SENSOR PERFORMANCE AFTER THE THREE MILE ISLAND (TMI) ACCIDENT

By:
H.M. Hashemian

IAEA TWG NPPIC Meeting

IAEA
Nuclear Power Engineering Section
Division of Nuclear Power
Wagramer Strasser 5,
Vienna, Austria
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1. BACKGROUND

In March 1979, the Three Mile Island (TMI-2) nuclear reactor suffered a loss-of-coolant accident (LOCA). Measurements made by reactor personnel during the accident showed some in-core thermocouples indicating temperatures at or above the melting point of the thermocouple materials (2550°F). The lowered water level in the reactor caused Platinum Resistance Thermometers (PRTs) installed in the hot legs of the coolant loops to be exposed to superheated steam. The PRTs in Loop B exceeded the upper recorder temperature indication limit of 800°F, which is significantly greater than the upper temperatures limit of 670°F specified for the PRTs.

During the accident the PRT connecting heads and signal cables are thought to have been subjected to escaping steam, and the PRT seals reached a temperature that was surely higher than normal. In addition, as the accident progressed the primary coolant became a saturated two-phase mixture of increasing void fraction that caused increasing vibration in the circulation pumps, with the result that the Loop B pumps were tripped 73 minutes into the accident and Loop A pumps were turned off 100 minutes into the accident in response to indications of low system pressure, high vibration, and low coolant flow.

2. POST ACCIDENT EXAMINATION

In order to assess the validity of the temperatures of the TMI-2 reactor coolant measured during and after the reactor accident, the three problems listed below had to be solved.

1. Possible Decalibration of PRTs

The temperatures of the primary coolant water in the TMI-2 reactor were measured by PRTs installed in thermowells. During the accident the PRTs were subjected to excessive temperatures, vibration, and radiation. After the reactor was shut down, the PRTs continued to be subjected to gamma radiation from the fission products deposited in the coolant loops.
2. Possible Response Time Degradation

Analysis of the coincidence of events during the accident requires knowledge of the response times of temperature sensors. The response times could have changed as a result of excessive temperature and/or vibration during the accident.

3. Possible Voltage Shunting

The validity of recorded temperatures depends on the assumption that the resistance measured is entirely that of the PRT sensing element. If there were, for example, an unaccounted for 0.1-MΩ leakage resistance in parallel with the PRT element, a 3°F error would result at the normal reactor operating temperature of 550°F. The output signal from the temperature transmitter may have been degraded by partial shorting between the PRT wires or the extension cable wires.

3. CONCLUSIONS FROM POST ACCIDENT EXAMINATION

The PRT test results lead us to conclude the following:

1. The low insulation resistance exhibited by many of the PRTs is attributable to failure of the cable conduits or the seals of the connecting heads. (All but two of the TMI PRTs show evidence of moisture). After PRT S/N 3670 was removed from the reactor, the insulation resistance increased to a value that exceeds factory specifications with the extension cable removed, indicating that the sheath seal was still intact.

2. The calibration of PRT S/N 3670 was not changed significantly by the accident. Since the in-situ tests showed the same element resistance for all PRTs in TMI-2 during nearly isothermal conditions, it is likely that none of the PRTs suffered a significant loss of calibration.

3. The response time of PRT S/N 3670 under benchmark conditions lies between the two sets of measurements reported in the original factory certifications. Therefore, there appears to be no degradation of response time in the PRT.
4. From laboratory measurements of response time at different temperatures and coolant flow rates, the response time of PRT S/N 3670 was calculated to be 4.5± 0.1 s under TMI-2 operating conditions. This value is even less that the 5 s required by the TMI-2 plant technical specifications.

5. The TMI-s PRT S/N 3670 met the technical specification response time requirement without the use of Never-Seez in the thermowell.

6. The self-heating index showed that PRT S/N 3670 had a poorer heat transfer to the surrounding reactor coolant water than did the other PRTs. This result provides an indication, but not proof, that the other TMI-2 PRTs probably have shorter response times under benchmark conditions that the one PRT that was removed.

4. TMI PROJECT STUDY

The TMI project had two main objectives: assessment of the current condition of the RTD/thermowell assemblies installed at TMI-2 and assessment of performance of RTDs during the accident.

Assessing the current condition of the RTD/thermowell assemblies provided an indication of their survivability under the conditions that they experienced during and after the accident. This was important for evaluating the suitability of sensor designs and qualification methods, to assess failure modes (if any), to permit design improvements, and to assess the reliability of the sensor readings during the recovery operations.

The second main objective to assess the performance of the RTDs during the accident provided information needed to ensure that the data records used to interpret system status during the accident were reliable.

The main conclusions from the TMI-2 testing were:

1. Resistance-to-ground measurements give abnormally low resistances for some sensors. Water ingress into the RTDs or cables is postulated. However, the change is not large enough to cause readily observable shifts in sensing element resistances used in temperature measurement.
2. The resistance of wires extending from the RTDs to their temperature transmitters located outside the containment is normal.

3. The heat transfer resistances between the platinum element and the fluid, and the RTD response times are within the expected range of values. This indicates that major physical damage to the sensor/thermowell system has not occurred.

4. A preliminary estimate of the time constants of the TMI-2 RTDs is approximately 13 seconds at full temperature and flow conditions. Verification of this result requires further testing at Crystal River 3.

5. It was demonstrated that the LCSR method can be used to determine whether an RTD is in moving water, stagnant water, or stagnant air.

A study was conducted by Analysis and Measurement Services Corporation (AMS) to evaluate the current steady state and transient status of the TMI-2 RTDs and to assess their performance during the accident. This study included bench testing of RTDs at TMI-3 and at Crystal River 3 (a sister plant to TMI-2). These tests consisted of measurements of electrical and heat transfer properties of the RTDs. The details are presented in this report. The main conclusions are:

1. The temperatures measured by the TMI-2 RTDs are reasonable. That is, the TMI-2 RTDs have not undergone a major shift in calibration.

2. The resistance of the wires extending from the RTDs to the temperature transmitters was found to be normal.

3. The resistance-to-ground has suffered degradation. A likely cause is moisture ingress into the RTDs or cables.

4. The heat transfer resistances and the response time of the TMI-2 RTDs appear to have experienced minor degradation. This indicated that the response of the RTDs to temperature perturbations during the accident was close to nominal performance.
5. ON-LINE MONITORING (OLM)

OLM technology developments date back to the accident at TMI-2 nuclear power plant near Harrisburg, Pennsylvania, in March 1979. The accident revealed the need for reliable signals from process instrumentation in nuclear power plants, and led to the development of signal validation techniques under projects sponsored by the Department of Energy (DOE), Electric Power Research Institute (EPRI), and others in the United States, as well as efforts carried out in a number of organizations outside the U.S., such as the Électricité de France and the Halden Reactor Project in Norway. The TMI accident changed the nuclear industry’s response in many areas of nuclear power plant operation, including emergency response planning, radiation protection, and reactor operator training. In the aftermath of the accident, the United States adopted a more precise approach for accident monitoring systems.

Nuclear plants’ Resistance Temperature Detectors (RTDs) are installed in thermowells located away from high radiation environments that exist within the reactor core. As such, the RTDs are not normally exposed to any direct radiation. We have done studies on RTDs that were removed from nuclear reactors after 20 or more years of service and saw little or no apparent radiation damage. Although the RTDs at TMI were exposed to much higher than normal radiation levels, no significant damage occurred in them due to the extensive radiation exposure. They suffered more from moisture intrusion than radiation. In contrast, the core-exit thermocouples were indicating erroneously high temperatures, and others had very low and implausible temperature indications (including negative values) at nearly the same locations. An evaluation of these thermocouples indicated that the inconsistent readings were due to inhomogeneity, which developed in the thermocouple wires after they were exposed to high radiation.

A “worst case” PRT removed from Unit 2 of the TMI reactor four years after the accident was found to conform to the original purchase specifications for calibration, response time, and electrical properties. In addition to verifying the benchmark response time (in 170°F water flowing at 3fps), it was confirmed that the response time of this PRT at full power conditions (550°F and 50 fps) met plant technical specifications. The particular PRT selected for removal on the basis of in situ tests had the lower insulation resistance and heat transfer coefficient of all seven PRTs tested in situ in the hot and cold legs of loops A and B of TMI-2. Since this PRT
met specifications in post-removal tests, we infer that the remainder of the PRTs would also meet specifications.

Although the PRTs apparently were not harmed by the accident, partial shorting of the extension cables during the accident may have caused erroneous temperature readings. The protective conduit connection to the thermometer head was found to be broken on the worst case PRT, allowing steam to enter the connecting terminal housing and the cable during the accident. All but two of the PRTs tested showed evidence of moisture in the measuring circuit.

Process instrumentation components, such as sensors and transmitters in the field, process-to-sensor interfaces, and cables and connectors are all subject to performance degradation as they age. This is particularly true for nuclear power plants, where these components are in harsh environments and are generally exposed to radiation, heat, humidity, vibration, chemical spray, and other taxing conditions which accelerate aging and cause performance degradation. At the same time, these aging nuclear power plants are becoming recognized for their low cost electricity generation, low carbon emissions, and even impressive safety records despite TMI-2 and Fukushima. Consequently, the future increased demand for low cost electricity will rely heavily on the continued uninterrupted performance of these nuclear power assets. As such, the nuclear industry has been directed by the Nuclear Regulatory Commission (NRC) to implement specific maintenance steps to ensure that the process instrumentation channels that feed the safety system of a plant operate properly at all times including under post-accident conditions. In response, nuclear power plants perform periodic surveillances on safety-related instrumentation that involve calibrations, response time measurements, cable testing, and other diagnostics. These steps are in addition to equipment qualification testing that is performed prior to installation of safety-related equipment in a nuclear power plant.

The TMI accident helped stimulate new R&D efforts in Instrumentation and Control (I&C) system design, signal validation, and the role played by human error in understanding and making decisions based on I&C data. Before TMI, Nuclear Power Plants (NPPs) had followed an “event-oriented” philosophy for responding to accidents in which operators first determined the cause of an event before ensuring that safety-critical parameters were not exceeded.
After TMI a “symptom-oriented” philosophy emerged, in which procedures were introduced that compelled operators to first ensure that safety-critical parameters were not violated before they determined causes. TMI-2 and the regulatory guidelines and industry standards it spawned, also hastened the development of methods and policies for more precisely and objectively determining the true aging trends of critical NPP I&C. In the U.S., for example, the NRC issued in 1996 its “maintenance rule,” requiring all NPPs to track the performance of equipment, including process instrumentation, to identify the onset of failures.

The evolution, application, and integration of non-invasive, continuous I&C monitoring techniques eventually formed the body of aging measurement techniques that comprise on-line monitoring systems for managing I&C aging.

Early I&C systems were designed to fulfill both control during normal operation and protection during upset and accident conditions. Safety concerns and regulatory initiatives stemming from incidents like Three Mile Island have led to a separate-system approach to I&C design: one independent system for normal operation and one for protection during accidents. In general terms, I&C consists of the sensors that measure the process; signal conversion and signal conditioning equipment to produce an electrical signal that is proportional to the measured process; the communications media that transmit the measurement; the microprocessors and other integrated circuits that process the signals; and the computers, programmable logic controllers, application-specific integrated circuits, and software that interpret the signals and respond with appropriate control and actuation commands.

Laboratory tests conducted on one resistance thermometer and thermowell removed from TMI-2 showed that neither its calibration nor its time response was adversely affected by the accident or post-accident conditions to which it has been exposed. A broken conduit fitting allowed moisture to enter the extension cables, which affected their insulation resistance.

7. RECOMMENDATIONS

The only failure mechanism in PRTs that resulted from the LOCA in TMI-2 apparently was caused by steam entering the wiring housings, condensing there, and shunting the signals
to an unknown extent. Consequently, it would seem advisable to require that signal cable conduits and connecting housings of PRTs for nuclear plants be (1) tested for ability to withstand the expected vibrations and (2) verified to be hermetically sealed after installation.

Diversity and defense in depth (D3) need to be looked at more closely for beyond design basis event/severe accident scenarios, in particular regarding what kind of diversity is needed and how much diversity is enough. Clearly, the D3 applied to post-accident power sources was inadequate for the extreme scenario that occurred at Fukushima.

Inexpensive, easily implemented sensors (perhaps with wireless capabilities) are needed to enable more extensive measurements of conditions for auxiliary and non-operations systems and components. The spent fuel pools at Fukushima (and many other plants) were under instrumented so it was not possible to monitor the level in the pools or the presence of adequate cooling. The availability of add-on, easily replaceable sensors could enable potentially valuable measurements of critical parameters for event monitoring (e.g., temperature measurements at different depths in the pool) without complicating the cabling infrastructure, increasing the complexity of permanently installed instrumentation systems or imposing significantly burdensome maintenance. Essentially, the capability to readily add (potentially) disposable sensors could promote more extensive condition measurement and greatly improve situation awareness and event management.

Techniques to parasitically extract power from high temperature, high vibration locations should be further investigated for use in providing robust sensors that can function in the event of power loss. In addition, the extremely harsh radiation conditions in several areas of the Fukushima plant suggests the value of radiation hardened electronics to support severe accident monitoring capabilities. Coupling rad-hard electronics with power harvesting techniques could enable sensors to withstand extreme conditions accompanied by significant disruption of normal I&C infrastructures.

Small Business Innovative Research (SBIR) work involving noise and related methods includes continuous high-sampling-rate monitoring of all significant instruments at many plants in hopes that the inevitable off-normal events that occur over time will provide interesting transients for post-event analyses. Inadequate computing capability has resulted in dismissing the value in this data in prior computer generations. Groups focusing on operations might be
interested in such data to supplement existing coarse time interval samplings for reconstructing the sequences of events, finding causes, etc.

OLM techniques enable plants to monitor the aging of their installed I&C systems while the plant is operating. These techniques include low- and high-frequency methods that use existing sensors, such as noise analysis; methods based on test or diagnostic sensors; and methods based on active measurements made by injecting a test signal into sensors.

A combination of improved encryption, cyber-security related standards (including those issued by International Electrotechnical Commission (IEC), Institute of Electrical and electronics Engineers (IEEE), and International Society of Automation (ISA)), vulnerability analyses, detailed procedures and installations, and intensive training will likely solve many of these potential security concerns. Due to the rapid pace of evolution in digital I&C, obsolescence, rather than aging, is the primary concern.

Manufacturers type-test representative transmitters and generically qualify them under simulated process conditions in a laboratory.