operators did not realize that the plant was experiencing a loss-of-coolant accident. They took a series of actions that made conditions worse. The water escaping through the stuck valve reduced primary system pressure so much that the reactor coolant pumps had to be turned off to prevent dangerous vibrations. To prevent the pressurizer from filling up completely, the staff reduced how much emergency cooling water was being pumped in to the primary system. These actions starved the reactor core of coolant, causing it to overheat.

Without the proper water flow, the nuclear fuel overheated to the point at which the zirconium cladding (the long metal tubes that hold the nuclear fuel pellets) ruptured and the fuel pellets began to melt. The 1A reactor coolant pump was turned on approximately 16 hours into the event, providing decay heat removal. It was later found that about half of the core melted during the early stages of the accident. Although TMI-2 suffered a severe core meltdown, the most dangerous kind of nuclear power accident, consequences outside the plant were minimal. Unlike the Chernobyl and Fukushima accidents, TMI-2's containment building remained intact and held almost all of the accident's radioactive material.

Federal and State authorities were initially concerned about the small releases of radioactive

gases that were measured off-site by the late morning of March 28 and even more concerned about the potential threat that the reactor posed to the surrounding population. They did not know that the core had melted, but they immediately took steps to try to gain control of the reactor and ensure adequate cooling to the core. The NRC's regional office in King of Prussia, PA., was notified at 7:45 a.m. on March 28. By 8 a.m., NRC Headquarters in Washington, D.C., was alerted and the NRC Operations Center in Bethesda, MD., was activated. The regional office promptly dispatched the first team of inspectors to the site and other agencies, such as the Department of Energy and the Environmental Protection Agency, also mobilized their response teams. Helicopters hired by TMI's owner, General Public Utilities Nuclear, and the Department of Energy were sampling radioactivity in the atmosphere above the plant by midday. A team from the Brookhaven National Laboratory was also sent to assist in radiation monitoring. At 9:15 a.m., the White House was notified and at 11 a.m., all non-essential personnel were ordered off the plant's premises.

By the evening of March 28, the core appeared to be adequately cooled and the reactor appeared to be stable. But new concerns arose by the morning of Friday, March 30. A significant release of radiation from the plant's auxiliary building, performed to relieve pressure on the primary system and avoid curtailing the flow of coolant to the core, caused a great deal of confusion and consternation. In an atmosphere of growing uncertainty about the condition of the plant, the governor of Pennsylvania, Richard L. Thornburgh, consulted with the NRC about evacuating the population near the plant. Eventually, he and NRC Chairman Joseph Hendrie agreed that it would be prudent for those members of society most vulnerable to radiation to evacuate the area. Thornburgh announced that he was advising pregnant women and pre-school-age children within a five-mile radius of the plant to leave the area.

Within a short time, chemical reactions in the melting fuel created a large hydrogen bubble in the dome of the pressure vessel, the container that holds the reactor core. NRC officials worried the hydrogen bubble might burn or even explode and rupture the pressure vessel. In that event, the core would fall into the containment building and perhaps cause a breach of containment. The hydrogen bubble was a source of intense scrutiny and great anxiety, both among government authorities and the population, throughout the day on Saturday, March 31. The crisis ended when experts determined on Sunday, April 1, that the bubble could not burn or explode because of the absence of oxygen in the pressure vessel. Further, by that time, the utility had succeeded in greatly reducing the size of the bubble. [4]

2. LEGAL FRAMEWORK

Once the initial emergency phase had passed and the accident was under control, attention was given to remediation. Many of the regulatory requirements to allow for decontamination and remediation of TMI-2 were handled through the NRC's existing licensing process, so an extensive change to the overall legal framework was not required. The NRC was clearly established as the regulatory body at the time of the accident, and regulations to protect public health and safety were in effect. Existing Technical Specifications, which were part of the licensing basis for the site, also remained in place unless they no longer applied to the damaged and non-operational facility. New "Recovery Technical Specifications" were generated to assist in remediation and were added to the licensing basis for the site. Even with an established regulatory body and licensing process, some adaptations to the regulatory approach (i.e., decision making and approval processes) were implemented to facilitate remediation, and these are discussed further below.

There was also an understanding that a collaborative effort between the Government and nuclear industry organizations would be useful, and that such a collaboration could foster important post-accident research to inform a prompt and safe remediation. In 1980, four organizations - the General Public Utilities (GPU), the Electric Power Research Institute (EPRI), the NRC, and the Department of Energy (DOE) - formed a group known as "GEND," signing a coordination agreement to implement the TMI-2 Information and Examination Program. This program executed R&D activities relating to the cleanup of TMI-2 and the study of the accident for the enhancement of nuclear power safety and reliability. The group documented the results of this program in a series of about 200 GEND and GEND-INF (informal) reports, as well as many other technical reports issued by the NRC, the DOE and its laboratories, EPRI, and GPU. [1]

As previously noted, some adaptations to the NRC's typical regulatory approach were required to address the unique challenges associated with the accident. At a high level, adaptations were required to address disposal of a unique waste stream and to allow the site to remain in an interim state awaiting decommissioning. These points are discussed below with respect to the Department of Energy Memorandum of Understanding on waste disposal and the NRC's "Post Defueling Monitored Storage" licensing basis for TMI-2.

In order to provide overall direction at the site level, a unique TMI Program Office (TMIPO) was established on April 1, 1980 to oversee TMI-2 recovery and cleanup operations. The TMIPO established a staff with management and technical expertise in key TMI-2 cleanup activities such as radiation protection, radiological assessment, radiological waste treatment, and nuclear safety. The TMIPO had the following regulatory responsibilities: (1) planning and managing all NRC involvement in TMI-2 cleanup activities; (2) obtaining information about and evaluating the current facility status; (3) analyzing and reviewing the licensee's proposed actions and procedures; (4) preparing technical review documents on the safety and environmental impacts of licensee-proposed cleanup actions; (5) approving or disapproving the licensee's proposed actions and procedures; (6) advising the NRC Commissioners on major cleanup actions; (7) coordinating the NRC's TMI-2 cleanup activities with other governmental agencies, as necessary, such as the DOE and EPA; (8) informing State and local governments and the public on the status and plans for cleanup activities; (9) overseeing day-to-day licensee activities to ensure that operations were implemented in accordance with NRC regulations, the facility's operating license, technical specifications, NRC orders, recovery plans, and approved procedures; (10) ensuring that activities are carried out in compliance with approved NRC limits and procedures; and (11) coordinating with the NRC Office of Inspection and Enforcement on its TMI-2 inspection activities. [1]

One noteworthy aspect of the regulatory approach taken with the TMI-2 accident was the usage of the Programmatic Environmental Impact Statement (PEIS) associated with cleanup activities. An Environmental Impact Statement (EIS) is a document required by the U.S. National Environmental Policy Act when a major federal action significantly affects the quality of the human environment. As such, the concept of an EIS is not unique to TMI-2, but the correlation of the TMI-2 PEIS to the cleanup decision making process was unique. The development of the PEIS and the application at TMI-2 are discussed below.

The PEIS was developed after the City of Lancaster, PA expressed concerns, and pursued litigation, regarding potential disposal of processed accident generated water into local waterways. Responding to a directive issued by the Commission on November 21, 1979, NRC staff prepared the draft PEIS dealing with the decontamination and disposal of radioactive waste resulting from the TMI accident. The draft statement (NUREG-0683, “Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive

Wastes Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2, Docket No. 50-320") was released for public comment on August 14, 1980. It discussed four fundamental activities necessary to the cleanup: (1) treatment of radioactive liquids, (2) decontamination of the building and equipment, (3) removal of fuel and decontamination of the coolant system, and (4) packaging, handling, storing, and transporting nuclear waste. The statement addressed the principal environmental impacts that can be expected to occur as a consequence of cleanup activities, including occupational and off-site radiation doses and resultant health effects, socioeconomic effects, and the effects of psychological stress. [1]

On February 27, 1981, the NRC issued the final version of the PEIS (NUREG-0683). NRC staff held 31 meetings with the public, media, and local officials, and the final PEIS included the staff's responses to nearly 1,000 comments received on the draft statement (following a 90-day comment period). The final PEIS reaffirmed the draft statement's conclusion that the decontamination of the TMI-2 facility, including the removal of the nuclear fuel and radioactive waste from the TMI site, was necessary for the long-term protection of public health and safety, and that methods exist or can be suitably adapted to perform the cleanup operations with minimal release of radioactivity to the environment. The final PEIS also concluded that the only environmental impact that might be of significance would be the cumulative radiation doses to the cleanup workers. [1]

On April 27, 1981, the Commission issued a policy statement endorsing the final PEIS (NUREG-0683), and concluded that the PEIS satisfied the Commission's obligations under the National Environmental Policy Act, with the exception of the disposal of processed accident-generated water. The Commission later issued a supplement stating that the PEIS allows staff to act on each major cleanup activity if the activity and associated impacts fall within the scope of those assessed in the PEIS. Based on this statement by the Commission, the PEIS became a crucial document in the regulatory approval process, as all cleanup methodologies proposed by the licensee would have to first be evaluated against the PEIS. This approach provided a clear framework in which the TMIPO could approve procedures and methodologies proposed by the licensee without further Commission approval. Even though the TMIPO was afforded this decision making authority, it is important to note that there was still accountability to and frequent communication with the Commission. For example, TMIPO weekly status reports were generated, which provided a detailed chronology of the plant status, environmental monitoring results, the licensee's recovery activities, NRC actions, and public meetings. [1]

Another unique adjustment to the NRC's regulatory framework was a requirement to utilize more restrictive design objective effluent criteria as mandatory limits for TMI-2. The PEIS noted that "throughout the cleanup, any anticipated releases to the environment must be controlled by the licensee in accordance with the staff's proposed effluent criteria to conform to the individual dose design objectives listed in 10 CFR Part 50, Appendix I, as mandatory limits." On June 26, 1981, NRC staff amended the Environmental Technical Specifications of the TMI-2 license to define the criteria in Appendix R of the final PEIS as limiting conditions of the cleanup operations. The Commission's policy statement declared that the cleanup should be expedited and activities carried out in accordance with the criteria in Appendix R of the PEIS, which limited the doses to off-site individuals from radioactive effluents resulting from cleanup activities. These effluent limits were numerically identical to the design objectives of radioactive effluents for operating power reactors contained in Appendix I of 10 CFR Part 50. The criteria in Appendix R of the PEIS for TMI-2 cleanup activities were more restrictive than those for the operating power reactors, since the Appendix R values were limits that could not be exceeded, whereas, for operating power reactors, they were design objectives to be met on the "as low as reasonably achievable (ALARA)" principle. [1, 2]

The NRC also implemented a number of broader, industry-wide regulatory actions resulting from investigations and lessons learned reviews, and completed many more "spin-off" actions in the decades following the accident. The first wave of actions that the NRC approved, included NRC orders to individual licensees and generic communications, such as bulletins and generic letters issued to nuclear power plant licensees. The NRC used these regulatory tools in the days to months following the accident. On April 1, 1979, the NRC's Office of Inspection and Enforcement issued a series of bulletins instructing all holders of operating licenses to take a number of immediate actions to avoid repeating several events that contributed significantly to the accident's severity (BL 79-05, 05A, 05B, 05C, 06, 06A, 06B, 06C, and 08). The bulletins and other related evaluations also provided substantial input on other staff activities, such as those associated with the generic study efforts and the Lessons Learned Task Force.

In the longer term, investigations and the implementation of lessons learned brought about sweeping changes in the U.S. nuclear industry. These included improvements in emergency response planning, reactor operator training, human factors engineering, radiation protection,

and many other areas of nuclear power plant operations. The NRC has intensively studied and documented the TMI-2 accident. Once the various investigative groups had documented their findings, the Commissioners considered the recommendations. The agency consolidated all of the recommendations that the Commission approved into NUREG-0737, "TMI Action Plan," published in 1980. The NRC found that many action plan items were applicable when reviewed against each specific licensed nuclear power plant. Some of these requirements involved changes to the internal NRC organization, processes, and practices. A few requirements caused the Commission to issue policy statements and specific changes to the NRC's regulations through the rulemaking process. Stakeholder input and public comments are solicited as a normal part of the NRC's regulatory process. As such, both of these long-term regulatory tools required extensive internal and external stakeholder involvement, and were completed during the 1980s. NUREG-0933, "Resolution of Generic Safety Issues," documents the prioritization and closeout of the TMI Action Plan requirements.

NRC Approvals Related to Reactor Defueling

Defueling of the reactor and removal of fuel debris was a major goal of the TMI-2 cleanup. In the near term, shortly after the accident and while the configuration of the fuel was unknown, there was a concern about potential recriticality of the fuel. In the longer term, one of the defueling goals was to prevent long term waste disposal at the site. This goal was clearly noted in the PEIS as follows:

> The staff has concluded that TMI should not become a permanent radioactive waste disposal site. If the damaged fuel and radioactive wastes are not removed, the Island would, in effect, become a permanent waste disposal site. The location, geology, and hydrology of Three Mile Island are among the factors that do not meet current criteria for a safe long-term waste disposal facility. Removing the damaged fuel and radioactive waste to suitable storage sites is the only reliable means for eliminating the risk of widespread uncontrolled contamination of the environment by the accident wastes.

Many interim remediation steps had to be evaluated and approved before defueling could begin, and procedures to remove the fuel and debris were evaluated to ensure safe operations and compliance with the PEIS. As previously noted, many of the requirements to allow for decontamination and remediation of TMI-2 were handled through the NRC's licensing process. As such, many of the NRC evaluations were issued for public comment, in accordance with

process requirements. Several examples of those licensing processes are provided below. [1, 2]

EPICOR II System Approved

On August 14, 1979, the NRC issued for public comment an environmental assessment for the use of EPICOR II in the decontamination of the intermediate level of contaminated water (less than 3.7 MBq/mL [100 µCi/mL]) in the auxiliary building. On October 3, 1979, the NRC issued NUREG-0951, "Environmental Assessment - Use of EPICOR II at Three Mile Island Unit 2." On October 16, 1979, the Commission issued a Memorandum and Order directing the use of EPICOR II. [1]

Recovery Technical Specifications Implemented

On June 25, 1979, the NRC provided draft Recovery Technical Specifications to the licensee for review. On February 11, 1980, the NRC issued NUREG-0647, "Safety Evaluation and Environmental Assessment, Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company, Docket No. 50-320, Three Mile Island Nuclear Station, Unit No. 2." This report contained an NRC Order for the Three Mile Island Nuclear Station, Unit 2, that (1) required that, effective immediately, the facility be maintained in accordance with the requirements of the attached proposed Technical Specifications; and (2) proposed to formally amend the Facility Operating License to include the proposed Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the unit would remain in a safe posture during the Recovery Mode. [1]

Purging of the Reactor Building Atmosphere

In March 1980, NRC staff issued for public comment a draft environmental assessment of a number of alternative options for the decontamination of the reactor building atmosphere. Approximately 800 responses were received from various federal, State, and local agencies and officials, as well as from non-governmental organizations and private individuals. Following appropriate revisions responding to the comments received, and additional reviews and analyses by NRC staff, NUREG-0662, "Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere," was issued in May 1980. The statement discussed several alternative options and the potential environmental impacts associated with each. [1]

NRC Approved the Use of the Submerged Demineralizer System

The NRC's review of a Submerged Demineralizer System (SDS), to provided controlled handling and treatment of highly contaminated wastewater generated during the accident, formally started when the licensee submitted the report "Technical Evaluation Report, Submerged Demineralizer System" in April 1980. However, due to a number of design changes and technical questions from the staff, formal NRC approval was not given until June 1981. [1]

NRC Issued a Safety Evaluation for Post-Defueling Monitored Storage

In August 1988, the licensee submitted a Safety Analysis Report to document and support their proposal to amend the TMI-2 license to a "possession-only" license, and to allow the facility to enter Post-Defueling Monitored Storage (PDMS). In February 1992, the NRC issued a Safety Evaluation for "post-defueling monitored storage," which addressed the license conditions and technical specifications necessary to implement PDMS following evaluations by NRC staff and contractor consultants from Battelle Memorial Institute's Pacific Northwest Laboratory. As part of the evaluation, the staff published a technical evaluation report, which appraised PDMS as an integrated process and assessed licensee commitments that were not within the technical specifications. These two documents and Final Supplement 3 to the Programmatic Environmental Impact Statement (NUREG-0683), which was issued in August 1989, formed the basis for the NRC's position on the PDMS. Later, on September 14, 1993, the NRC issued a possession-only license, and on December 28, 1993, Amendment No. 48 to the license was issued to extensively modify the technical specifications in ways consistent with the licensee's plans for post-defueling monitoring storage of the facility [1]

New Rules, Regulatory Guides, and Policy Statements

Generally speaking, the existing licensing process was utilized to accommodate the specific decontamination and remediation needs of TMI-2. However, lessons learned from the accident led to future regulatory and policy updates that would apply to the industry as a whole. Additional details on regulatory and policy updates are discussed below.

Rulemaking and Regulatory Guides

In most cases, the NRC implemented the TMI Action Plan items through the review of new licensee applications, and the imposition of confirmatory orders and specific license conditions, rather than through specific changes to the NRC's regulations. In several instances, implementation of the TMI Action Plan resulted in new or modified regulations. These

regulations, and a few others that could be considered relevant to the TMI accident, included the following: upgrading emergency planning regulations in 1980, requirements related to hydrogen control in Mark I and Mark II containments for boiling water reactors (BWRs) in 1981, upgrading operations personnel and staffing requirements at nuclear power plants in 1983, issuing a licensee event report rule in 1983, requirements related to hydrogen control in Mark III containments for BWRs and ice condenser containments for pressurized water reactors (PWRs) in 1985, improving the backfitting process for power reactors in 1985, improving personnel dosimetry processing in 1987, updating the operator licensing requirements in 1987, and mandating participation in the Emergency Response Data System program in 1991.

In general, a regulatory guide (RG) describes one acceptable method for implementing agency regulations. Regulatory guides are not substitutes for regulations, and the NRC does not require that licensees comply with them, but they do provide guidance to licensees on acceptable approaches to regulatory compliance. Notable regulatory guides that the NRC revised or created during the implementation of the TMI Action Plan included RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"; RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"; RG 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors"; and RG 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations." [1]

NRC Commission Policy Statements

Several policy statements that the Commission issued were directly or indirectly related to the TMI Accident. A policy statement is not a regulation and does not impose specific regulatory requirements, but rather provides the Commission's rationale and motivation for future regulatory positions. Two notable policy statements that continue to have far-reaching regulatory applications include "Severe Reactor Accidents Regarding Future Designs and Existing Plants," issued in 1985, and "Safety Goals for the Operations of Nuclear Power Plants," issued in 1986. The former provided the basis for redirecting NRC research programs and other regulatory programs, while the latter provided the basis for backfitting and regulatory analyses, and for the use of probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis. Examples of changes to the NRC's research programs include a greater focus on identifying the critical role of human performance in plant safety and the use of risk assessment to identify vulnerabilities of any plant to severe accidents. [1, 3]

Safeguards Considerations Following the Accident

Special Nuclear Material (SNM) accountability is required of all licensee holders of reactor fuel and other SNM, though the unique state and location of fuel debris at TMI-2 presented accountability challenges. Following the accident, damaged fuel debris was dispersed throughout the plant, and the origin of the debris could not be traced to specific fuel assemblies. The NRC and the DOE, who was set to receive the fuel debris, allowed the final SNM accountability for TMI-2 to be performed after defueling was completed. The NRC also granted the licensee exemptions regarding regulatory requirements for record keeping, inventorying, and reporting of special nuclear, source, and byproduct materials. Accountability of SNM was based on a thorough post-defueling survey of areas, systems, and components, and the licensee developed a plan that identified the methods and sequence of SNM accountability; the quality assurance program that was built into SNM measurement activities; and the areas, systems, and components that had undergone formal SNM measurement and those that did not require SNM assessment.

On February 1, 1993, the licensee notified the NRC in its last post-defueling survey report that the current best estimate of the residual fuel remaining in the reactor vessel was 925 kilograms with an uncertainty of plus or minus 40 percent as one standard deviation. The estimate of remaining fuel in the reactor vessel was based on underwater video inspections and passive neutron measurements. Video inspections were used to divide the reactor vessel into nine zones which separated the major fuel deposits by elevation. An array of helium-filled detectors were used to measure fast neutrons produced by the residual fuel as water was removed from the vessel in stages so that the water could be used as a shadow shield to separate the fuel deposits by zone. The estimate was derived from calculations made by onsite staff and an independent review by an offsite group headed by Dr. Norman Rasmussen of the Massachusetts Institute of Technology. The total residual fuel estimate (925 kilograms) from fast-neutron measurements was about 50 percent larger than the less accurate video estimate. For the balance of the facility outside the reactor vessel, earlier licensee estimates based on measurements, sample analyses, and visual observations indicated that no more than 174.6 kilograms (385 pounds) of residual fuel remained. [1]

The Applicability of Clearance Concepts to Remediation

The NRC Final PEIS was developed using surface contamination levels indicated in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors (published in June 1974,

and withdrawn in August 2016)," as those levels were considered complete for unrestricted access or unrestricted release of decontaminated equipment or facilities at the time of the NUREG publication (March 1981). However, it should be noted that the Regulatory Guide 1.86 limits were based on the capabilities of handheld survey instrumentation at the time, and the NRC later clarified policies on the release of materials and equipment by reactor licensees to note that there should be no detectable radioactivity released above background. NRC guidance has since been provided on the maximum acceptable detection limits that reactor licensees can use for surveys to release materials and equipment. Acceptable survey practices are described in NRC IE Circular No. 81-07, "Control of Radioactively Contaminated Material," and "Information Notice No. 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities." (Note: These guidance documents were not established as a result of the TMI-2 accident, but rather, as a result of ongoing communications to all reactor licensees.) As the NRC does not have established clearance limits, the policy on no detectable radioactivity would currently apply for any future release of materials and equipment from TMI-2.

Examples of Risk Informed Prioritization

The NRC utilized a risk informed approach to prioritize the approval of remediation efforts at TMI-2 while also ensuring that public interest was considered. There was an overarching goal, as described in the PEIS, to fully cleanup the TMI-2 facilities "as expeditiously as reasonably possible to reduce the potential for uncontrolled releases of radioactive materials to the environment." As previously discussed, removal of damaged fuel and radioactive waste to suitable storage sites was the ultimate cleanup goal, and this was presented in the PEIS as "the only reliable means for eliminating the risk of widespread uncontrolled contamination of the environment by the accident wastes." These underlying goals influenced day-to-day activities of the TMIPO, such as reviewing and approving (or disapproving) the licensee's proposed actions and procedures, as well as major efforts, such as the removal and treatment of accident generated water and the purging of krypton-85 from the reactor building. Along with the goals to expeditiously clean up and remove fuel and wastes, it was necessary to consider stakeholder concerns about the potential for public radiological exposures and releases to the environment. Because of the complexity and public interest surrounding the TMI-2 accident, comprehensive environmental assessments of both the purging and water treatment operations were required by the NRC. These activities could have been allowed under the same effluent limitations as would apply in the case of a normally-operating facility, had the Commission not determined that public interest warranted prohibiting these undertakings pending completion of an

environmental review.

With regard to the purging of krypton-85 from the reactor building, the NRC was able to make a risk informed decision to allow purging operations to occur, as this was a necessary step to allow workers to enter the reactor building to perform assessments, maintenance, and remediation. The Commission issued a Memorandum and Order on June 12, 1980, which authorized the licensee to clean the reactor building's atmosphere by means of a controlled purge. On the same day, the Commission issued a temporary modification of the TMI operating license setting offsite dose limits for the purge. The order included a one-time waiver of the environmental technical specifications requirements for the instantaneous and quarterly average limits for the release of noble gases. On July 24, 1980, Amendment No. 11 to the facility's operating license was issued to allow the bypassing of the interlocks from the reactor building's exhaust radiation monitors to the reactor building's exhaust purge dampers for the duration of the purge. During the purging event, radiological monitoring was performed by several government agencies and by local citizens. The maximum cumulative radiation dose and the maximum dose rate measured at offsite locations were a fraction of the limits allowed under NRC regulations.

With regard to radiologically contaminated water in building basements, its presence there constituted a continuing risk of leakage to the environment, and prevented or hindered the performance of the major decontamination activities. The NRC took actions in the first few months after the accident to assess treatment and storage options for contaminated water, while the ultimate disposal options of treated water were considered later. On August 14, 1979, in response to the Commission policy statement of May 25, 1979, NRC staff prepared and sent out for public comment an environmental assessment for the use of the EPICOR II filtration and ion-exchange decontamination system to remove radionuclides from intermediate-level radioactive waste water held in storage tanks in the TMI-2 auxiliary building (NUREG-0591). The proposed action was limited to cleanup and storage of waste water. The action also included the impacts of wet solid waste generated from EPICOR II operation, such as temporary storage, packaging, handling, transportation, and burial. On October 16, 1979, the Commission issued a "Memorandum and Order" directing the licensee to operate the EPICOR II to decontaminate intermediate-level radioactive waste water held in storage tanks in the TMI-2 auxiliary building. In response to that order, NRC staff issued an "Order for Modification of License" two days later to permit EPICOR II system operations. EPICOR II began operation on October 22. [1]

A risk informed approach to prioritizing and approving remediation actions such as these could be viewed as a part of the overall graded approach concept applied to decommissioning. The NRC also used a risk informed approach to evaluate regulatory exemption requests from the licensee, several of which were approved throughout the cleanup of TMI-2.

3. INSTITUTIONAL FRAMEWORK AND STRATEGIC PLANNING

3.1 Organizational Changes

Several organizational changes took place as a result of the TMI accident, affecting the licensee as well as the NRC as the regulator. These are discussed below as "Licensee Organizational Changes" and "Regulatory Organizational Changes."

Licensee Organizational Changes

With regard to industry resources and licensee organizational changes, the Electric Power Research Institute (EPRI), discusses organizational changes and resource requirements in their document titled "The Cleanup of Three Mile Island Unit 2, A Technical History: 1979 to 1990, EPRI NP-6931, (1991)." Within that document, EPRI discusses 4 different organizational structures which evolved over the life of the recovery and remediation projects. The initial organization included a staff of nearly 2000 people onsite (with around the clock coverage) focused on the immediate effects of the accident, and included leaders from across the U.S. nuclear industry. The second organization, in place by 1980, was described as being more "departmental in structure," with additional focus on radiological controls as personnel protection required much more attention than in a normally operating power plant. The report further describes the third organizational structure as focused more on the "growing sophistication of project management in terms of understanding the requirements for recovery, the overwhelming organizational need to make the project work efficiently, and the fact that, with the plant in effective cold shutdown, the need for redundant organizations was eliminated." The fourth organization, established around 1985, was mostly focused on defueling operations. [5]

Organizational changes were made to the nuclear industry as a whole based upon

recommendations of the Kemeny Commission (a 12-member commission appointed by President Jimmy Carter to investigate the TMI-2 accident). Specifically related to industry, the Kemeny Commission recommended the following:

- The (nuclear power) industry should establish a program that specifies appropriate safety standards including those for management, quality assurance, and operating procedures and practices, and that conducts independent evaluations.
- There must be a systematic gathering, review and analysis of operating experience at all nuclear power plants, coupled with an industrywide international communications network to facilitate the speedy flow of this information to affected parties.

To address these recommendations, the Institute of Nuclear Power Operations (INPO) was established with a mission "to promote the highest levels of safety and reliability – to promote excellence – in the operation of commercial nuclear power plants." [6]

Regulatory Organizational Changes

The regulatory organizational arrangements/structures shifted from operational NRC oversight prior to the accident, to an augmented on and off-site emergency response structure during the emergency phase, and to a new structure enhanced by lessons learned after the emergency.

There were many changes in the regulatory approach taken as a result of the accident, and these were based upon risk, safety significance and the condition of the facility or nuclear materials. A specialized U.S. NRC team was established soon after the accident. The team began to form with the arrival of NRC's Office of Inspection and Enforcement (IE) and Region I inspectors shortly after the accident, and continued to expand with the arrival of the first contingent from the Office of Nuclear Reactor Regulation (NRR) on March 29 and additional inspectors from all five regional offices. On March 30, the Director of NRR and additional NRR staff arrived at the site to assist in the recovery operation. A Public Affairs Office was also established in Middletown, PA, and staffed on a 24-hour basis to manage the flow of information to the public and the media. Initially, the NRC site team supported emergency response functions for the NRC and the U.S. Government. Within days of the accident, the site team performed on-site recovery activities, which can be broken down into the following four major areas:

1. Review system modifications and system additions
2. Review all procedures, both emergency and normal operation and maintenance, which were necessary to post-accident activities.
3. Provide close and continuous monitoring for the operations.
4. Provide consultation, review, and analysis of the ongoing radwaste, cleanup, and health physics activities. [1]

In addition to the TMIPO, which was formed specifically to provide overall direction for the TMI-2 cleanup operations and inspections, the NRC established the Office for Analysis and Evaluation of Operational Data (AEOD), with a broader reaching mission to analyze and evaluate operational safety data for all NRC-licensed activities (reactor and nonreactor) and to develop formal guidance for the agency on the collection, evaluation, and feedback of operational data. [7]

3.2 Funding

Funding is discussed in two parts – accident remediation and final decommissioning. Remediation of the accident is complete, and TMI-2 exists today in a post-defueling monitored storage state. TMI-2 is expected to be fully decommissioned at a later date, after expiration of the TMI-1 license.

Accident Remediation Funding

An early estimate of $500 to 600 million to "decontaminate and repair the damaged nuclear reactor and related facilities" was provided in a July 7, 1980 report by the Comptroller General of the United States titled "Three Mile Island: The Financial Fallout." The actual cost to remediate TMI-2 was around $1 billion. Approximately 2/3 of the $1 billion required for remediation was covered by the licensee (via the utility's shareholders, customers, and insurance underwriters), while the remaining amount was provided by the U.S. Department of Energy, the electric power industry, the States of Pennsylvania and New Jersey, and the Japanese government/nuclear industry (approximately $18 million - for research, knowledge management, and training purposes). [8, 9]

Final Decommissioning Funding

A decommissioning fund is maintained by the licensee to complete the final decommissioning of the site. An annual funding status report is provided to the NRC, and the latest report indicates

that the site-specific TMI-2 radiological decommissioning cost estimate is $1,180,928,000 (escalated to 2014 dollars).

Compensation of Parties Affected by the Accident

One financial consideration for damaged nuclear facilities, in addition to funding for remediation and decommissioning, is compensation that may need to be paid to affected members of the public. In the case of TMI-2, the Price-Anderson Act provided liability insurance to the public. The Price-Anderson Act became law on September 2, 1957, to cover liability claims of members of the public for personal injury and property damage caused by a nuclear accident involving a commercial nuclear power plant. The legislation helped encourage private investment in commercial nuclear power by placing a cap, or ceiling, on the total amount of liability each nuclear power plant licensee faced in the event of an accident. At the time of the accident, private insurers had $140 million of coverage available in the first tier pools. Insurance adjusters advanced money to evacuated families in order to cover their living expenses, requesting that unused funds be returned; recipients sent back several thousand dollars. The insurance pools also reimbursed more than 600 individuals and families for wages lost as a result of the accident. The insurance pools were later used to settle a class-action suit for economic loss filed on behalf of residents who lived near TMI. The Price-Anderson Act covered court fees as well. The last of the litigation surrounding the accident was resolved in 2003. Altogether, the insurance pools paid approximately $71 million in claims and litigation costs associated with the TMI-2 accident.

As a result of the TMI-2 accident, and out of concern that licensees may be unable to cover onsite cleanup costs resulting from a nuclear accident, the NRC established new regulations that require licensees to maintain a minimum of $1.06 billion in onsite property insurance at each reactor site. This insurance is required to cover the licensee's obligation to stabilize and decontaminate the reactor and site after an accident. Currently, only Nuclear Electric Insurance Limited provides this insurance for licensees. These regulations are found in 10 CFR 50.54(w), "Conditions of licenses." [10]

3.3 Strategy

Several adjustments in strategy occurred as a result of the TMI-2 accident. These are discussed below as Regulatory Oversight Strategy, Decommissioning and Waste Management Strategy, and Lessons Learned Strategy.

Regulatory Oversight Strategy

With regard to remediation activities that would lead to decommissioning, regulatory oversight practices were augmented by the formation of a special TMI Program Office (TMIPO) within the NRC's Office of Nuclear Reactor Regulation which provided overall direction for the TMI-2 cleanup operations and inspections. This office was staffed by approximately 30 employees during the first few years. Additional information is found in the NRC NUREG-0698 document. The TMI-2 Project Directorate was dissolved in February 1988, and the inspection program for TMI-2 was assumed by the TMI resident inspection staff. Technical review and project management functions were assumed by a NRC Headquarters project directorate. [1]

Decommissioning and Waste Management Strategy

As discussed in the "Legal Framework" section of this paper, the TMI-2 license was amended to a "possession-only" license after accident remediation. This allowed the facility to enter a post-defueling monitored storage (PDMS) state until eventual site decommissioning after TMI-1 ceases operations. This represents a unique licensing strategy that is currently held only by TMI-2.

Unique strategies were also required to address portions of the TMI-2 accident waste. A Memorandum of Understanding was required between the U.S. NRC and DOE to address certain types of wastes as described below.

<u>NRC and DOE Signed Memorandum of Understanding</u>

On July 15, 1981, the NRC and DOE signed a Memorandum of Understanding (MOU), which formalized the working relationship between the two agencies with respect to the removal and disposal of solid nuclear waste generated during the cleanup of TMI-2. This was a significant step toward ensuring that the TMI site would not become a long-term waste disposal facility. The MOU covered only solid nuclear waste, and did not cover liquid waste resulting from the cleanup activities. The MOU addressed three basic categories of TMI-2 waste: (1) waste determined by DOE to be of generic value in terms of beneficial information to be obtained from further research and development activities (the MOU calls for DOE to perform such activities at appropriate DOE facilities); (2) waste determined to be unsuitable for commercial land disposal because of high levels of contamination, but which DOE may also undertake to remove, store, and dispose of on a reimbursable basis from the licensee; and (3) waste

considered suitable for shallow land burial, to be disposed of by the licensee in licensed, commercial low-level waste burial facilities.

The MOU is provided in Appendix A to NUREG-0698, Revision 1, "NRC Plan for Cleanup Operations at Three Mile Island Unit 2." [1]

NRC and DOE Revised Memorandum of Understanding to Accept Fuel and Resins

The NRC and DOE agreed to a revision of the Memorandum of Understanding (MOU) in March 1982. Instead of taking only samples of the damaged fuel from TMI-2, DOE agreed to accept the entire core for research and development, and for storage at a DOE facility. The terms of ultimate disposal of the core will be negotiated between DOE and the utility operating the TMI facility. DOE also agreed to take possession of highly radioactive resins from the purification system, again on the basis of future reimbursement by the utility, and planned to take possession of zeolite waste from the Submerged Demineralizer System and retain it for research and testing with regard to waste immobilization.

The revised MOU is provided in Appendix A to NUREG-0698, Revision 2, "NRC Plan for Cleanup Operations at Three Mile Island Unit 2." [1]

Lessons Learned Strategy

Several investigations were completed and several committees were formed to develop new strategies and to address lessons learned from the TMI accident. Several of these are discussed below.

Two weeks after the accident, the President of the United States, Jimmy Carter, appointed a 12-member Presidential Commission to investigate the accident at Three Mile Island. This group, known as the "Kemeny Commission," conducted a comprehensive investigation of the accident and made recommendations based upon their findings. The "Kemeny Report" was issued in October 1979. The NRC published its initial response to the Presidential Commission's recommendations in November 1979 as NUREG-0632, "NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island." A subsequent report, NUREG-1355, "Status of Recommendations of the President's Commission on the Accident at Three Mile Island - A Ten-Year Review," updated that initial response to include each of the 44 recommendations made by the Kemeny Commission over a 10-year

period.

To help gain a comprehensive insight into the accident, the NRC sponsored both internal and external investigations. The NRC asked the independent Special Inquiry Group, known as the "Rogovin Committee" to perform an investigation. The Special Inquiry Group provided a thorough analysis and assessment of the causes and implications of the accident, and the NRC published this work in NUREG/CR-1250, "Three Mile Island, A Report to the Commissioners and to the Public." Working internally, the NRC created a Lessons-Learned Task Force. The NRC task force identified and evaluated safety concerns originating from the TMI-2 accident that required licensing actions at other nuclear power plants. The NRC published the task force's conclusions in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report.

Many other groups, both internal and external to the NRC, performed separate investigations. These included the U.S. Congress and its General Accounting Office (GAO); the Ad Hoc Dose Assessment Group, which comprised various Federal agencies (see NUREG-0558); the NRC's Advisory Committee on Reactor Safeguards; the Bulletins and Orders Task Force of the NRC's Office of Nuclear Reactor Regulation (see NUREG-0645); the NRC's former Office of Inspection and Enforcement's Special Review Group (see NUREG-0616), the Task Force on Lessons Learned (see NUREG-0600), and investigation of information flow during the accident (see NUREG-0760); the NRC's Siting Task Force (site location requirements for nuclear power plants); the NRC's Emergency Preparedness Task Force; the Staff Panel on the Commission's Determination of an Extraordinary Nuclear Occurrence (see NUREG-0637); the NRC's Office of Nuclear Regulatory Research; and the NRC's former Office of Standards Development. [1]

TMI-2 Lessons Learned Task Force Formed

In May 1979, an interdisciplinary team of engineers from the NRC's Offices of Nuclear Reactor Regulation, Nuclear Regulatory Research, Inspection and Enforcement, and Standards Development began to identify and evaluate those safety concerns originating from the TMI-2 accident that required licensing actions. The scope of the task force assignment covered the following general technical areas:

- Reactor operations, including operator training and licensing.

- Licensee technical qualifications.
- Reactor transient and accident analysis.
- Licensing requirements for safety and process equipment, instrumentation, and controls.
- On-site emergency preparations and procedures.
- NRR accident response role, capability, and management.
- Feedback, evaluation, and utilization of reactor operating experience.

The task force proceeded in two phases:

Short-Term Recommendations

The first phase culminated in the issuance of NUREG-0578, "TMI-2 Lessons Learned Task Force: Status Report and Short-Term Recommendations" (July 1979). The Director of NRR ordered the implementation of 23 short-term licensing requirements in September 1979, based on a favorable review by NRC's independent Advisory Committee on Reactor Safeguards (ACRS) received in August.

Final Recommendations

In the second phase of its work, the task force considered more fundamental questions in the design and operation of nuclear power plants, and in the licensing process. The issues were grouped into four general categories: (1) general safety criteria, (2) system design requirements, (3) nuclear power plant operations, and (4) nuclear power plant licensing. NUREG-0585, "TMI-2 Lessons Learned Task Force: Final Recommendations," was issued in October 1979 to complete this phase. [1]

4. TECHNICAL ISSUES

4.1 Initial Issues

The major technical issues affecting decontamination and remediation planning were the necessity to treat and dispose of accident contaminated water as well as to dispose of fuel debris and other highly contaminated materials. There was an overarching goal to remediate the site to a point where "normal" decommissioning could occur so that the facility did not effectively

become a long term disposal site. Certain steps had to occur before workers could enter some areas to begin remediation, such as purging the reactor building of krypton-85 and the removal and treatment of a large volume of contaminated coolant water. Once workers were able to enter the facility and begin remediation, worker safety was a main priority, as was evidenced by the attention to shielding, specialized dosimetry, and health physics practices.

Decontamination and Remediation

After the accident, a portion of the fuel within the reactor vessel had melted and contamination had dispersed within the facility. As such, contaminated coolant water existed as well as surface and gaseous contamination within the reactor building. Measures taken to remediate fuel debris and contamination are discussed below.

Purging of Reactor Building Atmosphere

Having reviewed the staff assessment and recommendations, together with the comments from the public, the Governor of Pennsylvania, and many others, the NRC's Commission issued a Memorandum and Order authorizing the licensee to clean the reactor building atmosphere by means of a controlled purge, or release of contaminated air through filter systems. On the same day, the Commission issued a modification of the TMI operating license setting off-site dose limits for the purge. The purging operation, which began on June 28, 1980, was carried out under detailed procedures approved by NRC staff. The purging operation, which began on June 28, 1980, was completed on July 11, 1980. Measurements showed that about 1.59E+15 becquerels [43,000 curies] of krypton-85 was released during this period. Samples from the release flow were analyzed to ascertain the presence of radionuclides other than krypton, and the amounts were determined to be insignificant.

During the entire operation, members of the NRC staff were on-site to monitor the licensee's activities. In addition, off-site radiation monitoring programs were conducted by the licensee, the NRC, the Environmental Protection Agency, the Department of Environmental Resources of the Commonwealth of Pennsylvania, and also by private individuals through the Community Radiation Monitoring Program set up by the U.S. Department of Energy and the Commonwealth of Pennsylvania. The maximum cumulative radiation dose and the maximum dose rate measured at off-site locations were a fraction of the limits allowed under NRC regulations. [1]

Decontamination of Intermediate Level Contaminated Water

It was necessary to decontaminate intermediate level contaminated water (defined as less than 3.7 MBq/mL) in the auxiliary building, and this was accomplished using the EPICOR II filtration system. The EPICOR II system, that was used to decontaminate around 1,438,456 liters [380,000 gallons] of intermediate-level radioactive water held in auxiliary building tanks, is shown in Figure 3. It consists of three process vessels (steel liners) shielded by 10.16 cm [4 inch] lead enclosures located in the chemical cleaning building. Each vessel contains ion-exchange resin. The vessel at the top of the photo at the left is the system prefilter/demineralizer, the center vessel is a cation ion-exchanger, and the third vessel is a mixed-bed polishing ion-exchanger. Each is fitted with three quick-disconnect hoses: a liquid waste influent line, a processed waste effluent line, and a vent line with attached overflow hose. Vented air from each vessel passes through a special filter and charcoal absorber. "Spent" ion-exchange resin liners containing radioactive material removed from the water are transferred by crane to cells (shown at top right) which are housed in modular concrete storage structures (shown at bottom right). The cells are concrete-shielded, galvanized corrugated steel cylinders 2.13 meters [7 feet] in diameter and 3.96 meters [13 feet] high. The storage module shown under construction has 1.22 meter [4 foot] thick walls and is 17.37 meters [57 feet] wide and 27.74 meters [91 feet] long. Each module holds about 60 storage cells. The modular design allowed additional storage modules that could be built on an as-needed basis. [1]

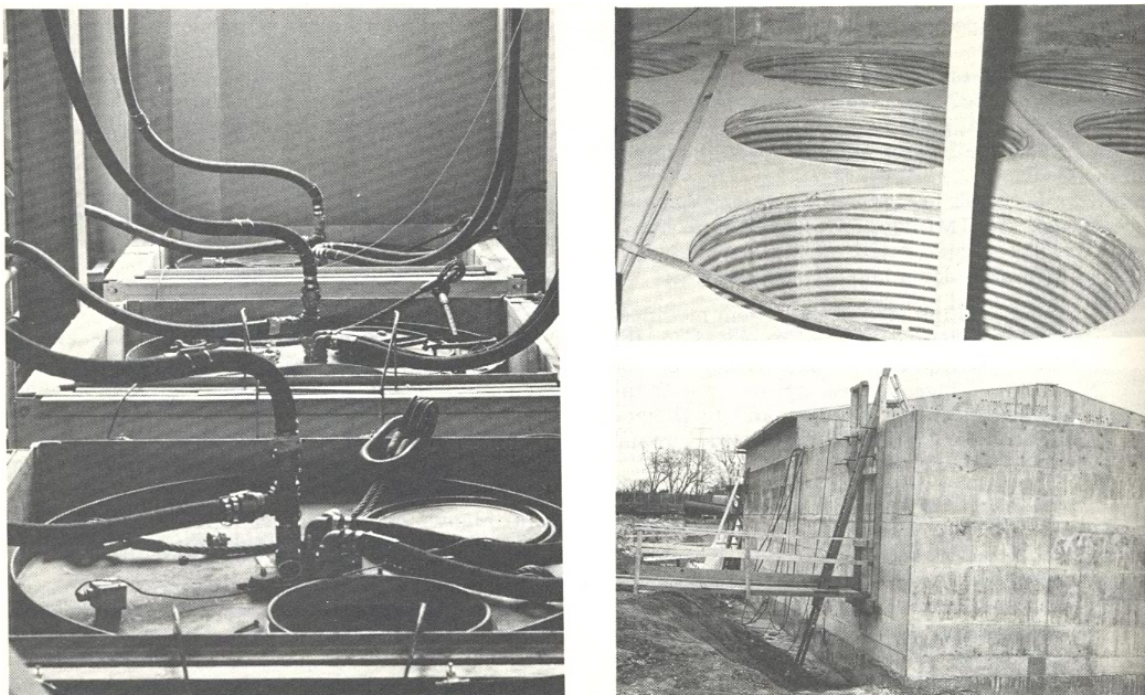

Figure 3 - EPICOR II System, Process Vessels and Storage Overview

Decontamination of Highly Contaminated Wastewater

On April 10, 1980, the licensee formally submitted its Technical Evaluation Report (TER) and requested permission to operate an underwater demineralization system. The Submerged Demineralizer System (SDS), as described in the licensee's TER, was designed to provide controlled handling and treatment of the highly contaminated wastewater generated during the accident. The SDS operated underwater, in one of the spent fuel pools of TMI Unit 2. It consisted of a liquid waste treatment subsystem, a gaseous waste treatment subsystem, and a solid waste handling subsystem. The liquid waste treatment subsystem was designed to decontaminate the high-activity wastewater by filtration and ion exchange. The primary components of the liquid waste treatment subsystem included two filters, and two parallel trains of four identical inorganic zeolite-filled ion exchanger vessels. In the event that additional cleanup of the effluent from SDS was required, it could be recycled through SDS or polished (refined) with the EPICOR II system.

On June 18, 1981, the licensee was directed to promptly commence and complete processing of the remaining intermediate-level contaminated water (less than 3.7 MBq/mL) in the auxiliary building tanks and the highly contaminated water in the reactor building sump and the reactor coolant system.

On August 9, 1981, the remaining 378,541 liters [100,000 gallons] of intermediate-level water were completely processed. The licensee started processing the high-activity water in September 1981. The approval to operate SDS did not include water disposal. All processed water was stored in existing on-site tanks. Decisions related to the disposal of processed water were made by the Commission at a later date (see NUREG-0683, Supplement 2, issued in June 1987). [1]

Removal of Fuel Debris from the Reactor

In October 1985, operators began to remove damaged fuel and structural debris from the reactor vessel by "pick and place" defueling of the loose TMI-2 core debris. Workers performed defueling operations from a shielded defueling work platform (DWP), which was located 2.74 meters [9 feet] above the reactor vessel flange (See Figure 4). The platform had a rotating 5.18 meter [17 foot] diameter surface with 15.24 cm [6 inch] steel shield plates, and was designed to provide access for defueling tools and equipment into the reactor vessel. The DWP supported defueling operators, specially designed long-handled tools, remote viewing equipment, and two

jib cranes used to manipulate the tools. Numerous manual and hydraulically powered long-handled tools were used to perform a variety of functions, such as pulling, grappling, cutting, scooping, and breaking up the core debris, several of which are shown in Figure 5. These tools were used to load debris into defueling canisters positioned underwater in the reactor vessel. The canisters were then sealed and transported using shielded canister transfer equipment to submerged storage racks in spent fuel pool "A" of the auxiliary and fuel handling building (AFHB). The canisters were designed and stored to prevent an inadvertent criticality event. Following dewatering to control the buildup of combustible gases, the canisters were loaded into a specially designed shipping cask and transported to a Department of Energy facility in Idaho for interim storage and research.

In December 1985, several defueling canisters were filled with debris consisting of fuel assembly end fittings, control rod spiders, and small pieces of fuel assemblies. In January 1986, the first group of defueling canisters was sealed, dewatered, and transferred to storage racks in spent fuel pool "A" in the AFHB.

The heavy-duty long handled tools were only marginally successful, so a drilling rig that was used for boring core samples was reinstalled as the primary tool for breaking up the hard mass of core debris, and drilling commenced in September 1986. Additional details on this drilling rig are provided below in the "Post-Accident Evaluations" discussion.

Dose rates to personnel during the initial phase of defueling were low and remained low throughout the year, averaging less than 0.1 millisievert per hour [10 millirem per hour] on the DWP and less than 0.4 millisievert per hour [40 millirem per hour] near the shielded canisters during transfer. The licensee discontinued the use of respirators during defueling activities, based on air sample data collected during the first month. [1]

Figure 4 - Shielded Defueling Work Platform (DWP)

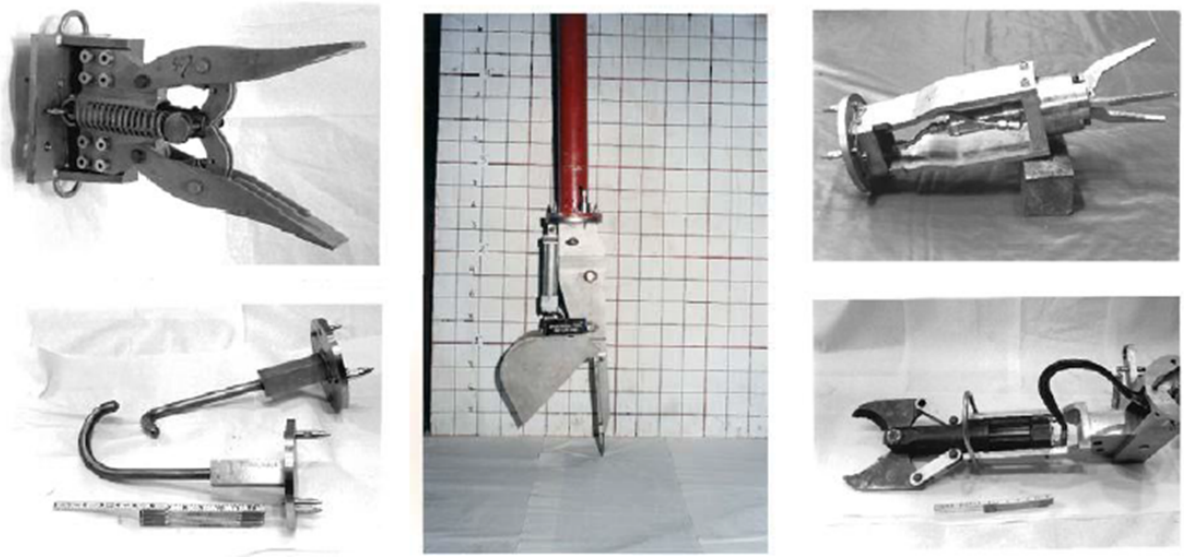

Figure 5 - Manual and Hydraulically Powered Long-Handled Tools Used to Load Debris into Defueling Canisters Positioned Underwater in the Reactor Vessel

Remote Robotic Equipment

In addition to the long handled tools discussed above, remote robotic technologies played a role in the remediation of TMI-2. Remote-controlled robotic vehicles and supporting control equipment were used extensively to perform work in the reactor building's basement, the makeup demineralizer room in the auxiliary building, the reactor coolant pump seal injection valve room in the fuel handling building, and the reactor vessel. These vehicles were both versatile and productive, and proved useful in many different tasks, including video camera inspections, radiation monitoring, sediment sampling, acquisition of concrete core samples, high pressure water flushing, concrete scabbling and scarification, and debris pickup and removal. In addition to the remote vehicles, fixed-position, remotely operated tools were developed for work inside the reactor vessel. The tools included a plasma arc cutting system to remove the stainless steel core support assembly, and several manipulator arms for handling damaged fuel and structural components. The use of robots at TMI-2 did not require any NRC licensing actions; however, activities in which robots were used, like most recovery and cleanup activities, required safety evaluations by the licensee and the NRC. Key robots used at TMI-2 are summarized below:

ROVER, or remote reconnaissance vehicle (RRV), was used in the reactor building's basement to perform video and radiation surveys, collect sludge samples from the floor, collect core samples from the wall surface, flush walls with high-pressure water, remove the surface of the walls using an ultrahigh-pressure scarification system, and remove sludge. The RRV was a tether-controlled, six-wheeled work platform with multiple attachments or modules to perform the various tasks. ROVER was operated by two operators in a control room located outside the reactor building.

LOUIE I remote vehicle was used to measure the radiation profiles of the two makeup demineralizer vessels, and to remove loose pre-accident debris and salt deposits on the floor inside the seal injection valve room. On loan from DOE, LOUIE I was a tether-controlled, tracked work platform with a telescoping boom-mounted manipulator that had been in use at DOE's Hanford Site since the 1950s. The control console was setup in the accessible hallway near the entrance to the room.

LOUIE II remote vehicle was used to perform remote floor scabbling in the seal injection valve room. An attached 3-piston pneumatic scabbler was used to pulverize the floor surface while

vacuuming loose concrete. LOUIE II was a tether-controlled, six-wheeled work platform with a heavy frame to withstand the stresses of the scabbler. The control console was setup in the accessible hallway near the entrance to the room.

WORKHORSE, or remote work vehicle (RWV), was a large, heavy-duty robot built for decontamination and demolition work in the basement of the reactor building. WORKHORSE was a tether-controlled, six-wheeled work platform that was 10 times heavier than the RRV and had a boom with a seven-meter vertical reach. A two-level control building was built in the turbine building for three operators and support staff to operate the RWV and manage the work activities. The RWV was successfully tested in mockups, but never used due to changes in cleanup direction.

Mini-Rover was a commercial submarine vehicle modified to remove larger fuel debris inside the pressurizer. The Mini-Rover had a color camera, pincer arm to break apart debris, and a scope. The submarine was fitted with a ballast tank to improve mobility.

A remote manipulator performed defueling operations in areas of the reactor vessel that were not directly below the defueling work platform's working slots. The manipulator could handle defueling tools and the in-vessel video viewing system, and pickup fuel debris. The manipulator was mounted to the manual tool positioner on the defueling work platform and had a 1.22 meter [4 foot] reach and six degrees of freedom of motion. The manipulator could be controlled from inside or outside the reactor building.

ACES, or automated cutting equipment system, was a remote-controlled plasma arc torch installed in the reactor vessel to cut the multi-layered lower core support assembly following bulk defueling. ACES consisted of a support bridge with a carriage and trolley to provide horizontal X-Y plane movement. A manipulator arm provided motion in the vertical Z-axis direction, including rotation and bending motions. At the lower end of the manipulator arm was a pneumatically operated gripper with a plasma torch and effector for cutting. A high-velocity stream of high-temperature ionized nitrogen gas plasma transferred an electric arc to melt the cut area. The two control consoles (one for the manipulator subsystem and the other for plasma subsystem) were located in the turbine building.

MANFRED, or manipulators for reactor defueling, was a robot manipulator system designed

and built for underwater disassembly and defueling of reactor vessel components. MANFRED consisted of a manipulator with various work attachments and grabber manipulators. The system was tested but never used due to the success of ACES (above) and manual handling of component pieces and loose fuel debris. [1]

Worker Protection Challenges

Several unique worker protection challenges existed during remediation of the TMI-2 accident. Post-accident radiological conditions at TMI-2 were substantially different from those normally encountered at commercial operating nuclear plants because of the magnitude and specific mix of the radionuclide contamination. Radiation surveys made shortly after the accident showed that general area radiation readings ranged from 1.5 to 5 millisievert per hour [150 to 500 millirem per hour] in the fuel handling building, and 0.5 to 50 millisievert per hour [50 to 5,000 millirem per hour] in the auxiliary building. Hot spots were measured in the auxiliary building reaching up to 1.25 sievert [125 rem] per hour, and exceeding 10 sievert [1,000 rem] per hour in some cubicles. The high-energy beta component was up to one hundred times the gamma component.

During the first entry into the reactor building in July 1980, dose rates at the 305-foot entry-level elevation ranged from 4 to 6 millisievert per hour [400 to 600 millirem per hour]. Localized areas of high radiation were measured at 0.18 sievert [18 rem] per hour over the open stairwell and 0.02 to 0.05 sievert [2 to 5 rem] per hour at floor drains. The general-area floor and wall beta-radiation readings ranged from 0.01 to 0.02 gray [1 to 2 rad] per hour. Surveys performed during the second entry at the next-higher level, the 347-foot operating-floor elevation, showed general radiation readings of 1 to 4 millisievert per hour [100 to 400 millirem per hour]. Below the 305-foot entry level elevation was the 282-foot basement-level elevation, which was flooded with highly-contaminated water and sludge. A telescoping radiation detector was inserted down through one of the reactor building stairwells and measured 0.40 to 0.45 sievert [40 to 45 rem] per hour at 1.52 to 2.13 meters [5 to 7 feet] from the surface of the basement water. Once the water was drained and processed through the submerged demineralizer system, dose rates from the remaining sludge ranged from 0.01 to 10 sievert [1 to 1,000 rem] per hour, depending on location and distance from the floor.

A unique concern at TMI-2 was high-energy beta contamination from fission products in the reactor coolant. The generation of activation products from corrosion such as cobalt-60 was

minimal because the new plant had less than a year of full-power operation. Areas in the auxiliary building that had experienced coolant leakage were measured in the 0.1 to 1 sievert [10 to 100 rem] per hour gamma range with associated beta dose rates in the 10 to 100 gray [1,000 to 10,000 rad] per-hour range. Similar gamma-to-beta ratios were measured on surfaces in the reactor building. The cesium isotopes have beta energies in the 0.5-million-electron-volt (0.5-MeV) range; strontium-89, which has the highest concentration of beta emissions, has a maximum energy of about 1.5 MeV; yttrium-90, the decay daughter of strontium-90, has a 2.3-MeV beta energy.

Typically, beta radiation from beta-emitting radionuclides at operating plants is low-energy, so protective clothing provides sufficient shielding. The high-energy beta emitters present at TMI-2 required special radiological protection practices, such as monitoring equipment, personnel dosimetry, heavy protective clothing, procedures, and training. Access into many areas in the plant with high levels of high-energy beta-emitting yttrium-90 required a combination of double-respirator face pieces or face shields and safety glasses. The need for these special worker protection practices became clear after several incidents of skin and extremity exposures exceeding regulatory limits. [1]

Disposal Challenges

There were several challenges associated with the disposal of various wastes generated during accident remediation. These included disposal and interim storage of solid wastes, disposal of the EPICOR II and SDS liners, and the ultimate treatment/disposal of slightly contaminated water, as discussed below.

Disposal and Interim Storage of Solid Wastes

Several onsite facilities were constructed for temporary storage of solid radioactive waste products from cleanup activities that were being readied for transportation. Solid waste included spent EPICOR I and EPICOR II resin liners; contaminated clothing, tools, and equipment; and decontamination materials. Note: The EPICOR I system was installed within a week after the accident to process low-activity, non-accident-generated liquid waste water, mainly generated from the TMI-1 outage. Over its 19-month lifetime, EPICOR I processed over 4,921,035 liters [1.3 million gallons] at an average flow rate of 37.85 liters [10 gallons] per minute.

The interim storage facility was built to temporarily store EPICOR resin liners and filters until

the construction of the solid waste staging facility was completed. The earthen facility was constructed underground from corrugated galvanized cylinders with a welded bottom. A 0.91 meter [3 foot] thick concrete cover provided shielding at the top. The ground provided radial shielding. The loaded liners were placed on a galvanized drip pan. The design requirements for the facility were provided to the licensee by the NRC. The facility was ready for use on November 5, 1979. The facility was abandoned in place after the liners were relocated to the new solid waste staging facility.

The interim solid waste staging facility (also known as the "car port") was used to collect and temporarily store (stage) low-level solid waste packages from both Units 1 and 2. All waste packages placed in the facility, such as liners, drums, and metal boxes, were already prepared for shipment. The facility performed passive functions of protecting waste packages from precipitation and provided the means to load packages onto trucks. The facility included a truck bay for loading and unloading. The facility was originally sized to accommodate the waste generated by both units over a 6-month period. Waste packages could be stored for up to 5 years. The facility was ready for use on December 16, 1982.

The solid waste staging facility was used to collect and temporarily stage radioactive waste, such as dewatered resins, filters, and sludge from both Units 1 and 2 before shipment. Two of six planned concrete storage modules were built, each consisting of 60 cells. Each cell was constructed of galvanized corrugated steel cylinders with a welded steel base plate, and was sized to accommodate any combination of waste containers, such as resin liners, metal boxes, and drums. A concrete cover 0.91 meters [3 feet] thick and weighing about 12.7 tonnes [14 tons] covered each cell. Both modules shared a sump compartment to collect drainage. The facility performed no active function other than storage. Storage Module "A" went into service in January 1980.

The waste handling and packing facility was used to process and package solid radioactive waste, such as contaminated clothing, tools, and equipment. Processing of contaminated material consisted of compaction, size reduction, and sometimes decontamination for reuse.
No radioactive waste was stored in this facility. The facility was ready for use in January 1987.

The TMI-2 spent fuel pools "A" and "B" were used to stage spent resin vessels from the submerged demineralizer system and loaded defueling canisters. The spent fuel pools were

filled with water for shielding. The submerged demineralizer system's liners were prepared for shipment and lowered into the shipping cask underwater. The loaded shipping cask was lifted out of the "B" spent fuel pool, moved by crane over to the truck bay in the fuel handling building, and lowered onto the truck bed. Defueling canisters had to be loaded individually into the shipping cask located on a special stand in the truck bay. The weight of the shipping cask prohibited loading it inside the "A" spent fuel pool where the defueling canisters were stored. Each defueling canister was lifted out of the pool, raised into the fuel transfer cask, moved to the truck bay, and lowered into one of seven canister slots in the shipping cask. The loaded shipping cask was then lowered to a horizontal position and mounted onto a railroad car. Each fuel shipment took about 1,000 person-hours to prepare. [1]

NRC Approved Disposal of EPICOR II Resin Liners

The licensee requested that the requirement for the solidification of spent EPICOR II resins be waived, and that those spent resin liners that were similar to normal reactor resin wastes be disposed of by shallow land burial at a commercial disposal site. NRC approval to dispose of these 22 liners in this manner was issued on March 25, 1981. The last of these liners was shipped on June 27, 1981 from the TMI site to the U.S. Ecology burial site at Richland, Washington, in which all 22 liners were successfully disposed.

The requirement to solidify the remaining 50 EPICOR II pre-filter spent resin liners was also waived, and a DOE program of research and development on waste characterization examined and characterized the condition of one of these liners and its contents at a DOE contractor facility. Research in resin radiation degradation was reported in several NRC and DOE reports. [1]

First SDS Liner Shipped to DOE

On May 21, 1982, the first waste vessel from the Submerged Demineralizer System (SDS) was shipped from TMI to DOE facilities in Hanford, Washington for disposal. This vessel was used to process wastewater from the reactor-coolant bleed tanks, and contained approximately 4.44E+14 becquerels [12,000 curies] of radioactive material on zeolite ion-exchange media. Subsequent shipments included liners containing more than 1.85E+15 becquerels [50,000 curies] of radioactive material removed from reactor building sump water. DOE conducted research on glass vitrification (solidification) of this type of solid waste at Hanford.

On July 27, 1982, one of the 49 high specific activity EPICOR II liners stored on-site was sampled for gas composition at TMI, and was shipped on August 17 to the Battelle Columbus Laboratories in West Jefferson, Ohio for radiation and chemical characterization tests. The liner contained approximately 6.66E+13 becquerels [1,800 curies] of radioactive material, and was shipped in a special cask designed to withstand severe transportation accidents. On August 25, 1982, a second liner was shipped from TMI to the Idaho National Engineering Laboratory for characterization tests. [1]

Last SDS Liners Shipped to DOE

The last two of the 50 EPICOR II prefilters of high specific activity were shipped from TMI-2 on July 12, 1983, and the last of the 13 highly contaminated Submerged Demineralizer System (SDS) liners left the TMI site on August 30, 1983. [1]

Disposal of Slightly Contaminated Water

The licensee submitted for NRC approval a proposal for disposing of approximately 7,949,365 liters [2.1 million gallons] of slightly radioactive water, contaminated during the accident and used in subsequent cleanup operations. Out of the proposed alternatives, the licensee requested approval for a method involving the forced evaporation of the water at the TMI site over a 2.5-year period. The residue from this operation, containing small amounts of the radioactive isotopes cesium-137 and strontium-90, and large volumes of boric acid and sodium hydroxide, would require solidification and disposal as low-level waste.

In June 1987, the NRC issued Final Supplement No. 2 to NUREG-0683, Programmatic Environmental Impact Statement, (PEIS) which dealt with the final disposal of 7,949,365 liters [2.1 million gallons] of slightly contaminated accident-generated water. The staff concluded that the licensee's proposal to dispose of the water by forced evaporation to the atmosphere, followed by the on-site solidification of the remaining solids and their disposal at a low-level waste facility, was an acceptable plan. The staff also concluded that no alternative method of disposing of the contaminated water was clearly preferable to the licensee's proposal. An opportunity for a prior hearing to consider removing the prohibition on the disposal of the contaminated water was offered, and the matter was pending before the Atomic Safety and Licensing Board at the end of fiscal year 1987. The NRC evaluated the licensee's proposal together with eight alternative approaches, giving consideration to the risk of radiation exposure to workers and to the general public; the probability and consequences of potential accidents;

the necessary commitment of resources, including costs; and regulatory constraints. On September 11, 1989, the NRC issued Amendment No. 35 to the facility's operating license, modifying the plant's technical specifications by deleting the prohibition for disposing of accident-generated water. Disposal was allowed in accordance with NRC-approved procedures.

The evaporator system began vaporizing accident-generated water on January 24, 1991, after a prolonged period of system testing, modification, and repair. On August 12, 1993, the decontamination and evaporation of 8,441,468 liters [2.23 million gallons] of accident-generated water was completed. The system evaporated about 99 percent of the initial pre-processing volume of 8,706,447 liters [2.3 million gallons]. The residual volume that remained in various tanks and building sumps was estimated to be about 70,030 liters [18,500 gallons]. [1]

Post-Accident Evaluations

Prior to entry into the reactor building, necessary evaluations and repairs had to be completed. Much of the reactor monitoring instrumentation was damaged, or assumed to be inaccurate, after the accident. Remote cameras were used to assess damage, and could generally be inserted through existing building penetrations.

The TMI-2 polar crane suffered severe damage as a result of the accident. Besides being highly contaminated, the crane's electrical components were damaged by hydrogen burns and exposure to the excessive moisture in the containment building atmosphere. Restoration of the crane was required to accomplish defueling (removal of the reactor vessel head and internal structure, and other cleanup activities). The NRC staff approved the licensee's safety evaluation for the refurbishment and use of the Reactor Building Polar Crane. The initial climb to the polar crane was made on May 14, 1981. Mechanical and electrical inspections were made in August 1982. The crane was successfully load-tested February 29, 1984, when a test assembly weighing 194.1 tonnes [214 tons] was lifted and moved along predetermined test paths. Details are documented in NUREG/CR-3884, "Evaluation of Nuclear Facility Decommissioning Projects: Summary Report - Three Mile Island Unit 2 Polar Crane Recovery." [1]

Several post-accident evaluations were performed to consider such things as the integrity of the facility, the extent of core damage, the potential for radiological exposure, and the potential for environmental contamination. These are discussed below.

Facility Evaluations

In order to prepare for reentry and decontamination of the TMI-2 containment facilities the licensee developed a "Planning Study for Containment Entry and Decontamination," published July 2, 1979. This study included a structural assessment of containment, and considered physical effects from the following:

- Elevated containment pressure and temperature,
- The hydrogen detonation,
- Containment spray actuation,
- High radiation levels and cumulative doses,
- Reactor coolant system thermal-hydraulic transient,
- Flooding the containment, and
- Extended operation at negative containment pressure.

Based upon this study, the licensee concluded that, aside from the conditions that occurred during the hydrogen detonation, the physical effects of elevated pressures and temperatures were probably minimal. The study notes that damage from the hydrogen detonation, or a sustained hydrogen burn, would probably be localized with floor damage possible in the immediate area. Containment spray actuation was considered to potentially affect components and equipment within containment via corrosion, though the effects would not likely be present if those items were kept dry. High radiation was considered to affect surface coatings on the concrete walls and floors, and also structural steel surfaces and the containment liner walls. The reactor coolant system thermal-hydraulic transient was considered to have affected major reactor components within containment, such as the reactor vessel/head, steam generators, RCS hot and cold leg piping, the pressurizer (and associated surge and spray lines), reactor coolant pumps/motors, and core flood tanks (and associated piping). Flooding of containment was concluded to have little effect on the structural integrity of containment, as the amount of water in the building was calculated to create less pressure on the walls than the pressures used for structural integrity testing before reactor startup. However, the flooded basement was considered to likely affect submerged equipment and components within containment as well as carbon steel surfaces. Extended operation at negative pressure was concluded to have no effect on containment integrity, as this was an acceptable mode of operation during normal cooldown operation and the containment building was designed accordingly. [11]

Before TMI-2 was placed into post-defueling monitored storage (PDMS), the NRC identified structures, systems, and components that provided reasonable assurance that the facility could be maintained in a defueled condition without undue risk to the health and safety of the public. These are known as the PDMS environmental protection systems, and the containment structure is one them. Several steps were taken to place the containment structure into an acceptable condition for PDMS. To maintain the integrity of the environmental barrier, inactive penetrations were closed off with isolation valves or with welded or bolted blind flanges. In addition, the PDMS technical specifications required routine surveillance inspections of containment penetration isolation. Isolation valves on active containment penetrations used by the containment atmospheric breather and the reactor building's purge system would close on a high containment pressure. [1]

A Reactor Building Integrity Assessment Program was established by the licensee, at the direction of the NRC, to monitor potential leakage paths from the TMI-2 reactor building sump. The leakage monitoring points, which were based on engineering evaluations, included groundwater monitoring wells; storm drainage areas; cork seals (concrete joint seals) in structures surrounding the reactor building; and the tendon access gallery (a passageway surrounding the reactor building below the basement, approximately 6.10 meters [20 feet] below the water surface in the reactor building). The system initially included eight monitoring wells located around the TMI-2 reactor building and other nearby locations. Seven observation wells were installed early in 1980 to determine the source of tritium found in the wells. Monitoring wells had pumps; observation wells had grab-sampling capability. Additional wells were installed in later years. [1]

Reactor Core Inspections

Of particular interest to all organizations involved in the cleanup and research, was the extent of damage to the reactor core and internal reactor vessel components. Data acquisition and analysis that were conducted inside the reactor vessel included visual "quick look" inspections using video cameras; ultrasonic measures; radiation dose measurement; grab sampling of loose core debris; and core samples using a core bore drilling machine. The most revealing examination, with the most far-reaching implications, was the first "quick look" visual inspection in July 1982, which showed extensive reactor core damage (fuel melting would not be identified and announced until four years later). Up to the time just before the quick look,

the licensee had held out hope that the plant would one day be restarted. After quick look, the licensee's primary focus was cleanup. A chronology of the reactor core inspection activities is summarized below:

- Insertion Test of Axial Power Shaping Rods

  In June 1982, an axial power shaping rod (APSR) insertion test was performed in an attempt to move each APSR leadscrew inside the control rod drive mechanism to its fully inserted position. During normal power operations, eight APSRs helped shape the power generation (neutron flux) uniformly across the reactor core to ensure even fuel burnup during the core's lifetime. These APSRs do not perform safety functions and do not drop into the reactor core during an automatic reactor shutdown. During the accident, the eight APSRs rods remained at a 25 percent withdrawn position while 75 percent of their length remained inserted in the fuel assemblies. One purpose of the test was thought to provide insight into the extent of damage to the reactor core and upper plenum. The analysis of the data concluded that the test provided little definitive information about the physical condition of the reactor core. [1]

- "Quick Look" Camera Inspection

  The first closed-circuit video inspection of the upper reactor core region was performed on July 21, 1982. A camera 3.81 cm [1.5 inches] in diameter and 30.48 cm [12 inches] long was inserted through the empty leadscrew support tube (inside a control rod drive mechanism), and then into a central control rod guide tube (inside the upper plenum). As the camera was lowered into the upper core region, it revealed a bed of rubble approximately 1.52 meters [5 feet] below the normal location of the top of the fuel assembly. It was believed that the rubble bed contained oxidized cladding, fuel fragments and/or pellets, poison material, and core structural components. No evidence of melted fuel pellets was found. Another inspection, on August 4, midway between the periphery and the center of the reactor core, also revealed a rubble bed approximately 1.52 meters [5 feet] below the top of the core region. Intact pellets, which might have been fuel or poison material, were visible on the top of the rubble. During a third inspection, which took place on August 12, a probe was poked through the rubble and it penetrated approximately 0.30 meters [1 foot] below the surface, indicating that the rubble in this region was composed of loose material. The visual inspections did not

find any apparent distortion of the upper plenum. [1]

- Underhead Characterization Study

  In late summer of 1983, a series of activities conducted for the underhead characterization study provided radiological data to support the radiological protection measures that would be needed to conceptualize and plan procedures for removal of the reactor vessel head. Previously, in December 1982, a simple vertical "quick scan" of the underhead region (using an ionization chamber lowered into an empty control rod drive mechanism opening) found higher-than-expected radiation levels, which ranged from 0.35 to 5.26 gray [40 to 600 roentgen] per hour with an unknown beta-radiation contribution. The results from the quick scan prompted a more detailed study.

  The initial underhead characterization study obtained dose rates around the reactor vessel head and service structure; re-measured dose rates under the head using a beta-shielded ionization chamber ("quick scan-2"), and a string of thermoluminescent dosimeters; performed remote visual inspections of the top of the upper plenum and underside of the reactor vessel head; obtained debris samples from the top of the plenum for pyrophoricity tests; examined the lower end of a leadscrew to characterize the stainless steel in the underhead environment; and evaluated the dose rate increase associated with moving several radioactive leadscrews from inside the reactor vessel to a position in the control rod drive mechanism's motor tubes in the reactor vessel head's service structure. The study required the depressurization of the reactor coolant system to atmospheric pressure, a slight draindown of the reactor vessel's water level to uncover the top 0.30 meters [1 foot] of the upper plenum (3.05 meters [10 feet] above the reactor core region), and the removal of a control rod drive mechanism. The NRC established a contract with DOE's Pacific Northwest Laboratory to review the radiation measurements, fission products plate-out (absorption) on the upper plenum, and other chemical phenomena.

  The results of the data analysis revealed the following: no visual structural damage on the upper plenum; areas inspected on top of the plenum appeared relatively free of debris (though some sedimentation appeared to be present on some horizontal surfaces); pyrophoricity tests on two samples of material from the plenum surface proved negative; gamma radiation fields in the range of 2.63 to 6.14 gray [300 to 700 roentgen] per hour

were measured in the space formed by the underside of the reactor vessel head and the top of the plenum; and no significant dose-rate increase was observed on the service structure platform following the withdrawal of four leadscrews. The analysis of the data from the underhead characterization study supported plans to remove the reactor vessel head without flooding the refueling canal (also called "dry lift"). [1]

- Reactor Core Debris Sampling Program

  The safety evaluation report for the underhead characterization study was amended to include two new activities that were sponsored by DOE's TMI Reactor Evaluation Program: a reactor core debris sampling program and reactor core topography program. The reactor core debris sampling program provided data that was essential to prepare for future reactor vessel defueling activities and for the design of water cleanup systems and defueling canisters. In September and October 1983, the program obtained six specimens of reactor core debris by lowering specially designed tools into the reactor. The tool scooped up small samples of loose debris and transferred them into small shielded casks for offsite shipment and analysis. The analysis of the samples included determining their particulate composition, particle size, fission-product content, and drying properties, as well as the fission-product leachability from the debris and the pyrophoricity of zirconium hydride or partially unoxidized zirconium fines.

  The first three samples were taken at various depths in the center of the reactor core. The radiation field at 0.30 meters [1 foot] from one sampler (which had a capacity of 16.39 cubic centimeters [1 cubic inch]) was approximately 0.026 gray [3 roentgen] per hour. The other three samples were taken midway between the center and the periphery of the reactor core at various depths in the debris bed. The six samples obtained during the grab-sample work were analyzed at the Idaho National Engineering Laboratory (INEL) and the Babcock & Wilcox (one sample) research facilities. Gamma-radiation levels of five samples, using a telescoping radiation instrument from 2.5 cm away, ranged from 0.026 to 0.316 gray [3 to 36 roentgen] per hour. Particle sizes ranged from about 0.6 cm to a fine debris. Five more samples were obtained in March 1984. Results from the sampling program were documented in the INEL report "TMI-2 Core Debris Grab Samples - Examination and Analysis (GEND-INF-075)." [1]

- Reactor Core Topography Program

  In August and September of 1983, the core topography system, designed and built by DOE at INEL, was used to conduct an ultrasonic profile of the void area in the upper reactor core region. A total of about 500,000 data points were obtained during system operation. The data and information obtained included the radial and axial extent of the reactor core cavity, the location of supported and unsupported fuel assembly end-fittings, and the location of the core cavity boundary with respect to structurally intact fuel assemblies. The analysis of this data supported upper plenum lift and defueling operations.

  A clear plastic scale model of the damaged upper reactor core region was constructed at INEL in late 1983 based on ultrasonic measurements. This topographic model provided the most accurate indication of the extent of reactor core damage at that time. The volume of the cavity in the damaged area of the reactor core was measured at 9.34 cubic meters [330 cubic feet] or 26 percent of the original core volume. The bottom of the cavity ranged from 1.52 to 1.83 meters [5 to 6 feet] below the top of the core. Of the original 177 fuel assemblies, 42 appeared to contain some full-length fuel rods, but 23 of those 42 had less than 50 percent of the rods intact. The sonic mapping also revealed several partial fuel assemblies hanging from the underside of the upper plenum and indicated some distortion of the core former wall. Each half of the plastic model was transferred to the NRC by DOE and the Smithsonian Institution. The complete model now resides at the NRC at Rockville, Maryland. [1]

- Reactor Core Video Mapping

  In April 1984, a comprehensive video mapping of the upper reactor core region between the plenum and rubble bed was completed. Video snapshots were assembled into a mosaic panoramic view of the rubble bed, core periphery, and the underside of the upper plenum grid section. Videos of the rubble bed showed broken fuel rods scattered around, fuel rod internal springs, intact fuel pellets, control rod assembly end couplings, and partially intact fuel assemblies around the periphery of the reactor core region. The video also showed unsupported partial fuel assemblies hanging from the underside of the upper plenum grid section, which had to be removed before plenum lifting. This map supported defueling operations planning, and eventually the removal of fuel debris. [1]

- Video Inspection of the Lower Head of the Reactor Vessel

  In February 1985, the first video inspection of the reactor vessel's lower head region was performed by guiding a small video camera and light through a gap between the upper plenum and core support flange during plenum jacking. The video inspection revealed re-solidified mass in the reactor vessel lower head on February 20, 1985. The camera followed a path down to the lower head region between the reactor vessel wall and the internal thermal shield. The video revealed the accumulation of a substantial quantity (estimated at 9.07 to 18.14 tonnes [10 to 20 tons]) of accident-generated debris with the appearance of a gravel pile. Over the next two years, four additional camera inspections of the lower head region were performed at different quadrants in the lower core support assembly (July 1985, December 1985, July 1986, and February 1987). The analysis of the video data, as documented in the INEL report "TMI-2 Lower Plenum Video Data Summary" (EGG-TMI-7429), revealed that large inhomogeneity existed in the physical appearance of the debris bed, ranging from a very fine dust-like and smooth surface, to a relatively flat, but coarse surface with large chunks, to a solid wall of lava-like debris. [1]

- Vertical Gamma Profiles of the Reactor Vessel's Lower Head Region

  In March 1985, an attempt was made to insert a miniature gamma-radiation sensor into several incore instrument tubes to obtain vertical gamma profiles for characterizing fuel deposits that had settled on the bottom of the reactor vessel. The incore instrument tubes penetrate the bottom of the reactor vessel, and extend upward into the reactor core region. A dummy detector wire of the same size and stiffness as the actual probes was inserted into 26 of the 52 incore instrument tubes; only one of the 26 wire probes reached the reactor vessel. The remaining 25 wires were blocked at various locations along the 36.58 meters [120 feet] of pipe between the bottom of the vessel and the pipe access at the incore seal table located in the reactor building. The single wire probe was inserted 55.88 cm [22 inches] into the lower reactor core region. A gamma sensor with a slightly larger diameter was inserted into the same instrument tube and reached the vicinity of the 13.65 cm [5-and-3/8-inch] thick lower head of the reactor vessel. [1]

- Core Stratification Sample Acquisition Program

  In July 1986, core boring operations were performed using a special computer-

controlled drilling machine. The core stratification sample acquisition program was conducted as part of DOE's TMI-2 Accident Evaluation Program to provide data on the material properties of the core debris. The core bore samples provided insights into fission-product release from the fuel, fission-product retention in the core, maximum temperature during the accident, and reactor core material interactions. A special commercially available drilling rig was assembled on top of the defueling work platform to bore into the hard crust. Ten full-length core bore samples were obtained from all regions of the lower reactor core; these samples (approximately 6.35 cm [2.5 inches] in diameter and 2.44 meters [8 feet] long) were analyzed at INEL, along with earlier samples of debris collected from the reactor vessel's lower head. Video inspections of the reactor core below the debris bed were performed through several of the bore holes created by the drilling operations. Initial inspections indicated that peripheral fuel assemblies were essentially intact below the hard crust layer, but that the central reactor core region consisted largely of a fused mass of material. The INEL report "TMI-2 Core Bore Examinations" (GEND-INF-092) documented results of the physical, metallurgical, and radiochemical examinations of the core bore samples and evaluated these results as they relate to the progression of core damage and fission-product behavior in the lower region of the reactor core. [1]

- <u>TMI-2 Vessel Investigation Project</u>

  In July 1989, a video inspection of the reactor vessel's lower head revealed several cracks that appeared to be associated with incore instrument penetration nozzles. Higher-quality color videos and a mechanical probe were used in August to obtain better information on the cracks. The cracks appeared to be up to approximately 15.24 cm [6 inches] in length, 0.64 cm [0.25 inches] wide, and about 0.48 cm [0.19 inches] deep. Penetration into the thick reactor vessel steel was later determined to be superficial.

  The TMI Vessel Investigation Project (VIP) was an international program sponsored jointly by the NRC and the Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA). Participants in this program included the U.S., Japan, Belgium, Germany, Finland, France, Italy, Spain, Sweden, Switzerland, and the United Kingdom. As described in the formal project agreement, the objectives of the VIP were to jointly carry out a study to evaluate the potential failure modes and the TMI-2 reactor vessel's margin for failure during the TMI-2 accident. In

February 1990, an international research effort obtained metallurgical samples from the reactor vessel's lower head after defueling was completed. The program evaluated the potential modes of failure and the reactor vessel's margin of failure during the TMI-2 accident. The condition and properties of material extracted from the reactor vessel's lower head were investigated to determine the extent of damage to the lower head by chemical and thermal attack, the thermal input to the reactor vessel, and the margin of structural integrity that remained during the accident. A total of 15 "boat" samples were obtained from the reactor vessel's lower head. In addition, 14 incore instrument penetration nozzles were cut off 2.54 to 5.08 cm [1 to 2 inches] above the reactor vessel's lower head, and obtained as samples. Two incore instrument guide tubes were cut free from the flow distributor head as samples. Results from the reactor vessel investigation project were documented in a series of NRC reports by the INEL and Argonne National Laboratory. [1]

Radiological Exposure Considerations and Ongoing Monitoring

The NRC conducted detailed studies of the accident's radiological consequences, as did the U.S. Environmental Protection Agency, the Department of Health, Education and Welfare (now Health and Human Services), the Department of Energy, and the Commonwealth of Pennsylvania. Several independent groups also conducted studies. The approximately 2 million people around TMI-2 during the accident are estimated to have received an average radiation dose of only about 0.01 millisievert [1 millirem] above the usual background dose. To put this into context, exposure from a chest X-ray is about 0.06 millisievert [6 millirem] and the area's natural radioactive background dose is about 1-1.25 millisievert [100-125 millirem] per year. The accident's maximum dose to a person at the site boundary would have been less than 1 millisievert [100 millirem] above background. [4]

In the months following the accident, although questions were raised about possible adverse effects from radiation on human, animal, and plant life in the TMI area, none could be directly correlated to the accident. Thousands of environmental samples of air, water, milk, vegetation, soil, and foodstuffs were collected by various government agencies monitoring the area. Very low levels of radionuclides could be attributed to releases from the accident. However, comprehensive investigations and assessments by several well respected organizations, such as Columbia University and the University of Pittsburgh, have concluded that in spite of serious damage to the reactor, the actual release had negligible effects on the physical health of

individuals or the environment. [4]

Additionally, ongoing radiological monitoring is required during the post-defueling monitored storage (PDMS) period. The NRC's safety evaluation of the PDMS Technical Specifications concluded that radiological surveillance of activities during PDMS could be conducted in accordance with the approved offsite-dose calculation manual and in compliance with the regulatory requirements of 10 CFR Part 20, "Standards for Protection against Radiation," which would, with the approved radiation protection plan, ensure adequate control of occupational exposure and protection of workers. The NRC evaluation also noted that the licensee's proposed surveillance program would adequately monitor PDMS environmental protection systems. [1]

4.2 Achieving a Standard Condition for Decommissioning

In August 1989, the NRC issued NUREG-0683 Supplement 3, Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2, Final Supplement Dealing with Post Defueling Monitored Storage and Subsequent Cleanup. This supplement evaluated the licensee's proposal to complete the current cleanup effort and place the facility into monitored storage for an unspecified period of time. The licensee had indicated that the facility would likely be decommissioned following the storage period, at the time that Unit 1 was decommissioned. Specifically, the supplement provided an environmental evaluation of the licensee's proposal and a number of alternative courses of action from the end of the current defueling effort to the beginning of decommissioning. However, it did not provide an evaluation of the environmental impacts associated with decommissioning. NRC staff had concluded that the licensee's proposal to place the facility in monitored storage would not significantly affect the quality of the human environment. Furthermore, any impacts associated with this action were outweighed by its benefits. The benefit of this action was the ultimate elimination of the small but continuing risk associated with the conditions of the facility, resulting from the March 28, 1979, accident. The draft supplement was issued for public comment in April 1988. [1]

Today, the TMI-2 reactor is permanently shut down. The reactor coolant system is fully drained and the radioactive water decontaminated and evaporated. The accident's radioactive waste was shipped off-site to an appropriate disposal area, and all spent fuel has been removed except for some debris in the reactor coolant system. The plant defueling was completed in April 1990, and the reactor fuel and core debris was shipped to the Department of Energy's Idaho National

Laboratory. The Department of Energy has taken title and possession of the fuel. [4]

5. CURRENT STATUS OF TMI-2 DECOMMISSIONING

TMI-2 has been defueled and decontaminated to the extent the plant is in a safe, inherently stable condition suitable for long-term management. This long-term management condition, termed post-defueling monitored storage, was approved in 1993. On June 28, 2013, the licensee submitted a Post-Shutdown Decommissioning Activities Report (PSDAR) to the NRC, which is a report required by all reactor licensees prior to decommissioning. This report recognized September 14, 1993 as the established date for permanent cessation of TMI-2 operations. There is no significant dismantlement or decommissioning currently underway at TMI-2. The plant shares equipment with the operating TMI - Unit 1. TMI-1 was sold to AmerGen (now Exelon) in 1999. GPU Nuclear retains the license for TMI-2 and is owned by FirstEnergy Corp. GPU contracts with Exelon for maintenance and surveillance activities. The licensee plans to actively decommission TMI-2 after expiration of the TMI-1 license. [12]

6. FURTHER DECOMMISSIONING PLANS

Since final decommissioning of the TMI-2 facility is deferred until after expiration of the TMI-1 license, both units will be subject to decommissioning/license termination regulations in effect at that point in time. Currently, license termination regulations are set forth in 10 CFR 20, Subpart E (Radiological Criteria for License Termination), as well as within regulations specific to the type of licensee. In the case of TMI-2, licensing criteria from 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities) apply, and more specifically, regulations from 10 CFR 50.82 (Termination of License) apply for license termination/decommissioning.

7. CONCLUSION

The Three Mile Island accident resulted in numerous investigations and lessons learned, many of which focused on preventing another accident. However, many lessons learned on remediation strategies and on the regulatory framework used to implement those strategies have also been captured – which are of particular relevance to the decommissioning and remediation of damaged nuclear facilities. Many of those lessons learned have been archived in the NRC's document collection titled "Three Mile Island Accident of 1979 Knowledge Management

Digest" (NUREG/KM-0001), which includes supplemental materials on DVD. This case study relies heavily on documents within that archive, and has reiterated excerpts from several of those documents.

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