INL/CON-14-33682
PREPRINT

QUALIFICATION OF DATA OBTAINED DURING A SEVERE ACCIDENT-ILLUSTRATIVE EXAMPLES FROM TMI-2 EVALUATIONS

9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies (NPIC & HMIT 2015)

Joy L. Rempe and Darrell L. Knudson

February 2015

This is a preprint of a paper intended for publication in a journal or proceedings. Since changes may be made before publication, this preprint should not be cited or reproduced without permission of the author. This document was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights. The views expressed in this paper are not necessarily those of the United States Government or the sponsoring agency.

The INL is a
U.S. Department of Energy
National Laboratory
operated by
Battelle Energy Alliance

Idaho National Laboratory

QUALIFICATION OF DATA OBTAINED DURING A SEVERE ACCIDENT-ILLUSTRATIVE EXAMPLES FROM TMI-2 EVALUATIONS

Joy L. Rempe*
Rempe and Associates, LLC
360 Stillwater, Idaho Falls, ID 83404
jlrempe@cableone.net

Darrell L. Knudson
Idaho National Laboratory
P.O. Box 1625, Idaho Falls, ID 83415
darrell.knudson@inl.gov

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 loss 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. Post-TMI-2 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. Prior efforts also focused on sensors providing data required for subsequent forensic evaluations and accident simulations. This paper 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: reactor coolant system (RCS) pressure; containment building temperature; and containment pressure. 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 described to facilitate implementation of a similar process using data and examinations at the Daiichi Units 1, 2, and 3 reactors so that BWR-specific benefits can be obtained

Key Words: Three Mile Island Unit 2, Data Qualification

1 INTRODUCTION

The accident at the Three Mile Island Unit 2 (TMI-2) pressurized water reactor (PWR) provided a unique opportunity to evaluate instrumentation exposed to severe accident conditions. Conditions associated with the loss 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 Reactor Safety Technology program initiated in 2012 by the Department of Energy Office of Nuclear Energy (DOE-NE), activities have been completed to gain insights from prior TMI-2 sensor survivability and data qualification efforts.[1,2,3] These activities focused on 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 paper provides additional details related to the process used to develop a qualified TMI-2 data base and presents data qualification details for three parameters, reactor coolant system (RCS) pressure, containment building temperature, and containment pressure. As shown below in Fig. 1, results

* Contact author.

from qualifying data for these parameters led to key insights related to TMI-2 accident progression. Understanding gained from TMI-2 accident simulations 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, these analyses helped establish a sound technical basis for post-TMI-2 regulatory actions.

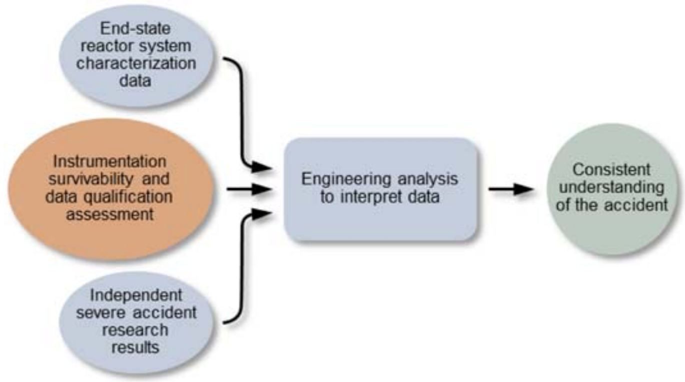

Figure 1. Integrated process including instrumentation survivability and data qualification assessment required to gain consistent understanding of the TMI-2 accident.

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 process was initiated that resulted in a qualified data base for TMI-2 post-accident evaluations.

2 BACKGROUND

This section summarizes relevant information related to the TMI-2 plant design, the accident, and the method used to qualify instrumentation data. More detailed information may be found in [1] through [5].

2.1 RCS and Containment Design

The TMI-2 power plant consisted of a Babcock & Wilcox, Inc. (B&W) PWR with a RCS that included the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping (see Fig. 2). 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 consisted of a large, domed, cylindrical steel shell surrounded by reinforced concrete; the inside diameter and height were approximately 130 ft (40 m) and 190 ft (68 m), respectively.

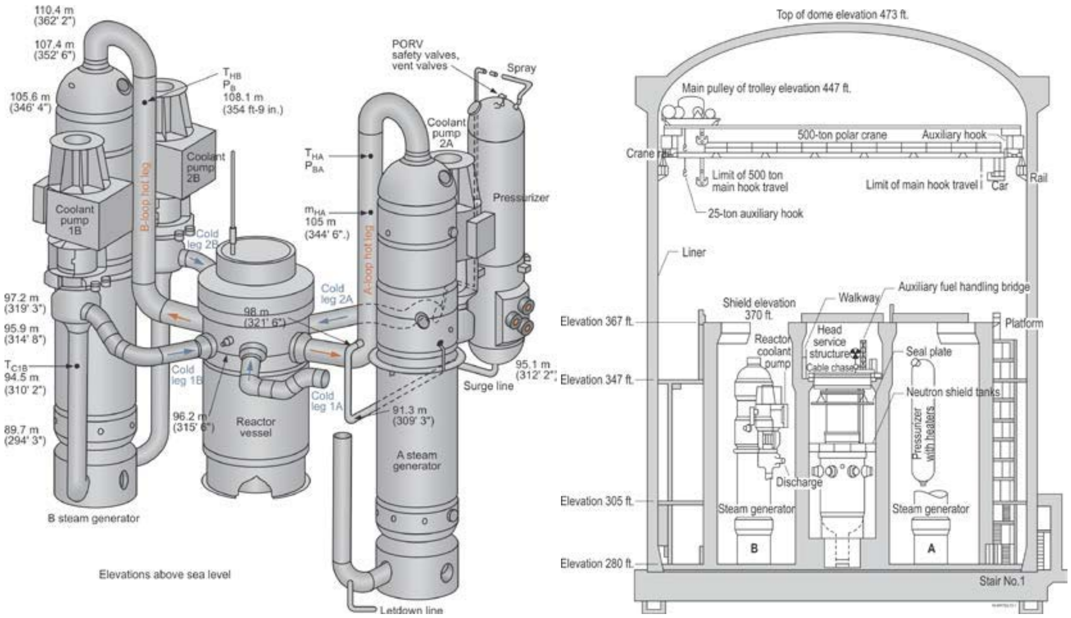

Figure 2. TMI-2 RCS and containment arrangement.

2.2 TMI-2 Accident Synopsis

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. Significant events occurring during the initial stages of the accident included turbine isolation (defined as time zero in Fig. 3), 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. Operators incorrectly interpreted this as indicating 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.

Instrumentation response suggests core uncovery began between 114 and 120 minutes and the vessel liquid level had dropped to the core midplane by approximately 140 minutes. When operators finally realized 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 Fig. 3) indicate core temperatures continued to increase between 150 and 165 minutes. Zircaloy-steam exothermic reactions produced large amounts of hydrogen and dramatically increased the core heatup rate.

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. 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. At 930 minutes, one of the A-loop primary coolant pumps was restarted, re-establishing heat removal from the vessel.

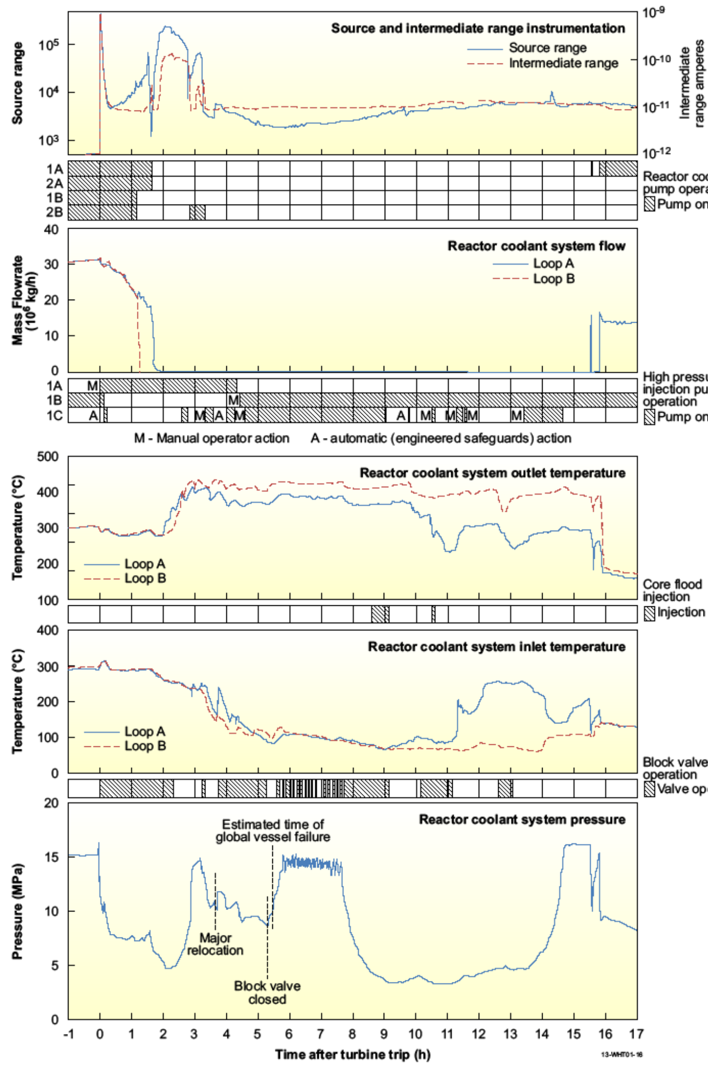

Figure 3. TMI-2 data from March 28, 1979.

2.3 TMI-2 Sensor Evaluation

An important aspect of the TMI-2 post-accident evaluation program was to provide a qualified data base for an analysis of the TMI-2 accident, known as the "TMI-2 Analysis Exercise." A TMI-2 Initial and Boundary Conditions Data Base was established to provide a qualified database for this analysis exercise as the sole source of data used by all participants in the TMI-2 Analysis Exercise. A data qualification process (see Fig. 4) 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. Several techniques were used to evaluate the sensor data, including comparisons with data from other sensors, analytical calculations, sensor 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 removed from the TMI-2 plant for evaluations. Methodologies were developed for performing the data uncertainty analyses, the criteria for determining the data quality categories, and the internal review process. 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 and deemed whether data were "Qualified", "Trend", or "Failed".

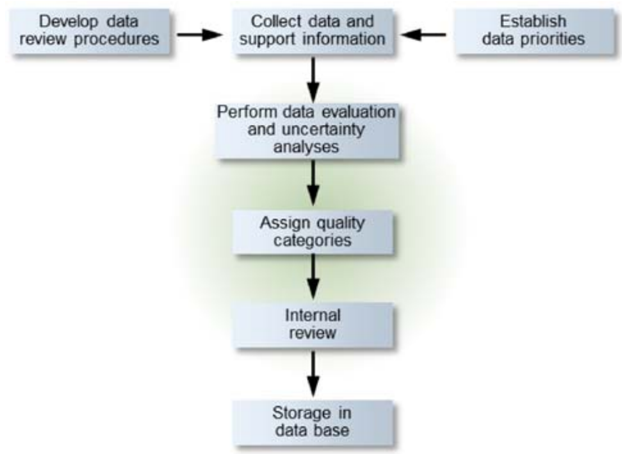

Figure 4. Activities completed to establish TMI-2 data.

3 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 paper 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: RCS pressure; containment building temperature; and containment pressure.

3.1 RCS Pressure

One essential parameter required for thermal-hydraulic analysis of the TMI-2 accident is RCS pressure. No single data source was available for estimating RCS pressure during the entire TMI-2 accident. Hence, a composite of various data sources was used to represent the RCS pressure.

There are two penetrations (see Fig. 2) for measuring the system pressure in each hot leg of the TMI-2 reactor. Connected to each of these penetrations, through sensing lines, are narrow range and wide range pressure transmitters, with measurement ranges of 11.7 to 17.2 MPa-gauge recorded on the plant reactimeter and 0 to 17.2 MPa-gauge recorded on the plant utility printer and on the strip chart. In addition, knowledge of reactor thermal-hydraulic conditions allowed the RCS pressure to be estimated from measured hot leg temperatures.

Fig. 5 compares A-loop wide range pressures recorded on the strip chart and utility printers (RC-3A-PT3-S and RC-3A-PT3-P) with B-loop narrow range pressure data (RC-3B-PT1-R). 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 B-loop narrow range pressure data (RC-3B-PT1-R).

The comprehensive TMI-2 data qualification effort provided important insights about the timing of accident phenomena. For example, 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 Fig. 6). In addition to the RCS pressure increase (between 224 and 226 minutes), the SRM count rate increases approximately 100% and measured cold leg temperatures also increase rapidly during this time.

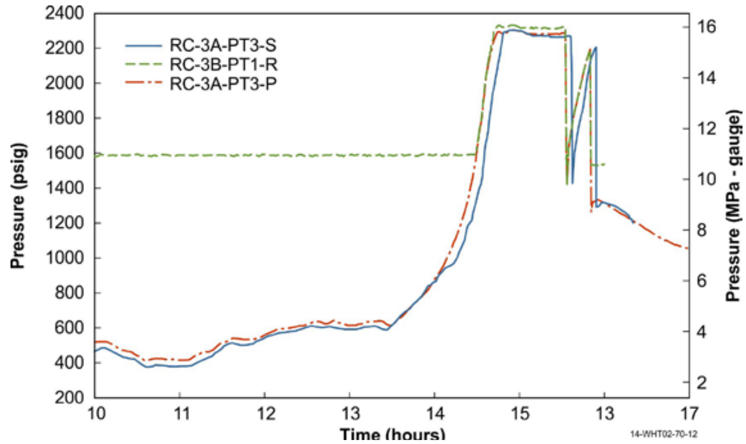

Figure 5. Comparison of A-loop wide range pressures recorded on strip chart and utility printer with the B-loop narrow range pressure on the reactimeter (between 10 and 17.5 hours).

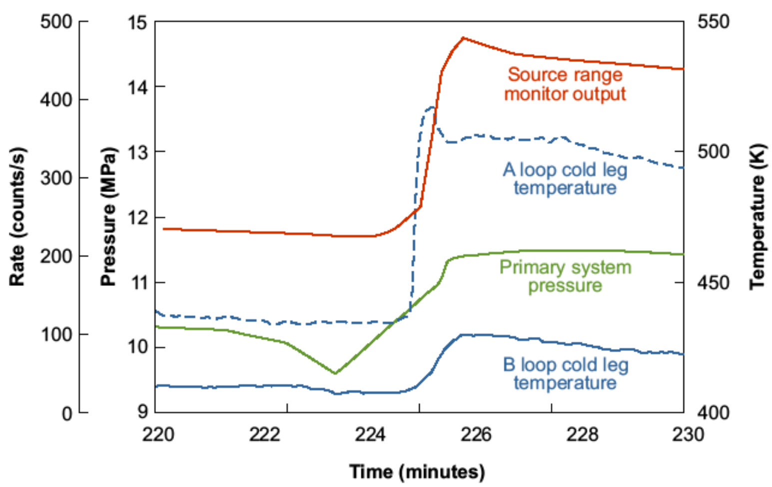

Fig. 6. Overlay of SRM count rate, RCS pressure measurements, and cold leg temperatures.

As noted above, the data qualification programs included estimating measurement uncertainties. The time-dependent RCS pressure uncertainty was estimated by considering the uncertainties associated with each component (e.g., the detector, the electronics, and recorder) of the measurement systems. The composite system pressure was assigned a classification of "Qualified" with a maximum calculated uncertainty of ± 0.28 MPa-gauge by the DIRC.

3.2 Containment Temperature

Containment air temperature was measured at 16 locations in the reactor building air handling system using Resistance Temperature Detectors (RTDs), with a measuring range from -100 to 660 °C and accuracy of ± 1 °C at 93 °C. RTD data were recorded on a strip chart in the control room. 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. Fig. 7 shows RTD data at selected containment locations. Certain locations, such as the reactor coolant drain tank room (AH-TE-5012), experienced many temperature changes. Other locations showed no temperature change, such as the primary shield RTDs (AH-TE-5016 through AH-TE-5019). None of the 16 RTDs indicated a peak temperature greater than 90 °C. During the hydrogen burn, the top ceiling RTD (AH-TE-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 positivetrending trace during this time. However, photographs of damaged materials within containment (Fig. 8) suggest much higher temperatures occurred during the accident. Furthermore, calculations assuming measured peak containment pressures (see Section 3.3) yielded peak containment temperatures of 650 °C, which are much higher than the measured 93 °C peak temperature data (Fig. 7).

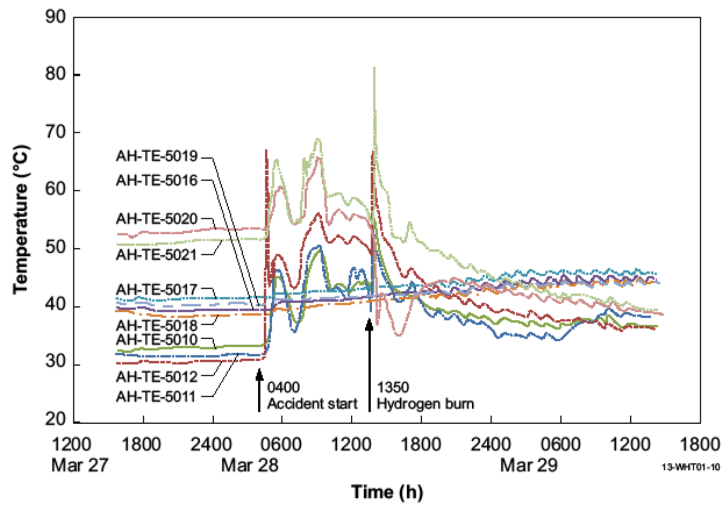

Fig. 7. Reactor building temperatures at selected locations.

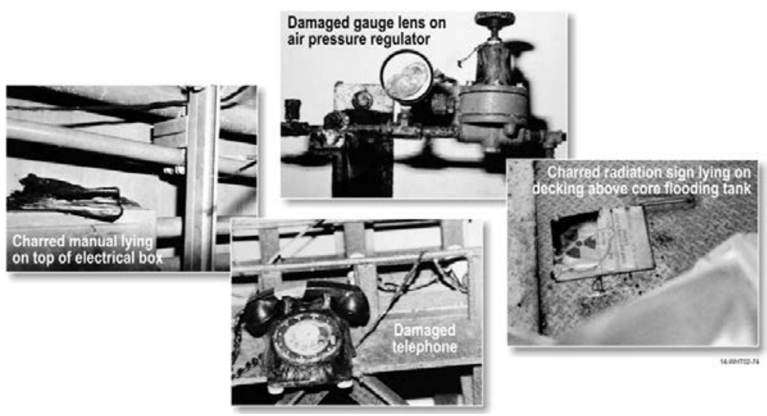

Fig. 8. Damage within TMI-2 containment attributed to the hydrogen combustion event.

Evaluations concluded RTDs remained operational during the accident. However, because of their slow sampling rate, recorded peak temperatures from the RTDs were much lower than the 650 °C estimates obtained from engineering calculations based on peak pressures measured in the reactor building. Physical damage to organic materials substantiated that containment temperatures exceeded 232 °C, which was also much higher than available data. RTD uncertainty evaluations indicated errors of ~1.5 °C before the hydrogen burn and 1.8 °C after the burn. Containment temperature data were categorized as "Qualified" except the time when peak temperatures were estimated to have occurred.

3.3 Containment Pressure

During the accident, the TMI-2 containment building pressure was continuously recorded on the strip chart recorder using two different 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. Measurements were within the narrow range measurements for most of the accident with the exception of the pressure spike during the hydrogen burn, which was captured by the wide range recorder (peak pressures of up to 0.2 MPa-gauge were measured). The containment pressure (Fig. 9) was estimated by combining narrow range data and the single data point for the pressure spike from the wide range channel.

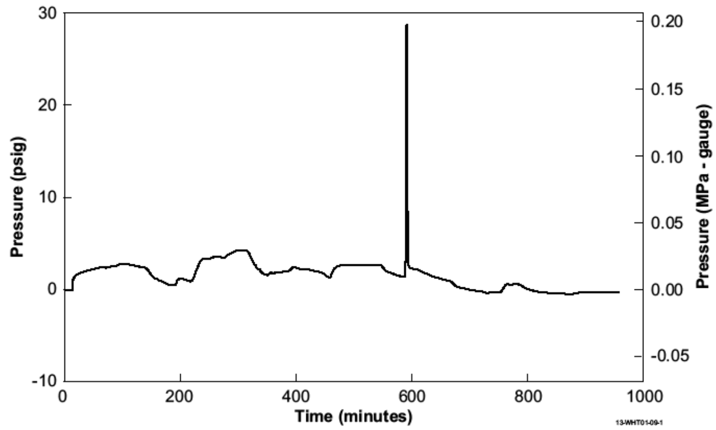

Fig. 9. Containment building pressure.

The time-dependent containment pressure uncertainty was estimated by considering the uncertainties associated with each component (transmitter, electronics, and recorder) of the measurement systems and comparisons between available data. The DIRC assigned the composite system pressure a classification of "Qualified." An uncertainty of ± 2.2 kPa-gauge was assigned to the data except during the pressure spike from the hydrogen burn when it was increased to ± 15 kPa-gauge.

4 SUMMARY

In summary, a comprehensive set of 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 to qualify sensor data and estimate uncertainties for TMI-2 accident simulations. The three examples described in this paper illustrate the types of activities completed in the TMI-2 data qualification process and the importance of such a qualification effort in the process of gaining a consistent understanding of events that occurred during the TMI-2 accident. A similar process should be followed at Daiichi Units 1, 2, and 3 to reap BWR-specific benefits.

5 ACKNOWLEDGMENTS

Work supported by the US Department of Energy, Office of Nuclear Energy, Science, and Technology, under DOE-NE Idaho Operations Office Contract DE AC07 05ID14517.

6 REFERENCES

1. J. Rempe and D. Knudson, *TMI-2 - A Case Study for PWR Instrumentation Performance during a Severe Accident*, INL/EXT-13-28043, Revision 1 (2014).
2. J. L. Rempe and D. L. Knudson, "Instrumentation Performance during the TMI-2 Accident," *IEEE Transactions on Nuclear Science*, **61**, Issue 4, pp 1963-1970 (2014).
3. J. Rempe and D. Knudson, *Qualification of Daiichi Units 1, 2, and 3 Data for Severe Accident Evaluations -Process and Illustrative Examples from Prior TMI-2 Evaluations*, INL/EXT-14-32628, (2014).
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**, pp 223-248 (2012).
5. J. Rempe, L. Stickler, S. Chàvez, G. Thinnes, R. Witt, and M. Corradini, "Margin-to-Failure Calculations for the TMI-2 Vessel," *Nuclear Safety*, **35**, No. 2, p 313, (1994).