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Qualification of Daiichi Units 1, 2, and 3 Data for Severe Accident Evaluations - Process and Illustrative Examples from Prior TMI-2 Evaluations

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ABSTRACT

The accidents at the Three Mile Island Unit 2 (TMI-2) Pressurized Water Reactor (PWR) and the Daiichi Units 1, 2, and 3 Boiling Water Reactors (BWRs) provide unique opportunities to evaluate instrumentation exposed to severe accident conditions. Conditions associated with the release of coolant and the hydrogen burn that occurred during the TMI-2 accident exposed instrumentation to harsh conditions, including direct radiation, radioactive contamination, and high humidity with elevated temperatures and pressures. As part of a program initiated in 2012 by the Department of Energy Office of Nuclear Energy (DOE-NE), a review was completed to gain insights from prior TMI-2 sensor survivability and data qualification efforts. This review focused on the set of sensors deemed most important by post-TMI-2 instrumentation evaluation programs. Instrumentation evaluation programs focused on data required by TMI-2 operators to assess the condition of the reactor and containment and the effect of mitigating actions taken by these operators. In addition, prior efforts focused on sensors providing data required for subsequent forensic evaluations and accident simulations.

To encourage the potential for similar activities to be completed for qualifying data from Daiichi Units 1, 2, and 3, this report provides additional details related to the formal process used to develop a qualified TMI-2 data base and presents data qualification details for three parameters: primary system pressure; containment building temperature; and containment pressure. As described within this report, sensor evaluations and data qualification required implementation of various processes, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors that were similar in design to instruments easily removed from the TMI-2 plant for evaluations. As documented in this report, results from qualifying data for these parameters led to key insights related to TMI-2 accident progression. Hence, these selected examples illustrate the types of activities completed in the TMI-2 data qualification process and the importance of such a qualification effort. These details are documented in this report to facilitate implementation of similar process using data and examinations at the Daiichi Units 1, 2, and 3 reactors so that BWR-specific benefits can be obtained.

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EXECUTIVE SUMMARY

The accident at the Three Mile Island Unit 2 (TMI-2) reactor provided a unique opportunity to evaluate instrumentation exposed to severe accident conditions. Conditions associated with the release of coolant and the hydrogen burn that occurred during this accident exposed instrumentation to harsh conditions, including direct radiation, radioactive contamination, and high humidity with elevated temperatures and pressures. As part of a program initiated in 2012 by the Department of Energy Office of Nuclear Energy (DOE-NE), a review was completed to gain insights from prior TMI-2 sensor survivability and data qualification efforts. This initial DOE review considered over 100 references related to the 170 measurements that were deemed to be of higher priority and documented conclusions related to the sensor survivability and data qualification.

Because there is the potential for similar activities to be completed for qualifying data from Daiichi Units 1, 2, and 3, this report was authored to provide additional details related to the process used to develop a qualified TMI-2 data base and present data qualification details for three parameters, primary system pressure, containment building temperature, and containment pressure. As described within this report, sensor evaluations and data qualification required implementation of various processes, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors that were similar in design to instruments easily removed from the TMI-2 plant for evaluations. Results from qualifying data for these parameters led to key insights related to TMI-2 accident progression. Hence, these selected examples illustrate the types of activities completed in the TMI-2 data e qualification process and the importance of such a qualification effort.

Many post-accident investigations concluded that actions taken by plant operators adversely contributed to the TMI-2 accident. However, the operators' ability to mitigate the accident was impacted by their limited access to accurate plant data. In addition, the ability to improve severe accident analysis codes is dependent on the quality of data used in accident simulations. After the event, an evaluation program was initiated to determine what data were available to the operators and the status of sensors from which such data were obtained As part of that effort, a process was initiated that resulted in a qualified data base for TMI-2 post-accident evaluations. Several techniques were used in the TMI-2 Accident Evaluation Program (AEP) to assess sensor status, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors that were similar in design to instruments easily removed from the TMI-2 plant for evaluations.

Background Information

As discussed in Section 2, the TMI-2 power plant contained a Babcock & Wilcox, Inc. (B&W) PWR with a Reactor Coolant System (RCS) that consisted of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The RCS was arranged into two heat transport loops, each with two pumps and a steam generator (often designated as the A and B loops). The TMI-2 containment building consists of a large, domed, cylindrical steel shell surrounded by reinforced concrete; the inside diameter and height are approximately 130 ft (40 m) and 190 ft (68 m), respectively.

The TMI-2 accident was initiated on March 28, 1979, by a shutdown of secondary feedwater flow due to condensate booster pump and feedwater pump trips that occurred when the plant staff was trying to unclog a pipe leading from the condenser demineralizers. Best estimates for plant data and events during the accident, as obtained from various post-accident evaluation programs, are depicted in Figure ES-1. As described in Section 2, significant events occurring during the initial stages of the accident included turbine isolation (defined as time zero in Figure ES-1), reactor trip (when reactor pressure reached 16.3 MPa at 10 seconds after turbine trip), RCS heat up and pressurization. The Pilot Operated Relief Valve (PORV) opened to relieve RCS pressure, but failed to close when RCS pressure decreased. This was incorrectly interpreted by the reactor operators, as indicating that the RCS was nearly full of water; when in fact, the RCS was continually losing its water inventory. Emergency core cooling was reduced by operators, and the coolant void fraction increased due to coolant loss through the PORV and decay heat generation in the fuel. The steam fraction in the primary system piping increased to such an extent that RCS pumps were tripped by the operators to prevent permanent damage from pump cavitation after 100 minutes.

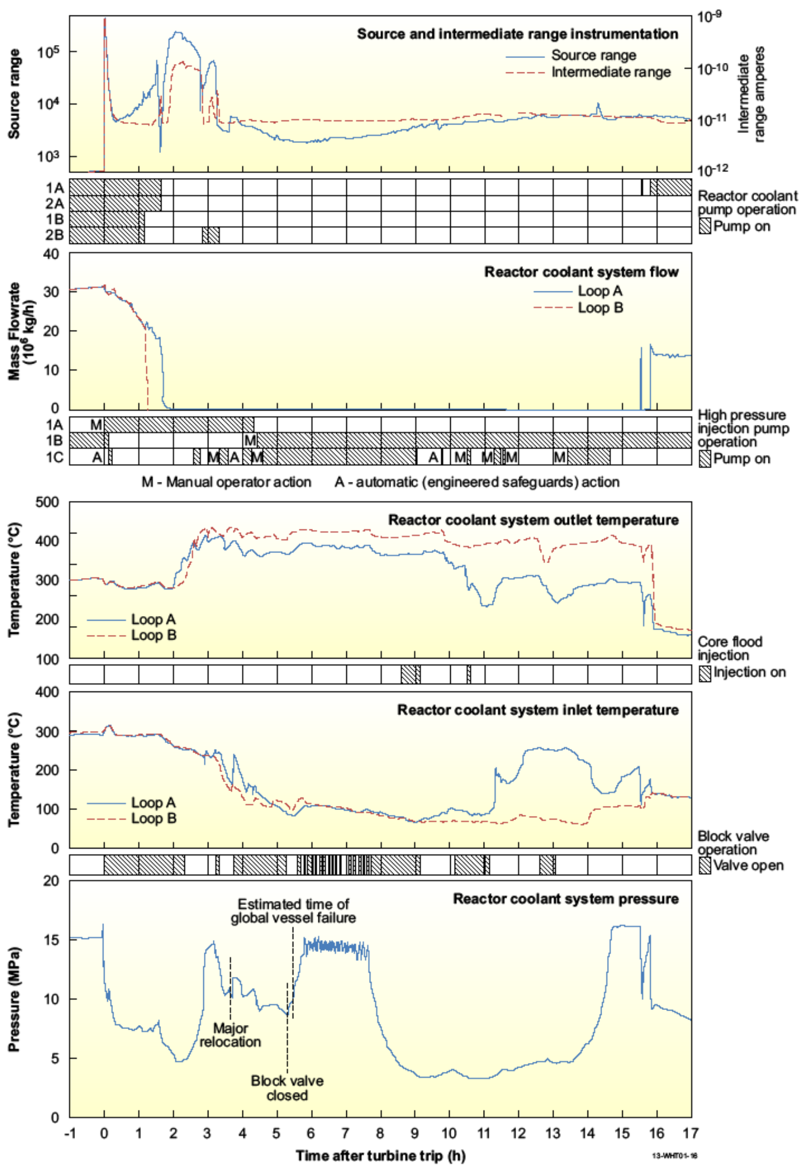

Figure ES-1. TMI-2 data from March 28, 1979.

Instrumentation response suggests that core uncovery began between 114 and 120 minutes and that the vessel liquid level had dropped to the core midplane by approximately 140 minutes. Insufficient decay heat removal associated with core uncovery is estimated to have led to upper regions of the core heating to temperatures that caused the cladding to overheat, balloon, and rupture. When operators finally realized that the PORV had failed in the open position, they closed the pressurizer block valve upstream of the PORV. In-core self powered neutron detector (SPND) output and RCS pressure data (see Figure ES-1) indicate that core temperatures continued to increase between 150 and 165 minutes. Zircaloy-steam exothermic reactions were initiated, producing large amounts of hydrogen and dramatically increasing the core heatup rate. When Zircaloy melting temperatures were exceeded, molten Zircaloy and some liquefied fuel relocated to lower core regions, solidifying near the coolant interface. This continued until 174 minutes, when a dense agglomeration of degraded core material formed in lower regions of the core, blocking core flow.

At 174 minutes, one of the reactor coolant pumps in the B-loop was turned on for approximately 19 minutes. This coolant injection into the vessel rapidly repressurized the RCS. At 200 minutes, the high pressure injection system was operated for 17 minutes, and the reactor vessel was refilled with water by approximately 207 minutes. Although the core was estimated to have been covered with coolant, analyses suggest that little coolant was able to penetrate into core regions with agglomerated debris and that these materials continued to heat up. Between 224 and 226 minutes after reactor scram, plant instrumentation (RCS pressure increases, Source Range Monitors (SRMs) count rate increases, cold leg temperature increases, and in-core SPND signal increases) indicated that the outer crust (resolidified molten material) surrounding the relocated core material failed; and molten core material relocated to the lower plenum. Increases in SRM count rates (see Figure ES-1) suggest that small quantities of molten debris may have continued to relocate to the lower head between 230 and 930 minutes, although peak count rates are considerably lower than values during the 224 to 226 minute relocation time period. At 930 minutes, one of the A-loop primary coolant pumps was restarted, re-establishing heat removal from the vessel.

TMI-2 Sensor Evaluations

An important aspect of the TMI-2 Accident Evaluation Program (AEP) was to provide a qualified data base for an analysis of the TMI-2 Accident, known as the “TMI-2 Analysis Exercise.” This analysis exercise was completed to assess the accuracy of available data and modeling tools, which were developed using small-scale experiments, when they were applied to a full scale PWR. A qualified database and a data qualification process were established for this analysis exercise. Prior to being entered into the database, the data and estimated uncertainties were reviewed by a Data Integrity Review Committee (DIRC). Understanding gained from the TMI-2 Accident Exercise was ultimately applied toward improving phenomenological models related to the chemical and materials interactions that occurred in the TMI-2 core and resolving applicable severe accident and source term issues. Hence, the analysis exercise contributed toward establishing a sound technical basis for post-TMI-2 regulatory actions.

A TMI-2 Initial and Boundary Conditions Data Base was established at around 6 years after the TMI-2 accident to provide a qualified database for this analysis exercise. This measurement data was the sole source of data used by all participants in the TMI-2 Analysis Exercise. A data qualification process (see Figure ES-2) was developed that included: collecting the TMI-2 measurement data and support information; establishing priorities and designing a formal approach for systematically performing the uncertainty analyses; and establishing quality categories of the data. Because of the limited time available for completing the data qualification process, the three items on the top of the chart were performed at approximately the same time, i.e., develop procedures, collect data, and establish priorities. It was necessary to define a methodology for performing the uncertainty analyses on the data, the criteria for determining the data quality categories, and the internal review process. These procedures were prepared at the beginning of the program and were modified as the realities of the task dictated. Prior to being entered into the database, the data and uncertainties were reviewed by a Data Integrity Review Committee (DIRC), which was composed of a panel of experienced persons knowledgeable in TMI-2 data analysis. The DIRC reviewed available information, including analyses, evaluations, and comparisons, to ensure that the data met established criteria.

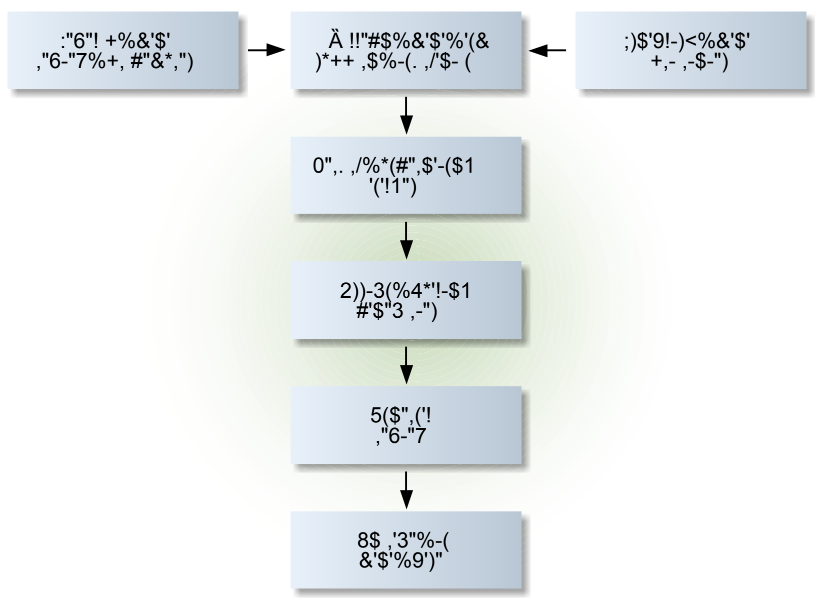

Figure ES-2. Activities completed to establish TMI-2 data base.

Selected TMI-2 Data Qualification Examples

Because there is the potential for similar activities to be completed for qualifying data from Daiichi Units 1, 2, and 3, this report provides additional details related to the formal process used to develop a qualified TMI-2 data base and present data qualification details for three parameters: primary system pressure; containment building temperature; and containment pressure. As described within this report, sensor evaluations and data qualification required implementation of various processes, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors that were similar in design to instruments easily removed from the TMI-2 plant for evaluations. Results from qualifying data for these parameters led to key insights related to TMI-2 accident progression. Hence, these selected examples illustrate the types of activities completed in the TMI-2 data e qualification process and the importance of such a qualification effort.

. Specific insights gained from the TMI-2 sensor and data evaluations include:

- The simultaneous increase in SRM count rate, RCS pressure, and cold leg temperatures provided confidence about the timing of a major relocation of materials from the reactor core to the lower head (see Figure ES-3).

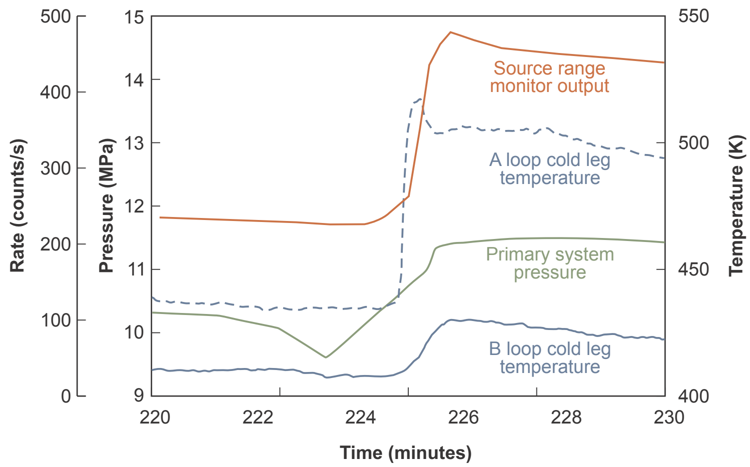

Figure ES-3. Overlay of SRM count rate, RCS pressure measurements, and cold leg temperatures.

- Peak values for containment building temperature would not have been obtained without considering data from other sensors such as the containment building pressure transmitters. Calculations assuming peak containment pressures (see Figure ES-4) yielded peak containment temperatures of 650 °C, which are much higher than the measured 93 °C peak temperature data (Figure ES-5). Recognizing that the TMI-2 containment temperature data had a limited sampling rate, experts qualified the containment pressure and a modified set of containment temperature data.

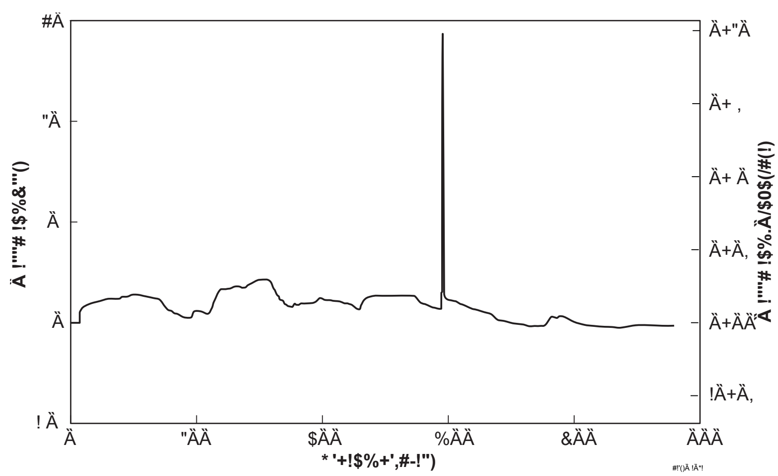

Figure ES-4. Containment building pressure.

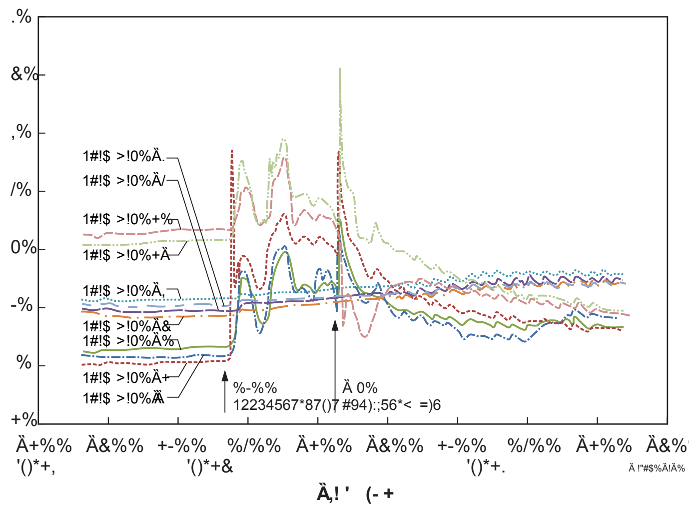

Figure ES-5. Reactor building temperatures at selected locations.

- Data unavailability was often due to computational limits, such as storage memory, inadequate paper or ink, insufficient sampling rates, and ‘preset’ limits associated with anticipated operating ranges (rather than sensor operating limits as illustrated by the pressure data shown in Figure ES-6). A wider range of limits and enhanced computational capabilities, with easy-to-read graphical displays, could alleviate such issues, as occurred with TMI-2 building resistance temperature detector (RTD) temperature and steam generator reference pressure transmitter sampling.

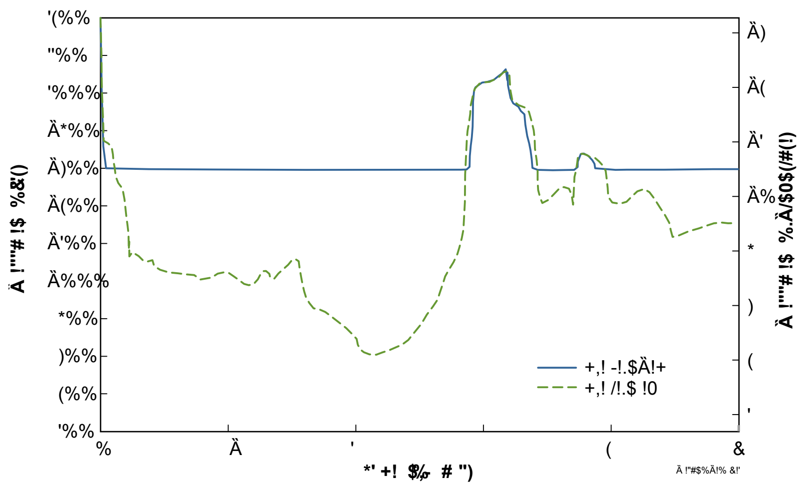

Figure ES-6. Comparison of A-loop wide range pressure recorded on the strip chart and B-loop narrow range pressure on the reactimeter (Note: narrow range pressure data limited to values above 1600 psig).

- Data unavailability was often due to sensor range limitations focused on assumed normal operating conditions. For example, sensors with ranges encompassing unanticipated accident conditions (e.g., at saturated conditions with steam voids present in a PWR) could have provided operators much needed information.
- Data unavailability was also attributed to inadequate status indicators. For example, the inability of the operators to detect that the PORV failed to close could have been rectified by the use of additional indicators and sensors. In this case, indicators were only available to show that the solenoid coil was energized (nothing about the status of the valve position). Since the TMI-2 accident, the US Nuclear Regulatory Commission (NRC) required that licensees make design changes so that positive indication of valve position is available in the control room. However, sensors could also have been included to measure the drain tank water level, which would have provided the operators information that the drain tank relief valve was open. A thorough investigation of other such situations could help alleviate similar occurrences in the future.
- Failures in sensors located in the vessel were often due to a combination of high temperatures and moisture ingress following sheath failure. In some cases, vibrations, moisture, and/or radiation exposure led to failures of sensors. In other cases, such as core exit thermocouples and loose parts monitoring system components, failed sensors continued to provide data that was erroneous. Clearly, it is important to select sensors with extended operating envelopes to consider more likely accident conditions (and to recognize when sensors have been exposed to conditions beyond their operating envelop).
- Failures were often related to transmission component exposure, rather than sensor exposure. For example, cabling and connectors located outside the RCS were subjected to higher than anticipated temperatures, moisture levels, and radiation levels. It was speculated that most of the moisture intru-

sion problems would have occurred eventually in the plant, even barring the accident. However, such limitations could be alleviated by better positioning and enhancements of components and/or shielding.

- Qualitative insights can be obtained by considering sensor response for alternate applications, e.g., ex-core source-range detector signals provide insights about real-time RCS water levels, in-core SPNDs provide insights about RCS temperature and water levels. However, such interpretations often require detailed analyses and assumptions related to the status of the core, the RCS and containment
- No functional damage to the nuclear plant instrumentation or electrical components from thermal effects of the hydrogen burn could be identified. Evaluations indicate that one Geiger-Mueller tube failed at the time of the hydrogen burn, but its failure was deemed to be shock-related, possibly caused by the pressure wave associated with the hydrogen burn.
- Evaluations emphasized the need to consider anticipated applications and more extensive inspection and maintenance programs for instrumentation and related systems. For example, data unavailability or high uncertainties could have been alleviated by the use of better installation and testing procedures with increased calibration checks. Such actions could have alleviated issues observed in dome monitor and RTD components.
- Careful evaluations of sensor data led to unexpected detection of instrumentation errors. For example, comparison of steam generator (SG) water level data led to the conclusion that the full range transmitter was incorrectly installed because readings were observed to be erroneously low when the SG was steaming.
- Surrogate testing of similar sensors and components that were more easily accessible and not required for plant safety monitoring, such as core drain tank water level and pressure measuring system components, allow insights related to instrumentation degradation to be obtained without adversely impacting systems essential to maintaining the TMI-2 plant in a safe condition.

Evaluations emphasized the need for 'applications analyses' to determine possible environments during which the devices must function (or not fail). These “environments” are not limited to just temperature, pressure, humidity (or steam), submersion (flooding), radiation, and vibration (both operational and seismic). They should also include the availability of power sources and the characteristics of supporting services such as instrument air, cooling water, lubrication (allowable contamination levels, moisture), calibration, and preventive maintenance. Such factors are often overlooked details of applications engineering that affect both equipment reliability and the interpretation of information received, as demonstrated at TMI-2.

In summary, a comprehensive set of instrumentation evaluations, that included careful integration of available data, analysis relying on basic engineering principals, operator information, laboratory evaluations, comparisons with accident simulation results and large integral tests, and post-accident inspection, was required for researchers to qualify sensor data for TMI-2 accident simulations. Knowledge gained from these evaluations offered important lessons for the industry with respect to PWR sensor survivability, the need for additional and/or enhanced sensors and indicators, and the identification of unanticipated failure modes for sensors when exposed to extreme accident conditions. A similar process should be followed at Daiichi Units 1, 2, and 3 to reap BWR-specific benefits.

ACRONYMS AND ABBREVIATIONS

AEP	Accident Evaluation Program
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
B&W	Babcock & Wilcox, Inc.
DC	Direct Current
DIRC	Data Integrity Review Committee
DOE-NE	Department of Energy Office of Nuclear Energy
GPU	General Public Utilities
IRM	Intermediate Range Monitors
KPPH	Thousand Pounds Per Hour
LOCA	Loss of Coolant Accident
LPM	Loose Parts Monitoring system
MILS	One Thousandth of an Inch
MPPH	Million Pounds Per Hour
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Science Advisory Committee
OECD	Organization for Economic Development
OTSG	Once-Through Steam Generator
PORV	Pilot Operated Relief Valve
PRM	Power Range Monitor
PRT	Platinum Resistance Thermocouple
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RTD	Resistance Temperature Detector
SG	Steam Generator
SPND	Self Powered Neutron Detector
SRM	Source Range Monitor
TMI-2	Three Mile Island Unit 2

1. INTRODUCTION

The accidents at the Three Mile Island Unit 2 (TMI-2) Pressurized Water Reactor (PWR) and Fukushima Daiichi Units 1, 2, and 3 Boiling Water Reactors (BWRs) demonstrate the critical importance of accurate, relevant, and timely information on the status of reactor systems during a severe accident. Conditions associated with the loss of coolant and the hydrogen burn that occurred during the TMI-2 accident exposed instrumentation[1] to harsh conditions, including direct radiation, radioactive contamination, and high humidity with elevated temperatures and pressures. Numerous insights were gained from the TMI-2 post-accident evaluations. Although there is still some debate about certain aspects of the TMI-2 accident,[3,4] the information obtained from post-accident evaluations and enhanced models provided a basis for improving plant design features, operator training, and accident mitigation strategies. At the time, the TMI-2 reactor was the only source of full-scale severe-accident data for addressing outstanding technical issues related to severe accident phenomena. Insights from TMI-2 post-accident inspections and accident simulations proved invaluable to the nuclear industry and led to significant post-accident safety improvements.

Many post-accident investigations concluded that actions taken by plant operators adversely contributed to the TMI-2 accident. However, the operators' ability to mitigate the accident was impacted by their limited access to accurate plant data. In addition, the ability to improve severe accident analysis codes is dependent on the quality of data used in accident simulations. After the event, an evaluation program was initiated to determine what data were available to the operators and the status of sensors from which such data were obtained. As part of that effort, a formal process was initiated that resulted in a qualified data base for TMI-2 post-accident evaluations. Several techniques were used in the TMI-2 Accident Evaluation Program (AEP) to assess sensor status, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors that were similar in design to instruments easily removed from the TMI-2 plant for evaluations. Reference 2 presents results from a recently completed review to gain insights about TMI-2 sensor survivability and data qualification efforts. The Reference 2 review focused on the set of sensors deemed most important by post-TMI-2 instrumentation evaluation programs. Typically, instrumentation evaluation programs focused on data required by TMI-2 operators to assess the condition of the reactor and containment and the effect of mitigating actions taken by these operators. In addition, prior efforts focused on sensors providing data required for subsequent forensic evaluations and accident simulations.

To encourage the potential for similar activities to be completed for qualifying data from Daiichi Units 1, 2, and 3, this document provides additional details about the process used to develop a qualified TMI-2 data base, that includes a formal uncertainty assessment and technical review, and present details for three parameters that illustrate what activities were completed to assess sensor survivability and obtain qualified data for post-accident simulations. Section 2 of this report provides background information related to the TMI-2 reactor and containment design, available knowledge about the TMI-2 accident progression, and the process used to develop a TMI-2 qualified data base. Sections 3 through 5 provide details related to the data qualification process for three parameters within the TMI-2 reactor vessel and containment building. Insights gained from this review are summarized in Section 6. References for this document are listed in Section 7.

2. BACKGROUND

Numerous insights were gained from the TMI-2 post-accident evaluations. Although there is still some debate about certain aspects of the TMI-2 accident,[3,4] the information obtained from post-accident evaluations and enhanced models provided a basis for improving plant design features, operator training, and accident mitigation strategies. At the time, the TMI-2 reactor was the only source of full-scale severe-accident data for addressing outstanding technical issues related to severe accident phenomena. Insights from TMI-2 proved invaluable to the nuclear industry and led to significant post-accident safety improvements.

As noted in Reference 5, such insights were not available until at least a decade after the event and required an integrated process that included several activities. This integration process, which is schematically shown in Figure 2-1, included information from plant instrumentation, post-accident evaluations and inspections to characterize the reactor endstate, severe accident research results from accident simulations, separate effects laboratory tests, and in some cases, data from large integral tests. Insufficient data were available from any single source to uniquely define a consistent understanding of the TMI-2 accident scenario. Hence, an engineering analysis to interpret and integrate these information sources was crucial

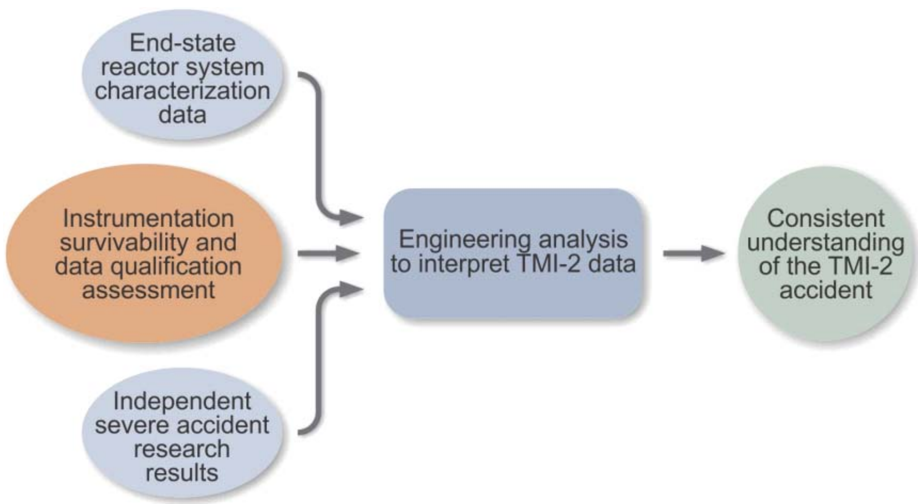

Figure 2-1. Integrated process used to develop TMI-2 accident insights.[5]

In the area of instrumentation, the accident at TMI-2 provided a unique opportunity to evaluate sensors exposed to unusual conditions,[1] i.e., direct radiation, radioactive contamination, moisture, high humidity with elevated temperatures and pressures, and pressure shock waves associated with hydrogen burns. Initially, various evaluation techniques were used to assess the accuracy of available plant instrumentation and improve accident simulations. As additional data from post-accident evaluations became available, accident descriptions were clarified; and accident simulation models were improved.

The remainder of this section provides relevant TMI-2 background. Plant design features are summarized, and significant accident events with representative plant parameters are presented.Then, the process used to develop a qualified data base for TMI-2 accident simulations is presented.

2.1. Plant Description

The TMI-2 power plant contained a PWR that was designed and manufactured by Babcock & Wilcox, Inc. (B&W). The core housed 177 fuel assemblies, corresponding to 93.1 metric tonnes of fuel. Core reactivity was controlled with control rod assemblies containing silver-indium-cadmium alloy and boron dissolved in the coolant. Reactivity was also controlled with burnable poison rod assemblies during the first fuel cycle. As shown in Figure 2-2, the Reactor Coolant System (RCS) consisted of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system was arranged into two heat transport loops, each with two pumps and a steam generator (often designated as the A and B loops).

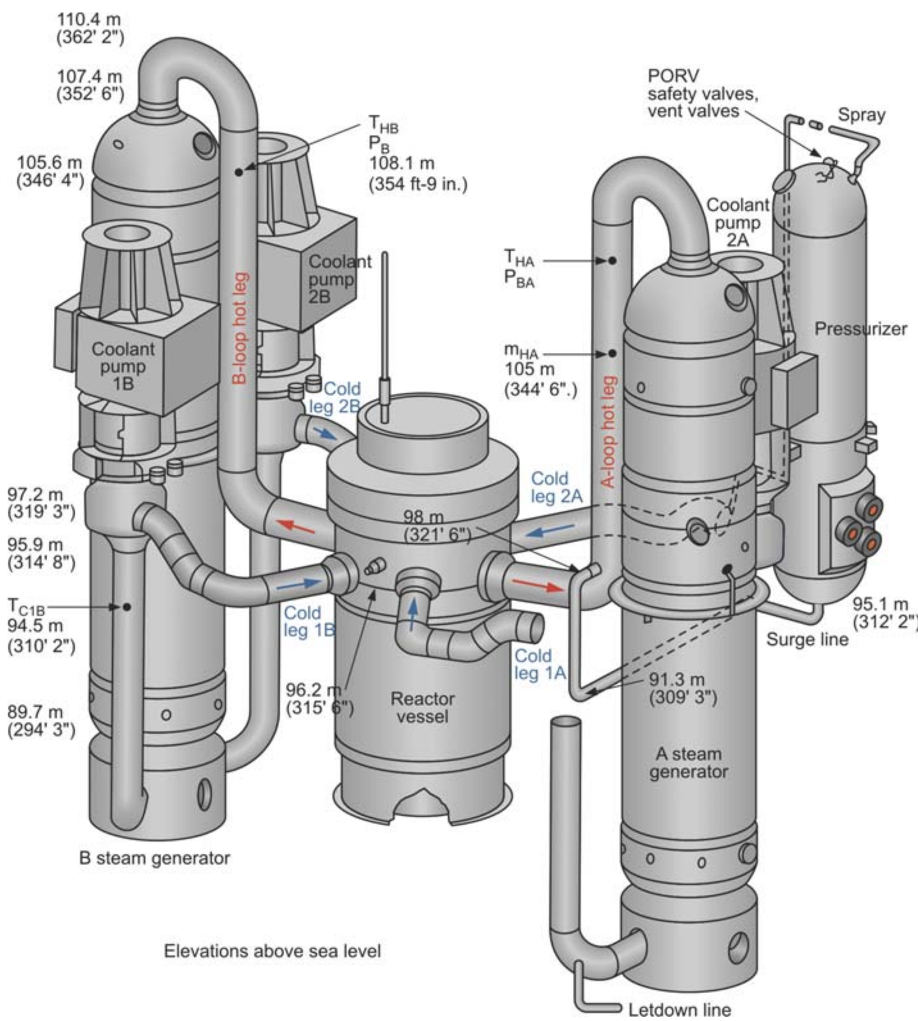

Figure 2-2. TMI-2 primary system layout.[5]

The TMI-2 containment building consists of a large, domed, cylindrical steel shell surrounded by reinforced concrete; the inside diameter and height are approximately 130 ft (40 m) and 190 ft (68 m), respectively. The basement floor is at Elevation 282, the main entry floor at Elevation 305, and the upper floor at Elevation 347 (see Figure 2-3). Plan views at each of the three floor levels and from the dome region at Elevation 450 (approximately) are shown in Figure 2-4.[6]

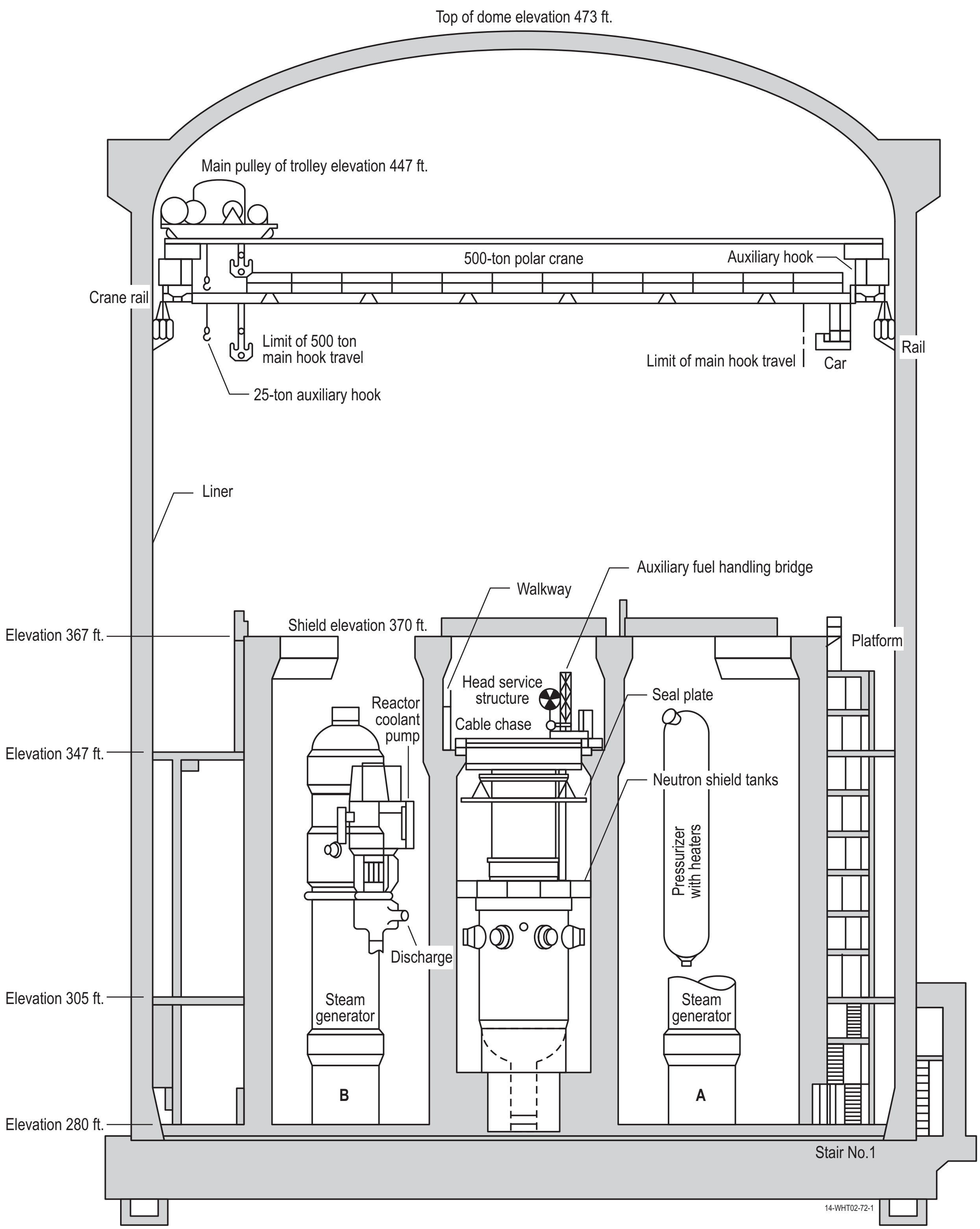

Figure 2-3. Cross section of the TMI-2 Reactor containment building (looking south).[6]

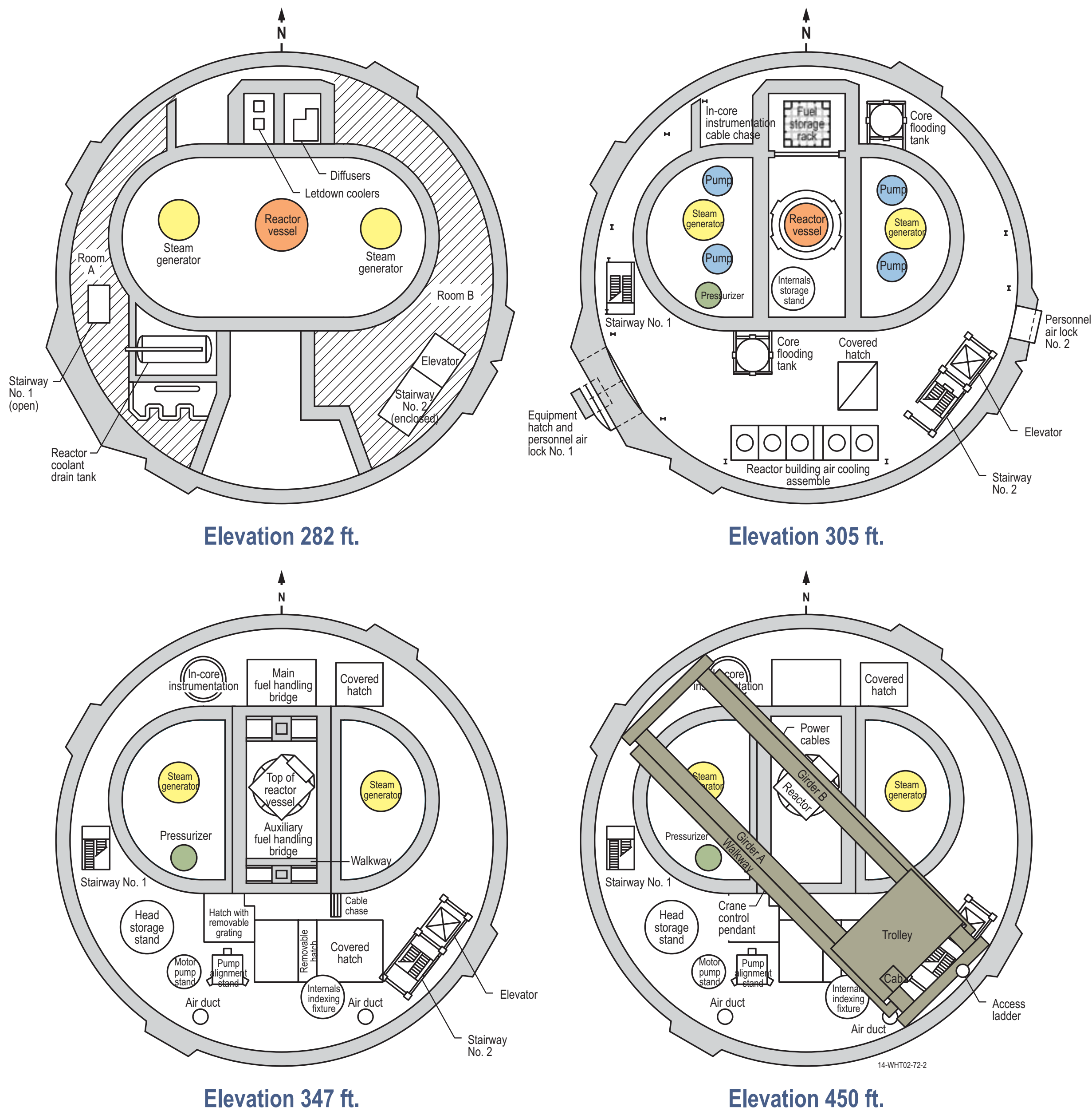

Figure 2-4. Floor plan views of the TMI-2 containment building at selected elevations.[6]

2.2. Synopsis of Accident

Numerous references provide descriptions of the TMI-2 accident sequence.[5, 7 through 12] Such descriptions were informed and updated as TMI-2 AEP[13 through 15] results became available. The scenario defined at the end of the TMI-2 post-accident examinations and selected ‘recommended’ data characterizing plant response are presented in this section. Many details pertaining to the core heatup and relocation scenario could only be obtained from post-accident examinations and testing. Likewise, as discussed in Section 2.3, instrumentation data to characterize the plant response could only be qualified after detailed evaluations were completed.

The TMI-2 accident started at about 4 a.m. on March 28, 1979. During attempts to unclog a pipe leading from the full-flow demineralizers downstream of the condenser, the accident was initiated by a shutdown of secondary feedwater flow due to a trip of the condensate booster pumps followed by a trip of the feedwater pumps. Best estimates for plant data and events during the accident, as obtained from various post-accident evaluation programs,[8] are depicted in Figure 2-5. Following turbine isolation (defined as time zero in Figure 2-5) and reactor trip (when reactor pressure reached 16.3 MPa at 10 seconds after turbine trip), the steam generator boiled dry; and the resultant reduction of primary-to-secondary heat exchange caused the primary coolant to heat up, surge into the pressurizer, and increase the primary system pressure. The Pilot Operated Relief Valve (PORV) opened to relieve pressure when the RCS pressure reached 15.7 MPa.[5] However, the PORV failed to close when RCS pressure decreased. The first 100 minutes of the accident can therefore be characterized as a small break loss-of-coolant accident (LOCA) (through the PORV) with a corresponding decrease in RCS inventory and pressure. The event differed from a typical small break LOCA in that the pressurizer liquid level remained high. This was incorrectly interpreted by the reactor operators[5,8] as indicating that the RCS was nearly full of water, when in fact, the RCS was continually losing its water inventory. Emergency core cooling was reduced by operators to address their concerns about a full RCS. However, the coolant void fraction increased due to coolant loss through the PORV and decay heat generation in the fuel. The steam inventory in the primary system piping increased to such an extent that RCS pumps were tripped by the operators to prevent permanent damage from pump cavitation after 100 minutes.[5,8]

At the time that pump operation ceased (see Figure 2-5), increases in Source Range Monitor (SRM) count rate and coolant system temperature and pressure, suggest that the reactor vessel liquid level had decreased. Studies correlating the response of the SRMs with the core liquid level suggest that core uncovery began between 114 and 120 minutes and that the liquid level had dropped to the core midplane by approximately 140 minutes.[8,9] Insufficient decay heat removal associated with core uncovery is estimated to have led to upper regions of the core heating to temperatures that caused the cladding to overheat, balloon, and rupture.[9,16] Such cladding failure, which results in the release of gaseous fission products, was substantiated by significant increases in containment radiation levels at 140 minutes. When operators finally realized the PORV failed in the open position, they closed the pressurizer block valve upstream of the PORV, terminating coolant loss and the release of fission products to the containment.

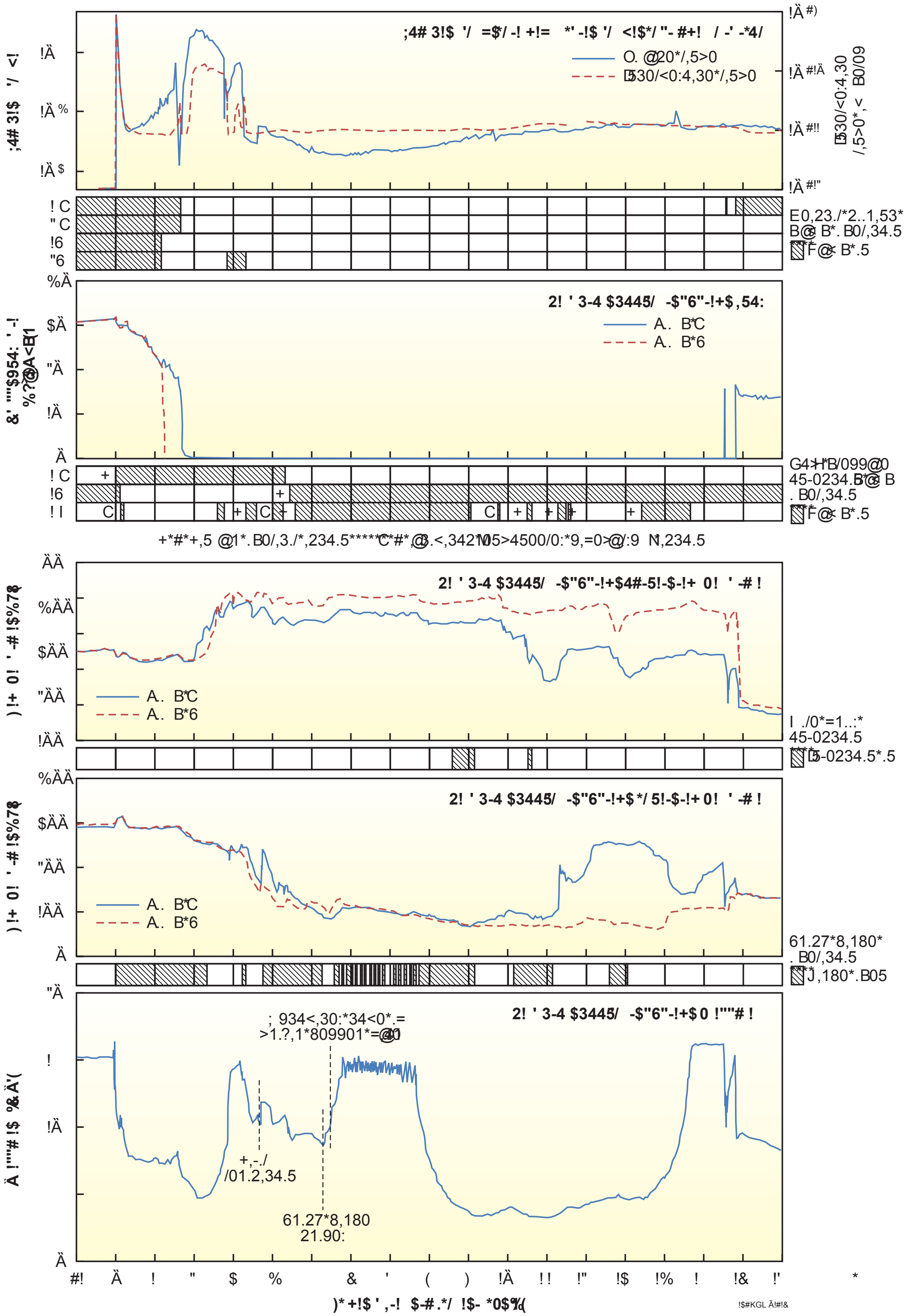

Figure 2-5. TMI-2 data from March 28, 1979.[8]

In-core self powered neutron detector (SPND) output and RCS pressure data (see Figure 2-5) indicate that core temperatures continued to increase between 150 and 165 minutes. Subsequent analysis of the SPND output indicated temperatures probably reached 1077 °C.[17] Insights gained from materials interaction and severe accident testing (e.g., see References 18 and 19) suggest that a Zircaloy-steam exothermic reaction was initiated, producing large amounts of hydrogen and dramatically increasing the core heatup rate. Zircaloy melting temperatures were exceeded, resulting in relocation of the molten Zircaloy and some liquefied fuel to the lower core regions, solidifying near the coolant interface. This continued until 174 minutes, when a dense agglomeration of degraded core material formed in the lower regions of the core and blocked core flow.

At 174 minutes, one of the reactor coolant pumps in the B-loop was turned on for approximately 19 minutes, and coolant was pumped into the reactor vessel. This coolant injection rapidly repressurized the RCS. Core exit thermocouples above peripheral fuel assemblies indicate cooling occurred, and SRM count rate decreased at the time of this injection (see Figure 2-5). Several references[5,9,20] indicate that the thermal-mechanical forces resulting from this injection and follow-on rapid steam formation may have shattered the oxidized fuel rod remnants in the upper regions of the core, forming a rubble bed on top of the consolidated core materials. At 200 minutes, the high pressure injection system was operated for 17 minutes. The reactor vessel was refilled with water by approximately 207 minutes.

2.3. TMI-2 Data Evaluation Process

The ability to improve severe accident analysis code models is dependent on the quality of data used in accident simulations. After the event, an evaluation program was initiated to determine what data were available to the operators and the status of sensors from which such data were obtained As part of that effort, a process was initiated that resulted in a qualified data base for TMI-2 post-accident evaluations. This section provides an overview of the process completed in the TMI-2 AEP to develop and qualify TMI-2 data for accident simulations.

2.3.1. Qualified Database Establishment

An important aspect of the TMI-2 AEP[13 through 15] was to provide a qualified data base for an analysis based on the TMI-2 accident, known as the "TMI-2 Analysis Exercise." This analysis exercise was completed to assess the accuracy of modeling tools, which were based upon data from small-scale experiments applied to a full scale PWR.[15] Understanding gained from this analysis exercise was ultimately applied toward improving phenomenological models related to the chemical and materials interactions that occurred in the TMI-2 core and resolving applicable severe accident and source term issues. By resolving such technical issues, the analysis exercise contributed toward establishing a sound technical basis for post-TMI-2 regulatory actions.

A TMI-2 Initial and Boundary Conditions Data Base was established at around 6 years after the TMI-2 accident to provide a qualified database for this analysis exercise. This measurement data was the sole source of data used by all participants in the TMI-2 Analysis Exercise. A data qualification process (see Figure 2-6) was developed that included: collecting the TMI-2 measurement data and support information; establishing priorities and designing a formal approach for systematically performing the uncertainty analyses; and establishing quality categories of the data. Because of the limited time available for completing the data qualification process, the three items on the top of the Figure 2-6 chart were performed at approximately the same time, i.e., develop procedures, collect data, and establish priorities. It was necessary to define a methodology for performing the uncertainty analyses on the data, the criteria for determining the data quality categories, and the internal review process. These procedures were prepared at the beginning of the program and were modified as the realities of the task dictated. Prior to being entered into the database, the data and uncertainties were reviewed by a Data Integrity Review Committee (DIRC), which was composed of a panel of experienced persons knowledgeable in TMI-2 data analysis. The DIRC reviewed available information, including analyses, evaluations, and comparisons, to ensure data met established criteria.[21]

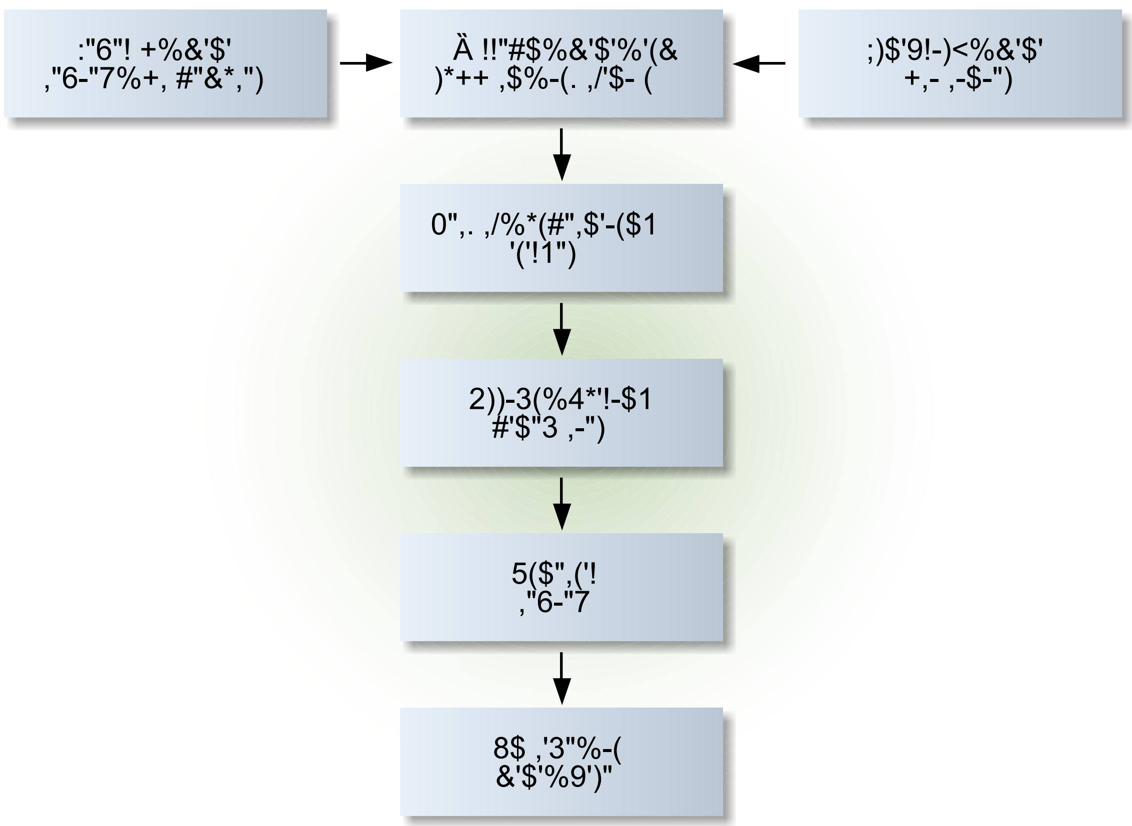

Figure 2-6. Activities completed to establish TMI-2 data base.[21]

Researchers performing TMI-2 data uncertainty analyses had to request most required reference materials, such as instrument calibrations, manufacturers specifications, operating manuals, etc. For most data, it was not possible to obtain the quantity and types of information necessary to do classical uncertainty analyses and data qualification. Hence, the uncertainty methodology was simplified to accommodate the information available, resulting in increased use of engineering judgment and consideration of the following factors:

- Consistency with respect to single channel analysis criteria (range, noise limits, time response, and correlation with the significant plant events and prior history);
- Agreement with other redundant information; and
- Agreement with thermal-hydraulic theory.

The DIRC also reviewed any underlying assumptions required to obtain the data. Prior to inclusion into the TMI-2 Initial and Boundary Conditions Data Base, the DIRC also approved qualification levels and uncertainty assigned to each set of data.

Basic information on the instrument systems was generally available, i.e., items such as system diagrams and manufacturer instruction sheets. In addition, analysts developing the TMI-2 data base usually knew and understood how the measurement systems worked. The major shortcoming was calibration information on the measurement systems and components, especially near the accident date. It was often impossible to find detailed information on items such as the exact location of a transducer or the functional details of a particular circuit. The lack of support information was an important contributor in the simplified uncertainty analysis methodology chosen.

2.3.1.1. Data Storage and Archival

Data at TMI-2 were recorded on computer print outs, magnetic tapes, and analog stripcharts. After the accident, all such records were impounded, and access was only allowed for copying or study. Sources of data included microfilm, photographs, and microfiche of the hard copy data and copies of magnetic tapes. Enlarged color photographs of selected multipoint recorder data and support documentation to be used in the uncertainty analyses (instrument calibrations, circuit diagrams, operating manuals, etc.) were also obtained to the extent possible. Computational systems where data were displayed and stored are highlighted in this section.

Reactimeter. Much of the available plant data recorded during the accident were stored on the plant "reactimeter," a 24 channel data acquisition system. Its name derives from its capability to record core reactivity data, which were normally used during reactor start-up testing. However, the reactimeter also recorded other data, such as pressurizer pressure, hot and cold leg temperatures, loop A and B coolant flow, etc. The 24 channels of data were recorded on magnetic tape in the form of voltage readings. These voltages were directly proportional to the parameters being monitored, e.g., pressure, temperature, and flow. Table 2-1 lists the 24 parameters recorded by the reactimeter at the time of the accident. Positions for some of the sensors monitored by the reactimeter are shown in Figure 2-7.

The reactimeter could sample each channel on any time interval from 0.2 second to 12.6 seconds. During the accident, the reactimeter was set to sample each channel on a 3 second interval, that is, it sampled all 24 channels once every three seconds. The availability of data from the reactimeter, which could be displayed in tables and graphs, made it a valuable resource for analyzing the TMI-2 accident.

Table 2-1. Parameters recorded on reactimeter.

Channel	Parameter (Range)
1	Power range level--nuclear instrument-5 (0-125%)
2	Loop A hot leg temperature--narrow range (520-620 °F)
3	Loop B hot leg temperature--narrow range (520-620 °F)
4	Loop A cold leg temperature--wide range (50-620 °F)
5	Loop B cold leg temperature--wide range (50-620 °F)
6	Loop A reactor coolant flow--temperature compensated; [0-90 million pounds per hour (MPPH)]
7	Pressurizer level--temperature compensated (0-400 in.)
8	Makeup tank level (0-100 inches)
9	Pressurizer spray valve position (open-closed)
10	Drain tank pressure (0-250 psig)
11	Loop B reactor coolant pressure--narrow range (1700-2500 psig)
12	Reactor trip (run-trip)
13	Loop B reactor coolant flow--temperature compensated [0-90 MPPH]
14	Feedwater temperature (0-500 °F)
15	Turbine header pressure--Loop A (600-1200 psig)
16	Steam generator A operate level (0-100%)
17	Steam generator A start-up level (0-250 in.)
18	Feedwater flow--Loop A[0-6500 thousand pounds per hour (KPPH)]
19	Feedwater flow--Loop B [0-6500 KPPH]
20	Turbine trip (run-trip)
21	Steam generator A steam pressure (0-1200 psig)
22	Steam generator B steam pressure (0-1200 psig)
23	Steam generator B operate level (0-100%)
24	Steam generator B start-up level (0-250 in.)

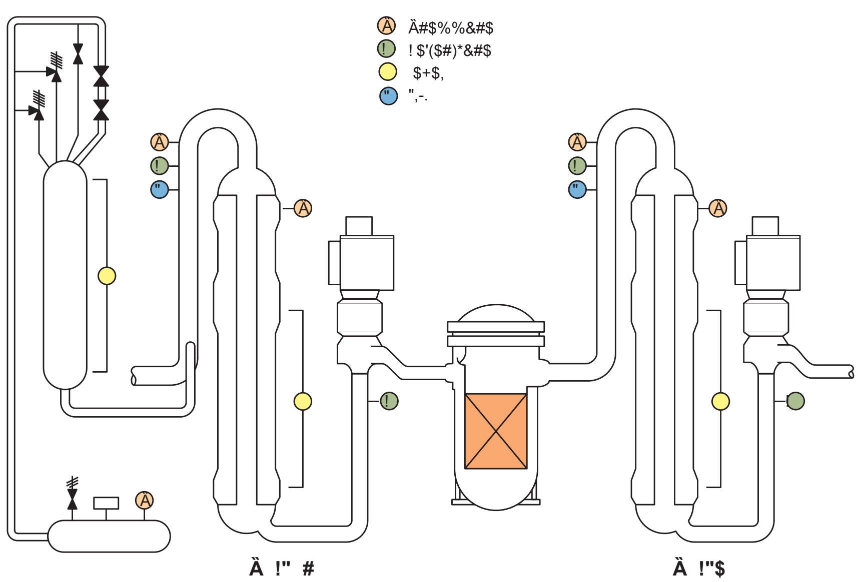

Figure 2-7. Location of selected sensors monitored by the reactimeter.

Stripchart Data. Data were also available on stripcharts at TMI-2. Unlike data stored on computers and magnetic tapes that could easily be extracted, stripchart data had to be digitized. Such digitalization was generally done on an apparatus which transferred the plot coordinates directly into the computer.

Some important parameters, such as the wide-range primary system pressure, containment pressure, and source and intermediate range neutron detector signals, were not available from the reactimeter. However, values for these and other primary and secondary plant parameters (see Table 2-2), were continuously recorded on stripchart recorders located in the control room. These recorders allowed the operators to observe and create historical records of trends in monitored parameters. There were basically two types of recorders used in the control room: pen recorders, which employed an ink pen to produce a continuous line plot of a parameter's value, and a multipoint recorder, which monitored several parameters and printed a code number identifying each parameter at a location on a strip chart representing the parameter's value.

Legibility was the biggest problem encountered in trying to extract information from the strip charts. This was especially true for the multipoint recorders when several parameter values were printed on top of each other and were difficult to read. The problem of legibility was compounded by the slow speed at which the strip chart traveled (e.g., typically 1 inch/hour for the pen plotters) and the large amount of data compressed onto them. In several cases, stripchart recorders had paper jams that were not corrected for several hours. In other instances, the measurement systems were over ranged by abnormally large input signals which caused the electrical systems to saturate. In addition, timing errors were introduced if the strip charts were not properly annotated when removed from the recorder.

Comparisons between stripchart and reactimeter values indicated strip chart data were generally less accurate. However, stripchart data were found to be a good source of trend information. The stripcharts were calibrated periodically and had acceptable accuracy for most purposes--especially as a source of trend information.

TMI-2 Computer System. The TMI-2 plant computer system was an additional source of information. The principal function of the computer system was to monitor plant parameters (approximately 3000) and to display them along with any related calculations.[5,21] The only permanent computer system record of instrumentation data was "hard-copy" from the utility printer (which only writes data if requested), the alarm printer (which writes data when an unusual occurrence happens such as a parameter exceeding an alarm setpoint or changing state), and the periodic log data (which were automatically printed out every hour and annotated to the minute).

The alarm inputs were stored by the computer in an alarm-backup-buffer until they were printed. This buffer could store up to 1365 alarm inputs. The alarm printer could only print one alarm every 4.2 seconds. If alarms were occurring at a faster rate, the printer got behind. At one point during the TMI-2 accident, the alarm printer was at least 161 minutes behind. After the buffer was filled (i.e., 1365 alarms were waiting to be printed), the computer program was designed to print the message "Alarm Monitor Holdup" indicating future alarms would not be stored until some of the 1365 backlogged alarms were printed. The operators had the option of suppressing the alarm sequence. This erased all prior alarms from the computer memory and caused it to start printing new alarms which originated after the suppression. At 167 minutes into the accident, the operators exercised this option in order to obtain current information (at this time, there was a data queue that would have taken 93 minutes to print). Because the operators needed timely information, they erased the memory buffer (destroying all alarm data between 74 and 167 minutes). This time period unfortunately corresponded to the time when the initial core heatup and uncovery occurred. The decision to exercise this option eliminated the possibility of printing backlogged and unstored alarms.

Table 2-2. Parameters recorded on strip charts.

Parameter, units
Reactor building pressure, psig
Makeup tank level, inches
Pressurizer level, in.
Loop A wide range pressure, psig.
Loop A narrow range pressure, psig.
Loop B narrow range pressure, psig.
Reactor coolant outlet temperature, °F
Reactor coolant average temperature, °F
Source and intermediate range power level, counts/second and amp
Intermediate range power level, amp
Power range level, percent
Selected turbine header pressure, psig
Steam generator A operate level, in.
Steam generator B operate level, in.
Steam generator A feedwater flow, KPPH
Steam generator B feedwater flow, KPPH
Steam generator A and B operate level, in.
Liquid waste discharge ΔT above river temperature, °F
Cooling tower makeup water flow, gpm
Reactor coolant pumps seal cavity pressure, psig.
Transfer flow from reactor coolant drain tank, gpm
Wind speed and direction, MPH and Degree
Outside air temperature and ΔT at different elevations, °F
Control rod drive motor temperature, °F
Valve stem leakage thermocouples, °F
Turbine generator temperatures, °F
Reactor coolant and SG temperature, °F
Radiation monitoring system, MR/HR and counts/min
Main turbine governor valve position, percent
Main turbine vibration, mils (one thousandth of an inch)
Main turbine casing temperatures, °F
Condenser vacuum, in HG (inches mercury)
Condenser circulating water temperature, °F
Main feedwater pumps speed and turbine governor valve position, rpm and percent
Main feedwater pump turbine vibration, mils
Main turbine header pressure, psig
Reactor building temperature, °F
Reactor building ventilation flows, ft^3/min
Auxiliary building exhaust ventilation, ft^3/min
Auxiliary building supply ventilation, ft^3/min
Fuel handling building exhaust ventilation, ft^3/min
Fuel handling building supply ventilation, ft^3/min
Control building ventilation flows, ft^3/min
Auxiliary building temperatures, °F
Fuel handling building temperatures, °F
Control building temperatures, °F

The utility printer provided output on request. The value or condition of any monitored parameter could be requested. The computer was also programmed to record automatically all changes in state of a predesignated group of parameters called “Sequence of Events” inputs. These event inputs were stored in the computer and could be printed on request. This particular computer function did not use the scan process described above, but used a continuous monitoring process which enabled it to print the exact time the “Sequence of Events” inputs occurred. Another feature programmed into the computer was the “Memory Trip Review.” Triggered by a reactor or turbine trip, this routine obtained a set of predesignated parameter inputs for 15 minutes before and 15 minutes after the trip and stored the data until the operator requested it be printed.

The plant computer provided the operator with an efficient means of keeping logs and showing trends on a large number of plant parameters under normal operating conditions. The computer was not designed to accommodate the data needs of the operator in an accident situation. Using the computer in an accident situation required the operator to leave his control panels in order to request computer output. It took the computer several seconds to supply the requested output; and the automatic alarm printout was often several minutes behind real time. All of these tended to limit the computer's usefulness in an accident situation.

2.3.2. Sensor Evaluations

Sensors were available to allow approximately 3000 measurements to be made at TMI-2. As documented in Reference 21, the DIRC identified 300 measurements of interest and developed a list of 170 measurements that were prioritized based on their ability to provide data required for subsequent accident simulations. As documented in Reference 21, only about half of these measurements were actually evaluated and included in the database at the completion of the TMI-2 data evaluation effort. In particular, the DIRC focussed upon:

- RCS temperature and pressure
- RCS coolant flow rates
- RCS makeup and letdown flow rates
- Operation periods of ECC injection
- Operation periods of PORV and block valves
- Containment radiation levels
- Source and intermediate-range detector data

Data were extracted and stored on a main frame computer. This process was accurate for computer printouts and magnetic tapes. Stripchart data, however, had to be digitized, which was generally done on an apparatus which transferred the plot coordinates directly into the computer.

The Data Review Task required not only the instrument measurement data recorded at TMI-2 but a great deal of supporting information on the reactor system and facility physical configurations. The sources of data and support information were documents ranging from formal reports to microfiche. A TMI-2 library was set up to store relevant reports, papers, drawings, etc. Included were over 100 microfilms and four books of microfiche. The microfilm contained most of the stripchart data used in this task as well as copies of computer printouts. The support information consisted of items such as design specifications, instrument calibration sheets, manufacturers equipment instruction sheets, drawings, and schematics. Often, additional support information needed for an analysis had to be requested from TMI-2 after an analysis was started.

Basic information on instrumentation systems, such as system diagrams and manufacturer instruction sheets, was generally available. In addition, analysts contributing to the data base usually knew and understood how the measurement systems worked. The major information lacking was calibration information on the measurement systems and components, especially near the accident date. Furthermore, it was often impossible to find detailed information on items such as the exact location of a transducer or the functional details of a particular circuit.

2.3.2.1. Data Uncertainty

Before completing any uncertainty analysis or quality evaluations, data were inspected. This preliminary examination was completed to detect such things as offsets, enigmas, and gross inconsistencies. Then, a value judgment was made regarding the uncertainty in components which make up the measurement circuit. For example, if there are no calibrations available for a stripchart recorder and no redundant data for comparison, the uncertainty in that data might be judged to be extremely large or impossible to calculate. In such cases, it might be concluded that an uncertainty analysis would be inappropriately expensive for a particular measurement and an alternate measurement should be sought.

In the TMI-2 data qualification effort, uncertainty was defined as the maximum probable amount of error in the data. Where possible, documentation (e.g., instrument calibrations, manufacturers specifications, operating manuals, etc.) for the uncertainty analyses was obtained. Most error information came from calibration sheets, which only listed the error tolerance of the circuits. Hence, most errors were treated as bias errors as a function of instrument range. In cases where bias errors were not applicable, conservative substitutions were made.

A rigorous calculation of confidence level or the coverage of the true value by the interval was not possible because the distribution of bias errors and limits could not be rigorously defined. Hence, a simplified uncertainty analysis approach was selected that was commensurate with the information available and that increased the reliance on engineering estimates and expert judgment. This simplified approach, which was derived from a method advocated by Abernathy,[22] was deemed to be conservative and simplified the uncertainty analyses and results presentation. Monte Carlo simulation studies were completed assuming various bias error distributions and bias limits. The result of these studies indicated that the uncertainty methodology used in the TMI-2 data analysis gave a reasonably accurate confidence level of 95% that the true value of a measurement would fall within the uncertainty interval.

The following two equations were the basis for the TMI-2 uncertainty analyses. The first is the basic equation for determining the uncertainty in a measurement, and the second is for a calculated parameter with three independent variables:

$$B = \left[\sum_{i=1}^{n} b_i^2\right]^{1/2} \tag{2-1}$$

$$B_w^2 = \left(\frac{\partial w}{\partial x}B_x\right)^2 + \left(\frac{\partial w}{\partial y}B_y\right)^2 + \left(\frac{\partial w}{\partial z}B_z\right)^2 \tag{2-2}$$

where

w = $f(x,y,z)$, a generic function of three independent variables

B = measurement uncertainty or error bounds

b = elemental measurement error

As noted above, the uncertainty analyses provided the error bounds for the data, i.e., the maximum and minimum probable error with 95% confidence. Because of the simplified approach taken, uncertainties were generally expressed as a symmetrical value which applied to the data at all times. In only a few cases were uncertainties calculated which had to be expressed as functions of time or data amplitude. Such cases were due to the nature of the data and increasing uncertainties in other parameters (for example, the uncertainties in mass flowrate due to increased voiding in the coolant).[21]

2.3.2.2. Sensor Data Qualification and Review

After the uncertainty was established for data, a quality category or label was assigned to each data set to generically describe its relative quality. This allowed users of the data to gain some concept of the data quality without referring to the uncertainty values in the data base.

All data were given a quality category of either "Qualified", "Trend", or "Failed". Data were deemed "Qualified' for cases where the data had reasonably sized uncertainty and were well-behaved. Data were deemed "Trend" for cases where data were deemed to contain some useful information but uncertainties were unreasonably large and data only approximated the phenomenon being measured. If it was not possible to do an uncertainty analysis on the data, it was automatically categorized as "Trend". "Failed" data contained no useful information and were not retained in the database. In addition, the terms, "Composite", "Computed", and "Estimated" were applied to data. The term "Composite" data indicated that the data set was composed from two or more sources. For example, data might be from a stripchart recorder for one time period and from the reactimeter for another time period. "Computed data" indicates that the parameter was calculated using measured data. "Computed" data could be classified as either "Qualified" or "Trend" data. An "Estimated" data set was calculated using assumptions and little real data. "Estimated" data were only classified as "Trend" data.

After the data had been categorized, the DIRC reviewed the data, the uncertainty analysis, and any underlying assumptions made before data were put into the TMI-2 Initial and Boundary Conditions Data Base. The size and composition of the DIRC review varied according to the type of data being reviewed, but generally consisted of members of the TMI-2 AEP. For some specific measurements, outside experts participated in the DIRC. Generally, the analyst who performed the data evaluation presented their work to the DIRC for approval. Very often it was necessary for the analyst to try several times before getting DIRC approval of their analysis and assigned quality category. After DIRC approval, the data were put into the dedicated TMI-2 Initial and Boundary Conditions Data Base.

2.4. Summary

Clearly, instrumentation data were required to understand the TMI-2 accident scenario. It is essential to have confidence in data during such accidents and in post-accident evaluations of such accidents. The curves presented in Figure 2-5 were only obtained after extensive studies were completed. In addition, the accident at TMI-2 provided a unique opportunity to evaluate instrumentation exposed to unusual conditions, i.e., direct radiation, radioactive contamination, moisture, and high humidity with elevated temperatures and pressures. As discussed in Reference 2, initial actions proposed by the regulators related to enhanced instrumentation were informed by examinations of sensors removed from TMI-2 and efforts to qualify data obtained from TMI-2 sensors.

In Sections 3 through 5 of this report, three parameters, that are a subset of the 170 higher priority measurements identified by the DIRC, are discussed in detail to provide insights related to activities completed in the data qualification process and the importance of such a qualification effort. These parameters are: primary system pressure (Section 3), containment building temperature (Section 4), and containment pressure (Section 5). As noted within these sections, several techniques were used to assess sensor status, including comparisons with data from other sensors, analytical calculations, laboratory testing, and comparisons with sensors subjected to similar conditions in large-scale integral tests and with sensors similar in design that are easily removed from the TMI-2 plant for evaluations. Results from qualifying data for these parameters led to key insights related to TMI-2 accident progression. Hence, the selected examples are not only useful for illustrating the process used in the TMI-2 data qualification efforts but are also useful for illustrating the importance of completing this qualification process.

3. PRIMARY SYSTEM PRESSURE

One essential parameter required for thermal-hydraulic analysis of the TMI-2 accident is primary system pressure. This pressure is required for input to computer code predictions of the TMI-2 accident and to obtain the phase properties of the fluid in the analysis. No single data source was available for estimating the pressure during the entire TMI-2 accident sequence. Hence, as described within this section, a composite of various data sources was used to represent the primary system pressure.

3.1. Instrumentation System Description

There are two penetrations for measuring the system pressure in each hot leg of the TMI-2 reactor. These penetrations are at an elevation of approximately 108 meters (separated by 90 degrees), and their locations are shown in an isometric of the TMI-2 system in Figure 2-2 (RC-P_A and RC-P_B). Connected to each of these penetrations (through 1.3 cm sensing lines) was a pressure transmitter. The narrow range pressure transmitter was a Rosemount model 1152GP (Figure 3-1) variable capacitance pressure transmitter (output 4-20 mA DC) and was set up for a 11.7 to 17.2 MPa-gauge measurement range.* The two narrow range transmitters in each loop were identified as: RC-3A-PT1 and RC-3A-PT2 for the A loop; and RC-3B-PT1 and RC-3B-PT2 for the B-loop. The other transmitter type connected to each sense line was a Foxboro model E11GH Bourdon tube/electronic force balance pressure transmitter (Figure 3-2) with an output of 10-50 mA DC and a measurement range of 0 to 17.2 MPa-gauge. These transmitters, which were referred to as the wide range pressure transmitters. were identified as: RC-3A-PT3 and RC-3A-PT4 for the A loop and RC-3B-PT3 and RC-3B-PT4 for the B loop.

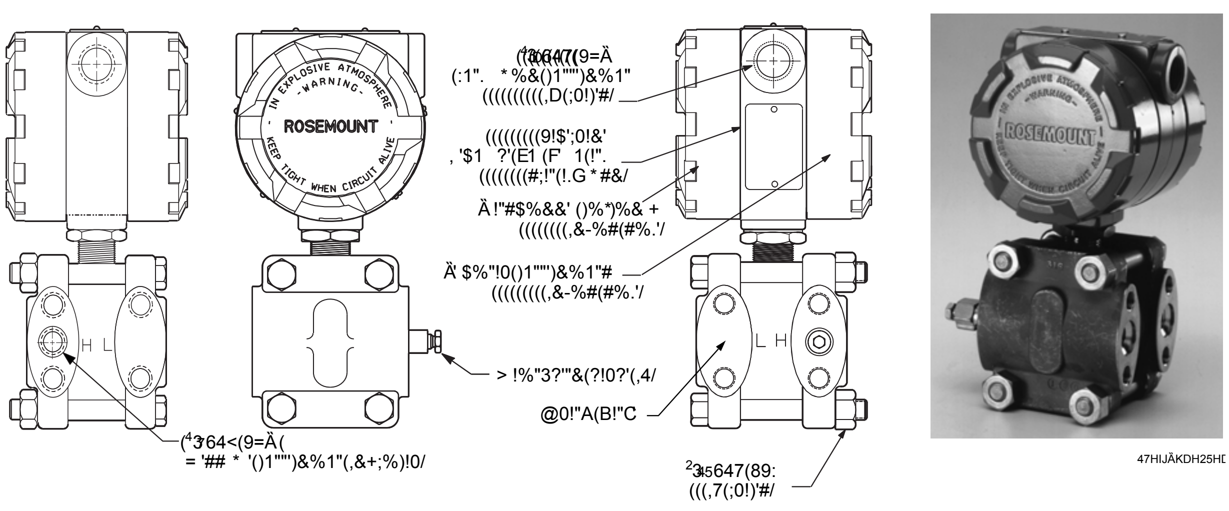

Figure 3-1. Rosemount Model 1152 GP pressure transmitter.

* Although the narrow range measurement was set-up for a range of 11.7 to 17.2 MPa-gauge, the measurement continued to produce readings slightly below 11.0 MPa-gauge. Therefore, the reactimeter data down to 11.0 MPa-gauge was used in the composite pressure.

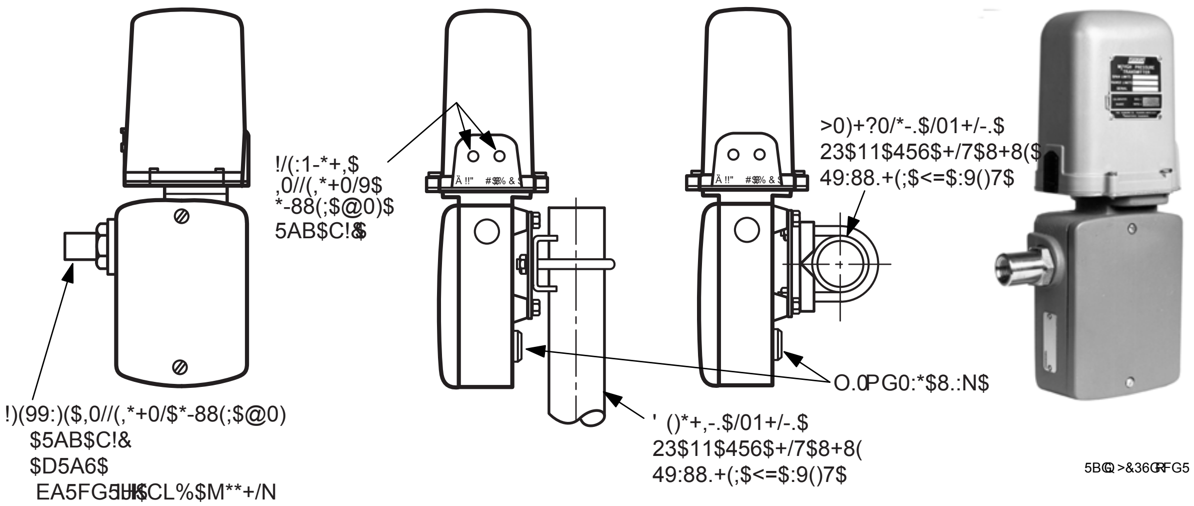

Figure 3-2. Foxboro Model E11GH pressure transmitter.

3.2. Data

There were several systems within the TMI-2 reactor for measuring and recording primary system pressure. Output from one of the narrow range pressure transmitters in the B-loop (RC-3B-PT1-R)[*] was recorded on the reactimeter at a sample rate of one sample every 3 seconds. A block diagram for this measurement is shown in Figure 3-3. These data were considered to be the best available TMI-2 primary system pressure data.[21] Following the reactor trip, the primary system pressure quickly dropped below the minimum range for this transmitter (by 2.2 minutes). With the exception of certain periods in which the system pressure increased to within the range of this transmitter (approximately 2.8 hours), other data sources were required for obtaining the primary system pressure.

Output from one of the wide range pressure transmitters installed in the A-loop (RC-3A-PT3) was recorded on the utility printer for two significant time periods. The first period was from -15 minutes to +15 minutes of the turbine trip, which was recorded on the utility printer as the Memory Trip Review (see Section 2.3.1.1). The second time period started at 570 minutes and continued throughout the remainder of the first day of the accident. The data were recorded on the utility printer as operator group trend C (e.g., data were recorded once every 2 minutes) Output from RC-3A-PT3 was also recorded on a strip chart mounted on one of the operators control panels (strip chart # 59). As discussed in Section 2.3.1.1, strip chart data were considered to be the least accurate data available and only used when no other data were available. Adjustment of this data was required to match the initial pressure and event timing in comparison to the reactimeter data. A block diagram of this measurement system is provided in Figure 3-4.

Knowledge of the thermal-hydraulic conditions in the reactor system (during the first 100 minutes) allowed researchers to also estimate primary system pressure from measured hot leg temperatures. By

* The TMI-2 AEP uses the basic measurement identifications originally assigned by GPU. However, a suffix is typically added which identifies the recording device. For example; -R is added for measurements recorded on the Reactimeter; -S is added for measurements recorded on Strip charts; and -P is added for measurements recorded on either the utility or alarm printers.

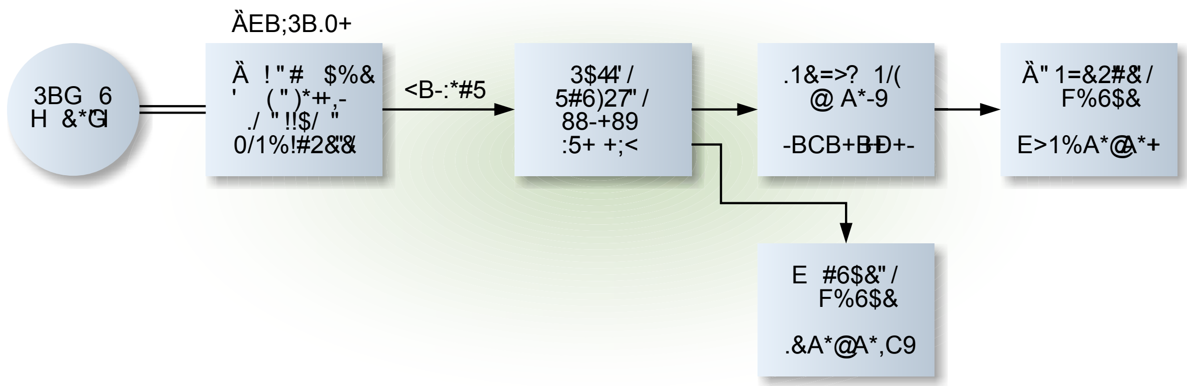

Figure 3-3. Narrow range reactor coolant pressure measurement block diagram.

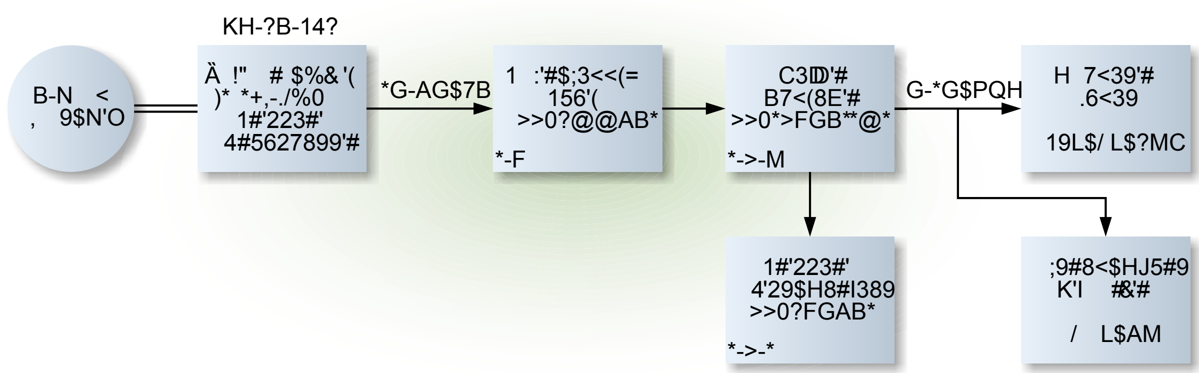

Figure 3-4. Wide range reactor coolant pressure measurement block diagram.

6 minutes into the accident, computer simulations indicated that the system had depressurized to the point where a two-phase mixture was exiting the core and flowing through the entire primary system (both steam generators had boiled dry by this time). This calculation result is supported by increased count rate measurements from source range neutron detectors within 20 minutes after the beginning of the accident (see Figure 3-5). During the period in which a two-phase mixture was flowing through the system, the system pressure had to have been at saturation pressure. The saturation pressure was obtained from the steam tables using the measured hot leg temperature which was recorded on the reactimeter (RC-4A-TE1-R).

The initial system pressure prior to the start of the accident was 14.91 MPa (2148 psig). Plots of primary system pressure data from various sources are compared in Figures 3-6 through 3-9. These data sources were used to create a best estimate composite primary system pressure. The time segments over which each data source were used are listed in Table 3-1. The composite primary system pressure is shown in Figure 3-10 for the first 300 minutes of the accident. This composite pressure is compared with the wide range pressure recorded on the utility printer memory trip review in Figure 3-11.

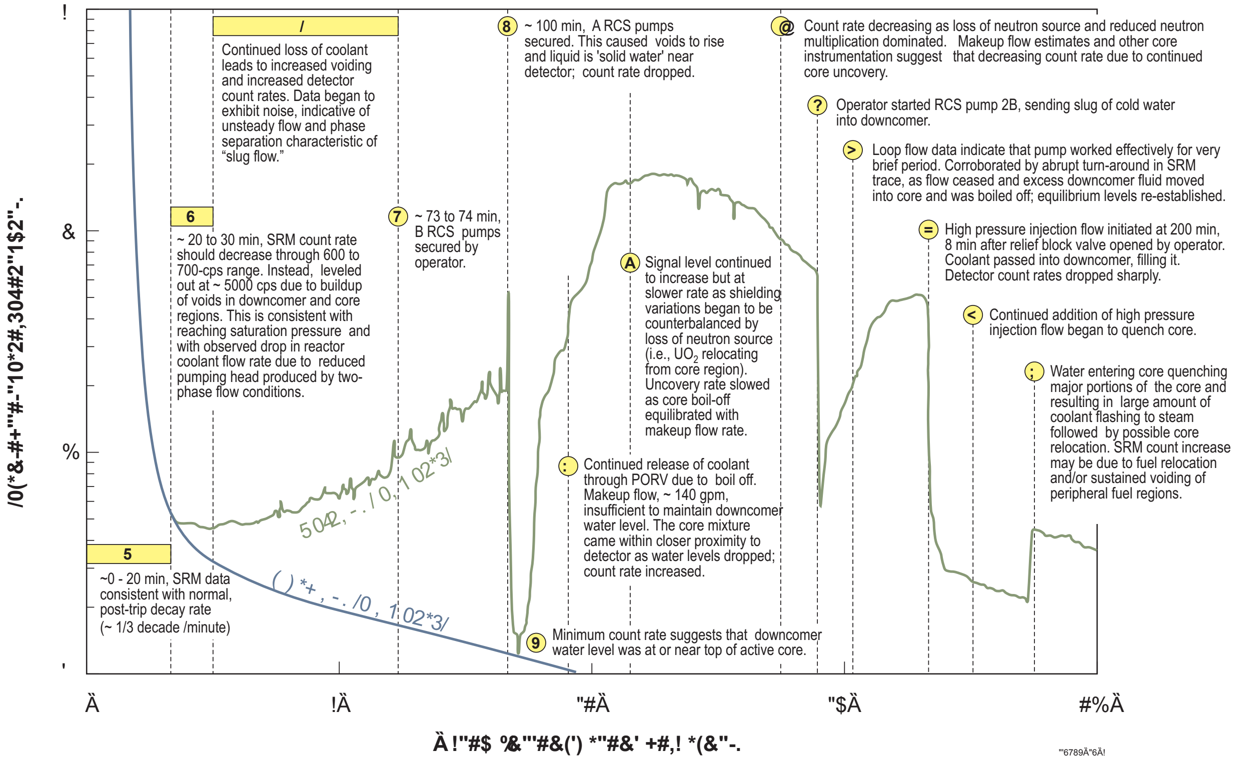

Figure 3-5. TMI-2 SRM response during the first 4 hours of the accident.

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Table 3-1. Data sources and time frame for developing the TMI-2 primary system composite pressure.[21]

Time Frame, minutes[a]	Data Source
-10. to 2.15	RC-3B-PT1-R, recorded on the Reactimeter
2.4 to 5.65	RC-3A-PT3-P, recorded on the Utility Printer as Memory Trip Review
6.0-100.	Saturation Pressure from RC-4A-TE1-R recorded on the Reactimeter
100.6 to 172.5	RC-3A-PT3-S, recorded on the strip chart
174.65 to 203.6	RC-3B-PT1-R, recorded on the Reactimeter
207. to 223.5	RC-3A-PT3-S, recorded on the strip chart
225.35 to 233.3	RC-3B-PT1-R, recorded on the Reactimeter
240.0 to 326.6	RC-3A-PT3-S, recorded on the strip chart
336.00 to 463.55	RC-3B-PT1-R, recorded on the Reactimeter
464. to 568.	RC-3A-PT3-S, recorded on the strip chart
570.3 to 869.6	RC-3A-PT3-P, recorded on the utility printer as operator group trend C
870.85 to 932.75	RC-3B-PT1-R, recorded on the Reactimeter
933.55	RC-3A-PT3-P, recorded on the utility printer
934.75 to 950.2	RC-3B-PT1-R, recorded on the Reactimeter
950.75 to 1000.	RC-3A-PT3-P, recorded on the utility printer

a. Timing uncertainties for the different data sources are as follows: Reactimeter 0.05 min.; Strip Chart 3 min.; Utility Printer = *0, -0.5 min.

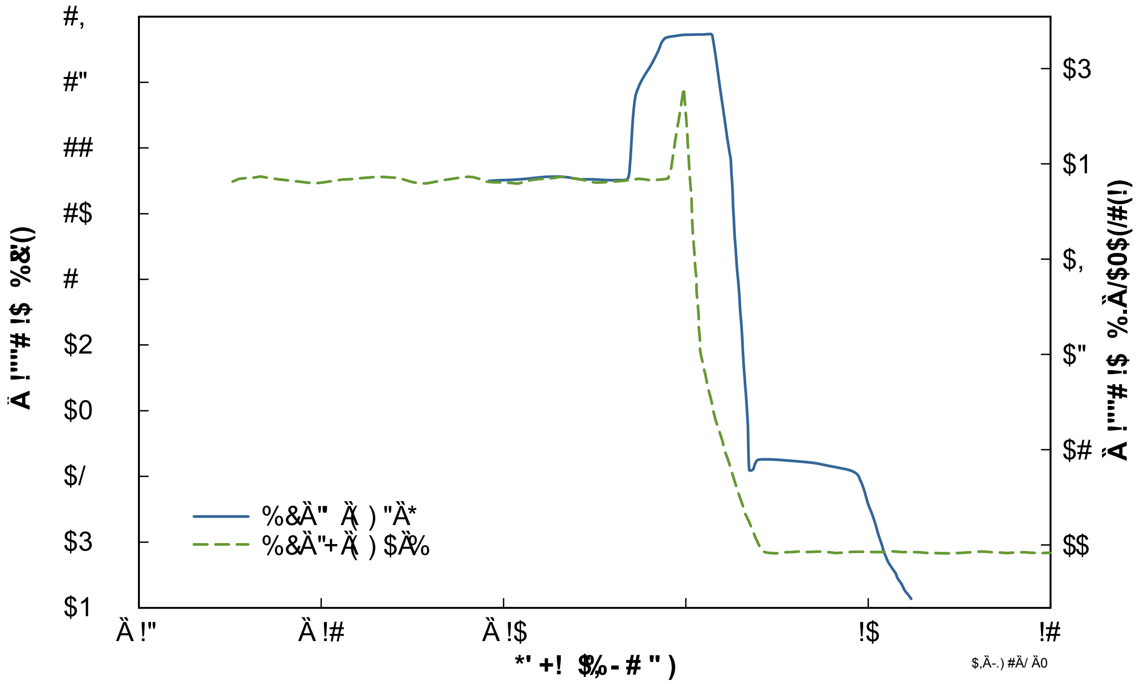

Figure 3-6. Comparison of A-loop wide range pressure recorded on strip chart No. 59 and the B-loop narrow range pressure on the reactimeter (up to 0.2 hours).[21]

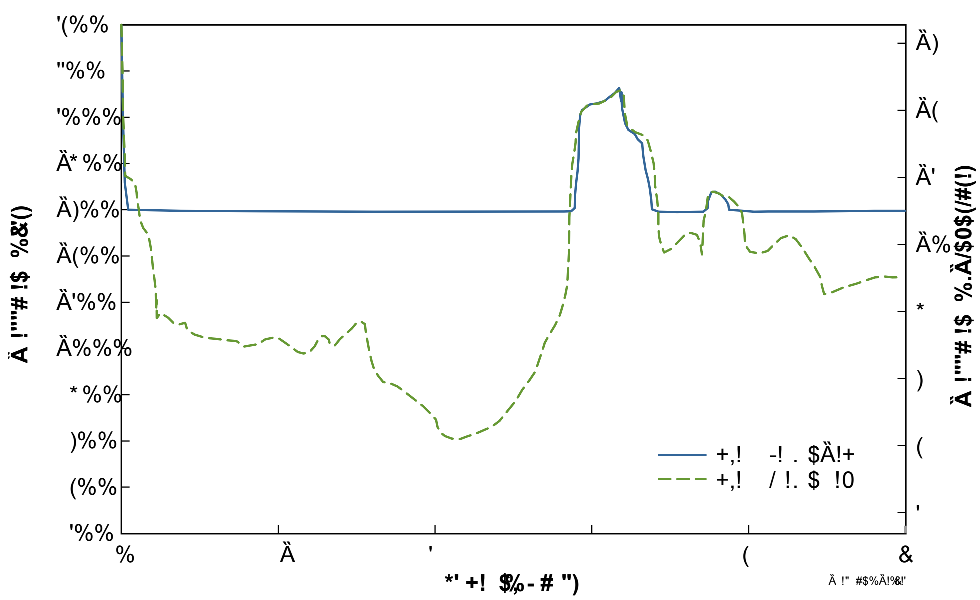

Figure 3-7. Comparison of A-loop wide range pressure recorded on the strip chart No. 59 and the B-loop narrow range pressure on the reactimeter (between 0 and 5 hours).[21]

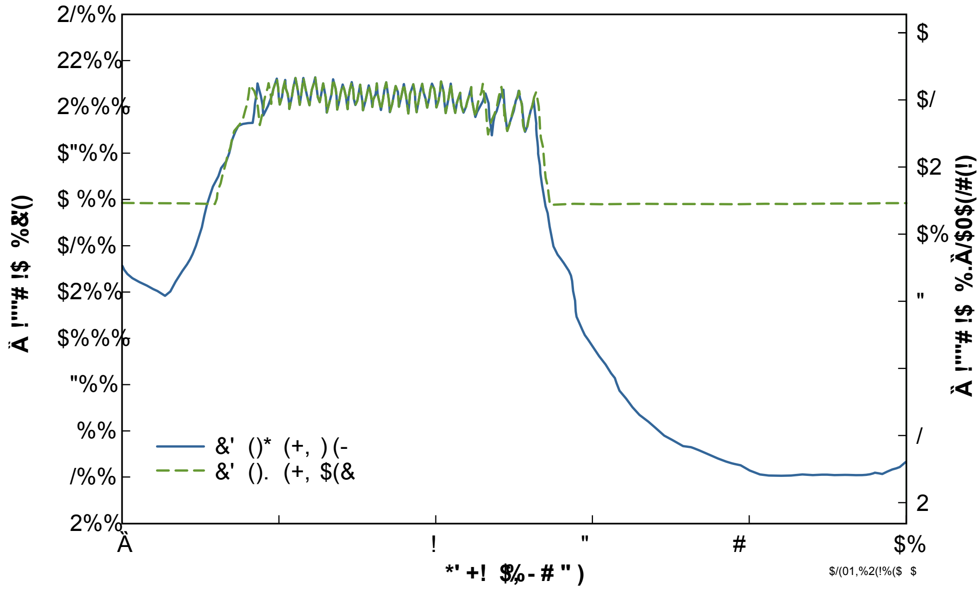

Figure 3-8. Comparison of A-loop wide range pressure recorded on the strip chart No. 59 and the B-loop narrow range pressure on the reactimeter (between 5 and 10 hours).[21]

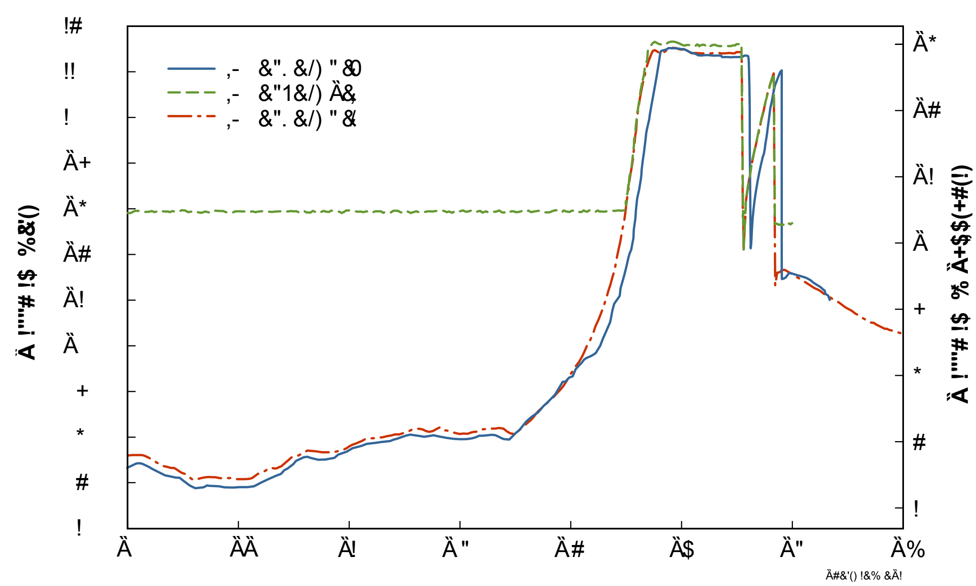

Figure 3-9. Comparison of A-loop wide range pressures recorded on strip chart and utility printer with the B-loop narrow range pressure on the reactimeter (between 10 and 17.5 hours).[21]

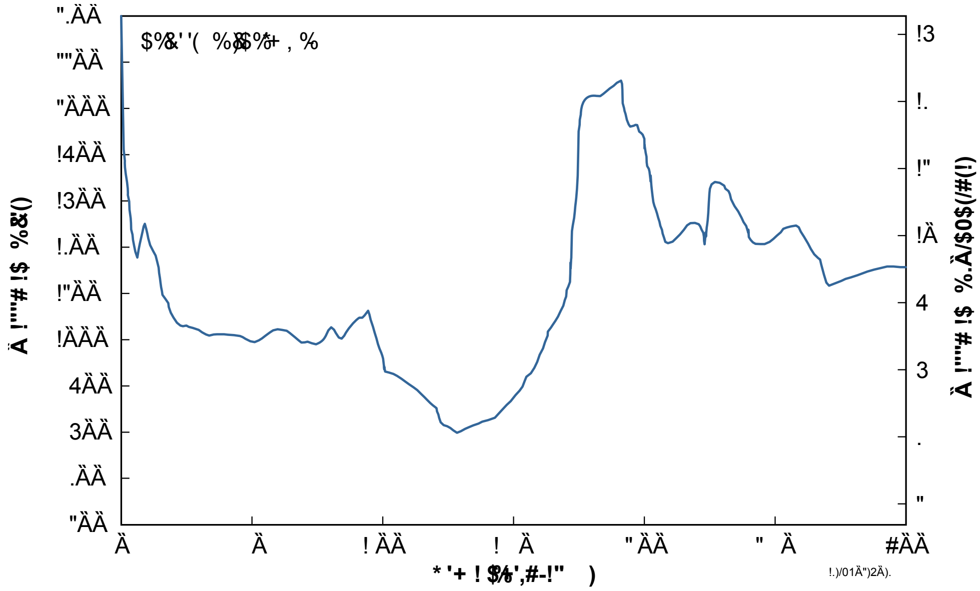

Figure 3-10. Composite primary system pressure (0 to 300 minutes).[21]

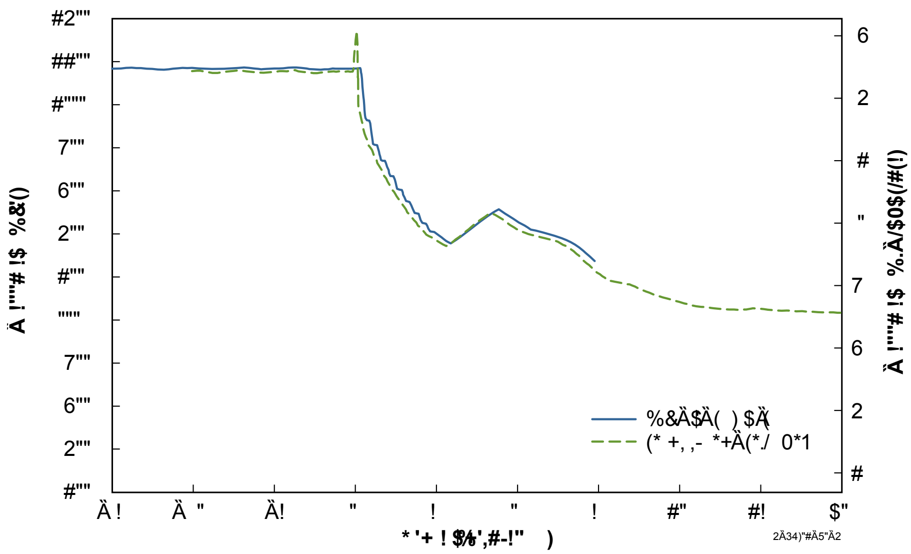

Figure 3-11. Comparison of composite primary pressure and wide range pressure recorded on the utility printer memory trip review.[21]

As discussed in Section 2, a comprehensive data qualification effort provides important insights about the timing of accident phenomena. In the case of primary system pressure, the qualified data provide insights about the timing of a major relocation of materials from the reactor core to the lower head. For example, the simultaneous increases from various sensors shown in Figure 3-12 suggest when relocation of core debris to the lower head occurred. The SRM count rate increases approximately 100% in less than 2 minutes (between 224 and 226 minutes) and then indicates a normal decay profile. Measured cold leg temperatures and RCS pressure shown in Figure 3-12 also increase rapidly at a time nearly coincident with the time when the abrupt increase in SRM response occurred.

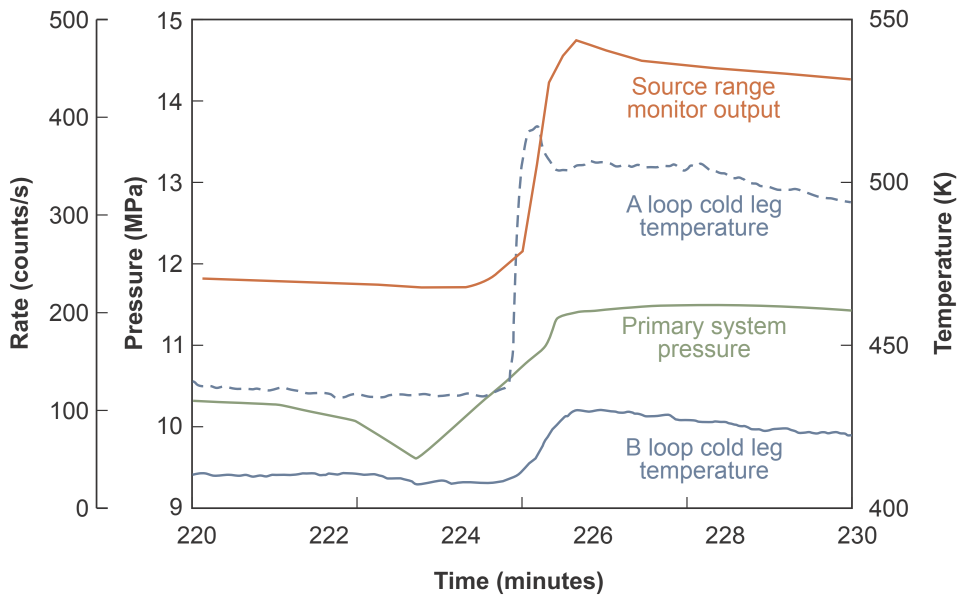

Figure 3-12. Overlay of SRM count rate, RCS pressure measurements, and cold leg temperatures.

3.3. Uncertainty Evaluation

As noted within this section, the time-dependent primary system pressure uncertainty was estimated by considering the uncertainties associated with each component of the measurement systems. Footnotes for Table 3-2 identify sources of component uncertainties and applied assumptions.

Table 3-2. Primary system pressure uncertainty analysis

Data Source[a]	Uncertainty Component		Uncertainty Estimate[b]	
			% of Range Span	Pressure (psig)
Reactimeter [c] RC-3B-PT1-R (Range = 1700-2500 psig)	Transmitter (Rosemount)[d]			
		Accuracy	±0.50%	± 4.0
		Temperature Sensitivity[e]	±1.0% per 100 °F	± 4.8
		Stability	± 0.25%	± 6.3
	Electronics (Tolerance)[f]		± 0.75%	± 6.3
	Recorder[g]		± 0.11%	± 0.9
	Total Uncertainty[h]			± 10.9
Utility Printer[i] RC-3A-PT3-P (Range = 0-2500 psig)	Transmitter (Foxboro)[j]			
		Accuracy	±0.50%	± 12.5
		Temperature Sensitivity	±1.0% per 65 °F	± 23.1
	Electronics (Tolerance)[f]		±0.50%	± 12.5
	Recorder (Computer)[k]		± 0.11%	± 2.5
	Total Uncertainty			± 29.2
Reactimeter-Saturation Pressure[l] RC-4A-TE1-R	P_{sat} due to Resistance Temperature Detector uncertainty			± 16.
Strip Chart[m] RC-3A-PT3-S (Range = 0-2500 psig)	Transmitter (Foxboro)			
		Accuracy	±0.50%	± 12.5
		Temperature Sensitivity	±1.0% per 65 °F	± 23.1
	Electronics (Tolerance)		±0.50%	± 12.5
	Strip Chart Recorder[n]			
		Recorder Setup		± 25.
		Digitalization of Strip Chart		± 10.
	Total Uncertainty			± 39.7

a. Composite primary pressure developed using the most accurate sources of data from multiple data sources.
b. The uncertainty estimates based on multiple sources listed in these footnotes. The uncertainty estimates are given for the 95% confidence level, and are all considered to be Bias estimates due to the lack of statistically significant data.
c. The narrow range pressure recorded on the Reactimeter (RC-3B-PT1-R), which was considered to be the most accurate data source, was used as the primary data source while within range.
d. Uncertainty estimates for the narrow range pressure transmitter based upon the Rosemount transmitter manual.
e. A 60 °F temperature increase in the containment building, near the location where the pressure transmitters were mounted, was used for obtaining uncertainty estimates due to temperature sensitivity of the pressure transmitter during the first 300 minutes.
f. The acceptable tolerance limits stated in the instrumentation calibration sheets and the surveillance procedure data sheets were used as the uncertainty estimates for the signal conditioning electronics.
g. The uncertainties associated with recording on the reactimeter were assumed to be the same as for recording on the plant computer. No reactimeter uncertainty information was found in manuals and drawings for the model and serial number of this unit.
h. Individual uncertainty components were combined using the Root-Sum -Square method (see Section 2.3.2.1).
i. Wide range pressure (RC-3A-PT3-P) information recorded on the utility printer was considered to be the most accurate available data source during periods in which the narrow range pressure transmitter was below its lower bound and used whenever available. Available utility printer wide range pressure data was from the Memory Trip Review (15 minutes of reactor trip) and the operator group trend C data (starting at 570 minutes).
j. Uncertainty estimates for the wide range transmitter are based on the Foxboro transmitter manual.
k. The uncertainty estimate for the data recorded on the computer (via the utility printer) was based on the individual uncertainty components of the analog- to-digital convertor given in the Bailey 855 Computer manual.
l. The saturation pressure, obtained using the hot leg RTD temperature measurement recorded on the Reactimeter, was considered to be the most accurate data source during the period of pumped two-phase flow in the A-loop (6-100 minutes). Uncertainty in the RTD temperature measurement of ± 1.1 °F (Reference 23) was used in conjunction with the ASME steam tables to obtain the stated uncertainty estimate.
m. Data obtained from the digitalization of the strip chart recorder were considered to be the least reliable data available and the last source of data used.
n. Uncertainty estimates for the strip chart recorder were based on engineering judgement.

3.4. Qualification

The composite system pressure was assigned a classification of “Qualified” with a maximum calculated uncertainty of ± 0.28 MPa-gauge by the DIRC during a meeting held on July 14, 1986.

3.5. Summary and General Insights

Qualification efforts to evaluate primary system pressure data provided confidence in an important input parameter for accident simulations. As illustrated in this section, data qualification efforts carefully considered data from various sensors within the vessel that were stored on various plant data acquisition systems. Several factors identified in Section 2 for developing qualified data were employed to develop a composite pressure. For example, analysts compared pressure data from various pressure sensors, reviewed SRM data for consistency with pressure sensor data, and performed engineering calculations to estimate pressure based on RCS temperature measurements. Recognizing that no single data source was appropriate for characterizing reactor pressure for the entire accident, a composite pressure and appropriate uncertainties were recommended. Ultimately, this composite pressure data was deemed “Qualified” by the DIRC.

4. CONTAINMENT TEMPERATURE

Because temperature measurements are key to understanding the TMI-2 accident, the survivability and performance of the 16 Resistance Temperature Detectors (RTDs) in the reactor building air handling system were investigated as part of the TMI-2 Data Qualification effort. As described within this section, evaluations[24,25] included analyzing data collected during and after the accident, observations of damage to materials within the containment, and data from in-situ tests conducted after the accident. In addition, comparisons were made with other data obtained from the containment and engineering analyses performed using such data.

4.1. Description

As discussed in Section 2, the TMI-2 containment building consists of a large, domed, cylindrical steel shell surrounded by reinforced concrete; the inside diameter and height are approximately 130 ft (40 m) and 190 ft (68 m), respectively (see Figures 2-3 and 2-4). Containment air temperature was measured at 16 locations in the reactor building air handling system (see Table 4-1) using RTDs.[25,26] These RTDs (see Figure 4-1) were Rosemount Series 78 sensors, having a single element with four lead wires. The RTDs conform to the International Platinum Temperature Scale No. IPTS-68 with an alpha coefficient of 0.00385 ohms/ohm/°C. The normal range of these RTDs was from -100 to 660°C with an accuracy of ± 1 °C at 93 °C. Nominally, these RTDs read 100 ohms at 0°C.

Table 4-1. Location of reactor building RTDs.[25]

RTD Identifier	Location	Elevation, ft
AH-TE-5010 (ambient air)	Sump pump	282
AH-TE-5011 (ambient air)	Letdown cooler	282
AH-TE-5012 (ambient air)	RC drain tank	282
AH-TE-5013 (ambient air)	Impinge barrier	282
AH-TE-5014 (ambient air)	Near equipment hatch	310
AH-TE-5015 (outlet temperature)	A/C plenum outlet	319
AH-TE-5016 (ambient air)	Primary shield concrete	282
AH-TE-5017 (ambient air)	Primary shield concrete	282
AH-TE-5018 (ambient air)	Primary shield concrete	282
AH-TE-5019 (ambient air)	Primary shield concrete	282
AH-TE-5020 (ambient air)	Top ceiling	353
AH-TE-5021 (ambient air)	Top ceiling	353
AH-TE-5022 (ambient air)	Southeast stairwell	330
AH-TE-5023 (ambient air)	West stairwell	330
AH-TE-5027 (outlet temperature)	A/C plenum outlet	305
AH-TE-5088(ambient air)	Southeast stairwell	310

RTD temperature measurements are based on the principle that changes in the resistance of the sensing element are related to changes in its temperature. This resistance change, which is precise and repeatable when circuit characteristics remain unchanged, is usually measured by passing a known current through the sensing element and measuring the voltage drop across it.

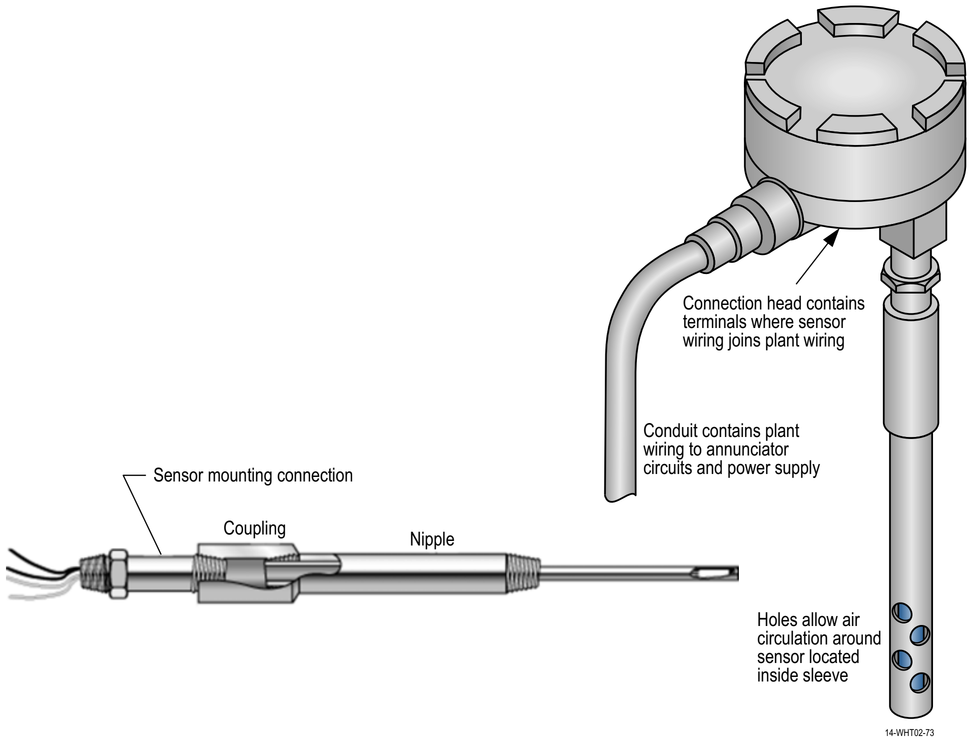

Figure 4-1. Air handling system RTDs in the TMI-2 containment and Rosemount 78 RTD design.[25]

RTD data were recorded on a strip chart in the Unit-2 Control Room. The recorder was a Bristol 550 Dynamaster multipoint unit, which was a servo-operated null balance potentiometer and bridge instrument. The recorder, calibrated in degrees Fahrenheit, sequentially recorded 24 variables on a 12-in. strip chart and was ranged for 0 to 200 °F. The recorder printed one temperature point each 15 seconds. With 24 points being printed for a complete cycle, it took 6 minutes to cycle and repeat an individual temperature point.

4.2. Data

The data from the plant strip chart recorders recorded during and after the accident were transcribed onto floppy discs. Researchers from the Electrical Engineering Department at the University of Idaho transcribed the data using the following process. Data from a 16mm film cartridge were projected and focused so that photo enlargements could be made. The enlargements were then recorded onto a graphics tablet. where the tablet stylus was used to identify the location of reference coordinates and the data of interest. Two individuals entered each data point, and a computer compared the two entries. Any differences in data between the two entries for a given time period was reviewed and corrections made as necessary.

Figure 4-2 shows RTD data at selected containment locations. Certain locations, such as the reactor coolant drain tank room (RTD 5012), experienced many temperature changes. Other locations showed no temperature change, such as the primary shield (RTDs 5016 through 5019). None of the 16 RTDs indicated a peak temperature greater than 90 °C. However, during the hydrogen burn, the top ceiling RTD 5020 behaved unexpectedly and recorded a negative trending trace, which is now attributed to activity from the containment sprays. Most other locations, including the other top ceiling RTD, recorded a positive-trending trace during this time.

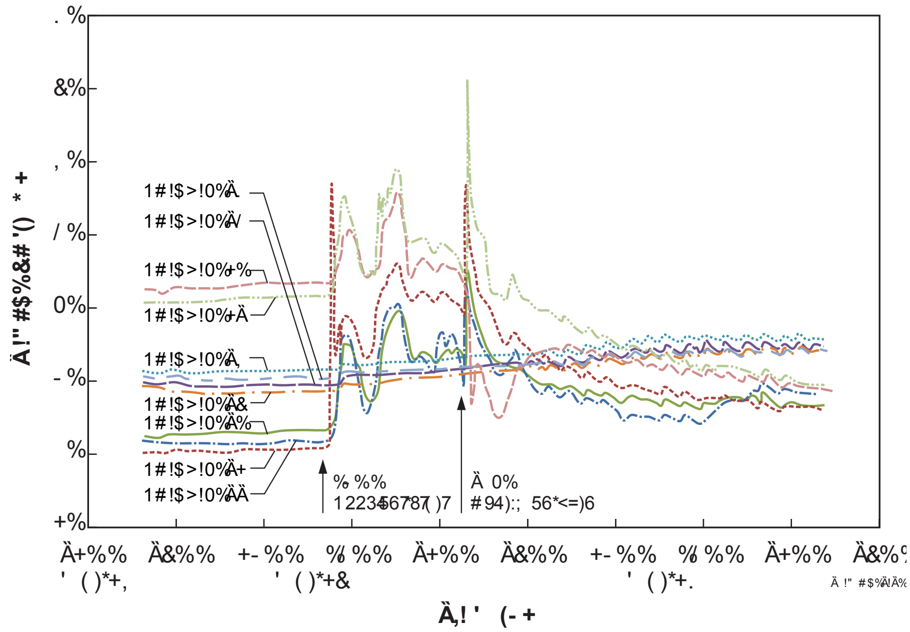

Figure 4-2. Reactor building temperatures at selected locations.

Photographs taken from within the TMI-2 containment[6] suggest much higher temperatures occurred during the TMI-2 accident. As shown in Figure 4-3, significant damage was sustained by thin materials (paper manuals, plastic sheeting, etc.), items made from thin plastics (e.g., telephones and buttons on instruments), and material with a low-thermal diffusivity (wooden planks, plastic rope, and the polar crane pendant cable). During the accident, some of these materials were exposed to temperatures that caused melting, pyrolysis, or burning. Materials with larger thickness and high-thermal diffusivity (metals) were not heated to temperatures high enough to undergo burn damage. Thus, as expected, most containment surfaces (painted steel or concrete) did not suffer apparent burn damage, and it is concluded that the materials, which were observably damaged, were those expected to be more susceptible to being heated to damage-threshold temperatures by the hydrogen burn.

4.3. Uncertainty Evaluation

Specifications on the Rosemount Incorporated Data Sheet for their Series 78 platinum RTD indicated accuracies of 1 °C. However, in 1982, there was a resistance reading error equivalent to 6°C in the RTDs. This was believed to be due to corrosion and surface contamination that occurred after the accident.

Table 4-2 estimates uncertainties for measuring the Letdown Cooler Temperature (AH-TE-5011-M) and the RC Drain Tank Temperature (AH-TE-5012-TE) by considering various sources contributing to

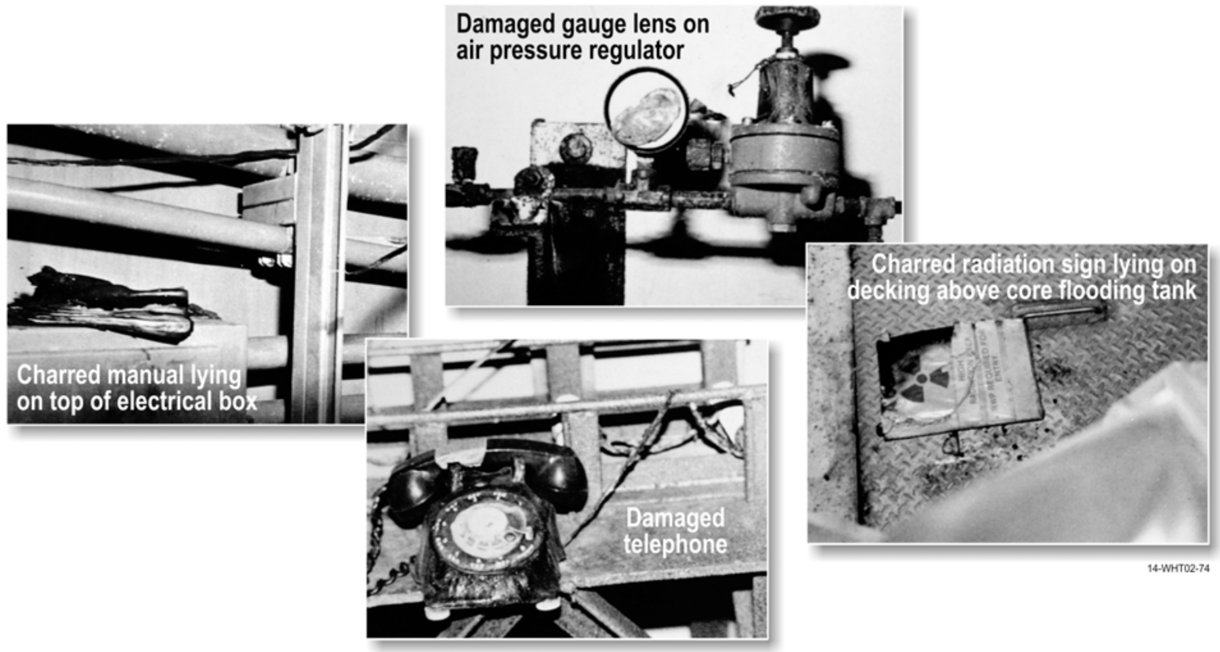

Figure 4-3. Damage within the TMI-2 containment attributed to the hydrogen combustion event.

uncertainties in the RTDs. Data from these RTDs were recorded on multipoint recorder AH-Y-MTR-5017, which was calibrated in November 1977 and again in March 1982. At the later time, a maximum error of 1 °F (0.6 °C) was measured in the recorder. According to the specifications on the RTD, the maximum possible error was approximately 1 °F (0.6 °C). Vendor information indicates that RTD stability was 0.2% of the maximum temperature. Post-accident evaluations, based on engineering analyses of the pressure data (see Section 5) indicate that the maximum temperature in the containment exceeded 650 °C.[24] The time uncertainty of data was ± 90 seconds, and data were printed every 6 minutes. The time constant was estimated to be 41 ± 24 seconds in air. As indicated in Table 4-2, analysts increased uncertainty estimates after the hydrogen burn because of high temperature exposures.

As part of efforts to understand instrumentation failure during the TMI-2 accident, air handling system RTDs were monitored and tested in-situ during the Unit-2 cleanup and recovery process. In-situ testing at TMI-2 indicated a problem with RTD temperature measurement. Specifically, RTD resistances were different when measured with the strip chart recorder system than when Fluke meters were used. A check of the instrument literature for all units showed that the strip chart recorder uses a constant current slightly greater than 1 mA, whereas the Fluke meters used constant currents of 3 to 3.5 mA. While such differences in current should not ordinarily cause differences in resistance and temperature readings, the environment at TMI-2, during and after the accident, may have caused abnormal conditions for the RTDs. The RTDs, when installed at TMI-2 were not required to be sealed against high humidity or protected from other conditions that existed during and after the accident. Although the actual conditions of the RTDs and associated wire and circuit components are not known, it is possible that they experienced chemical contamination and corrosion at terminals and penetrations, thus leading to different readings when measurements were made with different currents.

Table 4-2. Containment air temperature uncertainty analysis[a]

Item		Error	Comment
RTD[b]			
	Accuracy	0.95 °F	
	Stability[c]	0.46 °F	Before H_2 burn
		2.0 °F	After H_2 burn
Recorder[d]		1.0 °F	
Temperature		1.0 °F	
Digitizing Error[e]		2.0 °F	
Total Uncertainty			
Before hydrogen burn		2.7 °F	
After hydrogen burn		3.3 °F	
Time Uncertainty		± 1.5 minutes	

a. Based upon RTDs located at the Letdown Cooler Temperature (AH-TE-5011-M) and the RC Drain Tank Temperature (AH-TE-5012-TE)
b. Taken from Rosemount Inc. Product Data Sheet 2178 on the series 78 platinum RTD
c. Stability listed as 0.2% of exposed temperature hydrogen burn resulted in temperature greater than 1000 °F from GEND-INF-030.
d. Recorder error was measured in 1982 (EG&G Report ED-E3-82-017) and previously in 1977.
e. Estimated error of 1% for interpretation and reading. University of Idaho researchers projected 16 mm image onto a digitizing table.

4.4. Qualification

It was concluded that the RTDs remained operational during the accident. However, because of their slow sampling rate, recorded peak temperatures from the RTDs were less than 90 °C, which was much lower than the 650 °C estimates obtained from engineering calculations based on peak pressures measured in the reactor building.[24,27] Physical damage to organic materials substantiated that containment temperatures exceeded 232 °C, which was also much higher than available data from the RTDs. Hence, data were categorized as "Qualified" for all times except at the time when peak temperature were estimated to have occurred.

4.5. Summary

Qualification efforts to provide containment temperature data provided an important insight for analysts comparing results from TMI-2 accident simulations. As described within this section, several options identified in Section 2 for developing qualified data were employed to obtain qualified data, such as evaluating data collected during and after the accident, observations of damage to materials within the containment, obtaining data from in-situ measurements performed after the accident, comparisons made with data from other sensors within the containment, and engineering analyses were performed using such data. These comparisons ultimately led researchers to conclude that containment temperatures occurring during the hydrogen burn were not recorded due to the long time periods between measurements. Hence, containment temperature data were categorized as "Qualified" for all times except during the hydrogen burn.

5. CONTAINMENT PRESSURE

As discussed in Section 4, the peak temperature associated with the hydrogen burn was not measured by containment sensors because it occurred over a very short time period. Hence, containment pressure data were very important input for evaluations of the events that occurred during this phase of the accident. Several sources of data were available to estimate the reactor building pressure during the TMI-2 accident.[24,27] As described within this section, evaluations included analyzing data collected during and after the accident, observations of damage to materials within the containment, comparisons made with data obtained from other sensors within the containment, and engineering analyses performed using such data.

5.1. Description

As discussed in Section 2, the TMI-2 containment building consists of a large, domed, cylindrical steel shell surrounded by reinforced concrete; the inside diameter and height are approximately 130 ft (40 m) and 190 ft (68 m), respectively. During the accident, the containment building pressure was continuously recorded on the strip chart recorder using two different Foxboro pressure transmitters (Figure 3-2). In addition, pressure transmitters measured containment pressures as reference pressures for steam generator pressure monitors; and these data were recorded on the reactimeter every 3 seconds.[21,24] Furthermore, pressure events (pressure switch trips and resets) monitored by the TMI-2 plant computer system were timed to the second on the alarm printer. The sequence-of-events reports recorded on the utility printer also indicated the time of each event to the nearest millisecond. The availability of this recorded temperature and pressure data made the TMI-2 hydrogen burn a well-recorded, large-scale (57,600 m^3 or 2,033,000 ft^3), contained, premixed gas burn.

All 16 of the pressure switches were located in the auxiliary building and sensed the pressure in the Room B (east) side of the containment building through 28 to 65 ft-long tubes (see Figures 2-3 and 2-4). All of the pressure sensing points were located at Elevations 319 or 324 except for one at Elevation 293, which was the 3.58 lb/in^2 (25 kPa) (gage) switch and was the last to trip. The sensing point of this single switch was, therefore, separated from that of the other switches by the floor at Elevation 305. Further, the sensing point at Elevation 293 was located close to and in direct communication with the OTSG-B Bailey differential pressure transducer used to measure water level within the steam generator. Therefore, accurately timed pressure data were used to determine that the pressure-time delay across the Elevation 305 floor and the OTSG-B pressure response delay was approximately 3 seconds. As part of the TMI-2 data qualification effort, various mechanisms for this delay, which was longer than expected based on the sensor locations, were investigated. Results[24] indicate that it was due to a highly restrictive (more than 95%) inflow blockage at the screen (approximately 1 cm diameter) at the bottom of the transducer, which would delay its response to the changing reference pressure. This blockage was attributed to floating debris coming in contact with the bottom of the transducer as the water level in the basement area rose up to the level of the transducer.

Each strip chart recorder (BS-PT-4388-1 and BS-PT-4388-2) was a two-pen Taylor recorder, model 830 J, with input from two different Foxboro pressure transmitters. One transmitter on each recorder had a wide range of 0 to 0.7 MPa-gauge, and the other transmitter had a narrow range of -0.03 to 0.1 MPa-gauge. The measurement, BS-PT-4388-N-S, was stored on recorder SC-056, from the narrow range transmitter, SN 3259652. This measurement was within its range (-0.03 to 0.1 MPa-gauge) during the accident, with the exception of the pressure spike during the hydrogen burn. The only useful information from the wide range transmitter, BS-PT-4388-W-S (SN 3259653), is the magnitude of the pressure spike associated with the hydrogen burn (peak pressures of up to 0.2 MPa-gauge were measured). The narrow range pressure measurement on the other strip chart recorder (SC-055) BS-PR-1412-N-S was not recorded prior to the pressure spike due to failure of the pen to properly ink. The output from the wide range pressure transmitter BS-PT-1412-W-S was also routed to the plant computer. However, there was no indication that any data from this measurement were recorded during the accident.

5.2. Data

Data available from various sources within the containment consistently indicated that there was a sudden pressure increase in the reactor building. However, as discussed in Section 5.1, limitations associated with some of the instrumentation systems precluded some possible data options from being available. This section describes the manner in which each source was used to obtain qualified containment pressure data.

The containment pressure data was estimated by combining the narrow range data and the single data point for the pressure spike from the wide range channel. Comparison between the two recorded narrow range measurements were performed following the pressure spike to help in obtaining an estimate of the data uncertainty. This combined measurement was designated as BS-PT-4388-S. The recommended composite reactor building pressure are shown in Figure 5-1 (for 0 to 300 minutes) and in Figure 5-2 (for 0 to 1000 minutes).

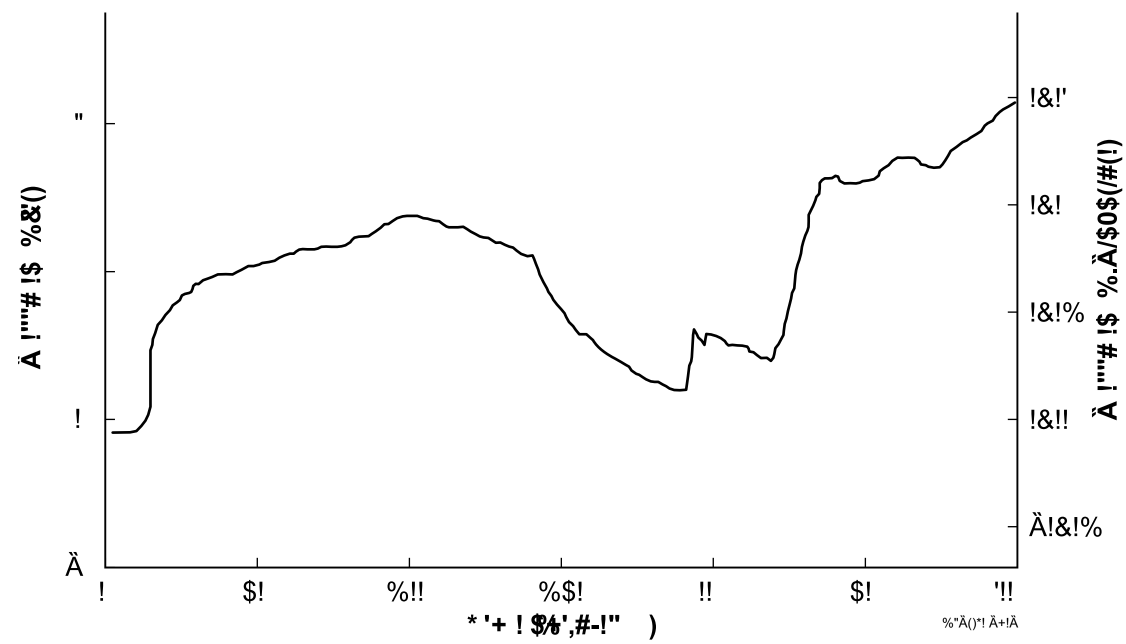

Figure 5-1. Containment building pressure (0 to 300 minutes)

The magnitude of the pressure spike, which occurred at the time of the hydrogen burn, was 0.20 MPa-gauge from the wide range pressure measurement, BS-PT-4388-W-S. This recorded spike was first considered a false electrical noise signal, such as might be caused by a ground fault. However, careful analysis of other recorded temperature and pressure data and of damage incurred to structures within the reactor building due to high temperatures (see Figure 4-3) and pressures (see Figures 5-3 and 5-4) led to

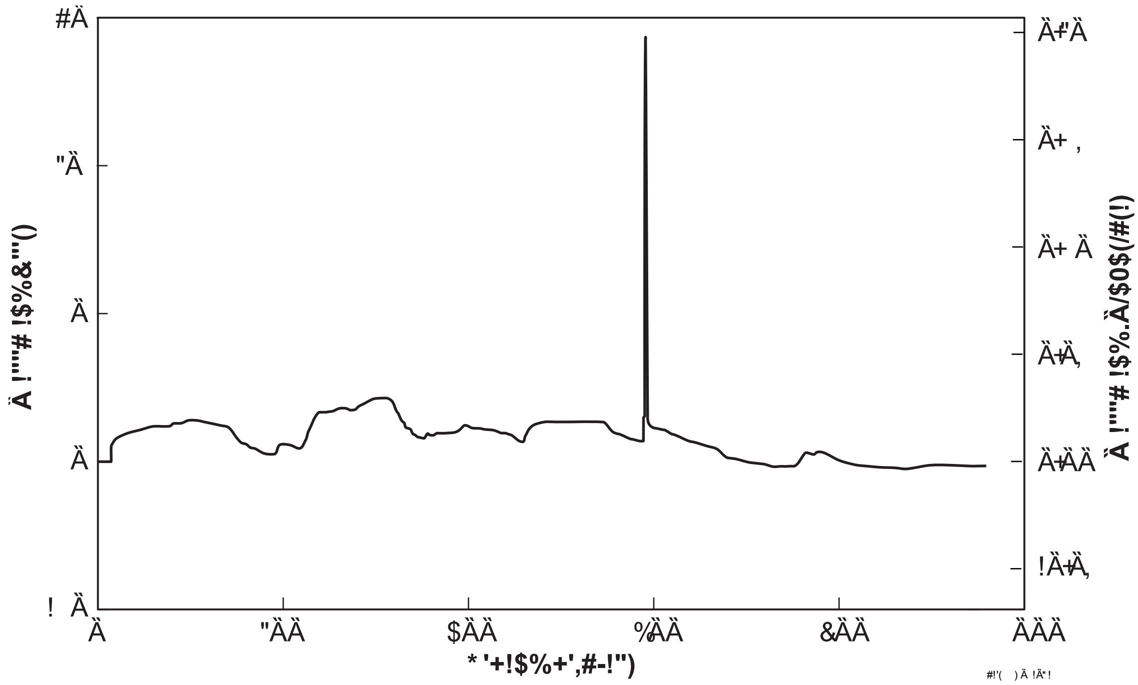

Figure 5-2. Containment building pressure (0 to 1000 minutes)

the conclusion that a hydrogen burn occurred in the containment building.[24]A composite of estimated containment gas temperatures and selected containment pressure data over the entire burn and cool down period is shown in Figure 5-5.[24]

Figure 5-3. Damage sustained to 55-gallon drum located in TMI-2 containment.

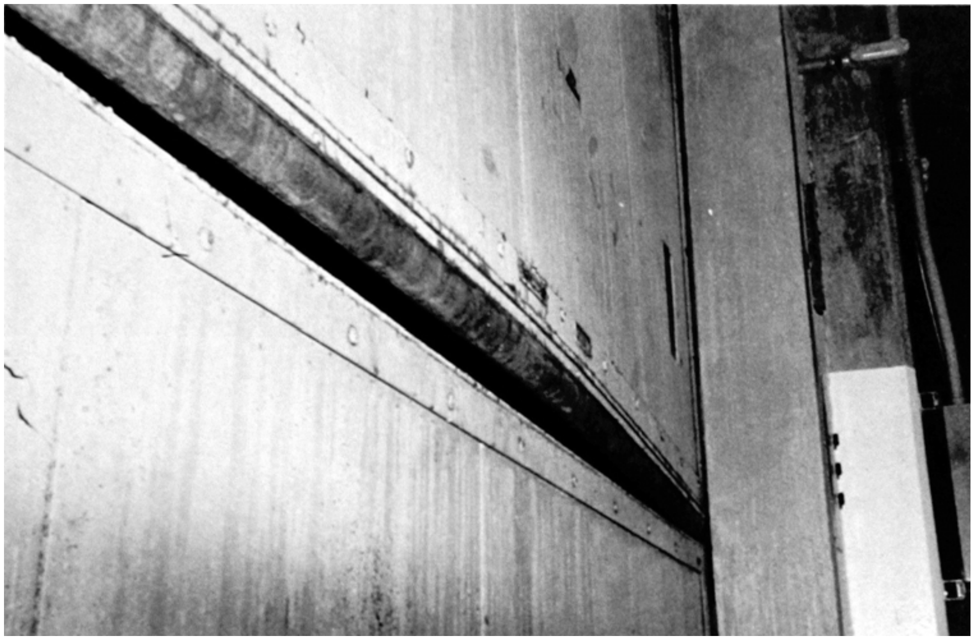

Figure 5-4. Close-up of gap in TMI-2 containment elevator door.

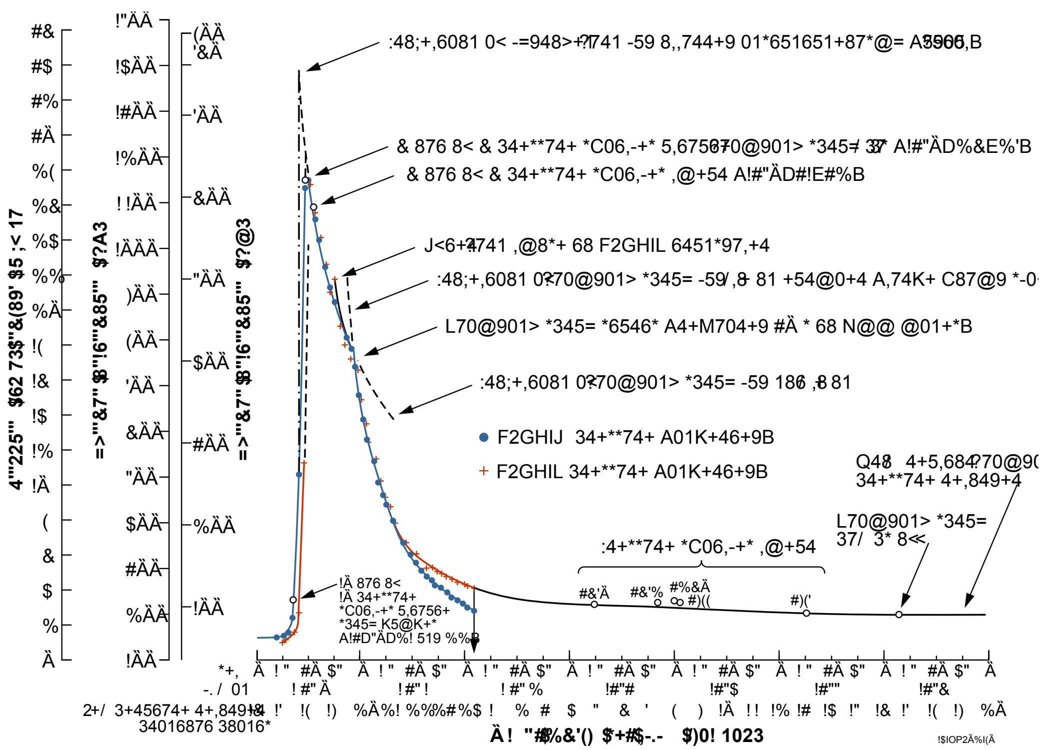

Figure 5-5. Composite TMI-2 containment pressure and calculated temperatures versus time.

The pressure transient caused by the hydrogen burn damaged structures within the containment, such as the 55-gallon drum (Figure 5-3). In addition, pressure differences developed between compartments due to the burn initiating at different times and propagating at different rates. The damage observed to the elevation shaft doors (Figure 5-4) is an example of a relatively isolated region where burning could be expected to be initiated at different times than when it occurred in the bulk of the containment. The observed damage indicates that a significant pressure difference existed between the elevator shaft and the containment building. However, it should be noted that the containment building was designed to safely withstand an internal pressure of 65 psia or 450 kPa (gage), and studies show that the containment could withstand much higher pressures. Therefore, the ~200 kPa (gage) pressure spike associated with hydrogen burn was not a serious threat to the containment building.

Although limited to single data points, the most accurate containment pressure data available were provided from 16 computer-monitored pressure switches. The alarm printer indicated the second these pressure switches actuate (trip or reset). When the operator requested a sequence of events involving these switches, the actuation times to the millisecond were recorded on the utility printer. These pressure switch data are correlated with the once-through steam generator (OTSG) pressure data and plotted in Figure 5-6. The 28 lb/in^2 (190 kPa) pressure switch data points are plotted as horizontal lines covering the time (second) of the reported actuations. The reactimeter data for OTSG-A were adjusted to match the accurate pressure switch data points.

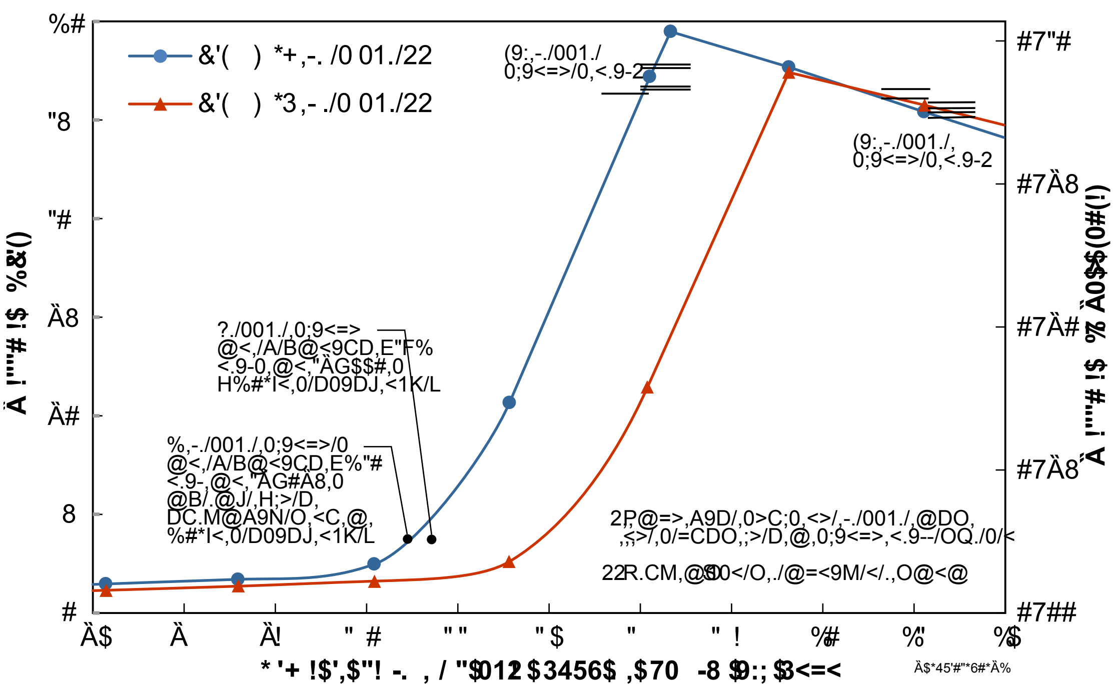

Figure 5-6. Comparison of sensor measurements during the hydrogen burn.

In addition, the alarm printer, which was limited to single data points, indicated that nine out of the ten 3.58 lb/in^2 (25 kPa) pressure switches were actuated during the second prior to 1350:21; the tenth switch actuated during the second prior to 1350:22. Additional information from a sequence-of-events report on the utility printer shows that four of these switches tripped between 1350:21:000 and 1350:21:440.

5.3. Uncertainty Evaluation

Estimates of the uncertainties from each data source are summarized in Tables 5-1 and 5-2. Containment building pressure uncertainty was estimated by considering the uncertainties associated with each component of the measurement systems. Footnotes are provided in these tables detailing the sources of the component uncertainties and assumptions used.

Table 5-1. Containment narrow range pressure uncertainty analysis (BS-PR-4388-N-S)

Uncertainty Component		Uncertainty Estimate[a]	
		% of Range Span	Pressure (psig)
Transmitter (Foxboro)[b]			
	Accuracy[c]	±0.50%	± 0.075
Electronics (Tolerance)[d]		±2.0%	± 0.300
Recorder (Strip Chart)			
	Set-up[e]	±0.50%	± 0.075
	Digitalization[f]	±0.33%	± 0.050
	Total Uncertainty[g]	±2.15%	± 0.32

a. The uncertainty estimates based on multiple sources listed in these footnotes.
b. Transmitter range for this measurement is -5 to 15 psig, which is used for the range span to obtain the pressure uncertainty estimates.
c. Uncertainties based on information in Foxboro transmitter manual.
d. Based on instrument loop test calibration sheet
e. Estimated by comparing independent measurements recorded on two strip charts.
f. Estimated from the recorder line width.
g. The total uncertainty estimate is obtained by combining the individual uncertainty components using the Root-Sum-Square method.

Table 5-2. Containment wide range pressure uncertainty analysis (BS-PR-4388-W-S)

Uncertainty Component		Uncertainty Estimate[a]	
		% of Range Span	Pressure (psig)
Transmitter (Foxboro)[b]			
	Accuracy[c]	±0.50%	± 0.50
Electronics (Tolerance)[d]		±2.0%	± 2.00
Recorder (Strip Chart)			
	Set-up[e]	±0.50%	± 0.50
	Digitalization[f]	±0.33%	± 0.33
	Total Uncertainty[g]	±2.15%	± 2.15

a. The uncertainty estimates based on multiple sources listed in these footnotes.
b. Transmitter range for this measurement is 0 to 100 psig, which is used for the range span to obtain the pressure uncertainty estimates.
c. Uncertainties based on information in Foxboro transmitter manual.
d. Based on instrument loop test calibration sheet
e. Estimated by comparing independent measurements recorded on two strip charts.
f. Estimated from the recorder line width.
g. The total uncertainty estimate is obtained by combining the individual uncertainty components using the Root-Sum-Square method.

Data uncertainty estimates are +2 kPa-gauge for the narrow range measurement and +14.8 kPa-gauge for the wide range estimate. This compares well with other recorded pressure data, considering uncertainty associated with digitizing data.

5.4. Qualification

During a meeting held on April 22, 1987, the DIRC assigned the composite system pressure a classification of "Qualified." An uncertainty of ± 0.32 psig (2.2 kPa-gauge) was assigned to the data except during the pressure spike when it was increased to ±2.2. psig (15 kPa-gauge).

5.5. Summary

Qualification efforts to evaluate containment pressure data provided an important insight for analysts comparing results from TMI-2 accident simulations. As described within this section, several options identified in Section 2 for developing qualified data were employed to obtain qualified data, such as evaluating data collected from various pressure instrumentation systems during and after the accident, observing damage to materials within the containment, and comparisons with other data obtained from within the containment. These comparisons ultimately led researchers to conclude measured containment pressures were correct and containment temperatures occurring during the hydrogen burn were not recorded due to the long time periods between measurements. Hence, containment pressure data were categorized as "Qualified" for all times during the accident with the uncertainty increasing after the hydrogen burn.

6. SUMMARY

The accidents at the TMI-2 PWR and Fukushima Daiichi Units 1, 2, and 3 BWRs demonstrate the critical importance of accurate, relevant, and timely information on the status of reactor systems during a severe accident. Many post-accident investigations concluded that actions taken by plant operators adversely contributed to the TMI-2 accident. However, the operators' ability to mitigate the accident was impacted by their limited access to accurate plant data. In addition, the ability to improve severe accident analysis codes is dependent on the quality of data used in accident simulations.

After the event, an evaluation program was initiated to determine what data were available to the operators and the status of sensors from which such data were obtained. As part of that effort, a formal process was initiated that resulted in a qualified data base for TMI-2 post-accident evaluations. Sensors allowed approximately 3000 measurements to be made at TMI-2. However, the TMI-2 Accident Evaluation Program (AEP) focused on data required by TMI-2 operators to assess the condition of the reactor and containment and the effect of mitigating actions taken by these operators. In addition, prior efforts focused on sensors providing data required for subsequent forensic evaluations and accident simulations.

Because there is the potential for similar activities to be completed for qualifying data from Daiichi Units 1, 2, and 3, this report provides additional details related to the formal process used to develop a qualified TMI-2 data base and present data qualification details for three parameters, primary system pressure, containment building temperature, and containment pressure. As discussed within this document, the effort invoked several options to qualify data from sensors for TMI-2 post-accident evaluations. Available instrumentation data was carefully evaluated using analysis relying on basic engineering principles, observations from post-accident inspections, operator information, laboratory evaluations of similar sensors, comparisons with accident simulation results, and results from large integral tests.

As discussed within this document, consistency comparisons with data from other sensors provided confidence about key accident phenomena, such as the occurrence of major relocation of materials from the reactor core and a hydrogen burn in the containment. As discussed in Section 3, the simultaneous increase in SRM count rate, primary system pressure, and cold leg temperatures, provided analysts confidence about the timing of a major relocation of materials from the reactor core to the lower head. Likewise, as discussed in Sections 4 and 5, peak values for parameters such as containment building temperature would not have been obtained without considering the results from other sensors such as the containment building pressure transmitters. Such results emphasize the benefits of the comprehensive sensor performance and data qualification efforts undertaken after the TMI-2 accident.

Evaluations of TMI-2 instrumentation demonstrated that both safety and non-safety-related sensors and components were affected by adverse environments and that many of these problems could have been avoided through applications engineering and more appropriate specifications, with only minor design changes being necessary. Several insights gained from developing qualified data for parameters discussed in this report, which were also identified in other data qualification efforts,[2] include:

- Data unavailability was often due to computational limits, such as storage memory, inadequate paper or ink, insufficient sampling rates, and 'preset' limits associated with anticipated operating ranges (rather than sensor operating limits). A wider range of limits and enhanced computational capabilities, with easy-to-read graphical displays, could easily alleviate such limitations.

- Data unavailability was often due to sensor range limitations that were focused on assumed normal operating conditions. For example, sensors with ranges that encompassed unanticipated accident conditions could have provided operators much needed information.
- Qualitative insights can be obtained by considering sensor response for alternate applications, e.g., ex-core source-range detector signals provide insights about real-time RCS water levels, in-core SPNDs provide insights about RCS temperature and water levels. However, such interpretations often requires detailed analyses and assumptions related to the status of the RCS and core.
- Although some materials within the containment were damaged due to temperatures and pressure pulses associated with the hydrogen burn, no functional damage to the nuclear plant instrumentation or electrical components from thermal effects of the hydrogen burn were identified.
- Evaluations emphasized the need to consider anticipated applications and more extensive inspection and maintenance programs for instrumentation and related systems. For example, data unavailability or high uncertainties could have been alleviated by the use of better installation and testing procedures with increased calibration checks.

In summary, a comprehensive set of instrumentation evaluations, that included careful integration of available data, analysis relying on basic engineering principles, operator information, laboratory evaluations, comparisons with accident simulation results and large integral tests, and post-accident inspection, was required for researchers to qualify sensor data for TMI-2 accident simulations. Knowledge gained from these evaluations offered important lessons for the industry with respect to sensor survivability, the need for additional and/or enhanced sensors and indicators, and the identification of unanticipated failure modes for sensors when exposed to extreme accident conditions. A similar process should be followed at Daiichi Units 1, 2, and 3 to reap BWR-specific benefits.

7. REFERENCES

1. C. A. NEGIN, M.E. LORENZI, C. F. BERGERON, P.H. RHINELANDER, R. D. MEININGER, and R. G. BOYD, "Instrumentation Performance under Severe Accident Conditions - Ways to Acquire Information from Instrumentation Affected by an Accident," EPRI TR-102371, November 1993.

2. J. REMPE and D. KNUDSON, *TMI-2 - A Case Study for PWR Instrumentation Performance during a Severe Accident,* INL/EXT-13-28043, Revision 1, May 2014.

3. R. HENRY, TMI-2: An Event in Accident Management for Light-Water-Moderated Reactors, American Nuclear Society, Chicago, IL, 2011.

4. J. REMPE, M. FARMER, M. CORRADINI, L. OTT, R. GAUNTT, and D. POWERS, "Revisiting Insights from Three Mile Island Unit 2 Post-Accident Examinations and Evaluations in View of the Fukushima Daiichi Accident," *Nuclear Science and Engineering,* **172**, November 2012, pp 223-248.

5. E. L. TOLMAN, J.P. ADAMS, J. L. ANDERSON, P. KUAN, R. K. MCCARDELL, and J. M. BROUGHTON, "TMI-2 Accident Scenario Update," EGG-TMI-7489, Idaho National Laboratory (1986).

6. EIDAM, G. R. and J. T. HORAN, 1981, "Color Photographs of the Three Mile Island Unit 2 Reactor Containment Building: Volume I - Entries 1, 2, 4, 5, 6," GEND-006, EG&G Idaho, Inc., Idaho Falls, Idaho, 1981.

7. M. L. RUSSELL and R. K. MCCARDELL, "Three Mile Island Unit 2 Core Geometry," *Nuclear Technology*, **87**, 865 (1989).

8. J.R. WOLF, J. L. REMPE, L.A. STICKLER, G. E KORTH, D. R. DIERCKS, L. A. NEIMARK, D. W. AKERS, B. K. SCHUETZ, T. L. SHEARER, S. A CHÁVEZ, G. L. THINNES, R. J. WITT. M. L. CORRADINI, and J. A. KOS, "TMI-2 Vessel Investigation Project Integration Report," NUREG/CR-6197, U.S. Nuclear Regulatory Commission (1994).

9. J. M. BROUGHTON, P. KUAN, D. A. PETTI, and E. L. TOLMAN, "A Scenario of the Three Mile Island Unit 2 Accident," *Nuclear Technology*, **87**, 35 (1989).

10. F. E. HASKIN, ET AL., "Perspectives on Reactor Safety," NUREG/CR-6042, Rev. 2, SAND-93-0971, March 2002.

11. J. G. KEMENY, ET AL., "Report of the President's Commission on the Accident at Three Mile Island," October 1979.

12. M. ROGOVIN and G. T. FRAMPTON, "Three Mile Island: A Report to the Commissioners and to the Public," NUREG/CR-1250, Vols. 1 and 2, January 1979.

13. W. PASEDAG, "DOE's TMI-2 Accident Evaluation Program," *Proceedings of an open forum sponsored by the OECD Nuclear Energy Agency and the U.S. Nuclear Regulatory Commission*, Boston, MA, October 20-22, 1993.

14. E.L. TOLMAN, “TMI-2 Accident Evaluation Program,” EGG-TMI-7048, Idaho National Laboratory (1986).

15. D. W. GOLDEN, K. AKAGANE, M. COLAGROASSI, P. DUMAZ, T. HAGA, K. HASHIMOTO, J. N. LILLINGTON, R. SAIRANEN. A. SHARON, R. O. WOOTON, T. VAN DER KAA, “Summary of the Three Mile Island Unit 2 Analysis Exercise,” *Nuclear Technology*, 87, 326 (1989).

16. R. L. MOORE, D. W. GOLDEN, and E.L. TOLMAN, “Three Mile Island Unit 2 Degraded Core Heatup and Cooldown Analysis,” *Nuclear Technology*, **87**, 990 (1989).

17. D. J. N. TAYLOR, “TMI SPND Interpretation,” *Proceedings of the First International Information Meeting on the TMI-2 Accident*, Germantown, MD, October 1985.

18. P. HOFMANN, S. J. L. HAGEN, G. SCHANZ, and A. SKOKAN, “Reactor Core Materials Interactions at Very High Temperatures,” *Nuclear Technology,* **87**, 146 (1989).

19. C. S. OLSEN, R.R. HOBBINS, and B. A. COOK, “Application of Severe Fuel Damage Experiments to Evaluating Three Mile Island Unit 2 Core Materials Behavior,” *Nuclear Technology*, **87**, 884 (1989).

20. A.D. KNIPE, S.A. PLOGER, and D.J. OSETEK, “PBF Severe Fuel Damage Scoping Test - Test Results Report,” NUREG/CR-4683, U.S. Nuclear Regulatory Commission (1986).

21. R. D. MCCORMICK, J.L. ANDERSON, and D.W. GOLDEN, “TMI-2 Data Summary Report,” EGG-TMI-7843, September 1987.

22. R. B. ABERNATHY, R. P. BENEDICT, “Measurement Uncertainty: A Standard Methodology,” *ISA Transactions*, Vol. 24, Number 1, 1985, 75.

23. R. D. MCCORMICK, “Initial Condition Temperatures of Primary Loop Coolant,” attachment to letter RDMc-7-86 to D. W. Golden, dated August 19, 1986.

24. J. O. HENRIE and A. K. POSTMA, “Lessons Learned from Hydrogen Generation and Burning during the TMI-2 Event,” GEND-061, May 1987.

25. J. W. MOCK, “Review of TMI-2 Resistance Temperature Detectors Accident Data and In-Situ Testing,” GEND-INF-024, November 1982.

26. M.O. Fryer, “Analysis of Air-Temperature Measurements from the Three Mile Island Unit 2 Reactor Building,” GEND-INF-030, April 1983.

27. J. K. JACOBY, R. A. NELSON, C. L. NALEZNY, and R. H. AVERIL, “Data Integrity Review of Three Mile Island Unit 2: Hydrogen Burn Data,” GEN-INF-023, Vol. 3, September 1983.