TMI Unit 2 Technical Information & Examination Program

UPDATE

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New Electrical Diagnostic System Supports Maintenance Activities

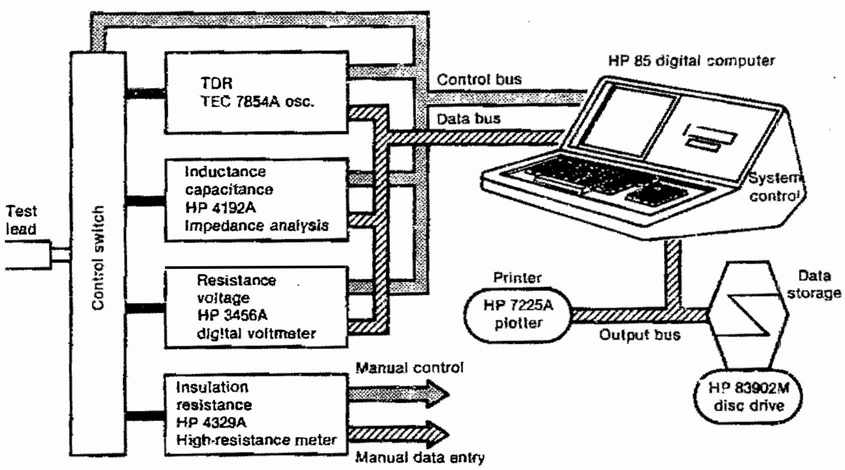

Figure 1. **Electrical circuit characterization and diagnostic system.**

A primary goal of the Three Mile Island (TMI) research and development program is to assess how the TMI-2 accident affected the general condition of all systems in the Reactor Building. During the accident, some of the instrument and control signals began deteriorating to the point of ambiguity, severely handicapping control room personnel. Because the hostile Reactor Building environment prevented direct access, it was necessary to assess system performance by the electrical characteristics gathered from remote locations. To perform the task, the Electrical Circuit Characterization and Diagnostic (ECCAD) system was designed as a means to acquire basic electrical data on electrical channels and to store and format the data for easy handling and analysis (see Figure 1).

Data Easily Accessed

The ECCAD system uses commercially available electronic test equipment with computerized control to provide a means to obtain a high-quality standard data set. The data set encompasses the electrical measurements typically performed in a plant maintenance program. The most significant feature of this system is that the data, which is stored and formatted, can be easily accessed for the quality assessment and diagnosis of electrical circuit and equipment conditions. Although the system is still under development, it could be used in its present configuration after some operator training in data analysis.

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Teams of personnel started working with basic data obtained from points outside the Reactor Building to establish the validity of signals coming from the Reactor Building and assigned a degree of confidence to these signals. Since then, electrical characteristics have been monitored, recorded, analyzed, and compared with laboratory test results to gain quantitative data from which analysts could assess the ability of instruments to function.

Forecasting More Reliable

Electrical characteristics monitored include indicated output signal, insulation resistance, circuit resistance, capacitance, time domain reflectometry, bias voltage, starting current, equipment actuation time response, operating current, and spectral content of output signals (feedback blanking or "noise"). Data acquisition consisted of passive and active surveillance of the above characteristics for the cable, junction points, and the end device. For the purpose of analysis, each circuit or channel was treated as a transmission line, with the end device being the load. Because the circuit's ability to function generally was not defined in terms of its electrical characteristics, analysts had to start with assumptions and arrive at qualitative results rather than make absolute, quantitative assessments. The results appear to be well suited to forecasting channel reliability.

Use of the ECCAD System at TMI-2 saved considerable test time and also resulted in high quality, repeatable data with minimum operator error. Direct storage of data on magnetic disks in the computer also eliminated paper work.

System Enhances Surveillance

The ECCAD System is still in its infancy, but at least one nuclear service vendor is planning to provide an identical system to support plant maintenance. In support of maintenance, ECCAD will enhance surveillance procedures to allow for a determination of the quality of operation and the likelihood of continued reliable operation of equipment being monitored. The System can quickly verify the accuracy of abnormal instrument indications, provide a verifiable and reproducible basis for operator action, and detect degraded circuits that, with maintenance, would be returned to proper operating condition. □

A Calculational Approach to Determining Combustible Gas Concentrations in Sealed Radioactive Waste

The Technical Integration & Examination Program (TI&EP) has developed a calculational method for determining the rates at which gas is generated in radioactive waste containers. The work is significant to facilities generating radioactive waste, because the method will decrease costs and reduce personnel radiation exposures during various venting and storage operations.

Gas Production A Safety Concern

The production of combustible gases in sealed radioactive waste containers has been identified as a significant safety concern relative to handling, shipping, and storage of radioactive waste. A Nuclear Regulatory Commission (NRC) evaluation of the hydrogen gas generation problem resulted in issuing new requirements for certain certificates of compliance related to radioactive waste shipment packages.

These new requirements address hydrogen gas generation and applicable safe storage and shipment periods. The requirements state that for waste containers that have the potential to radiolytically generate combustible gases, a determination must be made by tests and measurements of canisters for hydrogen and oxygen content.

Basically, hydrogen gas concentrations must be limited to no more than 5% by volume, or the container must be inerted to ensure that oxygen is limited to 5% by volume. Compliance with this requirement is unnecessary if the containers are shipped within 10 days of sealing or venting.

New Requirements Too Conservative

These new requirements affect most radioactive waste shipments from operating nuclear power plants. The TI&EP considered the new NRC requirements conservative and costly relative to financial expenditures and increased personnel radiation exposures and sought to improve predictive techniques.

In addition to the NRC requirements, utilities must consider that the determination of safe storage periods for radioactive waste containers is more significant with the enactment of the "Low-Level Radioactive Waste Policy Act" of 1980. The Act provides for the formation of interstate regional disposal facilities to relieve the present burden on the three states with low-level waste (LLW) disposal sites. After January 1, 1986, states with regional waste compacts will not accept LLW from nonmember states, thus requiring on-site storage for the affected utilities.

Task Force Organized

The Utility Nuclear Waste Management Group of the Edison Electric Institute formed a "Hydrogen Generation Task Force" to study and evaluate the new NRC requirements. The task force acquired direct technical and operational experience assistance from the TI&EP. This resulted in the development of a calculational method to quantify hydrogen gas generation in sealed containers.

The calculation model was developed by applying the results of the NRC and Department of Energy (DOE) funded research projects to the gas generation problem. A modified computer shielding code was used to reduce uncertainties associated with previous predictive models. Actual TMI EPICOR II measurements were compared to predicted values with excellent agreement.

Calculational Method Verified

Based on the work by the TI&EP and the Electric Power Research Institute (EPRI), the NRC acknowledged the validity of the calculational method. The Commission has modified certificates of compliance to allow calculation of hydrogen concentration as well as tests and measurements as an acceptable method of compliance to regulation.

Calculating combustible gas concentrations is now an acceptable means of determining quantities of gas in sealed radioactive containers. Waste generators will realize cost savings and reduce manrem exposures by eliminating special handling requirements for the majority of their radioactive wastes. Waste management safety will be enhanced by the ability to quickly identify those containers that present a potential hazard. □

Drop Tests Verify Design of Shipping Cask for Safety

Figure 2. **Height and orientation check before end drop.**

In spring 1985, engineers at Sandia National Laboratory, in Albuquerque, NM, dropped a quarter-scale model of the NuPac 125B transport cask onto a 526,000-lb mass of concrete faced with about 4 in. of battle ship armor plate. This model of a double-containment rail cask, designed by Nuclear Packaging, Inc. (NuPac), underwent a series of drop tests as a demonstration of the cask's structural integrity and capability to survive hypothetical accidents without rupture, leakage, gross deformation, or compromise to its payload.

EG&G Idaho, Inc., the U.S. Department of Energy's contracting manager of the TMI-2 cleanup project, selected the 125B rail cask to transport the damaged fuel to the Idaho National Engineering Laboratory (INEL). The results of the drop tests are a major chapter in the Safety Analysis Report that the Nuclear Regulatory Commission is now reviewing for cask licensing. The drop tests in fact confirmed the positive results of earlier computer analyses: that the cask can safely contain the TMI-2 core debris under the extreme conditions of hypothetical accidents.

Materials Effectively Protect Payload

Constructed principally of stainless steel and lead, the 125B rail cask has four basic components: foam-filled overpacks to absorb energy and protect the ends of the outer cask; the outer cask containment vessel, with lead shielding; the inner containment vessel, with borated concrete for neutron moderation and criticality control; and aluminum honeycomb energy absorbers at the ends of each canister tube in the inner vessel, to limit the axial "g" loads that could develop on the core debris canisters.

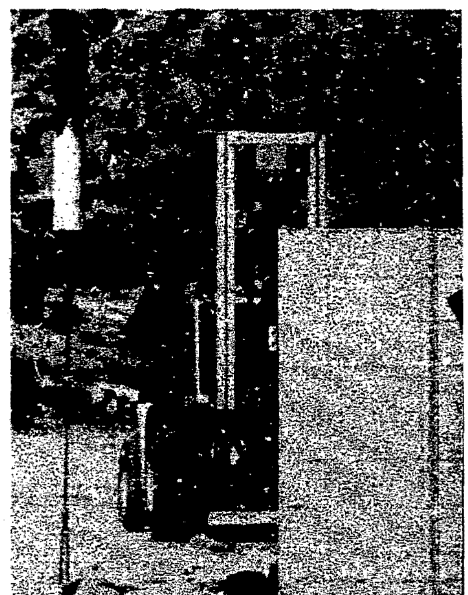
Additionally supporting the canister tubes are steel plates that make up a hub, spoke, and wheel arrangement in the inner vessel. Both vessel lids have rupture discs that contain pressure buildup in normal and hypothetical accidents. The discs are designed to rupture if the cask experiences a fire of longer duration and with temperatures significantly higher than considered even for hypothetical accidents.

While the full-scale NuPac rail cask will weigh approximately 183,000 lb, its quarter-scale model, a full representation of the actual cask, was 1/64th that weight, or 2,830 lb.

Regulation Establishes Accident Conditions

Federal regulation 10 CFR 71, subpart F, requires the evaluation of this package dropped onto a flat, essentially unyielding surface, given certain hypothetical accident conditions. The regulation specifies that the package strike the surface in a position for which maximum damage is expected.

Figure 3. **Instant prior to impact from oblique drop.**



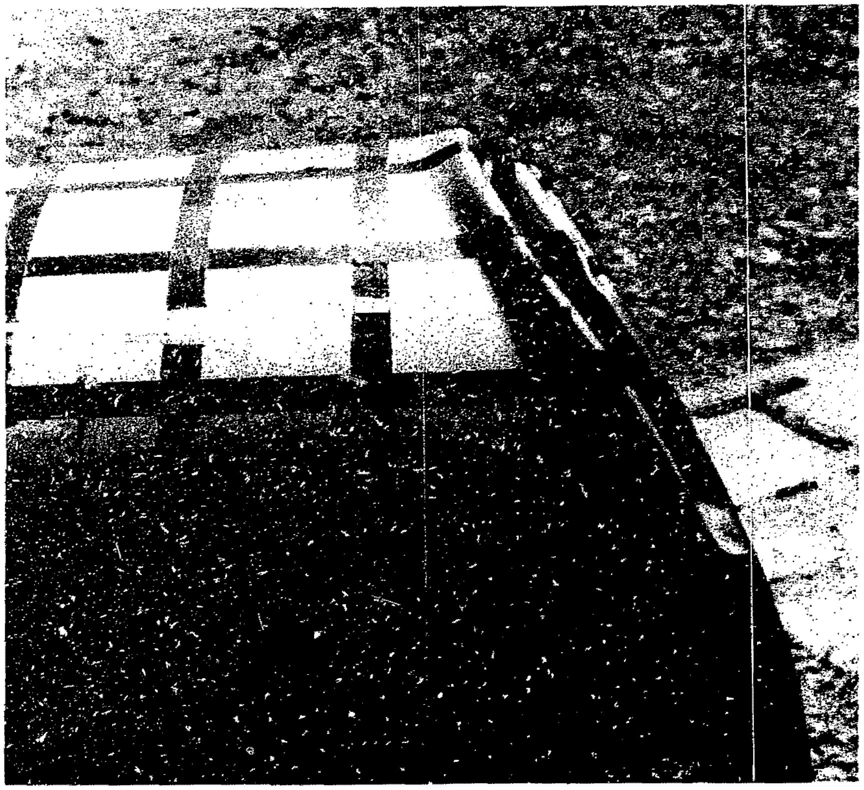
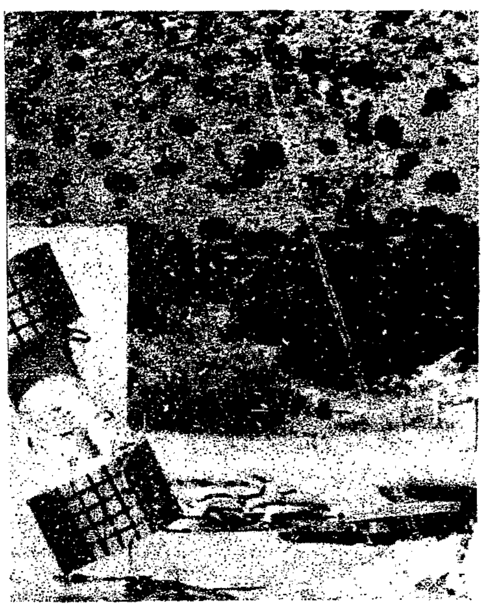


Figure 4. **Overpack damage from oblique drop.**



The model was dropped three times from 30 ft: flat on its bottom end, at a 62-1/2-degree (oblique) angle onto its lid, and on its side. It was then dropped twice from 40 in. onto a 1-1/2-in.-diameter pin. In these two puncture tests, the cask came down on its side and then on its lid onto the pin, demonstrating the integrity of the side wall and closure of the cask. Figures 2 through 4 show preparations, an actual drop, and a closeup of damage.

Nearly all permanent damage to the package was limited to the external overpacks and internal energy absorbers, as expected and desired. The only significant exception was damage to the outer cask outer shell and lead shielding on the side of the cask where it came in contact with the pin. The inner vessel experienced no damage.

Before the series of tests began, the model was instrumented with accelerometers, strain gauges, and thermocouples to obtain a detailed record of responses. Preceding each drop, workers swept, wet down, and reswept the target surface to eliminate dust that could cloud up and obscure the model upon impact. Portable, gridded stadia boards erected behind and to the sides of the landing surface provided a contrast for filming the experiment and for measuring velocity, elasticity, and deformation.

For the first two drops, the cask model was refrigerated to test its response at low temperatures, specifically to confirm that the unit is not subject to brittle fracture—a "worst-case" condition. By the time the model reached the drop site and was prepared for the test, its temperature was the desired -25°F. The test confirmed that the kind of stainless steel being used to construct the cask does not lose its ductility at temperatures this low, eliminating concerns about brittle fracture. The model was at ambient temperature for the side drop and puncture tests.

Damage Expected by Design

After the end and oblique drops, inspectors returned the model to the laboratory to examine the overpacks, leak-test containment seals for both vessels, and torque-check the lid bolts. After the last drop test, the cask model again was leak-tested and then X-rayed and measured to be sure there was no hidden damage. Also, the overpacks were sectioned and examined to see how well they performed.

Removing the cask lid, inspectors found only minor deformation to the seven top energy absorbers—damage that was intended by design. Just as expected, the lower seven energy absorbers clearly protected the payload; the seven quarter-scale canisters were undamaged. Leak tests performed before and after the drops confirmed that the seals maintained their integrity.

X-rays showed that even after repeated impacts, no quantifiable amount of lead slumped; only the side puncture drop reduced lead shielding, but to an extent consistent with federal regulations.

The puncture tests verified the equations used to determine the thicknesses of cask materials. In the side puncture test, most of the deformation occurred at the point of contact; the outer shell was indented by less than its thickness and maintained its integrity against puncture. Slight residual elastic stresses were induced in the package shells due to a modest inelastic deformation of the lead shield.

Consequently, the results of the actual drop tests verified the positive findings of earlier computer analyses conducted to determine cask safety. Most important, the stresses on the cask model were well below the yield stresses for cask materials. Also, the damage assumptions for input to the computer thermal analyses were found to be quite conservative compared to the actual damage from the drops. □

NuPac Rail Cask Featured in Videotape

"A Shipping Cask Developed for Safety" is the title of an 18-minute videotape produced by DOE contractor EG&G Idaho, Inc., now available for loan without charge. The program reviews the criteria behind the selection of the double containment rail cask designed by Nuclear Packaging, Inc., explains the cask design, and features various drop tests of a quarter-scale model, conducted to demonstrate cask safety.

To obtain a copy of the program, contact Kim Haddock, Administrator, EG&G Idaho, Inc., TMI Site Office, P.O. Box 88, Middletown, PA 17057, telephone FTS 590-1019 or (717) 948-1019. □

Program Highlights

CESIUM ELUTION COMPLETE

In mid-March 1985, GPU Nuclear, with technical support by DOE contractor EG&G Idaho, Inc., completed cesium elution of the two makeup and purification demineralizers in Unit 2. Cesium radioactivity was reduced by approximately 70% in demineralizer vessel A and by approximately 90% in vessel B.

The Submerged Demineralizer System liner that was used to decontaminate the eluant containing the cesium from the vessel was then shipped in April from TMI to Rockwell Hanford Operations in Washington State. It was buried there the following month, the last liner DOE accepted for its research and development monitored retrievable burial demonstration program.

DEFUELING BEGINS THIS FALL

GPU Nuclear will begin defueling the damaged reactor vessel at TMI this Fall. Debris will be loaded into canisters that will then go through several stages of transfer and storage before being shipped to the Idaho National Engineering Laboratory. For details of the defueling operation, see Update Volume 5, Number 2, dated February 1985.

PLENUM SUCCESSFULLY LIFTED

The TMI-2 reactor plenum was lifted successfully in May and now rests on its storage stand in the deep end of the refueling canal. The plenum was made accessable last summer by the removal of the reactor vessel head and is the last remaining component to be removed in preparation for the start of defueling this fall.

CORE TEMPERATURES ALLOWED FUEL MELTING

Metallurgical studies conducted by EG&G Idaho, Inc., indicate that portions of the core reached 5100°F during the TMI-2 accident, which is the melting point of uranium dioxide fuel. Although previous results showed the temperatures may have exceeded 4700°F during the accident, the latest study provides the first firm evidence that some fuel melting occurred. Studies are continuing to determine how much of the core reached temperatures that would allow melting of fuel.

FINAL LOFT TEST SIMULATES TMI ACCIDENT

The final test at the Loss-of-Fluid Test Facility (LOFT) at the Idaho National Engineering Laboratory simulated the partial meltdown conditions experienced during the TMI-2 accident. The partial LOFT meltdown took 4.5 minutes and produced temperatures of more than 4,400°F at the center of the core. Eight of the fuel rods at the center of the 100-rod assembly were deliberately deprived of cooling water for the simulation. During the 4.5-minute period, temperatures of the eight rods rose from 120°F to more than 4,400°F. ☐

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ADDRESS CORRECTION REQUESTED

UPDATE

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