GEND 050
Distribution Category: UC-78
TMI Supplement

THREE MILE ISLAND TECHNICAL INFORMATION AND EXAMINATION PROGRAM INSTRUMENTATION AND ELECTRICAL SUMMARY REPORT

GEND--050

DE85 015387

Richard D. Meininger

Published July 1985

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Idaho Operations Office
Under DOE Contract No. DE-AC07-76IDO1570

CONTENTS

THREE MILE ISLAND TECHNICAL INFORMATION AND EXAMINATION PROGRAM INSTRUMENTATION AND ELECTRICAL SUMMARY REPORT

INTRODUCTION

The accident at Three Mile Island Unit-2 (TMI-2) provided the opportunity to evaluate a full complement of instrumentation that had been exposed to the severe combined environmental effects of steam, Reactor Building spray, a hydrogen burn causing overpressurization to 290 kPa (28 psig), and release of fission products and traces of fuel to the Reactor Building. A cutaway model of the TMI-2 reactor is shown in Figure 1. The instrumentation has been further aged since the accident by a high radiation background and continuing decontamination efforts. The evaluation of the instrumentation and electrical components of TMI-2 has called upon expertise from the Westinghouse Hanford Development Laboratory, the Idaho National Engineering Laboratory (INEL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratory (SNL), and various private companies and consultants under the overall direction of EG&G Idaho, Inc. for the Department of Energy (DOE).

The results obtained from the investigations at TMI-2 support the conclusion that the basic design of nuclear plants is sound and the instrumentation and equipment is inherently rugged. The investigators did not find any problems that would jeopardize safe, normal operation of the plant due to design. The Nuclear Regulatory Commission, immediately after the accident, conducted studies and issued reports covering plant design changes necessary to correct deficiencies that either contributed to the cause of the accident or interfered with speedy recovery from the accident. The DOE Instrumentation and Electrical (I&E) program on the other hand, was established to evaluate the ability of the instrumentation and electrical components to accomplish their design function and to survive the accident environment. This program was intended to provide the maximum yield of data to (a) improve qualification standards, (b) assess the adequacy of existing standards, (c) improve future designs, (d) assess vulnerability of other plants that use similar equipment, and (e) better understand the accident itself.[1]

The major findings of the DOE I&E program are as follows:

- Most equipment failures occurred during the first 24 hours of the accident and were predominantly a result of moisture intrusion. Class 1E and safety-related equipment were generally more resistant to moisture than nonqualified equipment. However, there appeared to be a general lack of appreciation, from an installation standpoint, for the severity of the problem. It is the consensus of the investigators that most of the moisture intrusion problems would have occurred eventually in the plant without the accident. Moisture intrusion generally occurred at the electrical penetration to a device.

- No functional damage to the nuclear plant instrumentation and electrical components was apparent that could be identified as resulting from thermal effects of the hydrogen burn. One Geiger-Mueller tube was determined to have failed at the time of the hydrogen burn, but its actual failure was shock-related, possibly caused by the pressure wave associated with the hydrogen burn.

- Early failures of some equipment that were not qualified as Class 1E or safety-related were caused by improper installation or maintenance activities which in turn allowed moisture or spray penetration.

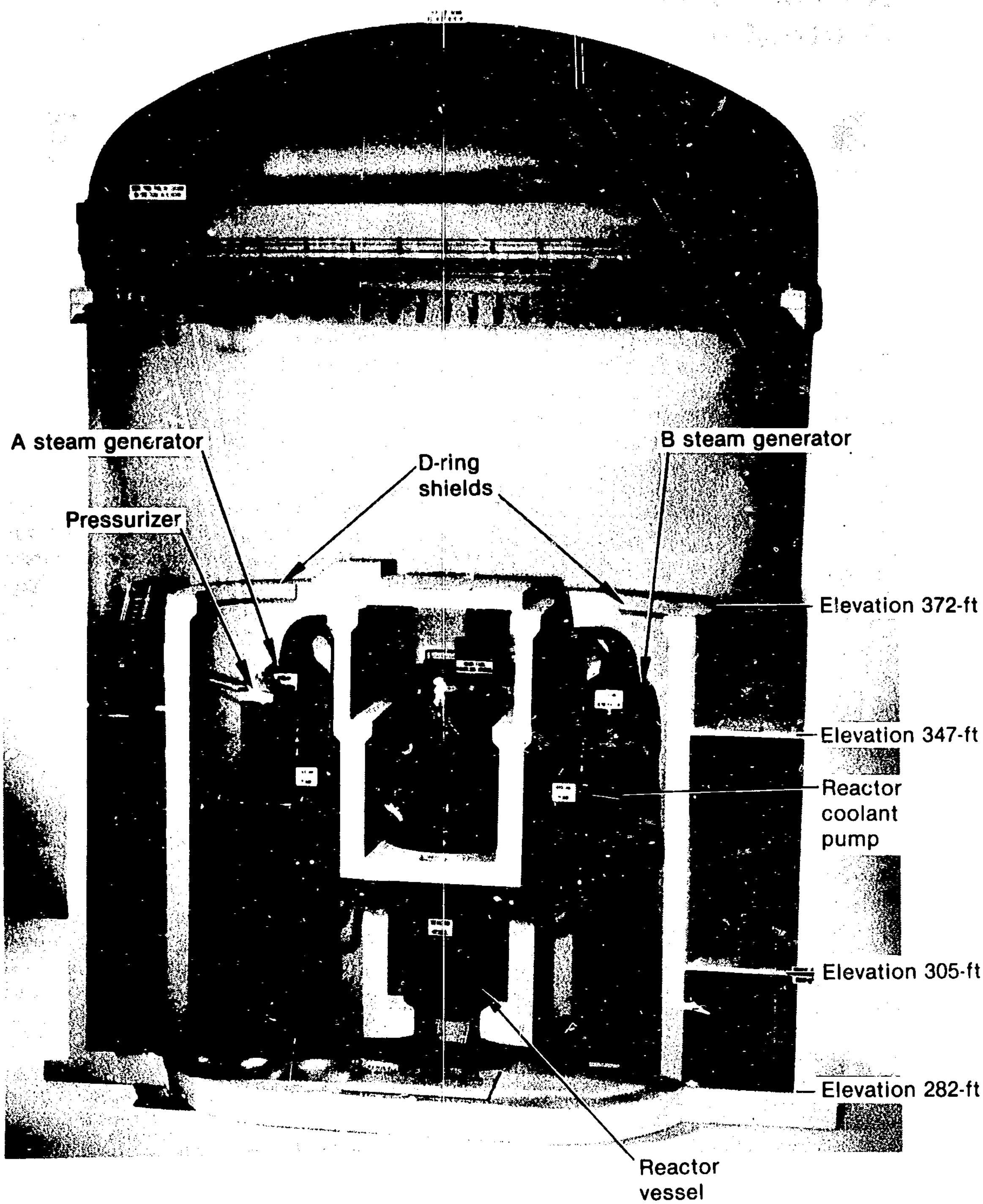

Figure 1. TMI-2 Reactor Building showing reactor and primary coolant system.

- The use of radiation-sensitive transistors in radiation monitors and loose parts monitors caused changes in sensitivity as early as the first 24 hours and ultimately led to functional failure.

- The use of off-the-shelf electrical components reduced the reliability of electronic circuits such as radiation monitors. The reliability of these components could be significantly improved by using Military Standard class components. Other practices, such as the application of conformal coatings on circuit boards (pioneered by defense and aerospace research) could further enhance the reliability of these components.

- The in-core thermocouples (TCs), although disregarded by the operators during the accident, presented reliable thermal data. The thermal characteristics of the TCs and associated cables are well known; when the temperature of the cable exceeds 900°C, a virtual junction is formed that provides as valid a measurement as the original junction. The only difference is that the location of the measurement point may be unknown.

INSTRUMENTATION

Pressure Transmitters

Seven pressure transmitters, of 58 available, were removed from the TMI-2 Reactor Building and evaluated at the INEL. Four were Bailey BY types, of which three were inoperable, having been damaged by water intrusion, and three were Foxboro E11GM types that were operational and within specifications. Additional testing at the INEL concluded that the Bailey type BY transmitter would have experienced a zero shift of 15% at a total integrated radiation dose of 2.4×10^6 R, while the Foxboro type E11GM would have experienced a negligible change. Otherwise, radiation did not contribute to functional degradation of these pressure transmitters. A selection of the transmitters that were evaluated was based primarily upon their accessibility and the extent to which they were representing instruments typically used by the industry. Most of the desired Class 1E transmitters were inaccessible or were impractical to remove because they were, and still are, in high radiation fields. Others continue to be essential to the safe maintenance of the plant and cannot be taken from service.

CF-1-PT3 and CF-2-LT3. Transmitters CF-1-PT3 and CF-2-LT3, a Foxboro E11GM gauge pressure unit and a Bailey Meter Company BY differential pressure unit, were removed from the Reactor Building in July 1981. Since the units were radioactively contaminated, they required special handling, storage, and shipping. They were shipped to the INEL in November 1981 and examined and tested in February 1982.

Foxboro unit CF-1-PT-3 was one of two transmitters used to monitor pressure in core flood tank B. No failure or degradation of the instrument was reported during or after the accident. However, contaminants were found in the junction box mounted externally on the transmitter housing, indicating that water had entered the electrical conduit. A seal between the transmitter housing and the junction box prevented water from damaging the transmitter's electrical and mechanical components. The transmitter was located at the 324-ft elevation, well above the high water mark.

The Bailey unit, CF-2-LT3, was one of two transmitters used to monitor differential pressure for measurement of the level of core flood tank B. It was also located on the 324-ft elevation. This unit survived the accident, but failed one year later. Examination of this transmitter indicated that it had failed as a result of water intrusion through its conduit or fittings.

CF-1-PT1, CF-2-LT1, and CF-2-LT2. A Foxboro E11GM gauge pressure unit and two Bailey Meter Company BY differential pressure units were removed from the Reactor Building in June 1983 and examined by EG&G Idaho at the INEL. The Foxboro transmitter, CF-1-PT1, was operational. However, the two Bailey transmitters, CF-2-LT1 and CF-2-LT2, failed as a result of water damage to the signal conditioning electronics located inside the transmitter housing.[3]

Foxboro unit CF-1-PT1 was used to measure pressure in core flood tank A. No failure or degradation of the transmitter was reported during or after the accident. This unit was located at the 324-ft elevation. GPU Nuclear Corporation performed in-place testing of the unit in January 1983 under the direction of EG&G Idaho. The measurements included monitoring the transmitter's output, performing time domain reflectometry (TDR) measurements of input and output cables, and measuring capacitance and resistance. The transmitter's output signal corresponded with the actual tank pressure.

Visual examination indicated that the transmitter was in good condition except for external corrosion. The interior of the unit did not receive any corrosion or radioactive contamination as a result of the accident.

Bailey transmitters CF-2-LT1 and CF-2-LT2, located at the 324-ft elevation, were used to measure the water level in core flood tank A. In situ testing on CF-2-LT2 in September 1980 by Technology for Energy Corporation (TEC) indicated that the unit had failed. Both units were tagged out of service in December 1980 because they did not respond to a known level change in tank A. A visual external examination revealed a heavy layer of rust on assembly nuts and conduit fittings. Internally, severe corrosion and degradation of components has rendered the units inoperable. In addition, the transmitters were contaminated with a high level of internal radiation. The units had failed from the intrusion of water into the housings through the electrical conduits. One of the transmitters had a faulty seal around the cover plate, also contributing to moisture intrusion.

CF-1-PT4 and CF-2-LT4. An additional Foxboro E11GM gauge pressure unit and a Bailey Meter Company BY differential pressure unit were removed from the Reactor Building in March 1984. Some of the conduit and the associated junction box were removed with the transmitters in an effort to understand more about the nature of water entry into the previously examined units. Examination of these units at the INEL indicated no apparent physical or functional degradation due to water damage.[4]

Foxboro unit CF-1-PT4, located at the 324-ft elevation, was used to measure the pressure in core flood tank B. The transmitter was in good condition except for some minor external corrosion and radioactive contamination. The junction box showed no signs of internal corrosion, however, some mineral deposits were apparent, indicating that moisture had been inside at some time. Since the junction box seal appeared to be in good condition, it is likely that the water entered through the conduit. A seal between the transmitter and the junction box had prevented moisture from entering the transmitter. The unit, as removed, was operational and within specification.

Bailey unit CF-2-LT4, located at the 324-ft elevation, was used to measure the level in flood tank B. The exterior of the transmitter had some rusting, but the interior was free of corrosion and rusting. There was no evidence of moisture in the housing, even though no special installation or sealing procedures were used. Internal radioactive contamination was present, but this was most likely a result of the cover being removed while the transmitter was in the Reactor Building.

The only differences between the installations of all the Bailey units examined was in where the conduit entered the junction box, either on the top or on the side and where a breather/drain was installed, either on the bottom or on the side. The use of a single drain without an associated breather was contrary to recommendation by the manufacturer. Also, conduits entering the junction box on the top tend to funnel any moisture directly into the junction box. Bailey unit CF-2-LT4, which suffered no moisture damage, had the conduit entering the side and the drain/breather was also on the side.

Irradiation Testing. The irradiation testing program was developed to determine the accuracy of pressure transmitters in radiation levels similar to levels generated during the accident. The transmitters tested were new units and had different sensitivity ranges than the units removed from TMI-2, but were electrically identical. The testing included the associated cables so as to maintain circuit conditions as close to actual conditions as practical. The units evaluated were two Foxboro pressure transmitters and one Bailey liquid level transmitter.[5]

Data obtained from the transmitters included calibration and response time. The cables were evaluated for capacitance, inductance, impedance, insulation resistance, and loop resistance. In addition, the cable resonant frequency was recorded and used to monitor changes in the dielectric constant of the cable insulation. Time domain reflectometry was recorded prior to and during the test.

The transmitters were subjected to irradiation fields on the order of 10 milli-rad. The transmitters remained operational during the test; however, a 16% of span change was observed in the zero of the Bailey transmitter. Response times increased approximately 15%, but this was for less than 0.2 s for a 90% step change. Physically, some darkening of components was evident in all the transmitters, and the internal insulated wiring of the Bailey unit became brittle. Since the wiring of the Bailey transmitters from TMI was still flexible, it is postulated that those units were subjected to lower radiation doses than the test units. The cable parameters that were measured showed little effect from radiation.

Radiation had little effect on operation of the transmitters during and following the accident. Failure of the Bailey transmitters resulted from moisture inside the transmitter housing. Similar damage would have occurred to the Foxboro units had it not been for the sealing around the transmitter leads where they exited the housing. Two possible sources of the moisture included:

- Water from the Reactor Building spray system, or condensate (rain) from the humidity in the building having direct access to the cables in the cable trays and to the ends of the conduits.

- **Humidity in the building, combined with the lack of adequate ventilation in some of the conduit, caused condensate to form on the inner walls of the conduits and drain into the transmitter housings. The conduits associated with CF-1-PT4 and CF-2-LT4 appeared to have a breather in the system and showed little evidence of moisture.**

This investigation has shown that the transmitters are capable of surviving a loss-of-coolant accident (LOCA), but proper installation of the conduit, junction boxes, and cabling associated with the transmitter is essential for protecting the transmitters from intrusion of water or moisture. Greater care should be given to these designs and installations to ensure that proper drains and adequate ventilation are provided. Consideration should also be given by the manufacturer to providing a seal around electrical leads as they exit the transmitter housings, similar to the sealing technique used by Foxboro on its transmitters. It is the opinion of the investigators that unless the seal is an integral part of the unit, verification of resistance to water will not be possible after installation.

In-Core Instruments

Testing was performed on the in-core instruments to establish the operational conditions and failure modes of the TCs, self-powered neutron detectors (SPNDs), and background detectors. All TCs failed and the majority of the SPNDs and background detectors failed.[6]

Postinstallation loop resistance data were available on all the TCs to provide a baseline for further analysis of the in-core TC test data to determine the possible extent of overall in-core damage.[7] Based on this analysis, the location of newly formed TC junctions, which occurred as a result of the accident, were identified. The locations of these newly formed junctions provide an early indication of the extent of core damage. This data showed a strong connection with the later video Quick Look data obtained during July 1983 and with other known core conditions.

Even though all the TCs are considered to have failed, they continued to be reliable indicators of temperature, although precise location of the measured temperature zone would be unclear. For temperatures less than 900°C, the measured zone would be at the end of the damaged cable. For temperatures greater than 900°C, the measured zone (hot spot) could be anywhere along the cable. This hot spot measurement is a basic characteristic of the TC called the virtual junction phenomenon. A virtual junction is the creation of a low resistance path (less than 10,000 ohms) between the TC leads in the cable by heating the magnesium oxide insulation; this phenomenon occurs at approximately 900°C. The tolerance of the measurement system to high-resistance junctions is due to the inherent characteristic that the input impedance of the measurement systems is much greater than 10,000 ohms; thus, little voltage drop is lost across the junction. This has been demonstrated during severe fuel damage experiments at the INEL and is consistent with results reported by R. L. Anderson,[8] even though this reference analyzes the core exit temperature accuracy and treats the virtual junction temperature as an error.

The in-core instrumentation consisted of 52 detector assemblies located in instrument tubes distributed throughout the core. Each of the assemblies contained seven SPNDs, one background detector (an SPND without a beta emitter), and one Type K TC. The SPNDs were equally spaced at different elevations throughout the active core region, while the TCs passed through the active core region with their junctions positioned 18 cm above the core. Each assembly has a total length of 39 m.

All TCs failed and the majority of the SPNDs and background detectors had moisture in their insulation as a result of the accident. Both a resistive model analysis and a statistical analysis indicated that the center area of the core experienced the major changes. The statistical analysis further characterized instrument damage by core location. Instrument damage was greatest in the center of the core, where the higher temperatures existed. The least amount of instrument damage occurred around the perimeter of the core, where lower temperatures prevailed. Therefore, a strong correlation exists between core damage and instrument damage.

Resistance Temperature Detector

A "worst-case" resistance temperature detector (RTD) removed from the reactor four years after the accident, was still within original specifications for calibration, response time, and electrical properties. The unit met the bench mark response time in 75°C water flowing at 0.9 m/s. In addition, it was confirmed that the RTDs response time at full power conditions (290°C and 15 m/s) met the plant technical specifications.[9]

The RTD examined was a Rosemont Engineering Company model 177HW removed from the hot leg of loop A of the reactor coolant system. It was removed along with the thermowell as a complete assembly. It was shipped to the INEL in April 1984, examined electrically, and later decontaminated. It was then shipped to the ORNL where it was checked as found, for response time, and then further disassembled and examined. No anomalies were discovered upon disassembly.

The removed RTD was selected because testing had shown that it had the lowest insulation resistance and heat transfer coefficient of seven RTDs tested while installed in the hot and cold legs of loops A and B. Examination determined that the low insulation resistance was a result of cable degradation, and further, that this was probably caused by a damaged conduit connection that allowed water and steam to enter. Since this RTD was the worst case RTD as determined from testing, and since it met plant specifications, it was concluded that all RTDs survived the accident environment without functional damage.

Radiation Monitors

The radiation instruments used as area monitors all failed during the time period of one hour to 218 days after the accident started. They were selected as the first class of equipment to be removed and examined for this reason and in an attempt to validate or add to the recorded information on radiation levels. One Geiger-Mueller (GM) tube detector failed naturally as a result of total dose gas depletion in the GM tube. One detector failed with a fractured GM tube during the hydrogen burn. Another detector failed as a result of improper installation and failure to mate properly with the electrical connector. The ion chamber detector used for the dome area radiation monitor failed with multiple problems. All of the GM tube detectors had an inherent design problem which caused them to indicate an apparent on-scale reading when subjected to a very high level off-scale radiation field. In addition, all of these instruments used radiation-sensitive transistors.

Transistor current gain (HFE) degradation and elastomeric material degradation properties, were used to estimate the total gamma radiation dose received by the radiation detectors. These doses, shown in the table below, are indicative of levels seen by other instruments and cables inside the Reactor Building. These estimates refer only to gamma-induced damage and not to beta damage, since beta damage is generally a surface phenomenon.

Reactor Building Elevation (ft)	Instrument	Dose (rad)
305	HP-RT-211	2.5 x 10+E5
305	HP-R-212	4.5 x 10+E5
347	HP-R-213	9.9 x 10+E5
372	HP-R-214 cable	7.9 x 10+E6
372	HP-R-214 detector	2.2 x 10+E5

Area Radiation Monitor HP-RT-211. HP-RT-211 was a Victoreen Instruments, Inc. model 857-2 GM detector installed at the 305-ft elevation in the Reactor Building. The monitor indicated a steep rise in radiation level six hours into the accident, peaked at 180 mR/h, and dropped rapidly until it was below scale. Subsequent testing indicated that the unit had a bad output drive transistor. The detector was removed from the Reactor Building and tested along with three new units at SNLs Gamma Irradiation Facility. The detectors were subjected to radiation fields up to 100,000 R/h. All detectors exhibited inconsistent behavior after being exposed to saturation or off-scale levels indicating on-scale levels. This inherent characteristic of the detectors constitutes a dangerous condition, since the monitors could indicate low radiation readings where an intense field may exist.[10]

This multivalued behavior of the monitors was determined to be the result of two factors not recognized in the design:

- The impedance mismatch between the detector output circuit and the coaxial cable used to transmit the signal to the remote indicator
- The GM tube/circuit interactions above the anti-jam point.

These deficiencies can be corrected with relatively simple design changes:

- Match the output signal circuit resistance to that of the coaxial cable

- Use radiation-tolerant output signal circuit transistors

- Increase the base drives in the output signal circuit transistors

- Disable the GM tube pulse output to the multivibrator when the anti-jam circuit is operational.

Two additional improvements in the ratemeter circuit design should be:

- Terminate the coaxial cable in 50 ohms to prevent reflections (gain changes in the ratemeter differential amplifier would be required)

- Use a zero-crossing comparator circuit to reconstruct the detector square wave and thus make the ratemeter input amplitude insensitive (a new printed circuit board would be required).

Similar problems could exist in other makes of GM detectors. ANSI N42.3-1969/IEEE Std. 309-1970, "Standard Test Procedure for Geiger-Mueller Counters," should be revised to incorporate a requirement for subjecting all untested GM detector models to 100,000% overload with a conservative length-of-signal transmission cable attached.

Substantial evidence indicates that the transistor failed as a result of spray or steam entering the connector assembly where the detector and cable mate, shorting the GM tube power line momentarily to the signal output line. To prevent future occurrences, these and similar detectors should be mounted with the connector below the housing and with the backshells potted. Further, only the sensor should be inside the reactor building; active electronics should be located outside the Reactor Building whenever practical.[11]

Area Radiation Monitor HP-R-213. HP-R-213 was located in the in-core instrument service area at the 347-ft elevation. It was operational during the accident until the hydrogen burn occurred, after which its output went to zero. The detector was removed from the Reactor Building in May 1981 for examination at SNL. The examination showed that the detector failed due to a cracked GM tube. Due to the presence of numerous chips and scratches around the glass-to-metal seal, the glass was apparently in a weakened state, and then failed as a result of the mechanical shock imparted to the assembly by the hydrogen burn. Since the glass thickness and printed circuit (PC) board mounting method should allow a good quality tube to withstand substantially higher shocks, tighter quality controls should be instituted for GM tube manufacturing.[12]

Radiation Detector HP-R-212. HP-R-212 was installed at the 305-ft elevation of the Reactor Building. The unit was not in use during the accident, but was energized three months later, after which it worked for five months before failing. The detector was then removed from the Reactor Building in November 1981. Its failure was the result of exhausting its quench gas and then continuously discharging. All of its transistors were functioning.[13]

Dome Radiation Monitor HP-R-214. The dome radiation monitor was the only instrument inside the Reactor Building capable of measuring the high radiation levels which resulted from the accident. Therefore, plant technical specifications required it to be operative throughout a LOCA. The readings from the dome monitor alone may be used to declare a general emergency. Since the accident, the dome monitor has been assigned a more important role in Regulatory Guide 1.97.

The dome monitor detector is located on top of the elevator shaft enclosure roof at the 372-ft elevation. It is a Victoreen model 847-1 detector, and consists of dual ion chambers and a fairly complex electronics package. These two components are housed inside a sealed container which is itself inside a sealed lead-lined pressure vessel. The failure modes described below were generally the result of the severe, but not unreasonable, Reactor Building environment.[14]

- *Moisture Intrusion into the Detector Electronics Package.* The protective stainless steel pressurized vessel seal leaked and allowed moisture from the Reactor Building atmosphere to enter the vessel. This moisture easily permeated into the detector electronics package because of an error in sealing the detector mounting bracket to the detector. This moisture reduced the resistance to ground in the high impedance ion chamber circuit, thus degrading the detector radiation measurement accuracy significantly. Moisture may have entered the electronics sometime within the first three hours of the accident.

- *Direct Current Feedback in the Preamplifier.* The effects of mositure were further accentuated by dc feedback paths in the two preamplifier circuits. Lowering preamplifier input impedances by the presence of moisture, coupled with the dc feedback paths caused the detector to at times indicate higher and lower levels of radiation than were actually present.

- *Metal Oxide Semiconductor Transistor Degradation.* Both ion chambers use 3N163 Solitron Metal Oxide Semiconductor (MOS) transistors to form high input impedance circuits. These MOS transistors were severely degraded by radiation exposure, and eventually caused irregular jumps in radiation readings.

- *Electrolytic Capacitor Failure.* Capacitor C17 leaked electrolyte onto the circuit board sometime after 416 days from the start of the accident. This leakage not only reduced the capacitance of C17, but also corroded completely through a transistor lead.

- *Reed Switch Reliability.* It is thought that neither reed switch in the preamplifier circuits failed during the accident. However, either they were both degraded, or they were unacceptably fragile as installed.

The following recommendations are provided to improve high-level radiation monitoring:

- The detector should be more nearly hermetically sealed. A single O-ring gasket of such a large circumference and with the particular sealing arrangement on HP-R-214 is not sufficient. The detector should be sealed periodically, and leak tested to verify that it is sealed.

- The detector should not be used inside a thick lead-shielded vessel since it is impossible to predict levels outside such a shield. If this recommendation is implemented, the detector electronics must either be redesigned to operate after accumulating extremely high total radiation doses, or must be removed from the Reactor Building altogether. It is difficult to design a radiation-hardened circuit to operate in the Mrad region; therefore, it is recommended that the electronics be placed outside the Reactor Building (the proper seals are still required for the ion chambers). If this is done, the maximum detection level should be increased from 10,000 R/h to at least 1 mR/h. The minimum detection level can be increased from 0.1 mR/h to 100 mR/h. This can be done because this instrument is intended to operate in a LOCA, and not simply to monitor normal low levels of radiation.

- MOS transistors or MOS-integrated circuits should not be used in any application where radiation exposure is a possibility. Most MOS devices are abnormally radiation-sensitive and degrade dramatically at reasonably low doses.

- Military grade or better components are recommended for electronics packages. Mil Std 883 class B components should be sufficient for this application. These components undergo rigorous inspection and testing procedures and have a much improved reliability over standard commercial-grade components. Commercial grade components such as the electrolytic capacitors, plastic-encapsulated transistors, and reed switches are not suited for use in such an important piece of equipment, particularly where severe environments are possible.

- All PC boards need to be uniformly coated to minimize effects in the event that moisture is able to circumvent a hermetic seal.

Loose Parts Monitoring System Charge Converters

The TMI loose parts monitoring system charge converters failed as a result of accumulated radiation exposure. In October 1981, two charge converters from the Tennessee Valley Authority's Sequoyah Unit 1 failed, and were examined to determine whether they had failed for the same reason as those at TMI. The Sequoyah converters had been mounted within the keyway, 3 m outside the mirror insulation underneath the reactor vessel, i.e., in the high radiation field of the reactor vessel itself. The TMI converters were mounted away from the normal radiation sources; however, the accident exposed them to moderate levels of radiation.

Analysis revealed that the gain select capacitor was the source of failure. The converters are neither designed nor manufactured to be radiation-tolerant; their sensitivity to radiation makes them unsuitable for nuclear applications. Regulatory Guide 1.133 should be modified to require the use of radiation-resistant transistors. Specified operating radiation levels should be consistent with those required for Class 1E equipment, since the loose parts monitoring system can, as stated in the Guide, "provide the time required to avoid or mitigate safety-related damage to or malfunctions of primary system components."[15,16]

CABLES AND CONNECTIONS

Cables and connections examination is an ongoing task. The status of the task is included in this summary report to present a complete picture of activities of the I&E Data Acquisition Program. Inplace test results have been obtained on 460 circuits, with 178 abnormalities identified. Of these, 36 circuits are failed, 38 circuits show significant changes, and 104 circuits show minor changes. The circuits represent a two-wire transmission line from the Reactor Building wall up to and including the end device. The final phase of this program is underway to obtain samples from these circuits to quantify the abnormalities and to assess the degree of functional impairment to the circuit.

Generally, the data show evidence of moisture and degradation which might be expected as a result of corrosion. Most of the cable analyses were based on comparisons of measured data to expected values which were obtained from laboratory measurements of identical or equivalent samples of the subject component. Due to manufacturing variations of the parameters measured (important for in-place testing, but not necessarily important to plant operation), these expected values are not precise. The ideal basis for comparison would be to have similar data taken at a known plant condition, preferably during plant startup.

Examination, testing, and analysis will provide information for better assessment of reliability and performance and for improvements in design, manufacture, and installation of instrumentation, electrical, and cable systems equipment. Three factors in particular are relevant for the cables and connections examinations at TMI: (a) all controls and information operate and flow via cables; (b) over 150,000 m of cable represent a substantial cost; and (c) analysis of these systems provides information for monitoring postaccident conditions, developing cleanup plan parameters, and ensuring greater plant safety.

HP-RT-211 Cable Analysis

A section of multiconductor from HP-RT-211 was examined to determine any effects from prolonged exposure to the radiation and thermal environment inside the reactor building. The cable was tested, along with three new samples of similar cable, for ultimate tensile strength, percent elongation at break, and insulation resistance. As received, the cable exhibited low radiation levels, but its heat shrink wrapping was resistant to decontamination. The cable was easily decontaminated after the heat shrink was removed.[17]

The examination indicated no substantial differences between the new cable and the TMI cable. Both were also compared to the manufacturer's specification for the cable, which they clearly surpassed. The manufacturer also requires its cable to withstand radiation levels up to 1 Mrad. Since the cable was probably exposed to less than 1 Mrad, and since manufacturing variations were as important as the TMI environment in establishing electrical properties, the cable could not be used to establish an independent assessment of the radiation levels to which its detector was exposed.

Penetration R607 Cables

R607 is a 137-channel instrumentation and control penetration. Of those available channels, 49 were initially chosen for testing based on the following:

- Results of prior testing
- End instrument removal
- Location in the Reactor Building
- Cable type representation
- Ability to provide supporting data for future testing.[18]

Mass screening tests indicated several broken wires and corroded contacts were in R607. Insulation resistance measurements between wires of different cables yielded evidence of "cross talk," an interference between wires in the penetration. The penetration is at the 292-ft elevation; it was therefore submerged until the Reactor Building basement water level was eventually lowered, and water remaining in the penetration may be the cause of the cross talk. The most predominant cable anomaly encountered is a shift in cable characteristic impedance, which could also be caused by moisture ingress through the insulation. Subsequently, three additional channels were tested. Of the 52 channels, 47 exhibited anomalous behavior, and 33 of these were determined to be inoperable.

Additional Testing

Five cables were tested in penetration R405 and all exhibited anomalous behavior; four were judged inoperable.

Fourteen instrumentation cables were tested in penetration R534; anomalies were observed in seven, of which five were judged inoperable. Cross talk voltages were observed, which suggested possible corrosion or water contamination. However, the data show that environmentally sealed splices survived well.

Penetration R506 contains reactor control circuits, including current transformers; level (pressure) transmitters; and temperature, pressure, and limit switches. Nineteen cables were tested; 16 exhibited anomalous behavior, of which six were judged inoperable.

Of 39 pressurizer heater cables, anomalous behavior was observed in 12. Five were determined to be inoperable: one with an open circuit, one with a short circuit, and three with low insulation resistance.

Since the penetrations evaluated were selected because of their high probability of impairment, the data reported are not statistically representative of the 1800 circuits in the Reactor Building, but should serve as an indication of the damage to be expected from this type of accident. Notably, except as discussed previously for one radiation detector, the hydrogen burn did not result in substantial instrumentation damage.[19]

ELECTRICAL

Electrical components were generally evaluated in situ. Radiation levels, large size, or the need to continue use of the component often made removal either impossible or prohibitively costly. Therefore, in situ testing techniques were developed at TMI-2 that have proved to be very informative. The tests are made at the connecting cable from a point outside the Reactor Building, generally at the outer penetration box. The testing consists of static measurements (resistance, capacitance, inductance, insulation resistance, and time domain reflectometry) and dynamic measurements (in-rush and steady state current measurements, signal spectral content analysis, and time response measurements). When it became possible to remove and examine components, the results were in agreement. The general conclusion reached from examination of the components was that moisture intrusion was the greatest problem.[20,21]

Level Switches

AH-LS-5006, -5007, and -5008 were three level switches used for leak detection of the Reactor Building air cooler coils. They furnished a signal for control room annunciation when a high level existed in the associated cooler sump. In situ tests revealed an abnormally high dc resistance on AH-LS-5006, and an indication of wetness at or close to the device. AH-LS-5007 also exhibited a slightly elevated dc resistance—an indication of an impending deterioration of the circuit. The elevated resistance, however, was not high enough to disable the circuits functionally. Subsequent removal and examination of AH-LS-5006 revealed a break in the level switch seal, a fault which rendered the device nonfunctional.

Solenoid Valves

AH-EP-5037 and -5039 are two ASCO solenoid valves used as the pilot valves of Reactor Building purge valves AH-V2B and AH-V2A. These valves were cited in an NRC IE bulletin to be unqualified for use in the Reactor Building. In situ tests revealed both valves to be normal in all aspects, statically as well as operationally. Laboratory examination of AH-EP-5039 confirmed the functionality of the device, except for a slight leak through the valve disk sealing surface, which was caused by rust flakes suspected to have originated from the air piping. The examination further revealed the absence of noticeable degradation of the purported weak parts (Buna-N and acetal plastic).

AH-V6 and AH-V74 are two VALCOR Class 1E solenoid valves, the former used as a Reactor Building isolation valve of the Reactor Building pressure instrument line, and the latter as pilot valve of the LOCA dampers. Both solenoid assemblies tested and operated normally; however, one limit switch of AH-V6 would not actuate, i.e., it remained closed whether the valve was open or closed. Examination of both solenoid assemblies revealed a gross rusting on the shell of AH-V6 caused by water intruding through the housing conduit opening. The water also intruded into the defective limit switches. Although the insulation of the solenoid assembly's internal wiring was intact, there was a slight discoloration and embrittlement. The intrusion of water into the AH-V6 solenoid assembly was attributed to leakage through the housing electrical penetration, which could have been prevented by sealing the interconnecting conduit.[21]

Power Operated Relief Valve RC-R2

RC-R2 is a solenoid-operated pressure control device used as part of the reactor pressure control system. It opened during the accident and then failed to close, leading to uncovering of the core and subsequent core damage. In situ tests on the valve revealed the solenoid plunger-actuated cut-out switch used to bypass the holding coil was open. The open position of the switch would prevent the actuation of solenoid plunger if the coil is energized. The electrical characteristics of the remainder of the valve circuit were normal.

Pressure Switches

NM-PS-1454, -4174, and -4175 were used in the Reactor Building nitrogen system to monitor the nitrogen manifold pressure and actuate an alarm when the pressure was outside the operating range. In situ tests of the three devices exhibited normal electrical characteristics of their circuits. NM-PS-4174, however, was found in a closed state instead of open. The wrong state of the switch was attributed to a mechanical binding of the pressure-actuated stem rather than to a failure of the switch. The low setting of the device coupled with the fact that the device was not cycled for a long period of time may have caused the instrument-actuated stem to bind slightly, hence, the wider dead band. All three pressure switches exhibited slight deterioration.

Reactor Coolant Motor Lube Oil Instruments

Two flow switches, two level switches, and a pressure switch were used on each reactor coolant pump motor to monitor the lube oil condition. These instruments provide input to the balance of plant (BOP) computer. When these instruments were tested, a high dc stray voltage was measured in their circuits. A dc resistive coupling occurring in the inner box of the electrical penetration which the circuits passed through probably caused the crosstalk. The cause of the fault is suspected to be contamination tracking that built up when the penetration was partially under water. The test also showed an impedance mismatch somewhere in the middle of the Reactor Building cables of the RC-P-1A and RC-P-2A instruments. The pattern of the mismatch is characteristic of that of a wet cable.

Vibration Switches

RC-67-VS1, -VS2, -VS3, and -VS4 are vibration switches mounted on reactor coolant pump motors RC-P-1A, -2A, -1B, and -2B, respectively, to monitor the associated motor vibration. RC-67-VS3 and -VS4 exhibited no abnormality in their respective circuits during their in situ testing. RC-67-VS1 and -VS2, on the contrary, exhibited a very high dc resistance in their respective reset coil circuit. TDR tests showed both instrument faults were located in the device proper, leading to speculation that the nature of the fault was corrosion on the connection point.

Reactor Coolant Pump Motors

The reactor coolant pump motors appeared to be electrically intact, in spite of the high vibration operation they experienced during the accident. Their insulation resistances, however, were marginal. The associated power monitoring current and potential transformers of RC-P-1A and RC-P-2A exhibited severe abnormalities. High series resistance occurred in these circuits and a ground wire of RC-P-1A's current transformer was missing. The high resistance faults, pinpointed by a TDR, were found in the inner box of the associated electrical penetration.

Reactor Coolant Pump Motor Lube Oil Pump Motors

The reactor coolant pump motor lube oil pump motors drive the pumps that provide lubricating oil to the associated reactor coolant pump motor backstop and guide bearings. Five units were in situ tested and each exhibited normal electrical characteristics which are compatible with each other. Their insulation resistances were also high, although two units had a low polarization index—an indication that the motors' insulations may be dirty.

Reactor Coolant Motor Backup Oil Lift Pump Motors

The backup oil lift pump motors are 10 hp, 250 Vdc shunt-wound motors. They drive the oil pumps that provide lubrication to the reactor pump motor thrust bearings. All four motors were in situ tested. The results showed a fault on the circuit of the motor associated with RC-P-1A. A high resistance existed on the armature circuit and on the field circuits. The fault on the armature circuit was pinpointed on the commutator brush interface and the field circuit in the inner penetration box. Like all unprotected junction points, the faults on the RC-P-1A backup oil lift pump motor were suspected to be due to corrosion.

Motor Operated Valves

Twenty-two motor operated valves, 18 of which were Class 1E, were tested. Seventeen were located in the Reactor Building basement. Five units have totally failed and a few others have degraded to a lesser degree. The five that failed have the same failure mode. They exhibited low insulation resistances, and their TDR traces are characteristic of wet circuits. There were also uncharacteristic changes on the limit switches, all of which indicate a wet circuit. Meanwhile, the degraded units also showed some minor wetting but the overall effect on their electrical characteristics was minimal. The five valves located above the ground elevation of the Reactor Building incurred no appreciable degradation.

General Electrical Conclusions and Recommendations

The tested electrical components exhibited anomalies ranging from mildly elevated switch contact resistance to a catastrophic break or discontinuity in AWG-2 and AWG-10 circuits. The Class 1E valves that failed were submersed in the basement. The other components that were tested are located in the 'A' D-ring or are associated with reactor coolant pumps RC-P-1A and RC-P-2A. The types of anomalies which can reasonably be attributed to the accident are as follows:

- High humidity and wetting brought about failure of Reactor Building components by lowering the insulation resistance and dielectric strength, and by corroding joints and splices.
- Buildup of corrosion products on motor commutator brushes enhanced by the chemical spray during the Reactor Building suppression spray event may have contributed to high circuit series resistance.
- Discontinuity in circuits in the southwest electrical penetrations can be attributed to steam originating from the reactor coolant drain tank; the steam could have enhanced the corrosion process on the ring tongue terminals used in penetration boxes, and eventually caused the connectors to break away from the terminal block.

Prevention or mitigation of nuclear reactor accidents can be enhanced by knowing the physical limitations of the I&E equipment exposed to that severe environment.

REFERENCES

1. *GEND Planning Report*, GEND-001, October 1980.

2. R. C. Strahm and M. E. Yancey, *TMI-2 Pressure Transmitter Examination Program Year-End Report: Examination and Evaluation of Pressure Transmitters CF-1-PT3 and CF-2-LT3*, GEND-INF-029, February 1983.

3. M. E. Yancey and R. C. Strahm, *TMI-2 Pressure Transmitter Examination and Evaluation of CF-1-PT1, CF-2-LT1, and CF-2-LT2*, GEND-INF-029 Vol. II, April 1984.

4. M. E. Yancey, *Examination and Evaluation of TMI-2 Transmitters CF-1-PT4 and CF-2-LT4*, GEND-INF-064, January 1985.

5. M. E. Yancey, *Irradiation Test Report—Foxboro E11GM, Bailey BY3X31A, and Flame Retardant Ethylene Propylene Instrumentation Cable*, GEND-INF-058, August 1984.

6. M. E. Yancey and N. Wilde, *Preliminary Report of TMI-2 In-Core Instrument Damage*, GEND-INF-031, January 1983.

7. M. E. Yancey, R. D. Meininger, and L. A. Hecker, *TMI-2 In-Core Instrument Damage: An Update*, GEND-INF-031 Vol. II, April 1984.

8. R. L. Anderson et al., *Interpretation of TMI-2 Data*, NSAC-28, Nuclear Safety Analysis Center, May 1982.

9. R. M. Carroll and R. L. Shepard, *Examination of Platinum Resistance Thermometers Installed in TMI-2 Reactor During the Accident*.

10. J. W. Mock, F. T. Soberano, and M. B. Murphy, *Quick Look Report on HP-RT-0211 Multivalued Behavior*, GEND-INF-008, July 1981.

11. M. B. Murphy, G. M. Mueller, and F. V. Thome, *Examination Results of the Three Mile Island Radiation Detector HP-RT-211*, GEND-014, October 1981.

12. G. M. Mueller, *Examination Results of Three Mile Island Radiation Detector HP-R-213*, GEND-019, November 1982.

13. G. M. Mueller, *Examination Results of the Three Mile Island Radiation Detector HP-R-212*, GEND-INF-049, January 1984.

14. M. B. Murphy, G. M. Mueller, and W. C. Jernigan, *Analysis of the TMI-2 Dome Radiation Monitor*, GEND-INF-063, February 1985.

15. M. B. Murphy and R. E. Heintzleman, *Examination Results on TMI-2 LPM Charge Converters YM-AMP-7023 and YM-AMP-7025*, GEND-020, November 1982.

16. M. B. Murphy, *Sequoyah Unit 1 Charge Converter Examination Results*, GEND-INF-046, January 1984.

17. V. J. Dandini and L. D. Bustard, *HP-RT-211 Cable Analysis*, GEND-INF-010, September 1981.

18. L. A. Hecker and H. J. Helbert, *TMI-2 Cable/Connection Program: A Look at In Situ Test Data*, GEND-INF-042, December 1983.

19. H. J. Helbert et al., *TMI-2 Cable/Connections Program: FY-84 Status Rcport*, GEND-INF-056, September 1984.

20. F. T. Soberano, *Testing and Examination of TMI-2 Electrical Components and Discrete Devices*, GEND-INF-034, November 1982.

21. F. T. Soberano, *Evaluation Results of TMI-2 Solenoids AH-V6 and AH-V74*, GEND-INF-045, January 1984.

ABSTRACT

This report summarizes the investigations on instrumentation and electrical systems that were subjected to a loss-of-coolant accident environment during and following the accident at Three Mile Island Unit-2 (TMI-2) on March 28, 1979. The report is a summary of information previously published in GEND-INF reports (see references), plus current knowledge of the investigators. The investigations reported here were funded by the Department of Energy and performed under the direction of EG&G Idaho. GPU Nuclear Corporation cooperated during the investigations by providing access to the plant for testing and by providing components for examination. The acquisition of data from TMI-2 is conducted under the GEND agreement between GPU Nuclear Corporation, the Electric Power Research Institute (EPRI), the Nuclear Regulatory Commission (NRC), and the Department of Energy (DOE).