
Task I.D: Control Room Design (Rev. 8)

The objective of this task was to improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them.

ITEM I.D.1: CONTROL ROOM DESIGN REVIEWS

This item was clarified in NUREG-0737,¹ requirements were issued, and MPA F-08 was established by DL/NRR for implementation purposes.

ITEM I.D.2: PLANT SAFETY PARAMETER DISPLAY CONSOLE

DESCRIPTION

This item was clarified in NUREG-0737,² requirements were issued, and MPA F-09 was established by DL/NRR for implementation purposes. Generic Letter No. 82-33³ transmitted Supplement 1 to NUREG-0737⁴ to further clarify the TMI action items related to emergency response capability, including Item I.D.2. This Supplement 1 included the fundamental requirements for emergency response capability from the wide range of regulatory documents issued on the subject. It was written at the conceptual level to allow for a high degree of flexibility in scheduling and design. In recognition of the interrelationships among the action items addressed in Supplement 1,⁵ the staff made allowance for each licensee to negotiate a reasonable schedule for implementing its emergency response capability. However, the staff identified the SPDS as an improvement to the control room that should not be delayed by progress on other initiatives.

CONCLUSION

The staff evaluated licensee/applicant implementation of the SPDS requirements at 57 units and found that a large percentage of designs did not satisfy requirements identified in Supplement 1 to NUREG-0737.⁶ Generic Letter 89-06⁷ (enclosing NUREG-1342⁸) was issued to inform licensees of the staff's findings to aid in implementing SPDS requirements. NUREG-1342⁹ describes: (1) methods used by some

¹ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

² NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

³ Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits from U.S. Nuclear Regulatory Commission, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982. [\[ML031080548\]](#)

⁴ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

⁵ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

⁶ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

⁷ Letter to All Licensees of Operating Plants, Applicants for Operating Licenses, and Holders of Construction Permits from U.S. Nuclear Regulatory Commission, "Task Action Plan I.D.2—Safety Parameter Display System—10 CFR "50.54(f)—(Generic Letter No. 89-06)," April 12, 1989. [\[ML031200729\]](#)

⁸ NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," U.S. Nuclear Regulatory Commission, April 1989.

⁹ NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," U.S. Nuclear Regulatory Commission, April 1989.

licensees/applicants to implement SPDS requirements in a manner found acceptable by the staff; and (2) design features that the staff found unacceptable, with the staff's reasons. The information in NUREG-1342¹⁰ did not constitute new requirements; Supplement 1 to NUREG-0737¹¹ contains NRC's requirements for SPDS.

ITEM I.D.3: SAFETY SYSTEM STATUS MONITORING

DESCRIPTION

Historical Background

This TMI Action Plan item¹² recommended that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47¹³ to install a bypass and inoperable status indication system or similar system.

Safety Significance

Implementation of a well-engineered bypass and inoperable status indication system could provide the operator with timely information on the status of the plant safety systems. This operator aid could help eliminate operator errors such as those resulting from valve misalignment due to maintenance or testing errors.

Possible Solutions

A study of existing industry (nuclear and others) practices could be undertaken to evaluate possible methods/ systems for verifying correct system alignment. In conjunction with this, a study of failures of systems due to pump or valve unavailability could be undertaken. Based on the results, a requirement to backfit or not backfit Regulatory Guide 1.47¹⁴ (or a revision thereof) would be set forth.

PRIORITY DETERMINATION

Assumptions

If the system is integrated with the overall control room, then it could be expected that it would reduce operator error which, in turn, will lower the risk associated with operation of the monitored safety systems. In some plants, this "new" system could result in a modest but significant reduction in operator error during an emergency whereas, in others, the system could have no discernible effect. An average of about 2% was applied to all operating plants; plants that were not yet licensed or were undergoing licensing were committed to Regulatory Guide 1.47.¹⁵

In an analysis of this issue performed by PNL,¹⁶ Oconee-3 was selected as the representative PWR. It was assumed that the fractional risk and core-melt frequency reductions for a representative BWR (Grand Gulf-1) were equivalent to those calculated for the representative PWR.

¹⁰ NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," U.S. Nuclear Regulatory Commission, April 1989.

¹¹ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

¹² NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

¹³ Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]

¹⁴ Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]

¹⁵ Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]

¹⁶ NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

Frequency Estimate

The reduction in core-melt frequency (F) for Oconee-3 was calculated to be $8.7 \times 10^{-7}/\text{RY}$, based on adjustments to the risk equation parameters affected by implementation of the possible solution and then a calculation of a core-melt frequency and comparison to the base case core-melt frequency. Based on a scaling calculation,¹⁷ the frequency reduction (F) for Grand Gulf-1 was $3.9 \times 10^{-7}/\text{RY}$.

Consequence Estimate

Assuming WASH-1400¹⁸ release categories, a typical midwest site meteorology, and a uniform population density of 340 people per square-mile, the reduction in public risk was calculated to be 5.9 man-rem/RY for Oconee-3 and 7.1 man-rem/RY for Grand Gulf-1. For 47 PWRs and 24 BWRs with average remaining lives of 28 years and 25 years, respectively, the total public risk reduction was calculated to be 1.2×10^4 man-rem.

Cost Estimate

Industry Cost: Installation costs (including labor and equipment) were estimated as follows:

	Equipment	Cost
(a)	Cable (30 miles @ \$6.00/100-Lft)	\$ 9,500
(b)	Electrical Penetration Limitations	300,000
(c)	Cable tray and Additional Termination	10,000
(d)	Intermediate Logic Panel	100,000
(e)	Control Room Alarms/Indications	10,000
	Total:	\$429,500
	Other	Cost
(a)	Design Labor (12 man-months)	\$ 75,000
(b)	Installation Labor (17 man-months)	100,000
(c)	QA	40,000
	Total:	\$215,000

Based on the above costs, the implementation cost was estimated to be \$644,500/plant. Maintenance of the solution was estimated to require 1 man-week/ plant; at \$1,000/RY, this amounted to a cost of \$1.9M. Thus, the total industry cost for implementation and maintenance of the possible solution was estimated to be \$48M.

NRC Cost: Development of the resolution was estimated to take 0.5 man-year. Review and implementation of the solution was estimated to take 4 man-weeks/plant. Therefore, the NRC cost was estimated to be \$0.6M.

Total Cost: The total industry and NRC cost associated with the possible solution was $$(48 + 0.6)\text{M}$ or

¹⁷ NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

¹⁸ WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Atomic Energy Commission, October 1975.

\$48.6M.

Value/Impact Assessment

Based on an estimated public risk reduction of 1.2×10^4 man-rem and a cost of \$48.6M for a possible solution, the value/impact score was given by:

Uncertainty

Because the estimate of the value/impact score relied heavily on the estimated value of the possible reduction in human error, there could be wide variance in the effective improvement.

Additional Considerations

- (1) This issue could be most effectively resolved in conjunction with Item I.D.1 which addressed control room design review. This issue was not explicitly included in the requirement for Control Room Design (Item I.D.1) which was implemented in accordance with SECY-82-111¹⁹ and a letter²⁰ issued to licensees of all operating plants.
- (2) Resolution of this issue was expected to provide a reduction in safety system unavailability due to the contribution of maintenance and testing.
- (3) DHFS/NRR contracted with various groups to study this issue.^{21,22} These studies were expected to better define the assumptions (for risk reduction) used in the calculation and provide better data for a benefit/cost study to determine implementation.

CONCLUSION

Based on the estimated public risk reduction and the value/impact score, this issue was given a medium priority ranking (See Appendix C) and RESOLVED with no new requirements.²³ In an RES evaluation,²⁴ it was concluded that consideration of a 20-year license renewal period did not affect the resolution of the issue.

ITEM I.D.4: CONTROL ROOM DESIGN STANDARD

DESCRIPTION

Historical Background

This TMI Action Plan²⁵ item emphasized a need for guidance on the design of control rooms to incorporate human factors considerations.

Safety Significance

Control rooms and control panels which incorporate human factors considerations can greatly enhance

¹⁹ SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]

²⁰ Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits from U.S. Nuclear Regulatory Commission, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982. [ML031080548]

²¹ NUREG/CR-2417, "Identification and Analysis of Human Errors Underlying Pump and Valve Related Events Reported by Nuclear Power Plant Licensees," U.S. Nuclear Regulatory Commission, February 1982.

²² NUREG/CR-3621, "Safety System Status Monitoring," U.S. Nuclear Regulatory Commission, March 1984.

²³ Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-I.D.3, 'Safety System Status Monitoring,'" August 20, 1993. [9502070295]

²⁴ Memorandum for W. Russell from E. Beckjord, "License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994," May 5, 1994. [9406170365]

²⁵ NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

operator performance. This could contribute to a reduction in operator error and, therefore, a potential reduction in the frequency of core-melt accidents.

Possible Solution

An NRC Regulatory Guide endorsing industry standard(s) could be developed with the intention of providing: (1) guidance for the design of control rooms; and (2) the evaluation criteria for use in the licensing process.

PRIORITY DETERMINATION

Assumptions

From the representative PWR (Oconee-3) and BWR (Grand Gulf-1), those parameters in the risk equations requiring direct operator actions were considered affected, i.e., it was assumed that the probability of operator error for these parameters was decreased by 3% based on resolution of the issue.²⁶ It was assumed that only plants to be licensed beyond 1986 would be affected.

Frequency Estimate

The affected accident sequences and associated base case frequencies were determined. From these frequencies, the new base case core-melt frequencies of 3.1×10^{-5} /RY and 6.1×10^{-6} /RY were calculated for PWRs and BWRs, respectively. The affected parameters were adjusted by 3% and the frequencies of the associated sequences and release categories were determined. New overall core-melt frequencies were then determined: 3.01×10^{-5} /RY for PWRs and 5.95×10^{-6} /RY for BWRs. Thus, the reduction in core-melt frequency (due to issue resolution) was calculated to be 9×10^{-7} /RY for PWRs and 1.8×10^{-7} /RY for BWRs.

Consequence Estimate

The base case public risk calculated for the affected parameters was 79.1 man-rem/Ry for PWRs and 40.4 man-rem/Ry for BWRs. The adjusted case public risk was calculated to be 76.9 man-rem/Ry for PWRs and 39.2 man-rem/Ry for BWRs. Thus, the public risk reduction was 2.2 man-rem/Ry for PWRs and 1.2 man-rem/Ry for BWRs. Based on 10 PWRs and 5 BWRs with an average remaining life of 30 years, the total public risk reduction was 840 man-rem.

Cost Estimate

Industry Cost: It was assumed that, for those plants expected to be completed after 1990, the cost to implement the standard would be part of the basic cost. For those plants expected to be completed between 1987 and 1990, the cost to redesign the control room was estimated to be \$100,000/plant. This was based on the assumption that, in all likelihood, draft standards would be available for use and only minor changes would be needed. Also, it was assumed that the standards would not require significant equipment additions but only reworking of preliminary designs. Since there were about 10 plants to be completed between 1987 and 1990, the total industry cost for implementation was estimated to be \$1M. No additional cost for yearly industry operation and maintenance was assumed.

NRC Cost: The NRC cost estimate was based on an assumed \$300,000 expenditure for regulatory guide development. It was assumed that additional NRC labor of about 4 man-weeks/plant would be necessary to review the modifications that would be required for the 10 plants completed between 1987 and 1990. This totaled a cost of about \$9,000/plant or \$90,000 total. Thus, the total NRC cost was estimated to be \$390,000.

Total Cost: The total industry and NRC cost associated with the possible solution was \$(1 + 0.39)M or \$1.39M.

²⁶ NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

Value/Impact Assessment

Based on an estimated public risk reduction of 840 man-rem and a cost of \$1.39M for a possible solution, the value/impact score was given by:

$$\begin{aligned} S &= \frac{840 \text{ man-rem}}{\$1.39\text{M}} \\ &= 600 \frac{\text{man-rem}}{\$M} \end{aligned}$$

Uncertainty

The human error reduction was not easily quantifiable; 3% was used here, but it was subject to large uncertainty.

Other Considerations

- (1) The issue was assumed to affect only future plants. NRC guidelines in NUREG-0700²⁷ were to be applied to all existing plants and NTOLs.
- (2) IEEE Standards were under development at the time the issue was evaluated.

CONCLUSION

Based on the above value/impact score, this issue was given a medium priority ranking (see Appendix C) and resolved. Although no action was taken on Item I.D.4, all commercial nuclear power plants in the U.S., whether operational or under construction, were subjected to a Detailed Control Room Design Review (DCRDR) in response to TMI Item I.D.1. NUREG-0700²⁸ and acceptable substitutes (e.g., the Boiling Water Reactor Owners' Group "Control Room Survey Program" and "Checklist Supplement") were used as control room design standards. In accordance with 10 CFR 50.34(g), all future applications for LWRs shall include an evaluation of the proposed facility against SRP²⁹ Section 18.1 which addresses control room design and references NUREG-0700³⁰ as appropriate guidance for control room design.

Thus, staff actions negated the need for evaluation of industry control room design standards and for the development of a Regulatory Guide endorsing those standards. NUREG-0700³¹ and acceptable substitutes are the de facto control room design standards for evaluating commercial nuclear power plants in the U.S. Design standards for advanced control rooms were to be addressed as a research issue under the Human Factors Research Program. Therefore, this issue was RESOLVED with no new requirements.³² In NUREG/CR-5382,³³ it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

²⁷ NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

²⁸ NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

²⁹ NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

³⁰ NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

³¹ NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

³² Memorandum for V. Stello from E. Beckjord, "Closure of Generic Issue I.D.4 `Control Room Design Standard,'" March 28, 1988. [9704160014]

³³ NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

ITEM I.D.5: IMPROVED CONTROL ROOM INSTRUMENTATION RESEARCH

ITEM I.D.5(1): OPERATOR-PROCESS COMMUNICATION

DESCRIPTION

This TMI Action Plan³⁴ item focused on the need to evaluate the operator-machine interface in reactor control rooms. The emphasis of this portion of the overall issue was the use of lights, alarms, and annunciators.

The method of presentation of information can significantly enhance the performance of the control room operators and thereby potentially affect operator error. It was proposed that existing practice and use of lights, alarms, and annunciators be reviewed to assess how well they facilitate operator-machine interaction and minimize errors.

CONCLUSION

RES studied the area of control room alarms and annunciators (through a contractor) and the results were reported in NUREG/CR-2147.³⁵ Based on this report, RES issued RIL-124³⁶ which provided a recommendation for further action. Thus, this item was RESOLVED with no new requirements. In NUREG/CR-5382,³⁷ it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

ITEM I.D.5(2): PLANT STATUS AND POST-ACCIDENT MONITORING

DESCRIPTION

This TMI Action Plan³⁸ item focused on the need to improve the ability of reactor operators to prevent, diagnose, and properly respond to accidents. The emphasis was on the information needs (i.e., indication of plant status) of the operator. In order for operators to perform their functions, it is necessary that they receive all the necessary information on the plant status. This can enhance operator performance (and therefore reduce operator error). Accident sequences should be analyzed to determine the information required to provide unambiguous indication of plant status. Specific instrumentation and ESF status monitoring needs would then be determined.

CONCLUSION

PWR instrumentation requirements were analyzed in NUREG/CR-1440³⁹ and BWR instrumentation requirements were analyzed in NUREG/CR-2100.⁴⁰ ESF Status Monitoring requirements were also studied in NUREG/CR-2278.⁴¹ RIL No. 98⁴² was issued in August 1980 and transmitted "the results of completed research describing an improved method for analyzing accident sequences." Revision 2 to

³⁴ NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

³⁵ NUREG/CR-2147, "Nuclear Control Room Annunciators," U.S. Nuclear Regulatory Commission, October 1981.

³⁶ Memorandum for H. Denton et al. from R. Minogue, "Research Information Letter #RIL-124, 'Control Room Alarms and Annunciators,'" October 20, 1981. [8111130045]

³⁷ NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

³⁸ NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

³⁹ NUREG/CR-1440, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1980.

⁴⁰ NUREG/CR-2100, "Boiling Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1981.

⁴¹ NUREG/CR-2278, "Light Water Reactor Engineered Safety Features Status Monitoring," U.S. Nuclear Regulatory Commission, October 1981.

⁴² RIL-98, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, August 18, 1980. [8104230867]

Regulatory Guide 1.97⁴³ was issued in December 1980. (See also Item II.F.3, "Instrumentation for Monitoring Accident Conditions.") The staff planned to have this guide implemented at all plants.^{44,45} This item was RESOLVED and new requirements were established.

ITEM I.D.5(3): ON-LINE REACTOR SURVEILLANCE SYSTEM

DESCRIPTION

This TMI Action Plan⁴⁶ item was based on work performed by ORNL. A continuous on-line automated surveillance system was installed at Sequoyah-1 (PWR) and information was obtained throughout the first fuel cycle. The demonstration at Sequoyah was to continue through the second fuel cycle (mid-1984). A similar demonstration at an operating BWR was planned for initiation in 1984. The system had the potential to provide diagnostic information to predict anomalous behavior of operating reactors which could be used to maintain safe conditions.

Noise surveillance and diagnostic techniques associated with the on-line reactor surveillance system have shown their safety significance and the results of the research were used by NRC in regulatory activities. Monitoring of neutron noise in BWRs was used to detect and monitor the impacting of instrument tubes against fuel boxes. The technique was used by NRC and its consultants to verify that partial power operation was safe until the next scheduled fuel outages for some 10 BWRs. Pressure noise surveillance was used at TMI-2 to monitor and guide degassification of the primary loop. The data obtained from the on-line surveillance demonstrated at Sequoyah-1 were used by NRC and its consultants in the assessment of loose thermal shields in Oconee Units 1, 2, and 3. In yet another example, NRR used results of this research in BWR stability determinations associated with regulatory actions pertaining to Dresden.

CONCLUSION

Based on the ongoing programs at the time this issue was evaluated, the technical resolution had been identified and the issue was considered nearly-resolved. As a result of the staff's work, RIL 171 was issued.⁴⁷ Thus, the issue was RESOLVED with no new requirements.⁴⁸ Consideration of a 20-year license renewal period would not affect the resolution.

ITEM I.D.5(4): PROCESS MONITORING INSTRUMENTATION

DESCRIPTION

This TMI Action Plan⁴⁹ item called for the staff to explore the feasibility of using new concepts for measuring certain reactor parameters. A directly related issue, Item II.F.2 in NUREG-0737,⁵⁰ mandated

⁴³ Regulatory Guide 1.97, "Instrumentation for Light- Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission, December 1975, (Rev. 1) August 1977 [8001240572], (Rev. 2) December 1980 [7912310387], (Rev. 3) May 1983. [8502060303]

⁴⁴ SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]

⁴⁵ Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits from U.S. Nuclear Regulatory Commission, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982. [ML031080548]

⁴⁶ NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

⁴⁷ Memorandum for T. Murley from E. Beckjord, "Research Information Letter Number 171, 'Continuous On-Line Reactor Surveillance System,'" May 4, 1993. [9305100271]

⁴⁸ Memorandum for J. Taylor from E. Beckjord, "Closure of Generic Issue I.D.5(3), 'On-Line Automated Continuous Reactor Surveillance Systems,'" November 12, 1993. [9502070301]

⁴⁹ NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

⁵⁰ NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission,

that industry develop and implement PWR liquid level detection systems. NRC evaluated a number of systems at the LOCA experiment facilities at ORNL and INEL.

CONCLUSION

This item was RESOLVED with no new requirements. In NUREG/CR-5382,⁵¹ it was concluded that consideration of a 20-year license renewal period did not affect the resolution of the issue.

ITEM I.D.5(5): DISTURBANCE ANALYSIS SYSTEMS

DESCRIPTION

Historical Background

This TMI Action Plan⁵² item called for the staff to explore advanced disturbance analysis systems for possible application to nuclear power plants.

Safety Significance

If potential transient events could be anticipated and terminated earlier and if operator response could be enhanced, then the core-melt frequency could be reduced. Advanced disturbance analysis systems could possibly provide the capabilities to achieve this.

Possible Solution

The purpose of this item was to assess the need, feasibility, and adequacy of advanced disturbance analysis systems. At the time that this issue was evaluated, research in this area was being conducted by EPRI.

PRIORITY DETERMINATION

Assumptions

It was assumed that the advanced disturbance analysis system would include the implementation of a continuous on-line surveillance system, as discussed in Item I.D.5(3). [A liquid level detection system was assumed available because it was already required - Items I.D.5(4) and II.F.2.]

The risk reduction was estimated by assuming a reduction of 2% in operator errors due to the implementation of this additional operator aid.⁵³ Also, a reduction in the number of transients requiring shutdown was assumed based on the potential that the operators will be able to terminate some transients before the need for shutdown. Reduced transient frequencies were calculated based on an EPRI analysis.⁵⁴ The basis for choosing the transients was that either the detection time leading up to the transient or the time from the transient occurrence to shutdown was perceived to be longer than 30 minutes, enabling the advanced diagnostic system to diagnose the problem and provide possible solutions for the operator.

Furthermore, it was assumed that an operator could only respond with actions to 80% of the transients listed that would occur during the remaining life of the subject plants. Of the 80%, only 25% of the operators' actions was assumed to prevent the need for shutdown. The average plant shutdown was assumed to last 0.75 day. Therefore, reduction in unscheduled outages was calculated as follows:

$$\text{PWR: } (4.63 \text{ transients/R Y})(0.80)(0.25)(0.75 \text{ day/shutdown}) = 0.69 \text{ day/R Y}$$

November 1980, (Supplement 1) January 1983.

⁵¹ NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

⁵² NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

⁵³ NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

⁵⁴ EPRI NP-2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.

BWR: $(5.20 \text{ transients/RY})(0.80)(0.25)(0.75 \text{ day/shutdown}) = 0.78 \text{ day/RY}$

Frequency Estimate

The parameters which included direct operator action were adjusted based on the 2% operator error reduction. In addition, the reduced transient frequency calculated from above were divided by the total PWR and BWR transient frequencies (i.e., 9.8 events/RY for PWRs and 8.9 events/RY for BWRs) to give a percent transient reduction. Then the parameters for transients (T2 and T3 for PWRs and T23 for BWRs) were adjusted.

Combining the reduction in operator error and the reduction in transient frequencies, the reductions in core-melt frequencies were $4.4 \times 10^{-6} \text{ event/RY}$ for PWRs and $2.6 \times 10^{-6} \text{ event/RY}$ for BWRs.

Consequence Estimate

The associated reduction in public risk was calculated (assuming 340 people per square-mile) to be 12 man-rem/RY for PWRs and 18 man-rem/RY for BWRs. Assuming 90 PWRs and 44 BWRs with remaining lives of 28.8 and 27.4 years, respectively, the total public risk reduction was calculated to be 53,000 man-rem.

Cost Estimate

Industry Cost: For the advanced diagnostic system, implementation costs (hardware and installation) were estimated to be \$1.5M/plant. The on-line surveillance system was estimated to cost \$125,000/plant for hardware and \$375,000/plant for installation. For 134 plants, the total implementation cost was approximately \$270M.

Operation and maintenance was estimated to be about 10 man-weeks/RY beyond that required for control room instrumentation. Therefore, this cost would be $(10 \text{ man-weeks/RY})(\$2,270/\text{man-week})(134 \text{ plants})(30 \text{ years})$ or \$91M. Therefore, the total industry cost was estimated to be \$360M.

NRC Cost: NRC costs for resolution were considered to be relatively minor (\$2M), based on the assumption that EPRI would continue to do the major portion of the research on the issue. Labor to approve and monitor hardware changes for the backfit of the affected plants was based on an average of 4 man-weeks/plant. The total cost for this effort was given by $(4 \text{ man-weeks/backfit plant})(\$2,270/\text{man-week})(71 \text{ plants})$ or \$650,000. Therefore, the total NRC cost was \$2.65M.

Total Cost: The total industry and NRC cost associated with the possible solution was $\$(360 + 2.65)\text{M}$ or \$362.65M.

Value/Impact Assessment

Based on an estimated public risk reduction of 53,000 man-rem and a cost of \$362.65M for a possible solution, the value/impact score was given by:

$$\begin{aligned} S &= \frac{53,000 \text{ man-rem}}{\$362.65\text{M}} \\ &= 150 \frac{\text{man-rem}}{\$M} \end{aligned}$$

Uncertainty

The assumed benefits of resolution and cost for implementation were extremely hard to quantify because of the uncertain nature of possible future developments in this area.

Other Considerations

- (1) Assuming that replacement power costs were \$300,000/day and, as previously calculated, resolution would reduce down time by 0.69 day/RY for PWRs and 0.78 day/RY for BWRs, the industry cost saving would be:

$$(\$300,000/\text{day})[(0.69 \text{ day/RY})(90 \text{ plants})(30 \text{ years}) + (0.78 \text{ day/RY})(44 \text{ plants})(30 \text{ years})] = \$870\text{M}$$

Combining this with the industry costs (implementation and operation) would show an industry

saving of about \$500M. Including accident avoidance costs would further increase this saving.

(2) EPRI was conducting research in this area which was being followed by NRC.

CONCLUSION

Based on the judgment that a disturbance analysis system could reduce operator errors by 2% and the number of transients by a factor of 2, the issue was given a medium priority ranking (see Appendix C). After a more detailed review, the staff concluded that, although disturbance analysis systems might decrease plant shutdowns and thereby reduce plant costs, this economic benefit should not be a reason for requiring installations of such systems because the assumed safety benefit was too uncertain. The staff further concluded that, in order to determine whether or not a specific safety problem existed, more research was necessary to determine the effect that disturbance analysis systems could have on operator performance.⁵⁵ As a result, the issue was reclassified as a Licensing Issue and integrated into the research activity, Human Factors Aspects of Advanced Controls and Instrumentation.⁵⁶

Guidelines for the verification and validation of expert systems that could be used in the development and review of disturbance analysis systems were developed from a joint EPRI/NRC research program; these were published in NUREG/CR-6316.⁵⁷ Thus, this Licensing Issue was resolved.⁵⁸

ITEM I.D.6: TECHNOLOGY TRANSFER CONFERENCE

DESCRIPTION

In January 1980, the NRC and IEEE jointly sponsored a technology transfer conference entitled "Advanced Electrotechnology Applications to Nuclear Power Plants" which had as its objective consideration of the practicality of applying advanced technologies from other industries (e.g. aerospace, defense, aviation) to the nuclear power industry.

During the conference, eight parallel workshops were held including: Systems Management Techniques; Reliability Engineering; Risk Assessment; Software Reliability Verification and Validation; Smart Instrumentation; Operational Aids-Command Control and Communications; Education, Training, and Simulators; and Simulation and Analysis. The conference report⁵⁹ was issued in June 1980. This item was related to increasing knowledge and understanding of safety issues and, therefore, was considered a Licensing Issue.

CONCLUSION

This Licensing Issue was resolved with the completion of the conference.

⁵⁵ Memorandum for B. Morris from B. Sheron, "Updated GIMCS for GI I.D.5(5)," February 2, 1988. [9704150145]

⁵⁶ Memorandum for V. Stello from E. Beckjord, "Redesignation of Generic Issue I.D.5(5), 'Disturbance Analysis Systems,'" February 22, 1988. [8809190312]

⁵⁷ NUREG/CR-6316, "Guidelines for the Verification and Validation of Expert System Software and Conventional Software," U.S. Nuclear Regulatory Commission, (Vols. 1, 2, 3, 4, 5, 6, 7, and 8) March 1995

⁵⁸ Memorandum for J. Taylor from E. Beckjord, "Closeout of TMI Action Plan Task I.D.5(5), Research on Disturbance Analysis Systems," April 17, 1995. [9705190216]

⁵⁹ IEEE Catalog No. TH0073-7, "Record of the Working Conference on Advanced Electrotechnology Applications to Nuclear Power Plants, January 15– 17, 1980, Washington, D.C.," The Institute of Electrical and Electronics Engineers, Inc.