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Memo For: Rich Smith, DSB

From: P. Larkins, TIDC
W. Olin, TIDC

Subject: Transmittal of Speeches

Attached are (two) copies of a speech to be sent to the PDR and TERA. We have filed the 426 Form.

P. Larkins

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THERMAL HYDRAULICS

THERMAL HYDRAULICS: CHANGES IN METHODS IN LIGHT OF TMI

Sponsored by Thermal-Hydraulics Technical Group

All Papers Invited

1. The Frequency and Consequences of Small Loss-of-Coolant Accidents, *Zoltan R. Rosztoczy (NRC)*

Small breaks in the coolant systems of nuclear reactors have not, until recently, been subjected to detailed analytical study comparable to the attention devoted to large breaks. Typically, small breaks have been analyzed down to the smallest break size that would produce system depressurization without core uncovery. Although the analyses, in general, predicted acceptable fuel cladding temperatures, they considered only single active failures, and did not provide sufficient information for operator training and for the preparation of plant emergency procedures. Furthermore, only the consequences of small loss-of-coolant accidents were evaluated; little attention was paid to their frequency.

Following the accident at Three Mile Island (TMI) the emphasis on small breaks increased greatly. Licensees of operating power plants were requested to provide information on both the frequency and consequences of small breaks. Suppliers of nuclear reactors were required to provide additional guidance to the operators of the plants through the issuance of operating guidelines. The licensees had to initiate plant modifications, retrain the operators, and modify the emergency procedures based on the new information. The purpose of the effort was to substantially reduce the likelihood for accidents of the type that occurred at TMI.

The new evaluation of small breaks produced one major surprise and focused attention on a number of shortcomings. Throughout the years, applicants and licensees have maintained that the worst break size was a large break with a relatively low probability of occurrence. The new analysis revealed that, without any restriction on the operation of the reactor coolant pumps, the small breaks are limiting and the calculated cladding temperatures are well in excess of the 2200°F limit specified by the ECCS Acceptance Criteria. Immediate action was taken to eliminate this possibility by requiring prompt tripping of the reactor coolant pumps in case of a loss-of-coolant accident. While tripping of the pumps is not an ideal solution, it solved the immediate problem. A better, more permanent solution should be forthcoming from the industry.

The frequency of relief valve challenges was evaluated based on operating experience and also based on available plant safety evaluations. In all but one case, there was good agreement between the two evaluations. The results show that BWRs designed by GE are expected to experience approximately 15 valve openings per reactor year, and that PWRs designed by Westinghouse, CE, and B&W are expected to experience 2, 1, and 0.2 valve openings per reactor year, respectively. The relatively low number associated with the B&W design results from the design changes initiated after the TMI-2 accident (changes in relief valve and reactor trip setpoints; anticipatory trips on loss of feedwater and on turbine trip), and is supported by the operating experience accumulated since the change. The observed failure rate per challenge varies between $\frac{1}{20}$ and $\frac{1}{60}$. Consequently, a stuck-open relief valve is the most probable cause of a small loss-of-coolant accident. The probability of this event is at least an order of magnitude higher than the probability of a small pipe break (10^{-3} per reactor year) for Westinghouse and CE plants and two orders of magnitude higher than a small pipe break for GE plants.

The consequences of a small break depend on break size. A typical stuck-open PWR relief valve is equivalent to a 1.4-in.^{2} break. With a single failure assumption, this break size does not result in reactor core uncovery. BWRs have significantly larger valves, which are ~14 in.^{2} This break size with a single failure partially uncovers the reactor core. Other design features such as a tight upper-head design, as used by Westinghouse, can reduce the water level in or above the core because of slow upper-head drainage.

The evaluation also showed that protection in the case of small breaks often requires prompt operator action. Design changes were required to automate some of the operator actions; for example, tripping of the reactor coolant pumps. The information obtained from the small-break evaluation was also used to improve operator training and to upgrade plant emergency procedures. Special attention was given to requirements permitting termination of high-pressure safety injection.

The uncertainty of the calculations has not yet been evaluated. It could be rather large; for example, equivalent to a 4- or 5-ft level difference in the core. Should this be the case, a predicted 5-ft uncovery could actually be a 10-ft uncovery. As we learned from TMI, deep uncoveries for extended time intervals are unacceptable.

In addition to the single failure assumption, two degraded conditions were also evaluated: loss of auxiliary feedwater (RCIC in the case of BWRs) for an extended period of time, and loss of natural circulation for an extended period of time. PWRs with a high cutoff head, high-pressure injection system (HPI) were shown to have sufficient time (at least 20 or 30 min) to initiate emergency core cooling and were protected for these cases. BWRs needed operator action within a few minutes to prevent core uncovery. The time available was found to be insufficient and a design change to automatic initiation of the automatic depressurization system (ADS) was required. PWRs with low cutoff head HPI are possible unprotected for these events. Present emergency procedures require opening of relief valves and initiation of HPI. It is not known whether the system would sufficiently depressurize to permit HPI flow.

In summary, the risk associated with small breaks was found to be larger than previously reported. This was mainly the result of high relief valve challenge and failure rates, and a faster than expected mass depletion caused by the running reactor coolant pumps. New requirements on tripping the pumps solved the latter problem. Corrective measures taken on B&W plants produced a significant reduction in the frequency of small loss-of-coolant accidents. GE, Westinghouse, and CE plants should take appropriate steps to improve their respective designs. Furthermore, PWRs with low HPI cutoff head are possibly not protected for the extended loss of auxiliary feedwater or natural circulation. Appropriate design changes should be considered. Finally, the current state-of-the-art of small loss-of-coolant accident evaluation is unsatisfactory. Uncertainties are large, and experimental evidence is rather limited. Natural-circulation tests (two-phase) and integral small-break tests are needed to verify the analytical methods and results.

2. Post-TMI Thermal-Hydraulic Aspects of LWR Safety Analysis, *B. R. Sehgal, R. B. Duffey, W. B. Loewenstein (EPRI)*

The accident at Three Mile Island (TMI) has spawned examinations of the practices and attitudes followed in LWR safety and licensing by a number of bodies. Some of these have been published and are receiving attention at various levels. In this paper we will concern ourselves only with the thermal-hydraulic aspects of issues in LWR safety emphasized since the TMI accident.

Perhaps it is worth repeating that the pre-TMI LWR safety evaluations were largely dominated by those concerning the hypothetical large-break LOCA design basis accident. Thus the main R&D efforts were focused toward understanding and predicting the thermal-hydraulic behavior during the various phases of the large LOCA; i.e., a very rapid blowdown, ECC injection and bypass, lower plenum refill, and core reflood. The peak cladding temperature and the cladding oxidation were the two main safety limits and the accident lasting ~3 min, in fact, is a minor one because of the engineered safeguards. Regardless of what form an accident may take, heat must be transported from the fuel to an ultimate sink for extended periods. An analysis of the path that the heat must be transported through gives a good indication of the effectiveness of the reactor and safety heat transport systems during an accident.

The degree of reliability and integrity demanded of the various heat transport paths can be determined independently from an overall reactor risk analysis.

A whole class of accident scenarios, previously considered but not particularly emphasized in LWR safety evaluations, should be considered in light of the TMI accident. These accidents may be broadly classed as degraded-heat-removal accidents. The duration of such accidents may be on the order of hours or days and operator action plays a crucial role throughout.

The degradation in heat removal may occur due to decrease in (a) the core-to-coolant heat transfer; e.g., when the core gets uncovered, (b) degradation in core geometry, when blockages occur, (c) the system heat transport; e.g., when the flow from the core to steam generators is reduced, (d) system coolant inventory; e.g., in swell break LOCA, and (e) the heat sink; e.g., when the secondary side of steam generators becomes empty.

The thermal-hydraulic phenomena of interest include (a) mass flow-through relief valves, pipe cracks, valve seals, etc., (b) phase separation and movement of a two-phase level, (c) fuel cladding dryout and DNB, (d) post-DNB heat transfer, (e) vapor generation rate, (f) steam cooling, (g) liquid entrainment in vapor, (h) quenching, (i) condensation heat transfer and condensation-induced pressure changes, (j) hydrogen generation, (k) heat transfer reduction due to the presence of noncondensible gases, (l) reflux boiling, (m) heat transfer in degraded core geometry, (n) natural-convection flows in the heat transport system with and without presence of noncondensibles, (o) mass transfer characteristics of the noncondensibles, (p) steam generator secondary-side flow distribution, two-phase level, boilout, (q) heat rejection in suppression pools, (r) heat rejection in containment through sprays, (s) containment pressures, (t) ultimate and long-term heat rejection.

Analysis of the possible transient scenarios will require the understanding of the various thermal-hydraulic phenomena mentioned above. A start has been made at EPRI by conducting large- and small-scale experiments with analytical modeling.[1-4] It was found that the primary and secondary heat removal capabilities are coupled through the two-phase levels in the core and in the secondary side of the steam generator.[5] The results of the analysis of the TMI accident[6,7] showed that reasonably satisfactory post-test predictions can be made; however, many assumptions had to be made and the computer running times were very large. We are presently in the process of initiating tests on full-scale relief and safety valves to obtain mass discharge data.

Reliable prediction of the actual reactor conditions during such long-term transients is the challenge we face. It will require the integration of the information obtained from experimental and analytical studies into prediction codes. It will require the development of stable, accurate, and fast-running system analysis codes, which perhaps is the greatest challenge, since the thermal hydraulic phenomena of interest involve both mechanical and thermal nonequilibrium between phases. The recently developed system analysis codes, e.g., RETRAN, TRAC, and RELAP-5, having models for such nonequilibria, generally require large computer times as well as the scarce information for the interphase exchange processes during different parts of such transients. We at EPRI have started studies on efficient and stable numerical methods and models for the nonequilibrium processes, besides the experimental studies mentioned above. The goal is to develop the required code(s) and to qualify them with as much experimental data as possible at different scales.

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1. K. H. SUN, R. B. DUFFEY, and C. M. PENG, "A Thermal-Hydraulic Analysis of Core Uncovery," ASME-AIChE National Heat Transfer Conference, Orlando, FL (1980).

2. S. P. KALRA, R. B. DUFFEY, and G. ADAMS, "Loss of Feedwater Transients in PWR U-Tube Steam Generators," ASME-AIChE National Heat Transfer Conference, Orlando, FL (1980).

3. U. ZVERIN, P. R. JEUCK, C. W. SULLIVAN, R. B. DUFFEY, and P. BAILEY, "Experimental and Analytical Studies of a PWR Natural Circulation Loop," ANS Topical Meeting on Thermal Reactor Safety, Knoxville, TN (1980).

IMPACTS OF TMI ON REGULATORY REQUIREMENTS – I

Sponsored by Power Division

All Papers Invited

1. NRC Action Plans, *Harold R. Denton (NRC)*

The accident at Three Mile Island has had unprecedented immediate and long-range consequences. Many organizations were assembled with the sole purpose of reviewing the accident, the response of utility, industry, state, and federal personnel, and identifying concerns of potential safety significance that need to be resolved by modifying the plant or procedures, or demonstrating by analysis that corrective actions are not required. Among the several special NRC groups established were "The Bulletins and Orders Task Force," "The Lessons Learned Task Force," "Unresolved Issues Group," "Emergency Preparedness Task Force," and the "Special Inquiry Group." President Carter also established a Commission under the chairmanship of John G. Kemeny.

These groups developed recommendations that would directly or indirectly improve reactor safety and reduce the risk to the public. After the report of the President's Commission on Three Mile Island was released, the NRC staff prepared an analysis of the recommendations and identified ongoing staff efforts. We then developed, from the numerous recommendations of the various Task Forces, a plan which was to integrate all the TMI-related recommendations. The resulting document, NUREG-0660, "Action Plan for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident," has four chapters.

Chapter I deals with plant operators, operator training, and the use of operating experience. The second chapter includes reliability and risk assessment, and design improvements. The third chapter deals with emergency plans and improving the understanding among state, local government, and utilities. The last chapter deals with internal NRC reorganization and utilization of NRC staff.

The draft action plan was first published in early December 1979. Since then it has undergone refinement, such as assuring recommendations in the Special Inquiry have been considered. We will be incorporating these TMI-related items into the existing reactor review procedures. When this is accomplished, the licensing basis for nuclear power reactors will have been improved and the overall risk to the public reduced.

2. Impact of TMI—A Change in Attitude, *L. M. Mills (TVA)*

This past year seems to have been "the year of task forces and reports." We have become familiar with Kemeny, Lessons Learned, Bulletins and Orders, Action Plans, and Rogovin. In addition, many involved in the nuclear industry have performed self-evaluations in quest of a safer and more acceptable nuclear program.

We, in TVA, believe the future of the nuclear industry in this country is much more dependent on what we do, and require of ourselves, than on requirements put forth by any regulatory agency. We have seen many new NRC requirements related to physical plant modifications, operator training, additional plant personnel, and upgrading of plant and off-site management. All of these are important; however, the most significant impact on nuclear power in this country is the change in attitude.

This change in attitude can best be exemplified by contrasting the current NRC-industry method of resolving problems with the method of the past. Until recently, the scenario went something like this: The NRC identified a problem and set a requirement; industry reacted in many cases by taking the position that existing designs had sufficient margin or that NRC's requirements had little or no technical merit. After much argument back and forth, industry conceded and proposed a fix which met minimum NRC requirements and which would be implemented at some convenient future refueling outage. NRC then applied the "ratchet" and industry reluctantly implemented the fix. In contrast, in the post-Three Mile Island era, problems have been identified by both industry and NRC; fixes have been volunteered by industry as well as required by NRC and, although there are still some throwbacks to the old way, in an increasing number of situations industry has gone beyond NRC requirements both in intent and in spirit. We expect a similar change in attitude from the regulators. The NRC technical reviewers, in addition to NRC top management, must recognize that it is possible to provide a completely satisfactory response on the first try.

There are certain aspects of the nuclear safety program that can best be served by industry response as a whole. Good examples of this are the formation of the Institute of Nuclear Power (INPO) and the Nuclear Safety Analysis Center (NSAC). Neither of these institutions was a result of imposed requirements, but were formed and financed by the industry on its own initiatives.

Even more important are the attitude and initiative of each utility that operates a nuclear unit. I will devote the remaining portion of my paper to items that I believe reflect TVA's current attitude and display our desire to be innovative in the ever important area of nuclear plant safety.

Before the NRC issued its first post-TMI requirements, the TVA Board of Directors established a task force to make recommendations relating to TVA's nuclear program in light of the Three Mile Island incident. One of the most significant improvements recommended was the establishment of a Nuclear Safety Review Staff. This staff, which has been functioning for about a year, is completely independent of our operating, design, and construction divisions. Reporting directly to the General Manager and Board of Directors, the staff serves as a mechanism for keeping the highest level of management informed of nuclear safety matters.

A separate nuclear operating division has been established in recognition that nuclear plants must be operated under a different set of rules than fossil and hydro plants. In addition, TVA has established a Nuclear Engineering Branch in the Division of Engineering Design. Personnel with nuclear backgrounds have been placed in high-level management positions in both the operating and design organizations. These organizational and personnel changes enhance the awareness and ensure the consideration of nuclear safety throughout the TVA organization. Last December, TVA volunteered to

ASSESSMENT OF PLANT INSTRUMENTATION: MEETING THE OPERATOR'S NEEDS

Cosponsored by Reactor Operations Division and Human Factors Technical Group

1. Nuclear Power Plant Saf .y in the Man-Machine Interface, *R. M. Satterfield, L. Beltracchi (NRC), invited*

The automatic portion of a nuclear power plant safety system consists of the reactor trip system and the engineered safety features. These systems act automatically in response to off-normal events to ensure their quick termination and to mitigate their effects.

The reactor operator is another important component of the plant safety system. The reactor operator is required to assess plant safety by evaluating control panel displays and to take corrective action on detection of an approach to unsafe operating conditions. He also must assist in mitigating the effects of transients and accidents, should they occur. The capability of the operator to perform these actions effectively is a function of the operator's training, the adequacy of the control panel to depict the status of the plant, and the availability of operator aids to assist in the detection, analysis, and correction of anomalous operations.

The NRC has recently developed requirements which, when implemented, should enhance the capability of the operator as a component of the plant's safety system. These include the use of human factors engineering to improve control room design, and improvements in control room displays to facilitate the operator's capability to respond to upset conditions. Research is also under way to develop computer-based operator aids such as disturbance analysis systems.

2. Improving the Operator's Role in Future Control Room Designs, *D. G. Cain, A. B. Long, L. C. Oakes (EPRI), invited*

Present-day nuclear reactors rely heavily on human operators for both startup and steady-state operation. In addition, operators are responsible for bringing the reactor to a safe shutdown condition following system transients when such transients are outside the capability of the automatic control devices.

Data have been collected over the past few years which suggest that limitations on human response capabilities have not been properly considered as a possible constraint on system performance, especially during unusual occurrences or large off-normal excursions from plant steady-state conditions.

OPERATOR LIMITATIONS

Studies have been made which attempt to quantify human response time and reliability. Fullwood and Gilbert[1] report that the optimum data transmission rate for human comprehension is approximately 1 bit/s, at which point the minimum error rate of 5.5×10^{-3} is achieved. At a rate of ~45 bits/s the probability of error reaches D. The proposed ANSI Standard N-660 (Ref. 2) uses a value of 10 min as the minimum time for human thoughtful response. It should be noted that the 10-min response is a minimum time. Some off-normal conditions may never be discovered by a particular observer, giving rise to a possible infinite response time. Even though one may question the uncertainty associated with these data, they nonetheless suggest that humans are bandpass limited to relatively low frequency even for repetitive functions and that their response time deteriorates rapidly when faced with new situations requiring reasoning.

OPERATOR ASSISTANCE

From the previous observations it would seem to be worthwhile to try to augment the operator's capabilities in areas where he is already performing near the limit of his ability. An area having potential for improving operator performance that comes immediately to mind is the use of additional automation. Extensive automation can now be easily accomplished because of the availability of inexpensive components for which software development costs are reasonable. Also, complex sequences can now be automated reliably. However, if one attempts to carry the concept of automation to the extreme, it soon becomes obvious that the success of such an effort requires that all possible accident scenarios be accurately anticipated in advance. Thus, complete automation suffers from the same lack of *a priori* information that has severely limited the effectiveness of the event approach now used by operators in arresting the progress of accidents. Basically, neither can wholly succeed without a precise understanding of how an accident sequence is going to proceed before it happens.

Despite the inherent shortcomings of a totally automated approach, a new look should be taken to see where its use might be employed to advantage in particular applications. For example, more automation of predetermined sequences and procedures and in fast transients can be a good first step toward freeing the operator from manual operation in preparing him for a more sophisticated role.

REAL-TIME SYSTEMS ANALYSIS

The foregoing discussion suggests there is a fundamental problem that must be solved if significant improvement is to be exepcted in the man-machine interface. Namely, we must develop a system capable of providing the operator with high-quality information during upset conditions that may never have been previously anticipated and analyzed. The system must have predictive capability to aid the operator in his decision-making process so he will know in advance of taking corrective action which of the several options available to him will produce success.

The approach outlined here is actually a variation on limited form of automation. It would provide for an automated or semiautomated means of analyzing and integrating plant data. This is presently done "manually" in the operator's head, handicapped by the operator's rather modest information bandpass capability.

TABLE I

Variables Recorded at Time of TMI-2 Accident

1.	Nuclear Instrumentation Power Range, Channel 5.
2.	Reactor Coolant Loop A, T_h, T_c and Flow
3.	Reactor Coolant Loop B, T_h, T_c and Flow
4.	Pressurizer Level
5.	Makeup Tank Level
6.	Pressurizer Spray Valve Position
7.	RC Drain Tank Pressure
8.	Reactor Coolant System narrow range pressure (1600-2500 psig)
9.	Reactor trip contacts
10.	Turbine header pressure
11.	Feedwater temperature
12.	OTSG A startup and operate levels
13.	OTSG B startup and operate levels
14.	Loop A and B feedwater flows
15.	OTSG A & B steam pressure
16.	Turbine trip contacts

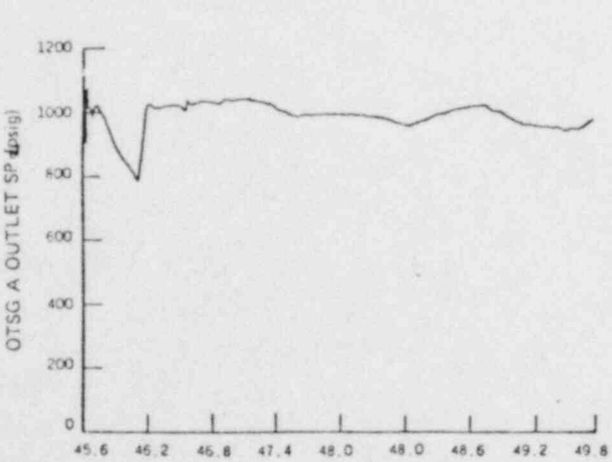

Fig. 2.

continuous information about transients and was not dependent on the standard plant computer. As a result, a Reactimeter[c] system has been part of the startup of every B&W plant and some version of data acquisition used during operation. A typical Reactimeter[c] cross plot is shown in Fig. 2.

In retrospect, several improvements and extension of the Reactimeter[c] would have reduced remaining ambiguities associated with TMI-2 events. Two of these are: faster transmission of the information to the point of usage in chart form, and recording a greater number of parameters. A recent tabulation of desired parameters for the RECALL system now available from B&W identified 125 of these. The extension of permanent memory storage capability and transmission techniques should add greatly to the use of such systems.

The expanded reactimeter (RECALL system) will include a highly reliable recording device designed to be in operation at all times to provide endless loop "flight recorder style" recording of the data. The objective here is to ensure transient recording as opposed to relying on a fortuitous stroke of luck, such as having the reactimeter connected and operating as it was at TMI-2. The paper will describe other features of the expanded reactimeter data recording device.

3. Lessons Learned from TMI In-Reactor Instrumentation: An NRC Viewpoint, *John C. Voglewede (NRC)*

During the TMI-2 accident, a condition of inadequate core cooling existed and was not recognized for many hours. This resulted from a combination of factors including insufficient indicating range for existing instrumentation, unfavorable location of instrument readout, and perhaps insufficient instrumentation. The Nuclear Regulatory Commission staff has analyzed each of these perceived deficiencies and has made a number of new regulatory requirements and technical recommendations.

The regulatory viewpoint of the lessons learned from the in-reactor instrumentation at TMI-2 is similar to that of the owner, the reactor vendor, and the nuclear industry as a whole. That is, the in-reactor instrumentation played a prominent role during the first few hours of the accident as well as the period of time leading to cold shutdown of the plant. Because of this, the NRC Lessons Learned Task Force concluded that the as-designed and field-modified instruments at TMI-2 provided sufficient information to indicate reduced reactor coolant level, core voiding, and deteriorated core thermal conditions.

Three Mile Island was one of the best instrumented reactors in operation because of the large number of in-core thermocouples and other features. Unfortunately, these positive features did not result in the prompt recognition of, and the rapid recovery from, a condition of inadequate core cooling.

In response to this and other findings from the Three Mile Island accident, the NRC staff initiated a number of short-term requirements, based on "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578). The status of the industry response concerning in-reactor instrumentation is summarized in this presentation, and the Metropolitan Edison Restart Submittal on Three Mile Island Unit 1 is used as an example of the industry response.

Metropolitan Edison, like other utilities, is required to review its present in-reactor instrumentation to determine how the condition of reduced coolant level and core voiding can be detected. The existing instruments examined by Metropolitan Edison measure coolant temperature, flow, and pressure as well as neutron flux and motor current of the coolant pumps. The staff supports such diversity. However, these instruments existed at TMI-2 at the time of the accident. New instruments, such as the PWR vessel coolant level detector, have also been required and long-term improvements in other areas are being actively pursued by the staff. The results of these efforts, however, have not yet been implemented.

With regard to existing instrumentation, the staff has noted that many instruments at TMI-2 lacked sufficient range of indication. A notable example was the range of the core exit thermocouples. As a result, Metropolitan Edison has extended the indicating range of these thermocouples as well as the indicating range of the reactor outlet resistance temperature detectors (RTDs) in the Unit 1 facility. This was done without replacing the sensors. Access to the in-core thermocouple signals from outside the containment building has also been provided. This feature did not exist on Unit 1 at the time of the Unit 2 accident. The indicating range of these instruments now encompasses the physical limitations of the sensor rather than the expected response of the device during normal operation. The change from expected to extended instrument range reflects the staff position in the proposed revision to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Access Plant Conditions During and Following an Accident," issued December 4, 1979.

All PWR owners are also required to install a saturation meter in the control room to provide an indication of the degree of subcooling in the primary coolant. However, the use of in-core thermocouples, extended range reactor outlet temperature RTDs, and the new saturation meter are enhancements of existing instruments. The staff previously concluded that reduced coolant level and the existence of core voiding could be determined with these instruments. Longer term improvements in the instrumentation will make this determination easier, provided the operator is aware of the available information and interprets it correctly.

An important lesson learned from TMI-2 on the subject of in-reactor instrumentation is that the operator must be made aware of the available information and must know how to interpret it correctly. Marked improvement in an operator's ability to quickly recognize a condition of inadequate core cooling, and his ability to act on this information, will, in my judgment, lie more with improvement to the operator's training and instruction than with improvement of the instrumentation. However, both approaches have been required by the NRC to ensure that conditions of inadequate core cooling do not go undetected in the future.

4. PWR Instrumentation: Recommendations for Improvement from TMI-2 Experience, *R. L. Shepard, J. L. Anderson (ORNL)*

Instruments in the Three Mile Island plant, Unit 2 (TMI-2), were inadequate to unequivocally indicate critical plant parameters to the plant operators during the accident in March 1979. As a result, the operators were uncertain about the level of coolant in the reactor vessel, the total inventory of coolant in the system, whether there was boiling in the core, the volume and composition of the gas phase above the core, and whether coolant was flowing in feedwater and relief lines during the onset of the accident as well as during the attempts to regain control and stabilize the plant during the postaccident period. The instruments had been judged adequate to license the TMI-2 plant and to operate it under normal conditions. However, during the accident, the information from these instruments was either erroneous or conflicting or both. The instruments were not required to meet Regulatory Guide 1.97 (Ref. 1) for postaccident monitoring, and, indeed, many failed later in the accident sequence owing to effects of moisture or radiation. Uncertainties in the readings from (and the condition of) in-core, self-powered neutron detectors and thermocouples, differential-pressure gauges used to measure coolant level in the pressurizer, primary-leg resistance thermometers, and valve position indicators limited efforts to maneuver the reactor from an accident condition to a safe shutdown. In some cases, the instrument readings could not be verified because provision for the use of *in situ* calibration verification had not been made in the installation and cabling of the sensors.

In response to the alarming and costly TMI-2 accident, many organizations began to restudy and reevaluate many factors to which the severity of the accident may be related: containment, plant equipment, instruments, procedures, operator training, plant supervision and regulation, control room layout, and man-machine interaction. Some programs acknowledge the need demonstrated in the post-accident recovery work at TMI-2 for improved instrumentation and the value of diagnostic and surveillance methods.

We submit our view that the development and qualification of improved instrumentation are of paramount importance in response to TMI-2, and are prerequisites to meeting several of the needs identified above. An operator cannot be expected to assess properly the severity of an upset condition or to respond effectively unless the critical plant parameters are displayed accurately and in a comprehensible form. Equally important, the value of a sophisticated information display system can be no greater than permitted by the validity of its input information. Verified plant data are essential to effective man-machine interaction and therefore to skillful plant operation. Verified data can be obtained, we believe, only by the application of a higher state-of-the-art of instrumentation design philosophy; namely, (a) direct measurement of some plant parameters that are now merely inferred, (b) development or application of developed methods of sensor signal verification, and (c) application of a computerized model of the plant process for displaying to the operator, in real time, information about the status of the plant.

To be more specific, we suggest six approaches that may be undertaken effectively for improving plant instrumentation by retrofit or for new designs.

1. Design instruments for direct measurement of plant parameters that are now inferred from other measurements; for example, in-core liquid-level detectors and relief-valve flow indicators.

2. Develop instruments for diverse measurement of critical, safety-related plant parameters where redundant sensors susceptible to common-mode failure are now employed, such as resistive or ultrasonic level detectors for the pressurizer or a combined thermocouple-resistance thermometer for temperature measurement of the primary coolant.

3. Develop multifunctional instruments that could measure more than one process condition, such as a resistance thermometer that would also serve as a level detector for the pressurizer.

4. Improve instrument sensors, cables, connectors, and transmitters so they would operate reliably under environmental conditions of high moisture, temperature, and radiation exposure. Reconsider the location of critical instruments in the reactor containment.

5. Develop, qualify, and apply methods for *in situ* verification of the continuity, isolation, calibration, and response of plant instruments. Some verification methods already developed for plant thermometers to meet requirements of Regulatory Guides 1.118 (Ref. 2) and 1.105