Section 1. TMI Action Plan Items

Contents

Section 1. TMI Action Plan Items ( )

This section contains the TMI Action Plan items that were documented in NUREG-0660.[48] All items in Chapters I, II, III, and IV that were identified for prioritization and listed in this section follow the numbering system established in NUREG-0660.[48] Items found to be closely related have been combined where possible to form single issues for prioritization purposes. As a result, some of these combined issues contain items with the lead responsibility assigned to several offices. However, the lead responsibility and a summary of the findings for each item listed can be found in Table II of the Introduction. Items clarified in NUREG-0737[98] are listed in this section for accounting purposes only.

Chapters I, II, III, and IV presented a detailing of plans for NRC staff or licensee action whereas Chapter V addressed NRC policy, organization, and management and originally called for 17 specific actions to be taken by the Commissioners. In recognition of the interrelationships that required correlated planning, these 17 items were later grouped into seven subject areas by the staff and forwarded to the Commission in SECY-80-230B.[972] This revision to Chapter V was agreed upon by the Commission and was published as Rev. 1 to NUREG-0660[48] in July 1980. All items of Chapter V listed in this section follow the numbering system established in NUREG-0660,[48] Rev. 1.

Task I.A: Operating Personnel (Rev. 3) ( )

TASK I.A.1: OPERATING PERSONNEL AND STAFFING

Complex transients in nuclear power plants place high demands on the operators in the control room. The objective of the actions described in this task was to increase the capability of the shift crews in the control room to operate the facility in a safe and competent manner, by assuring that a proper number of individuals with the proper qualifications and fitness are on shift at all times. The work to improve the design of control rooms is described elsewhere in this plan.

ITEM I.A.1.1: SHIFT TECHNICAL ADVISOR

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-01 was established by DL/NRR for implementation purposes.

ITEM I.A.1.2: SHIFT SUPERVISOR ADMINISTRATIVE DUTIES

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.A.1.3: SHIFT MANNING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-02 was established by DL/NRR for implementation purposes.

ITEM I.A.1.4: LONG-TERM UPGRADING

DESCRIPTION

The purpose of this item was to develop changes to 10 CFR 50.54 concerning shift staffing with licensed operators and working hours of licensed operators. As described in NUREG-0660,[48] the staff was expected to "develop proposed changes to 10 CFR 50 for consideration by the Commission to effect appropriate changes concerning plant staffing, including shift manning, control room presence, and working hours."

CONCLUSION

SECY-81-440[250] was prepared by the staff in July 1981 and resulted in a Commission policy statement on working hour limitations which was issued in the Federal Register on February 17, 1982. Working hour limitations were to be incorporated into Regulatory Guide 1.33[225] [see Issue 75]. The specific issues to be considered were: (1) the number of licensed operators, based on the number of reactors, control room configuration, and operating model; (2) whether existing rulemaking should be expanded to include non-licensed operators; (3) whether existing rulemaking should be expanded to include "position titles," in addition to the type of NRC license; (4) whether shift technical advisors (STAs) or shift engineers (SEs) should be required on shift; and (5) whether shift supervisors (SSs) should be licensed.

A proposed rule was published on August 30, 1982 and, after the comment period expired, the final rule was submitted to the Commissioners in SECY-83-52A[595] on March 14, 1983. In response to the TMI Action Plan, licensing has required, through technical specification (TS), the great majority of the substantive features of the expected changes to regulation. Therefore, adoption of the rule was expected to have the effect of codifying existing requirements with minimal impact on licensees. The final rule amending 10 CFR 50.54 was approved[596] by the Commission on April 28, 1983. Thus, this issue was RESOLVED and new requirements were established.[956]

REFERENCES

0048.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

0225.	Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, November 1972, (Rev. 1) February 1977, (Rev. 2) February 1978. [7907100144]
0250.	SECY-81-440, "Nuclear Power Plant Staff Working Hours," U.S. Nuclear Regulatory Commission, July 22, 1981. [8107290183]
0595.	SECY-83-52A, "Final Rulemaking Concerning Licensed Operator Staffing at Nuclear Power Units and Draft Policy Statement on Shift Crew Qualifications," U.S. Nuclear Regulatory Commission, March 14, 1983. [8304010029]
0596.	Memorandum for W. Dircks from S. Chilk, "Staff Requirements—Affirmation/Discussion and Vote, 3:35 p.m., Thursday, April 21, 1983, Commissioners' Conference Room (Open to Public Attendance)," April 28, 1983. [9705190263]
0956.	Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987. [8701140115]

Task I.A.2: Training and Qualifications of Operating Personnel (Rev. 6) ( )

The objectives of this task were to: (1) improve the capability of operators and supervisors to understand and control complex reactor transients and accidents; (2) improve the general capability of an operations organization to respond rapidly and effectively to upset conditions; and (3) increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capability to perform their duties.

ITEM I.A.2.1: IMMEDIATE UPGRADING OF OPERATOR AND SENIOR OPERATOR TRAINING AND QUALIFICATIONS

This item required all operating plant licensees and all license applicants to provide specific improvements in training and qualifications of senior operators and control room operators. The three parts of this item are listed below.

ITEM I.A.2.1(1): QUALIFICATIONS - EXPERIENCE DESCRIPTION

This NUREG-0660[48] item set specific experience requirements that were to be met by applicants for senior operator licenses by May 1, 1980. Applicants for senior operator licenses were required to have been a licensed operator for one year effective December 1, 1980.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were issued, and MPA F-03 was established by DL/NRR for implementation purposes.

ITEM I.A.2.1(2): TRAINING

DESCRIPTION

This NUREG-0660[48] item set the following specific requirements:

(1) Effective August 1, 1980, senior operator applicants were required to have 3 months of continuous on-the-job training as an extra person on shift.

(2) Effective August 1, 1980, control room operator applicants were required to have 3 months training on shift as an extra person in the control room.

(3) Training programs were to be modified to provide: (a) training in heat transfer, fluid flow, and thermodynamics; (b) training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged; and (c) increased emphasis on reactor and plant transients.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were issued, and MPA F-03 was established by DL/NRR for implementation purposes.

ITEM I.A.2.1(3): FACILITY CERTIFICATION OF COMPETENCE AND FITNESS OF APPLICANTS FOR OPERATOR AND SENIOR OPERATOR LICENSES

DESCRIPTION

This NUREG-0660[48] item required all applicants for operator and senior operator licenses, pursuant to 10 CFR 55.10(a)(6), 10 CFR 55.33(a)(4), and 10 CFR 55.33(a)(5), to be certified by the highest level of the corporate management of their respective plants. This requirement was effective May 1, 1980.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were issued, and MPA F-03 was established by DL/NRR for implementation purposes.

ITEM I.A.2.2: TRAINING AND QUALIFICATIONS OF OPERATIONS PERSONNEL

DESCRIPTION

Under the TMI Action Plan,[48] the NRC could require reactor licensees to review their training and qualification programs for all operations personnel. This was interpreted to include licensed and auxiliary operators, technicians, maintenance personnel, and supervisors. The review was to be conducted to examine existing practices in light of the safety significance of the duties of the operations staff. If the review determined that the existing practices adequately assured proper safety-related staff conduct, then documentation of the justification for this determination was required; this documentation did not require submittal to the NRC but was required to be maintained on site. If the review uncovered inadequacies, the licensee was required to upgrade the training and qualification practices to ensure adequate performance of operations personnel. The evaluation of this issue included the consideration of Item I.A.2.6(3).

PRIORITY DETERMINATION

To estimate the effect of training reviews on operator-error contributions to plant risk, a panel of PNL experts was assembled with considerable experience in reactor operations, utility training programs, and reactor plant systems. The panel included members with utility field experience and reactor operator licensing examiners. The judgments of the panel, as detailed below, were based on the two following considerations:[64]

(1) The potential effect of this issue was limited by its semi-voluntary nature, i.e., the judgment of adequacy was in the hands of the individual utilities. Furthermore, the existing Institute for Nuclear Power Operations (INPO) and NRC research work in task analysis dealt with generic routine operations. Plant-specific operation and operation under upset conditions were left to the individual utilities. This diluted the effectiveness of the task analysis efforts in providing the basis for the training and qualifications review.

Related issues which were supported by and, in turn, supported this issue were the conduct of plant drills and accreditation of training programs. While neither of these were directly required by the training and qualifications review, both could have been a part of the response and both would have had a positive effect on personnel performance.

(2) There was a wide variation among utilities in both the training programs and the performance of operations staff. In many facilities, there was much room for improvement. Therefore, while the potential effect of the training and qualifications review effort was limited, a significant overall reduction in safety-related human error for operations personnel was expected because of the wide margin available for improvement.

Assumptions

The PNL panel divided licensees into three groups:

(1) Minimally-Affected: These utilities had a good, effective training and qualification program and good operations personnel performance. They were to be minimally affected by this issue. The fractional population of this group was estimated to be 15% of the reactor licensees.

(2) Intermediately-Affected: These utilities' training and qualification programs and/or operations performance had room for improvement. This group, estimated to be 60% of the population, had to undergo improvements and, therefore, were affected by the issue.

(3) Maximally-Affected: These utilities had deficiencies in their training and qualification programs and in operations personnel performance. They would be significantly affected by this issue and major restructuring of programs were expected. This group was estimated to contain 25% of reactor licensees.

From the estimates for these groups, weighted composite estimates were derived. NUREG/CR-2800[64] shows the safety benefit estimated by the panel for each of the groups and also gives the weighted averages.

The values given in NUREG/CR-2800[64] are in terms of percent changes. For inclusion into the value/impact score formula, they were converted to other measures. The reduction in human error was transformed into the resulting reduction in risk, as measured by change in probabilistic risk exposure (man-rem/RY). The change in annual ORE was also transformed from percent improvement into man-rem/RY.

The reduction in risk was developed by examining the quantitative impact on accident event frequencies of human error rates in key scenarios. The reduction in human error translated into a reduction in accident frequency. No additional reduction due to accident mitigation was assumed. The values given in NUREG/CR-2800[64] were used for the best estimate of improvement: 17% for operator error and 28% for maintenance.

Frequency Estimate

This issue centered around operator and maintenance training programs to improve personnel performance. The issue related generically to both BWRs and PWRs and, ideally, the risk reduction attributable to its resolution was estimated by selecting a representative plant of each type. However, maintenance and operator performance essentially impacted accident sequences in the risk equations. The calculations were performed for one representative PWR and inferences drawn for all reactors. The Oconee-3 (a RSSMAP PWR) plant risk equations developed in NUREG/CR-1659,[54] Vol. 4, were used for this analysis.

It was assumed that the 17% reduction in operator error could be applied directly to elements containing an operator error frequency and the 28% reduction could be applied directly to maintenance variables. This assumption introduced some error in the maintenance contribution because some maintenance operations on nuclear systems have fixed times associated with cooldown and preparation, etc., in addition to the actual hands-on time for maintenance that would be subject to improvement through training. Maintenance done properly the first time also reduces the frequency of maintenance outage and downtime for proper repairs at some future date. Thus, fixed time periods in maintenance outages are indirectly reduced over the long run with improved maintenance performance simply because the need for maintenance may be reduced, except for systems that undergo preventive maintenance at set intervals.

Consequence Estimate

It was assumed that the resolution applied to all plants existing and planned, as given in NUREG/CR-2800, Appendix C.[64] This represented a total of 4,000 RY of operation (143 plants with an average remaining life of 28 years). Implementation of the solution would provide a reduction of 9 man-rem/RY. For all plants, assuming a typical midwest-type meteorology and an average population density of 340 people per square-mile at U.S. reactor sites, the total public risk reduction was estimated to be 122,400 man-rem.

Cost Estimate

Industry Cost: In estimating the costs to industry of implementing and operating under the resolution of this issue, the PNL panel divided the industry once again into three categories; these groups and their estimates are shown in NUREG/CR-2800.[64] The cost for implementation was the product of the number of plants and the cost/plant: (143)($0.335M) or $48M. The operation cost was the product of the number of plants, the average remaining life, and the annual plant cost: (143)(28)($0.16M) or $640M. Thus, the total industry cost was $(640 + 48)M or $688M.

NRC Cost: The NRC cost to implement the resolution was taken from NUREG-0660.[48] This called for 1.1 man-years of NRC effort which was equivalent to $110,000. The annual NRC effort through OIE to review the justification documentation and new training programs was estimated to be one man-year or $100,000/year. Over the lifetime of the completed and planned reactors, this cost was estimated to be $2.8M. Therefore, the total NRC cost was $[0.11 + 2.8]M or $2.9M.

Total Cost: The total industry and NRC cost associated with the possible solution was $[688 + 2.9]M or approximately $691M.

Value/Impact Assessment

Based on an estimated public risk reduction of 122,400 man-rem and a cost of $691M for a possible solution, the value/impact score was given by:

$$S = \frac{122{,}400\,\text{man-rem}}{\$691\text{M}}$$

$$= 177\,\text{man-rem}/\$\text{M}$$

Other Considerations

It was estimated that, with improved training, the operational doses could be reduced by 2.4×10^5 man-rem for 143 plants over the average remaining plant life. Including the occupational dose reduction (2.4×10^5 man-rem) in the above equation would increase the value/impact score to 524 man-rem/$M. PNL calculated[64] the occupational risk reduction for accident-related ORE to be 880 man-rem.

CONCLUSION

Because of the extensive number of sequences considered to be affected by this issue, the base case risk was high with a calculated range from 60 to 73 man-rem/RY. Based on the potential reduction in public risk and ORE, the issue was given a high priority ranking (see Appendix C). However, in June 1985, the Commission recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry.[777] Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.2.3: ADMINISTRATION OF TRAINING PROGRAMS

DESCRIPTION

This NUREG-0660[48] item required the staff to develop criteria and procedures to be used in auditing training programs, including those provided by reactor vendors, and to increase the amount of auditing. Specifically, NRR was expected to: (1) audit training programs to ensure training was formalized and, eventually, in conformance with accreditation; (2) conduct cold operator licensing certification at simulators; and (3) pending accreditation, require certain instructors to be SRO-certified.

CONCLUSION

Elements (2) and (3) were implemented and were incorporated into the Examiner Standards and Inspection Procedures. The issue of training audits was addressed by the Commission's Policy Statement[966] on Training and Qualification of Nuclear Power Plant Personnel which endorsed the INPO-managed accreditation program.[956] Thus, this item was clarified in NUREG-0737[98] and new requirements were issued.

ITEM I.A.2.4: NRR PARTICIPATION IN INSPECTOR TRAINING

DESCRIPTION

Based on NUREG-0660,[48] the NRR licensing and human factors staff was required to provide supplemental instruction to the OIE inspectors as an addition to the previously established OIE inspector training program. The purpose of such instruction was to focus the inspectors' attention on problems associated with human factors. With such training, it was expected that the inspectors would become more sensitive to such problems and, hence, more apt to initiate corrective action and thereby improve plant safety in this area. This would provide a means of responding to the TMI-related concern on human factors problems for plant operations staff.

The principal benefit to be derived from NRR participation in OIE inspector training was the improvements the inspectors would gain from enhanced training. This training would increase inspector awareness in human factors and personnel-related problems. In areas such as emergency procedures reviews, routine operational practices and hardware-to-human interface deficiencies could be found by inspectors and corrected. The potential significance of this issue was explored by a panel of PNL experts that included three reactor operator license examiners with utility field experience in training as well as general reactor safety.

The panel envisioned that the solution to this issue would be the addition of one week of instruction in human factors to the OIE inspector training course. The staff from NRR would participate in the instruction but would probably rely on a qualified consultant to conduct the majority of the instruction. It was assumed that the principal target of the training would be the resident inspectors. The potential effect of the training upon the OIE review of emergency procedures, plant hardware, and routine practices could be significant, but the overall effect was thought to be limited because of two factors: the short exposure of the inspector to human factors training, and the indirect nature of the safety benefit. That is, a marginal improvement in inspector awareness could result in some corrective actions which would result in some safety improvement. The separation between initial action and the safety benefit complicated assessment of the effectiveness of the proposed resolution of the issue.

PNL estimated[64] a human-error rate reduction of 2% for operators and maintenance personnel (operations staff assumed most likely to affect plant safety). This was an overall industry-wide estimate; some isolated actions could be highly significant. PNL estimated the cost for this additional training to be about $1,000.

Capabilities of inspectors could clearly be improved through the proposed training. There would be an indirect effect on risk, since better-trained inspectors would identify more cost-effective improvements in plant operations. However, there was no reasonable way that the magnitude of the safety significance and cost of the improvements could be estimated quantitatively. This additional training would enhance the capabilities and thus contribute to the effectiveness and efficiency of the NRC in performing its regulatory safety mission. Thus, this training proposal was determined to be a Licensing Issue.

CONCLUSION

This Licensing Issue was resolved in September 1983 with the regionalization of the operator licensing function which provided for training and guidance of the regional operator licensing personnel.[956]

ITEM I.A.2.5: PLANT DRILLS

DESCRIPTION

The intent of this TMI Action Plan[48] item was to upgrade operator training by requiring operating personnel to conduct plant drills during shifts. Normal and off-normal operating maneuvers would be simulated for walk-through drills on a plant-wide basis. Drills would also be required to test the adequacy of reactor and plant operating procedures. This was an effort to reduce the risk of off-normal operating conditions by improving the capability of operators and supervisors to understand and control complex reactor transients and accidents, and also to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions.

PRIORITY DETERMINATION

Assumptions

It was assumed that the frequency of a core-melt incident was 5×10^{-5}/RY, based on WASH-1400.[16] Also, it was assumed that operator error accounted for 50% of these events, but plant drills would improve operator performance by 2%. In addition, it was assumed that the release associated with a core-melt was the value averaged over the probabilities of the WASH-1400[16] accident categories for PWRs and BWRs and weighted by the number of PWRs (95) and BWRs (48). This resulted in a total of 2.4×10^{6} man-rem/accident. The average remaining plant life was assumed to be 28 years.

Frequency Estimate

Based on the assumptions above, the reduction in the core-melt frequency resulting from plant drills was calculated to be $(0.02)(0.50)(5 \times 10^{-5})$/RY or 5×10^{-7}/RY.

Consequence Estimate

For 143 affected plants with an average remaining life of 28 years, the public risk reduction was estimated to be $(5 \times 10^{-7}$/RY$)(2.4 \times 10^{6}$ man-rem)(28 years)(143 reactors) or 4,805 man-rem.

Cost Estimate

Industry Cost: The industry resources required for implementation were estimated to be one man-month/plant. This was the estimated personnel requirement associated with the utility staff time for attendance at the drill, preparation by staff and management, and staff time dedicated to the dissemination of insights gained from the drills. At a cost of $100,000/man-year and with 4.33 weeks/month, this yielded a cost of $8,333/plant. For the 143 affected plants, the cost was estimated to be $1.2M.

The industry resources required annually to participate in the plant drills were estimated to be 2 man-months/plant and included drill attendance, preparation before the drill, and dissemination of information afterward; this cost was $16,660/RY. For the 143 affected plants, this cost was $2.38M/year. Over the average remaining life of 28 years, this cost was estimated to be $67M.

Thus, the total industry implementation and operational cost was $(1.2 + 67)M or approximately $68.2M.

NRC Cost: The total NRC cost to implement the resolution of this issue included NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor were to some degree interchangeable, no attempt was made to provide separate estimates so that the total implementation cost was estimated to be

$300,000. The annual cost to the NRC was also estimated to be $300,000; again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life of 28 years, the operational cost was estimated to be $8.4M. Therefore, the total NRC implementation and operation cost was $(8.4 + 0.3)M or $8.7M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(68.2 + 8.7)M or $76.9M.

Value/Impact Assessment

Based on an estimated public risk reduction of 4,805 man-rem and a cost of $76.9M for a possible solution, the value/impact score was given by:

$$S = \frac{4{,}805 \text{ man} - \text{rem}}{\$76.9\text{M}}$$

$$= 62 \text{ man} - \text{rem} / \$\text{M}$$

CONCLUSION

Based on the above value/impact score, the ranking of this issue would have been low to medium. Because the risk may have been estimated to be well on the conservative side, the issue was given a low priority ranking (see Appendix C). However, ongoing work by DHFS/NRR on the subject was completed in July 1985 and published for information only as NUREG/CR-4258.[800] Thus, this item was RESOLVED and no new requirements were established.[801]

ITEM I.A.2.6: LONG-TERM UPGRADING OF TRAINING AND QUALIFICATIONS

ITEM I.A.2.6(1): REVISE REGULATORY GUIDE 1.8

Items I.A.2.6(1), I.A.2.6(2), I.A.2.6(3), and I.A.2.6(5) were combined and evaluated together.

DESCRIPTION

Historical Background

Item I.A.2.6 of the TMI Action Plan[48] called for the long-term upgrading of training and qualifications of operations personnel. The specific paragraphs of this item in NUREG-0660[48] called for a revision of Regulatory Guide 1.8,[226] (ANSI/ANS 3.1),[253] in order to incorporate short-term requirements into this issue and any other changes resulting from a national standards effort. Also, it was stated that more explicit guidance regarding exercises in simulator requalification programs would be included in the regulatory guide (Recommendation 8 of SECY-79-330E[251]) as would qualifications of shift supervisors and senior reactor operators [NUREG-0585,[174] Recommendations 1.6(1) and (2)]. In addition, based on the NRC staff review of NRR-80-117,[252] recommendations were to be made to the Commission and Commission decisions factored into the regulatory guide or regulation changes. Moreover, appropriate revisions to 10 CFR 55, Operator Licenses, were to be recommended for action by the Commission in order to incorporate the applicable short-term changes plus requirements based on Commission action on SECY-79-330E[251] for mandatory simulator training for applicants for licenses (Recommendation 4); mandatory simulator training in requalification programs (Recommendation 7); NRC administration of requalification examinations (Recommendation 9, as modified by the Commission); and mandatory operating tests at simulators (Recommendation 11).

Finally, the Nuclear Waste Policy Act of 1982 (Public Law 97-425, Section 306) authorized and directed NRC to promulgate regulations or guidance for the training and qualification of civilian nuclear power plant personnel. A task force was formed within NRC as a result of this bill. As part of the task force objectives, Items I.A.2.6(1, 2, and 3) were to be addressed.

Safety Significance

A public risk reduction was anticipated as a result of a reduction in core-melt frequency which follows from a reduction in operator error rates. Reduction in operator errors was expected to result from the upgraded training and qualifications which formed the assumed resolution of this issue.

Possible Solutions

The upgrades were assumed to include an increase in time spent in simulator operation, both in training and in requalification. The simulator time was assumed to improve in quality as well as quantity. Emphasis on improvements on the operators' diagnostic capability was felt to be especially important in contributing to a reduction in core-melt frequency. Furthermore, the enforcement activities in terms of NRC-administered examinations and OIE inspection of training programs were likely to emphasize the value of this long-term training and qualification of reactor operators.

PRIORITY DETERMINATION

The numerical assessment of this issue was conducted by PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provided utility training services and had significant experience in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

Assumptions

It was assumed that the resolution of this issue would take the form of upgrading utility training and qualification programs that would represent a major enhancement of the training and qualification programs.

It was noted that many of the TMI Action Plan[48] items associated with operator training were interrelated and it was, therefore, difficult to assess them independently. For example, this issue was related to I.A.4.1 which addressed the improvement of simulators and provided for more realistic modeling of a plant, whereas this issue, [I.A.2.6(1,2,3,5)], dealt with training improvements, including the enhanced use of existing simulators. Either issue, by itself, would improve operator performance; however, there could have been significant overlaps in improving operator performance if both items were implemented. Even though it was recognized that the total improvement would be less than the sum of the individual contributions when each is assessed separately, the extent of any overlap was not identified here.

Based on engineering judgment, it was estimated by the PNL panel that the resolution of this issue would result in a 30% reduction in operator error rates. The number of plants to which this issue was applicable was assumed to be 95 PWRs and 49 BWRs with average remaining lives of 28.5 years and 27 years, respectively.

For the PNL analysis,[64] Oconee-3 was selected as the representative PWR plant. It was assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf-1) would be equivalent to those for the representative PWR. Therefore, the analysis was conducted only for a PWR, but the fractional risk and core-melt frequency reductions were also applicable to a BWR. The dose calculations were based on a reactor site population density of 340 people per square-mile and a typical midwest meteorology was assumed.

Frequency Estimate

Based on the affected accident sequences and the parameters affected by the possible solution, the original core-melt frequencies of 8.2×10^{-5}/RY for PWRs and 3.71×10^{-5}/RY for BWRs were calculated to be reduced by about 16%.

Consequence Estimate

The associated reduction in public risk was 31 man-rem/RY for PWRs and 37.4 man rem/RY for BWRs, resulting in a total public risk reduction of 132,600 man-rem for all plants.

Cost Estimate

Industry Cost: The resolution of this issue was assumed to be a major enhancement of the training and qualification programs. The programs would have to be upgraded in order to meet the requirements of INPO accreditation. These requirements were assumed to be far-reaching and required significant effort on the part of utility training staffs. The amount of effort would vary among the utilities, depending on the existing state of their programs. The effort to implement the program was estimated by the PNL panel to require 10 to 20 man-years of effort at each plant. The mean value was expected to be shifted toward the lower end since, at the time of this evaluation, many utilities were improving their training programs. A 12 man-year effort was taken as the mean estimate.

Operation under the upgraded programs would require enhanced training activities and more operator time in training; the training staff was estimated to require three additional people. It was assumed the major cost of additional operator time could be estimated from increased time at simulators. It was estimated that 40 hours of simulator time would be added to operator training and requalification. For 20 operators/year passing through these programs, this was equivalent to 800 additional hours. It was further assumed that operators could be trained three at a time on the simulator and that simulator time could be acquired for $600/hour. This additional simulator cost was estimated to be $160,000/year. The industry costs were estimated as follows:

(1) Implementation

(12 man-years/plant)(143 plants)($100,000/man-year) = $173M

(2) Operation and Maintenance

(a) Labor

Training Staff = (3 man-year/RY)(52 man-weeks/man-year)

= 156 man-weeks/RY

Operators = (800 man-hour/RY)/(40 man-hours/man-week)

= 20 man-weeks/RY

Thus, the total labor was 176 man-weeks/RY.

(b) Simulator Time (Operators)

(800 man-hours/RY)/(3 man-hours/simulator-hour) = 267 simulator-hour/RY

Therefore, the industry cost/plant-year for operation and maintenance was given by:

For all affected plants, the total industry cost for operation and maintenance was ($500,000/RY)[(49)(27) + (95)(28.5)]RY or $2,000M.

The total industry cost for implementation, operation, and maintenance of the solution was then $(173 + 2,000)M or $2,173M.

NRC Cost: The NRC effort to implement the resolution of this issue would be significant. It was estimated in NUREG-0660[48] that 5.4 man-years plus $259,000 would be required. At the time of the evaluation of the issue, some of the development activities had been completed; however, much work remained to be done. The remaining effort was estimated to be 4.5 man-years and $100,000.

The operational activities of the NRC would include reviews of training program, increased inspection, and additional examination. The annual labor for reviews and inspections was estimated to be equivalent to 3 man-years. The principal addition in examinations was assumed to be NRC conduct of a portion of requalification examinations. It was assumed that the NRC would conduct 25% of the requalification examinations and that 20 operators would be requalified at each plant every year. It was estimated that one man-month was required for each plant based on the assumption that the five (25% of 20) operators selected for NRC examination at each plant would be tested at the same time. NRC costs were estimated as follows:

(1) Implementation	
Staff Labor + Other Costs	
	= (1.4 man-week/plant)($1,600/man-week) + ($100,000/144 plants)
	= $3,386/plant

Total cost for all affected plants was ($3,386/plant)(144 plants) or $488,000.	
(2) Review of Maintenance and Operation	
(a) Review and Inspection = (3 man-year/yr)(52 man-wk/man-yr)/144 plants	
	= 1.08 man-wk/RY
(b) Examination = (1 man-month/RY)(3.7 man-wk/man-month)	
= 3.7 man-wk/RY	

Thus, the total time spent was 4.78 man-wk/RY.

The cost/plant-year for the review of operation and maintenance was (4.78 man-week/RY)($1,900/man-week) or $9,088/RY. For the 144 affected plants, this cost was ($9,088)[(49)(27) + (95)(28.5)] or $36.6M.

Thus, the total NRC cost for implementation, operation, and maintenance was $(0.488 + 36.6)M or $37.1M.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $(2,173 + 37.1)M or $2,210M.

Value/Impact Assessment

Based on an estimated public risk reduction of 132,600 man-rem and a cost of $2,210M for a possible solution, the value/impact score was given by:

$$S = \frac{132{,}600\text{man} - \text{rem}}{\$2{,}210\text{M}}$$

$$= 60\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

The total occupational risk reduction was associated only with accident avoidance inasmuch as there was no dose associated with implementation or maintenance of the solution. With a dose of 20,000 man-rem associated with accident cleanup and with the calculated reductions in core-melt frequencies of 1.3×10^{-5}/RY and 5.9×10^{-5}/RY for PWRs and BWRs, respectively, the total occupational dose reduction associated with accident avoidance was calculated to be 860 man-rem.

CONCLUSION

Although the value/impact score was low, this issue was given a high priority ranking because of the large potential public risk reduction (see Appendix C). Resolution of the issue included the consideration of Items I.B.1.1(6,7) regarding changes to Regulatory Guide 1.8.[226]

In November 1986, SECY-86-348[1043] was submitted to the Commission with recommended revisions to Regulatory Guide 1.8[226] to endorse ANSI/ANS 3.1-1981 for the positions of shift supervisor, senior operator, licensed operator, shift technical advisor, and radiation protection manager. These revisions to Regulatory Guide 1.8[226] were subsequently approved by the Commission and published in May 1987.[1044] Thus, this issue was RESOLVED and new requirements were established.[1045]

ITEM I.A.2.6(2): STAFF REVIEW OF NRR 80-117

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an RES memorandum,[437] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(3): REVISE 10 CFR 55

This item was evaluated in Item I.A.2.6(1) above and, as a result of the Nuclear Waste Policy Act of 1982 (Public Law 97-425), was determined to be covered in Item I.A.2.2.[438]

ITEM I.A.2.6(4): OPERATOR WORKSHOPS

DESCRIPTION

Historical Background

On the basis of NUREG-0660,[48] NRR was required to develop a Commission paper on training workshops for licensed personnel. NUREG-0585,[174] the source of this issue, states that the intent of the issue was to conduct seminar-type workshops to exchange information on operations experience between the NRC and licensees and among licensees. This would assist in the improvement of operator performance and in improvements to reactor regulation, both resulting in improved safety. The proposed requirements would have one representative for each shift at each unit attend such a workshop annually.

Safety Significance

It was expected that there would be two potential pathways to improved safety benefit that would emerge from this issue: (1) improved operator performance through the sharing of safety-related experiences; and (2) the effect of improved regulation arising out of interaction between the operators and the NRC attending the workshops. The second pathway was considered to be a second-order effect and very difficult to quantify. Therefore, it was assumed that all the benefit would be derived through the reduction in operator-error rates.

PRIORITY DETERMINATION

Assumptions

It was assumed that major gains in reactor safety would come through the improvement in operator performance, i.e., a reduction in their error rates. There was also a pathway to improve safety by means other than human performance through improved regulations developed from operator input at the workshops. The latter would be extremely difficult to quantify so that only the human error rate-reduction pathway to improved safety was treated.

A panel of PNL experts was assembled and included staff that conduct operator licensing examinations, staff with experience in reactor operations, reactor safety and risk assessment, and the staff responsible for the conduct of the operator feedback workshops for NRR. This panel produced the estimates that formed the basis of this analysis. The analysis was based on the following additional assumptions:

(1) Applicable Plants: 95 PWRs and 48 BWRs.

(2) Selected Analysis Plant: Oconee-3 - representative PWR. It was assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf-1) would be equivalent to those for the representative PWR. Therefore, the analysis was conducted only for a PWR, but the fractional risk and core-melt frequency reductions were also applied to a BWR.

(3) Affected Accident Sequences and Base Case Frequencies: Most sequences were affected. The affected sequences and the base case frequencies are shown in NUREG/CR-2800.[64]

(4) Affected Release Categories and Base Case Frequencies: All release categories were affected by the resolution. The original base case frequencies were used as given below.

Oconee-3	Grand Gulf-1
PWR-1 = 1.10×10^{-7}/RY	BWR-1 = 1.09×10^{-7}/RY
PWR-2 = 1.00×10^{-5}/RY	BWR-2 = 3.35×10^{-5}/RY
PWR-3 = 2.86×10^{-5}/RY	BWR-3 = 1.44×10^{-6}/RY

Frequency Estimate

The PNL panel estimated[64] the most likely reduction in human error rates for operators due to the conduct of the proposed workshops would be 3%, assuming that the workshops were conducted in the manner perceived, i.e., to focus on data-gathering for the NRC. This reduced the amount of time that could be devoted to inter-licensee sharing of operational experiences which would have had a more direct effect on safety-related operational performance in the plants. The possible reduction ranged from 1% to 10%. If the focus could have been shifted toward the inter-licensee exchange of operational experiences, the most likely reduction in error rate would shift upward; however, it was not expected to exceed 10%.

Consequence Estimate

Based on the PNL estimates and calculations[64] and assuming a typical midwest-type meteorology and an average population density of 340 people per square-mile at U.S. reactor sites, the public risk reduction was estimated to be 7,140 man-rem for 143 plants with an average remaining life of 28 years. The occupational dose reduction was minor at a calculated value of 46 man-rem.

Cost Estimate

Industry Cost: The industry resources required for implementation were estimated to be one man-month/plant. This was the estimated personnel requirement associated with the trial workshops that were being conducted. It included utility staff time for attendance at the workshop, preparation by staff and management, and staff time dedicated to the dissemination of insights gained at the workshop. At a cost of $100,000/man-year and with 4.33 weeks/month, this yielded a cost of $8,333/plant. For the 143 affected plants, this cost was estimated to be $1.19M.

The industry resources required annually to participate in the training workshops were estimated to be the same as those for implementation, i.e., one man-month/plant, which included workshop attendance, preparation before the workshop, and dissemination of information afterward. This was equivalent to $8,333/RY. For 143 plants, this cost was estimated to be 143 man-months/year or $1.19M/year. Given the average remaining life of the plants, the operational cost was $33.3M. Therefore, the total industry cost associated with the solution to this issue was $34.5M.

NRC Cost: The NRC cost to implement the resolution of this issue was estimated to be $0.3M and included NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor were to some degree interchangeable, no attempt was made to provide separate estimates. The annual cost to the NRC was also estimated to be $0.3M; again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining plant life, the operational cost was estimated to be $8.4M. While not specific, these estimates for implementation and operation were firmly based on the experience of conducting the trial workshops. Therefore, the NRC implementation and operation cost was estimated to be $8.7M.

Total Cost: The total industry and NRC cost for the possible solution was estimated to be $(34.5 + 8.7)M or $43.2M.

Value/Impact Assessment

Based on the estimated public risk reduction of 7,140 man-rem and a cost of $43.2M for a possible solution, the value/impact score was given by:

$$S = \frac{7{,}140\,\text{man-rem}}{\$43.2\text{m}}$$

$$= 165\,\text{man-rem}/\$\text{M}$$

Other Considerations

The accident avoidance cost was estimated by calculating the product of the change in accident frequency (ΔF) and the estimated cost to the utility of a major accident (A); the latter term was estimated to be $1.65 Billion. Thus, the cost/plant-year was estimated to be:

PWRs: (ΔF)(A) = (7 x 10)($1,650M)/RY = $1,200/RY

BWRs: (ΔF)(A) = (3.2 x 10)($1,650M)/RY = $530/RY

The total cost for all plants was the product of the cost/plant-year, the number of plants (N), and the average remaining life (T) of each type of plant:

$\Sigma(NT)(\Delta F)(A)$ = \$(95)(28.5)(1,200)M + \$(48)(27.0)(530)M = \$3.9M

CONCLUSION

Because of the extensive number of sequences considered by PNL to be affected by this issue, the base case risk was high at a calculated range from 60 to 73 man-rem/RY. With a value/impact score of 165 man-rem/$M and an estimated risk reduction of 7,140 man-rem, this issue was given a medium priority ranking (see Appendix C).

The staff conducted three workshops and a mail survey in order to evaluate the effectiveness of both mechanisms for obtaining feedback to the NRC from utility operating staffs. The results of these two approaches were documented in NUREG/CR-3739[802] and NUREG/CR-4139,[803] respectively. The staff concluded that both feedback mechanisms proved to be effective methods of gathering data from operations personnel and did not recommend conducting workshops or surveys on an annual basis; it was preferable to use such mechanisms judiciously when a real need existed.[804] Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.2.6(5): DEVELOP INSPECTION PROCEDURES FOR TRAINING PROGRAM

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an OIE memorandum,[379] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(6): NUCLEAR POWER FUNDAMENTALS

DESCRIPTION

Historical Background

This NUREG-0660[48] item called for NRR to develop requirements for the inclusion of nuclear power fundamentals in the instruction given to reactor operators. This arose out of a concern[174] that the 12 weeks of fundamentals training that were being given to operators were insufficient.

Safety Significance

Safety issues that deal with operator training can affect public risk by improvements in the operator safety-related performance. This can lead to a reduction in core-melt frequency and a reduced probabilistic risk.

Possible Solution

The additional nuclear power fundamentals training would add 4 weeks to the training period.

PRIORITY DETERMINATION

In order to assess this issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas. The results of the PNL assessment are contained in NUREG/CR-2800.[64]

The PNL panel felt there had been significant progress across the industry in the area of instruction in nuclear power fundamentals since the issuance of NUREG0585[174] in 1979. Further increase in emphasis on fundamentals was felt to be unlikely to improve operator performance. The trend in operator licensing examinations was to stress operational knowledge and de-emphasize fundamentals. This supported the view that further fundamental training would not add to plant safety.

Frequency/Consequence Estimate

The PNL panel felt that the existing level of instruction in nuclear power fundamentals was adequate. Further emphasis on fundamentals was viewed as not likely to improve operator safety performance. Therefore, there would be no measurable public risk reduction associated with the possible solution to this issue. The PNL panel also saw no reduction in occupational dose associated with the implementation of the solution.

Cost Estimate

Industry Cost: It was assumed that 20 operators would complete the training course each year at every plant. In addition, one full-time instructor was assumed to be required. This yielded 80 man-weeks for the operators and 44 man-weeks for the instructors, or 124 man-weeks/plant overall each year. To implement this practice, an effort equivalent to one year of operation (124 man-weeks) was estimated to be required.

NRC Cost: Implementation of the solution was estimated[48] to take 0.4 man-year or approximately 18 man-weeks; no added costs were estimated for operation. The review of the additional instruction could be contained in the existing routine function thereby causing no added expense.

Value/Impact Assessment

Based on the judgment that there would be no risk reduction resulting from this issue, the value/impact score was zero.

CONCLUSION

In view of the fact that it was believed that the existing level of instruction in nuclear power fundamentals was adequate for reactor operators, further emphasis on fundamentals as required by this issue was viewed as not likely to improve operator safety performance. With the resulting value/impact score of zero, this issue was DROPPED from further consideration.

ITEM I.A.2.7: ACCREDITATION OF TRAINING INSTITUTIONS

DESCRIPTION

Historical Background

Based on the requirements of NUREG-0660,[48] this item required NRR to complete a study to establish the procedures and requirements for NRC accreditation of reactor operator training programs. The resulting study was to be developed into a Commission paper describing the various options for accreditation.

Safety Significance

There were two aspects to the safety benefit of this issue: (1) the reduction of public risk through the improvement of operator performance, which was expected from the improved training accreditation; and (2) a reduction in occupational exposure, primarily for operators who often supervise maintenance or perform other duties in radiation zones. However, some reduction in routine occupational exposure could also be expected for other operations personnel as a result of the increased awareness by the operators.

Possible Solution

In order to assess this issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas. The panel envisioned the resolution of this issue as the formation of an accreditation board consisting of representatives from the NRC, industry, and academia. This board would develop and apply criteria for accreditation including training programs of utilities, university-related programs, and independent training institutions. While theoretically applying to training for all operations staff, the PNL panel felt the existing thrust was focused on reactor operators. Therefore, this assessment was made assuming only operators would be affected.[64]

PRIORITY DETERMINATION

Assumptions

The views of the PNL panel included an awareness of the fact that, at the time this issue was evaluated, some training programs were very near to accreditation. Either through association with the universities or through other means of providing high quality instruction, these programs would be likely to acquire accreditation from the board easily. Other training programs were not so well prepared for accreditation and may have required significant effort and expense to upgrade them. Some savings may have been gained for multi-unit sites by sharing costs.

Therefore, the resolution of this safety issue was assumed to be an improvement in operator performance. For some utilities (approximately 10% of the total), this issue essentially had no effect because: (1) their existing training programs would be accredited with little effort; and (2) the quality of their programs was sufficiently high that accreditation would result in no discernible improvement in their operators' performance. Other utilities were expected to see varying degrees of improvement. Those with training programs that were below the accreditation standards were to be brought closer to the high quality enjoyed by the outstanding utilities. Overall, the effect on operator human error was estimated to be a reduction of 10% across the affected portion of the industry. The detailed assumptions for this analysis were as follows:

(1) Applicable Plants: 90% of all plants - 43 BWRs and 86 PWRs, or 129 plants.

(2) Selected Analysis Plant: Oconee-3 - representative PWR. It was assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf-1) would be equivalent to those for the representative PWR. Therefore, the analysis was conducted only for a PWR, but the fractional risk and core-melt frequency reductions were also applied to a BWR.

Frequency/Consequence Estimate

Based on the PNL analysis[64] and assuming a typical midwest-type meteorology and an average population density of 340 people per square-mile at U.S. reactor sites, the estimated public risk reduction was 26,180 man-rem.

Cost Estimate

Industry Cost: The PNL panel estimated[64] the one-time industry cost to implement the change initially to be in the range of \$0.1M to \$1M/reactor. Those plants with training programs closer to accreditable status would enjoy the smaller costs. The best estimate for the average plant was taken to be \$0.3M. Operation under the accreditation program was estimated to cost between \$0.05M and \$0.25M/plant annually for additional funding to maintain an accredited training program; the best estimate was \$0.1M/plant annually. The following is a breakdown of the industry cost:

(1) Implementation: Approximately 3 man-years (\$300,000/plant) to: (1) review accreditation standards; (2) compare the existing utility practices with the developed standards; and (3) plan the necessary upgrades and implement the program upgrades to fulfill the accreditation requirements. For 129 affected plants, this cost was estimated to be \$39M.

(2) Operation and Maintenance: \$100,000/plant-year for: (1) the time invested by the staff in upgraded training (increased course time, quality, etc.); and (2) instruction upgrade (time, quality, etc.). For 129 affected plants with an average remaining life of 28 years, this cost was estimated to be \$360M.

Thus, the total industry implementation, operation, and maintenance cost for the possible solution was estimated to be \$399M.

NRC Cost: The NRC cost to implement the accreditation was estimated to be \$0.635M which was equivalent to 330 man-weeks to: (1) accredit, predicated on the possibility that INP0 accreditation would not be forthcoming; and (2) develop accreditation standards and regulations for adoption by the industry. The annual operational cost to the NRC was estimated[64] to be \$100,000 or one man-year for additional OIE efforts to ensure industry maintenance of standards (at all plants). For an average remaining plant life of 28 years, this operation and maintenance cost was estimated to be \$2.8M. Thus, the total NRC cost for a solution to this issue was \$3.435M.

Total Cost: The total industry and NRC cost associated with the possible solution was \$(399 + 3.435)M or \$402.4M.

Value/Impact Assessment

Based on an estimated public risk reduction of 26,180 man-rem and a cost of \$402.4M for a possible solution, the value/impact score was given by:

$$S = \frac{26{,}180\text{man} - \text{rem}}{\$402.\text{M}}$$

$$= 65\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

The industry accident avoidance cost was estimated by PNL[64] to be $14M. The occupational risk reduction was estimated to be 22,170 man-rem resulting from accident avoidance (170 man-rem) and from operation and maintenance of the solution (22,000 man-rem).

CONCLUSION

Although the value/impact score was low, this issue was given a medium priority ranking (see Appendix C) because of the magnitude of the potential public risk reduction. However, in June 1985, the Commission recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry.[777] Thus, this item was RESOLVED and no new requirements were established.

REFERENCES

0016.	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.
0048.	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.
0054.	NUREG/CR#1659, "Reactor Safety Study Methodology Applications Program," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1981, (Vol. 2) May 1981, (Vol. 3) June 1982, (Vol. 4) November 1981.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0174.	*NUREG-0585*, "TMI Lessons Learned Task Force Final Report," U.S. Nuclear Regulatory Commission, October 1979.
0226.	Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1971, (Rev. 1) September 1975 [8801130111], (Rev. 1#R) May 1977 [7907100073], (Rev. 2) April 1987. [8907180147]
0251.	SECY-79-330E, "Qualifications of Reactor Operators," U.S. Nuclear Regulatory Commission, July 30, 1979. [7910020256, 7910020279]
0252.	NRR-80-117, "Study of Requirements for Operator Licensing," U.S. Nuclear Regulatory Commission, February 4, 1982. [8203180234]
0253.	ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," American National Standards Institute, 1981.
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0437.	Memorandum for H. Denton from R. Minogue, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," March 29, 1983. [8401160475]

0438.	Memorandum for G. Cunningham et al. from W. Dircks, "NRC Actions Required by Enactment of the Nuclear Waste Policy Act of 1982," January 19, 1983. [*ML040440387*]
0777.	Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan Items I.A.2.2 and I.A.2.7 Training and Qualifications of Operating Personnel," June 24, 1985. [8507020587]
0800.	NUREG/CR-4258, "An Approach to Team Skills Training of Nuclear Power Plant Control Room Crews," U.S. Nuclear Regulatory Commission, July 1985.
0801.	Memorandum for W. Dircks from H. Denton, "Team Training for Nuclear Power Plant Control Room Crews," July 10, 1985. [8507220495]
0802.	NUREG/CR-3739, "The Operator Feedback Workshop: A Technique for Obtaining Feedback from Operations Personnel," U.S. Nuclear Regulatory Commission, September 1984.
0803.	NUREG/CR-4139, "The Mailed Survey: A Technique for Obtaining Feedback from Operations Personnel," U.S. Nuclear Regulatory Commission, May 1985.
0804.	Memorandum for W. Dircks from H. Denton, "TMI Action Plan Item I.A.2.6(4)," September 25, 1985. [8510030079]
0956.	Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987. [8701140115]
0966.	*Federal Register* Notice 50 FR 11147, "10 CFR Ch. 1, Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel," March 20, 1985.
1043.	SECY-86-348, "Final Rulemaking for Revisions to Operator Licensing—10 CFR 55 and Conforming Amendments," U.S. Nuclear Regulatory Commission, November 21, 1986. [8701020003]
1044.	*Federal Register* Notice 52 FR 16007, "Regulatory Guides; Issuance and Availability," May 1, 1987.
1045.	Memorandum for V. Stello from E. Beckjord, "Resolution of TMI Action Plan Items and Human Factors Issues," May 18, 1987. [8710280270]

Task I.A.3: Licensing and Requalification of Operating Personnel (Rev. 6) ( )

The objectives of this task were to: (1) upgrade the requirements and procedures for nuclear power plants operator and supervisor licensing to assure that safe and competent operators and senior operators are in charge of the day-to-day operation of nuclear power plants; and (2) increase the requirements for initial issuance of licenses and for license renewals and provide closer NRC monitoring of licensed activities.

ITEM I.A.3.1: REVISE SCOPE OF CRITERIA FOR LICENSING EXAMINATIONS

DESCRIPTION

This NUREG-0660[48] item called for NRR to notify all operator license holders and applicants of the new scope of examinations and criteria for issuance of reactor operator (RO) and senior reactor operator (SRO) licenses and renewal of licenses. Simulator examinations were to be included as part of the license examination. As a result of Public Law 97-425, it was determined that additional staff work on the issue was required and a proposed rule for operator licensing was presented to the Commission in SECY-84-76.[593] Approval of this rule was expected to effectively close out this item.

CONCLUSION

This item was clarified in NUREG-0737[98] and new requirements were established.

ITEM I.A.3.2: OPERATOR LICENSING PROGRAM CHANGES

DESCRIPTION

This NUREG-0660[48] item called for NRR to take the following actions:

(1) Develop and implement a plan to relocate Operator Licensing Branch (OLB) examiners at Nuclear Power Plant Simulator Training Centers or in Inspection and Enforcement Regions.

(2) Conduct a study of the staffing of the operator licensing program and the qualifications and training of examiners.

(3) Develop and implement a plan to report operator errors and to act on operator errors with respect to continuation of licensing.

In response to the above actions, the following were accomplished:

(1) The administering of examinations and issuance/renewal of operator licensing were transferred to Region III in FY 1982 and to Region II in FY 1983. As a result of these changes, all regions had operator licensing authority in FY 1984. NRR provided oversight and guidance, including examination procedures and criteria.[88]

(2) A study of the staffing of the operator licensing program and the qualifications and training of examiners was completed in November 1980 and documented in NUREG/CR-175O.[89] Examiner standards were published in NUREG-1021.[962]

(3) A plan for reporting operator errors and for acting on operator errors with respect to continuation of licensing was developed in NUREG/CR-1750.[89] However, after review of this recommended plan, DHFS/NRR concluded that no further action was required.[440]

CONCLUSION

This item was RESOLVED and no new requirements were established.[956]

ITEM I.A.3.3: REQUIREMENTS FOR OPERATOR FITNESS

DESCRIPTION

Historical Background

This NUREG-0660[48] item called for the staff to develop a regulatory approach to: (1) provide assurance that applicants for RO and SRO licenses were psychologically fit; and (2) prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds. The regulations were to be applied to all operating and future power plants.

A proposed rule addressing alcohol and drug use and the broader issue of fitness for duty of operating licensee personnel and contractors was forwarded to the EDO on April 16, 1982, with the concurrence of several NRC offices. The proposed Fitness for Duty Rule was issued for public comment in the Federal Register on August 15, 1982, with the public comment period extending to October 5, 1982. A final rule package was completed on December 1, 1982, and a final rule was expected to be published by April 1, 1983. The rule, when promulgated, would have required facilities licensed under 10 CFR 50.21(b) or 10 CFR 50.22 to establish and implement adequate written procedures to provide reasonable assurance that persons with unescorted access to protected areas of nuclear power plants, while in those areas, are not under the influence of alcohol, other drugs, or otherwise unfit for duty due to mental or physical impairments. Secondly, a proposed rule amending 10 CFR 73.56 regarding access authorization for nuclear power plants had not been completed, although a value/impact analysis in support of the proposed rule had been prepared. Staff studies of the issue were published in NUREG/CR-2075[289] and NUREG/CR-2076.[290]

Safety Significance

There could be significant damage if impaired personnel were performing critical safety operations. Legal and institutional problems could limit a thorough implementation of the proposed program. Given that there was an adequate program implemented at all power plants and integrated into overall plant operations, the new program would reduce operator error which, in turn, would lower the risk associated with operation of the plants.

Possible Solutions

This issue had two components: the first involved initial access to protected areas of nuclear power plants and the second involved continuing fitness for duty once initial access has been granted. The proposed Fitness for Duty Rule, issued for public comment on August 15, 1982, was directed toward the second component of this issue, mandating behavioral observation programs for power plants licensed by the NRC. Behavioral observation was also a part of the proposed Access Authorization Rule directed toward the first component of this issue.

The second component of this issue dealt with limiting access of psychologically unstable individuals to vital plant areas. This component was expected to have a major cost impact on the industry because this access authorization program was comprehensive in that it was aimed at limiting the access to vital plant areas of disgruntled employees, psychologically unsuitable employees, as well as personnel under the influence of drugs or alcohol.

The access authorization program had the following three parts: (1) background search; (2) psychological assessment; and (3) behavior observation. The first two parts would occur prior to granting an individual an unescorted access authorization to protected and vital areas, and the last part would be an ongoing activity for individuals who have been granted an unescorted access authorization. The background check would examine an individual's past for unstable activities, a criminal record, credit problems, and previous employment problems. It was established by NRC personnel that data on psychological screening showed that 2% to 3% of white-collar workers were identified as unstable and, for blue-collar employees, the rate ranged from 7% to 10%. These figures provided the background for the assumptions made in the evaluation below.

PRIORITY DETERMINATION

This issue was assessed by PNL[64] in consultation with a number of engineers with expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

Assumptions

The major result of this issue was assumed to be a reduction in operator error. For some utilities, this new system would result in some reduction in operator error whereas, in others, it would have no discernible effect. Based on engineering judgment, an average of 2% was arrived at by PNL to apply to all operating and future plants. Thus, this issue assumed the implementation of the access authorization system at all 134 plants, either under construction (63) or in operation (71), with average remaining lives of 28.8 years for 90 PWRs and

27.4 years for 44 BWRs. Thus, the total remaining life of the affected plants was [(28.8)(90) + (27.4)(44)]RY or 3,798 RY. Neither the implementation, operation, or maintenance of the solution would involve any changes in occupational dose.

For the analysis performed by PNL,[64] Oconee-3 was taken as the representative PWR. It was assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf-1) would be equivalent to those for the representative PWR. Therefore, the analysis was conducted only for a PWR, but the fractional risk and core-melt frequency reductions were also applied to a BWR.

Frequency Estimate

All release categories were affected by this issue, but the principal Release Categories affected by the solution were 3, 5, and 7; the numerical calculations were based on these categories. The calculated[64] reductions in core-melt frequency were 4 x 10^{-7}/RY for PWRs and 1.8 x 10^{-7} /RY for BWRs.

Consequence Estimate

The dose calculations were based on a reactor site population density of 340 people per square-mile and a typical midwest meteorology. Based on the above core-melt frequency reduction and Release Categories, the total estimated public risk reduction was 16,000 man-rem. The occupational risk reduction for implementation, operation, and maintenance was zero.

Cost Estimate

Industry Cost: A value/impact analysis in support of the anticipated Access Authorization Rule had been prepared by the staff and industry cost estimates had been developed. These cost estimates, which were reviewed and accepted by the Atomic Industrial Forum (AIF), were as follows:

(1) For all operating plants, the implementation cost was $140,000/plant and included the preparation of plant and associated procedures ($33,000), licensee management and clerical staff ($63,000), training to implement the behavioral observation program ($34,000), and storage for files ($10,000). The total industry implementation cost for operating plants was $(140,000)(71) or $9.94M.

(2) For all future plants (at which none of the employees were to be grandfathered), the implementation costs were estimated to be $590,000/plant. In addition to the costs noted above for operating plants, this implementation cost included background investigations ($375,000), review process and appeals procedures ($36,000), increased file storage requirements ($30,000), and miscellaneous criminal checks with the FBI, etc. ($9,000). The total industry cost for future plants was ($590,000)(63) or $37.2M.

(3) The cost of operating of the access authorization system at each plant was estimated to be $300,000/ year. This cost included background investigations for new people as a result of employee turnover ($94,000), professional management and clerical staff ($63,000), a review and appeal process ($67,000), refresher training for old supervisors ($19,000), training of new supervisors ($9,000), plan maintenance and updates ($8,000), file storage ($39,000), and criminal history checks with the FBI for new people ($2,000). The total industry cost for operation and maintenance of the access authorization system was ($0.3M/RY)(3,798 RY) or $1,140M.

The total industry cost for the possible solution was $[9.94 + 37.2 + 1,140]M or $1,187M.

NRC Cost: Further development and issuance of the proposed plan was estimated to take 1.5 man-years; at a rate of $100,000/man-year, the cost for this effort was $150,000. The review and modification of the utilities' implementation plans was estimated to take 1.5 man-years. For the 134 affected plants, this amounted to 0.6 man-week/plant. At a cost of $2,270/man-week, the implementation cost was $182,500. Review of the operation and maintenance of the possible solution was estimated to require 1 man-week/RY for all plants. At a cost of $2,270/man-week, the total operation and maintenance of the solution was $8.6M. Thus, the total NRC cost for the possible solution was $[0.15 + 0.1825 + 8.6]M or $8.9M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(1,187 + 8.9)M or $1,196M.

Value/Impact Assessment

Based on an estimated public risk reduction of 16,000 man-rem and a cost of $1,196M for a possible solution, the value/impact score was given by:

$$S = \frac{16{,}000\text{man} - \text{rem}}{\$1{,}196M}$$
$$= 13.4\text{man} - \text{rem} / \$M$$

Other Considerations

It was estimated by cognizant NRC personnel that the Fitness for Duty Rule would have a negative cost impact on operating licensees in the long run. The staff estimated that initial licensee burden to develop written procedures required by the rule would be approximately 1,200 man-hours over a six-month period at a total cost between $50,000 and $75,000, if no fitness for duty program existed at a licensee's facility. While utilities such as TVA claimed that alcohol abuse alone cost them approximately $18.5M annually, fitness for duty programs of the type envisioned by the Fitness for Duty Rule were expected to save costs through quicker identification of employees not fit for duty and through assisting these employees, in whom considerable resources had been invested, in returning to high levels of productivity.

Nationwide, absenteeism due to alcohol and drug abuse cost U.S. industries an average of $300 annually for every worker. Alcohol drug-abusers lose an additional 25% of their productive time when on the job, at an average annual cost to U.S. industries of approximately $2,900/abuser. At the time this issue was evaluated, the total annual cost to U.S. industries was between $12 Billion and $15 Billion. Wrich, in "The Employee Assistance Program; Updated for the 1980's," Hazelden, 1980, reported that U.S. industries received a return of $10 in decreased absenteeism, accidents, and increased productivity for every dollar spent on fitness for duty.

CONCLUSION

Although the estimated risk reduction was 16,000 man-rem and the value/impact score was only 13.4 man-rem/$M, this issue was given a high priority ranking (see Appendix C) because of its advanced state of completion.

On October 24, 1984, the Commission notified the staff that it would not promulgate a rule on fitness for duty for a minimum of two years but would issue a policy statement on the subject. A proposed policy statement was submitted to the Commission in SECY-85-21.[963] In a separate action, a notice withdrawing the final Fitness for Duty Rule was submitted to the Commission in SECY-85-21A.[964] The proposed policy statement (SECY-85-21[963]) was reaffirmed by the staff in SECY-85-21B.[965] In recognition of the industry's efforts in establishing fitness for duty programs, the Commission approved a Policy Statement[967] in July 1986. Thus, this issue was RESOLVED and no new requirements were established.[956]

ITEM I.A.3.4: LICENSING OF ADDITIONAL OPERATIONS PERSONNEL

DESCRIPTION

Historical Background

This NUREG-0660[48] item sought to upgrade the operations performance in nuclear power plants by imposing licensing requirements upon other operations personnel in addition to ROs and SROs.

Safety Significance

It was possible that, by undergoing licensing, personnel such as managers, engineers, and technicians would be better qualified and less likely to commit errors in performing their functions.

Possible Solution

A study could be undertaken to determine which, if any, personnel should be licensed. Licensing would then be required by the NRC for those additional personnel.

PRIORITY DETERMINATION

Assumptions

It was estimated that the effects of resolution of this issue would be minimal for many utilities since there were existing practices that went a long way toward ensuring that qualified and trained individuals were in responsible positions. It was assumed that additional licensing requirements would produce some improvement by assisting in the screening of potentially poor performers from the operations staff. The net effect was estimated to be equivalent to a 2% reduction in human error rates for reactor operators and maintenance personnel.[64]

Frequency Estimate

Based on the 2% reduction in human error rate, the Oconee 3 (representative PWR) risk equation parameters were adjusted. All Accident Sequences except V were assumed to be affected and all Release Categories were affected. The reduction in core-melt frequency for Oconee 3 was calculated to be 1.4 x $1O^{-6}$/RY. The reduction in core-melt frequency for Grand Gulf 1 was then calculated by assuming that the fractional core-melt frequency reduction for the representative BWR would be equivalent to the fractional reduction for a PWR. Therefore, since the Oconee-3 fractional reduction was 0.017, the core-melt frequency reduction for Grand Gulf-1 was calculated to be 6.3 x 10^{-7}/RY.

Consequence Estimate

The corresponding reduction in public risk for Oconee-3 was calculated to be 2.4 man-rem/RY and the public risk reduction for Grand Gulf-1 was calculated to be 2.7 man-rem/RY. The total risk reduction for each type of plant was given as follows:

PWRs: (28.5 yrs)(95 reactors)(2.4 man-rem/RY) = 6.5 x 10^3 man-rem

BWRs: (27 yrs)(49 reactors)(2.7 man-rem/RY) = 3.6 x 10^3 man-rem

Therefore, the total risk reduction for this issue was 1.01 x 10^4 man-rem.

Cost Estimate

Industry Cost: It was assumed that the required additional effort to license the majority of the operations personnel at a plant would be roughly equivalent to the existing licensing efforts for ROs and SROs; this was estimated to be $250,000/plant. For operation, industry would have to provide new training staff, staff time for training and exams, and administration; this was estimated to be $50,000/RY. Therefore, the total industry cost was $250M.

NRC Cost: To implement the solution, qualification criteria, licensing exams, and procedures would have to be prepared and would be a major undertaking. The implementation cost was estimated to range from $20M to $50M; for analysis purposes, $35M was used. To operate with the new licensing requirements, 50 additional staff members would be required at a total cost of $5M/year. To perform the annual operational needs of the program, funds would be needed for travel, publications, etc. This was estimated to be an additional $2M/year. Therefore, the total NRC cost was approximately $240M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(250 + 240)M or $490M.

Value/Impact Assessment

Based on an estimated public risk reduction of 10,100 man-rem and a cost of $490M, the value/impact score was given by:

$$S = \frac{10{,}100\text{man} - \text{rem}}{\$490\text{M}}$$
$$= 20\text{man} - \text{rem} / \$\text{M}$$

Uncertainty

Because the estimate of the value/impact score relied heavily on the estimated value of the possible reduction in human error rate, the effective improvement could vary significantly.

Other Considerations

DHFS/NRR had been pursuing this issue and the Commission concluded[181] that licensing of managers should not be required. The other portion of the issue (i.e., licensing of other personnel - engineers, maintenance personnel, etc.) was still under study and was to be concluded in FY 1983.

CONCLUSION

Although the value/impact score was low, the potential risk reduction was considered and this issue was given a medium priority ranking (see Appendix C). However, in February 1985, the staff determined that there was insufficient evidence to support the licensing of additional plant personnel.[778] Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.3.5: ESTABLISH STATEMENT OF UNDERSTANDING WITH INPO

DESCRIPTION

As part of the overall evaluation of the TMI incident, it was determined[48] that a statement of understanding was needed to address the mutual intent of NRC and INPO concerning the extent to which NRC should review or rely upon training, certification, and other INPO activities. Consideration was also to be given to providing alternative mechanisms for industry to inform NRC of its general progress on needed safety reforms. It was intended that the statement of understanding would provide a basis for evaluation of any safety reforms or programs. Since there was no direct risk that could be attributed to this issue, it was considered to be a Licensing Issue.

CONCLUSION

A Memorandum of Agreement[148] between INPO and NRC was issued in April 1982; however, it did not specifically address training and certification. Following further revision, the EDO agreed[594] with the Coordination Plan for NRC/INPO Training-Related Activities (Appendix Four to the Memorandum of Agreement) in November 1983. With the issuance of the Memorandum of Understanding, this Licensing Issue was resolved.

REFERENCES

0006.	IE Bulletin 79-12, "Short Period Scrams at BWR Facilities," U.S. Nuclear Regulatory Commission, May 31, 1979. [ML080310488]
0048.	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.
0064.	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.
0088.	Memorandum for All NRR Employees from H. Denton, "Regionalization of Selected NRR Functions," June 15, 1982. [9507280052]
0089.	NUREG/CR#1750, "Analysis, Conclusions, and Recommendations Concerning Operator Licensing," U.S. Nuclear Regulatory Commission, January 1981.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0148.	"Memorandum of Agreement between the Institute of Nuclear Power Operations and the U.S. Nuclear Regulatory Commission," (Rev. 1) April 1, 1982. [8207010053]
0181.	SECY-82-155, "Public Law 96-295, Section 307(B), Study of the Feasibility and Value of Licensing of Nuclear Plant Managers and Senior Licensee Officers," U.S. Nuclear Regulatory Commission, April 12, 1982. [8205050080]
0289.	NUREG/CR#2075, "Standards for Psychological Assessment of Nuclear Facility Personnel," U.S. Nuclear Regulatory Commission, July 1981.
0290.	NUREG/CR#2076, "Behavioral Reliability Program for the Nuclear Industry," U.S. Nuclear Regulatory Commission, July 1981.

0440.	Memorandum for W. Minners from D. Ziemann, "Schedules for Resolving and Completing Generic Issues," April 5, 1983. [8304180758]
0593.	SECY-84-76, "Proposed Rulemaking for Operator Licensing and for Training and Qualifications of Civilian Nuclear Power Plant Personnel," U.S. Nuclear Regulatory Commission, February 13, 1984. [8403260357]
0594.	Letter to E. Wilkinson (Institute of Nuclear Power Operations) from W. Dircks (U.S. Nuclear Regulatory Commission), November 23, 1983. [8312090099]
0778.	Memorandum for W. Dircks from H. Denton, "TMI Action Item I.A.3.4," February 12, 1985. [8502260084]
0956.	Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987. [8701140115]
0962.	*NUREG-1021*, "Operator Licensing Examiner Standards," U.S. Nuclear Regulatory Commission, October 1983.
0963.	SECY-85-21, "Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel," U.S. Nuclear Regulatory Commission, January 17, 1985. [8502280427]
0964.	SECY-85-21A, "Withdrawal Notice: Fitness for Duty of Nuclear Power Plant Personnel," U.S. Nuclear Regulatory Commission, April 12, 1985. [8505030703]
0965.	SECY-85-21B, "Fitness for Duty of Nuclear Power Plant Personnel," U.S. Nuclear Regulatory Commission, August 26, 1985. [8510150472]
0967.	*Federal Register* Notice 51 FR 27921, "Commission Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel," August 4, 1986.

Task I.A.4: Simulator Use and Development (Rev. 6) ( )

The objectives of this task were to: (1) establish and sustain a high level of realism in the training and retraining of operators, including dealing with complex transients involving multiple permutations and combinations of failures and errors; and (2) improve operators' diagnostic capability and general knowledge of nuclear power plant systems.

ITEM I.A.4.1: INITIAL SIMULATOR IMPROVEMENT

ITEM I.A.4.1(1): SHORT-TERM STUDY OF TRAINING SIMULATORS DESCRIPTION

This TMI Action Plan[48] item called for a short-term study of training simulators to collect and develop corrections for identified weaknesses.

CONCLUSION

A study of training simulators was undertaken and NUREG/CR-1482[299] was published in June 1980. Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.4.1(2): INTERIM CHANGES IN TRAINING SIMULATORS

DESCRIPTION

This TMI Action Plan[48] item called for the development of requirements to correct specific training simulator weaknesses, based on the short-term study results from Item I.A.4.1(1).

CONCLUSION

This item was RESOLVED with the issuance of Regulatory Guide 1.149[439] in April 1981 and new requirements were established.

ITEM I.A.4.2: LONG-TERM TRAINING SIMULATOR UPGRADE

The four parts of this item were combined and evaluated together.

DESCRIPTION

Historical Background

Nuclear power plant simulators were recognized as an important part of reactor operator training and this TMI Action Plan[48] item called for a number of actions to improve simulators and their use.

There was significant interaction among the simulator-related action items and clear separation of this item was difficult. Item I.A.4.2 had a number of components dealing with long-term upgrades. The NUREG-0660[48] description called for research to: (1) improve the use of simulators in training operators; (2) develop guidance on the need for and nature of operator action during accidents; and (3) gather data on operator performance. Specific research items mentioned included simulator capabilities, safety-related operator action, and simulator experiments. The item also called for the upgrading of training simulator standards, specifically the updating of ANSI/ANS 3.5-1979. A regulatory guide endorsing this standard and giving the criteria for acceptability was also mentioned. The final portion of Item I.A.4.2 called for a review of simulators to ensure their conformance to the criteria.

At the time the issue was initially evaluated, a significant portion of the activities to be conducted had been completed. For example, ANSI/ANS 3.5 was revised and issued in 1981 and Regulatory Guide 1.149,[439] which endorsed this standard, had been published along with numerous research reports. It was clear that the regulations, the ANS standard, and the regulatory guide did not require a site-specific simulator. 10 CFR 55 states that, if a simulator is used in training, it "shall accurately reproduce the operating characteristics of the facility involved and the arrangement of the instrumentation and controls of the simulator shall closely parallel that of the facility involved." ANSI/ANS 3.5-1981 called for a high degree of fidelity between the simulator and the "reference plant." However, there was no requirement that the reference plant be the same facility that the personnel in training would operate. Regulatory Guide 1.149[439] explicitly made the distinction stating "the similarity that must exist between a simulator and the facility that the operators are being trained to operate is not addressed in the guide and should not be confused with the guidance provided that specifies the similarity that should exist between a simulator and its reference plant."

The work that had been completed for Item I.A.4.2(1) included the issuance of NUREG/CR-2353[300] (Volumes I and II), NUREG/CR-1908,[416] NUREG/CR-2598,[417] NUREG/CR-2534,[418] NUREG/CR-3092,[419] and NUREG/CR-3123.[653] This item, however, had long-range requirements calling for: (1) the review of operating experience to provide data on operator responses; and (2) the design and conduct of experiments to determine operator error rates under controlled conditions. Items I.A.4.2(2) and I.A.4.2(3) were completed with the issuance of Regulatory Guide 1.149.[439] Item I.A.4.2(4) addressed the long-term training simulator improvement criteria which were established in Regulatory Guide 1.149[439] and initiated in FY 1982. However, the staff review of submittals from simulator owners for conformance with the criteria was an ongoing task in 1983. Therefore, the outstanding portions of this issue (the continuation of simulator research and the review for conformance to acceptability criteria) were evaluated.

Safety Significance

Use of simulators with high fidelity to the reference plant would significantly improve operator training in dealing with abnormal conditions thereby reducing operator error. Operators' performance under accident conditions was expected to be enhanced. Thus, a potential core-melt would be avoided and overall core-melt frequency reduced.

Possible Solution

A possible solution was to establish a high level of realism in the training and retraining of plant operators by developing simulators with a high degree of fidelity to the reference plant.

PRIORITY DETERMINATION

The assessment of this issue was conducted by PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provided utility training services and had experience in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

In the assessment of this issue, it was necessary to acknowledge that many of the TMI items associated with operator training were interrelated and that ranking problems surface when an attempt is made to assess these independently. For example, this item was related to Items I.A.2.6(1,2,3, and 5), which dealt with training improvements, including the enhanced use of existing simulators. I.A.4.1, dealt with initial simulator improvement, including short-term and interim changes in training simulators. However, the final safety ranking of this issue was relatively insensitive to changes in the basic assumptions used to distinguish these interrelated issues by the very nature of the ranking matrix. Therefore, it was possible to establish a priority ranking for this issue, despite the possible overlapping of potential benefits and costs with the other interrelated issues.

Assumptions

It was assumed that the major effect of these issues, both in terms of safety benefit and cost incurred, would be the enhancement of the level of realism imparted by simulators. The modeling capabilities given under Item I.A.4.1(2) and in ANSI/ANS 3.5-1981 reflected this feature.

It was assumed that, in order to provide the intended level of realism, site-specific simulators would be acquired. Such simulators would be significantly more realistic when compared to the specific facilities, both in layout and operation, than existing generic simulators. In addition, they were assumed to enhance transient and accident modeling capabilities.

It was clear that provision of site-specific simulators, while not explicitly required, would meet the requirements of Item I.A.4.1(2), the fidelity requirements of ANSI/ANS 3.5-1981, and the accurate reproduction requirements of 10 CFR 55. Less sweeping simulator enhancements might also fulfill these requirements but would have to be decided on a case-by-case basis. Therefore, it was assumed that the enhancement would be effected by the introduction of site-specific simulators.

The public risk reduction (and occupational dose reduction due to accident avoidance) were associated with the reduction in operator error expected to result from the training and requalification of operators on improved simulators. Inasmuch as any studies relating human error rates to the realism of simulator training were not available, this assessment was based primarily on PNL engineering judgment. Therefore, it was estimated that a reduction in operator error rate of 30% would result from the resolution of this issue. This estimate implied that, for specific instances, the improvement could be much greater, but the 30% reduction was used as an estimate of the average improvement.

There were 90 PWRs and 44 BWRs affected by this issue with average remaining lives of 28.8 years and 27.4 years, respectively. The representative plants selected for analysis were Oconee-3 and Grand Gulf-1 for PWRs and BWRs, respectively. (It was assumed that the fractional risk and core-melt frequency reductions for Grand Gulf would be equivalent to those for a PWR which was calculated directly.)

Frequency Estimate

All release categories were affected by the resolution of the issue. The calculated core-melt frequencies were 8.2×10^{-5}/RY for PWRs and 3.7×10^{-5}/RY for BWRs. The reduction in these frequencies, based on the 30% reduction estimated for operator error, was 1.3×10^{-5}/RY for PWRs and 5.9×10^{-6}/RY for BWRs.

Consequence Estimate

The dose calculations were based on a reactor site population density of 340 people/square-mile and a typical midwest meteorology. The resulting total reduction in public risk was 150,000 man-rem.[64]

Cost Estimate

Industry Cost: The major effect of the resolution of these issues was assumed to be the acquisition and use of site-specific simulators. The cost of such an undertaking would be substantial. If improved modeling changes were possible on existing simulators, the cost to industry would be substantially smaller. However, this was not clear at the time of the evaluation and it was assumed that new simulators would be required. (The impact of this assumption could be weighed subsequently in the final safety priority ranking. The assumption could be reevaluated at that time for any appropriate modifications.)

Assuming that new simulators would be required, the principal implementation cost would be the purchase of the simulators and provision of the new training materials. The capital cost of a simulator was estimated to be $7M. The provision of training materials was estimated to be equivalent to a 7 man-year effort.

It was assumed that all reactors, both operating and planned, would be affected. However, not every reactor would require a simulator. Many reactor sites have two or more reactors located together. If these reactors were sufficiently similar, a single simulator could serve them. Examining the list of 134 operating and planned power reactors, it was estimated that 62 additional site-specific simulators would be adequate. This assumed that 20% of the potential simulators were not required because either a site-specific simulator already existed or the plant in question was an older facility with limited remaining life.

The cost of the 62 new simulators spread over 134 reactors yielded $3.2M/reactor in capital cost and 3.2 man-year/reactor to provide new training materials. The operation and maintenance of the new simulators was estimated to require 3 man-years/simulator. Again, sharing the expense for 62 simulators over 134 reactors yielded 1.4 man-years/reactor. Industry may also experience costs stemming from participation in simulator experiments and research; however, these costs would be small in comparison to the costs related to new simulators. Based on these assumptions, the total industry cost was obtained as follows:

(1) Implementation

Labor: $$\frac{[7\,man-year]}{simulator}\,\frac{[62\,simulators]}{134\,plants}\,\frac{[\$100{,}000]}{man-year} = \$320{,}000\,/\,plant$$

Equipment: $$\frac{[62\,simulators]}{134\,plants}\frac{[\$7\,M]}{simulator} = \$3.2\,M\,/\,plant$$	
Thus, the total industry implementation cost was (134 plants)$(0.32 + 32)M/plant or $470M.	

(2) Operation and Maintenance

$$\frac{[1.4\,man-year]}{reactor}\frac{[\$100{,}000]}{man-year}\left[(90\,PWRs)(28.8\,years) + (44\,BWRs)(27.4\,years)\right]$$

= $530M

Therefore, the total industry cost was $(470 + 530)M or $1,000M.

NRC Cost: There was no cost for development of a solution since all work was essentially complete and a solution had been identified. The principal costs were the continuation of research and the conduct of the confirmatory reviews. No additional development costs were foreseen as ANSI/ANS 3.5 was being revised and necessitated a revision to Regulatory Guide 1.149.[439]

The continuing research was treated as an implementation cost. It was estimated to require one NRC man-year and $1M in contractor support. (This included the remaining costs associated with Item I.E.8.) The confirmatory reviews were also treated as an implementation cost and were estimated to require 4 man-weeks/ simulator, or a total of 248 man-weeks for the assumed 62 new simulators.

The operational review cost to the NRC was minimal. It was assumed that annually each simulator would be audited to ensure that reference plant updates had been adequately represented on the simulator. Such an annual review was estimated to require 2 man-weeks/simulator or 124 man-weeks/year for all 62 new simulators assumed. NRC costs were estimated as follows:

(1) Implementation

Continuing Research:	**1 man-year =**	**0.33 man-week**
	134 plants	plant

Initial Simulator Reviews:	**248 man-weeks =**	**1.9 man-weeks**
	134 plants	plant

Based on a total NRC manpower of 2.23 man-weeks/plant, the implementation cost was given by:

$$\frac{(2.23\,man-weeks)}{plant}\frac{(\$2{,}270)}{man-week}(134\,plants) = \$678{,}300$$

With contractor support estimated to be $1M, the total implementation cost was $(0.6783 + 1)M or $1.7M.

(2) Review of Operation and Maintenance

$$\frac{(2\,man-weeks)}{simulator-year}\frac{(67\,simulators)}{134\,plants}\frac{(\$2{,}270)}{man-week} = \$2{,}100\,/\,RY$$

The cost for review of operation and maintenance for all affected plants was [(90 PWRs)(28.8 years) + (44 BWRs)(27.4 years)]($2,100/RY) or $8M.

Thus, the total implementation, operation, and maintenance cost was $(1.7 + 8)M or $9.7M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(1,000 + 9.7)M or $1,010M.

Value/Impact Assessment

Based on an estimated public risk reduction of 150,000 man-rem and a cost of $1,010M for a possible solution, the value/impact score was given by:

$$S = \frac{150{,}000\text{man} - \text{rem}}{\$1{,}010\text{M}}$$

$$= 148.7\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

The estimated reduction in occupational dose was 820 man-rem, based on accident avoidance only, since there were no implementation or maintenance dose reductions associated with resolution.

CONCLUSION

Based on the estimated risk reduction of 150,000 man-rem and the value/impact score of approximately 150 man-rem/$M, this issue was given a high priority ranking (see Appendix C). In view of the large estimated risk reduction, this ranking was essentially unaffected by any reasonable uncertainties in the cost estimates.

ITEM I.A.4.2(1): RESEARCH ON TRAINING SIMULATORS

This item was evaluated in Item I.A.4.2 above and was given a high priority ranking (see Appendix C). In April 1987, the issue was RESOLVED with the publication of Revision 1 to Regulatory Guide 1.149[439] and new requirements were established.[1045]

ITEM I.A.4.2(2): UPGRADE TRAINING SIMULATOR STANDARDS

This item was and RESOLVED with the issuance of Regulatory Guide 1.149[439] in April 1981 and new requirements were established.

ITEM I.A.4.2(3): REGULATORY GUIDE ON TRAINING SIMULATORS

This item was RESOLVED with the issuance of Regulatory Guide 1.149[439] in April 1981 and new requirements were established.

ITEM I.A.4.2(4): REVIEW SIMULATORS FOR CONFORMANCE TO CRITERIA

This item was evaluated in Item I.A.4.2 above and was given a high priority ranking (see Appendix C). Staff efforts in resolving the issue resulted in the publication of a rule and a simulation facility evaluation procedure.

When this item was originally identified in 1980, the staff's approach was to require a submittal from each licensee in compliance with a regulatory guide (which later was issued as Regulatory Guide 1.149[439]) and to conduct a review of each simulator; there was no simulator regulation in effect at that time. However, in 1983, Section 306 of the Nuclear Waste Policy Act (Public Law 97-425) directed the NRC, in part, to establish "requirements for operating tests at civilian nuclear power plant simulators." This Congressional mandate had the effect of superseding the original intent of Item I.A.4.2(4) and required the staff to develop regulations for simulators. As a result, the approach taken by the staff for the resolution of Item I.A.4.2(4) was modified to comply with the Congressional mandate. The work scope was changed to reflect the fact that licensees, under the proposed regulation, would be required to certify their plant-referenced simulators to the NRC, and that NRC would perform an audit only when a need was identified, or upon request. Only in the case of those few licensees (estimated to be six), which were expected to seek NRC approval for a simulation facility that did not include a plant-referenced simulator, would the staff be obligated to review simulator documentation.

The final rule was published[1077] as 10 CFR 55.45 and states, in part: "The operating test will be administered in a plant walkthrough and in either (i) a simulation facility which the Commission has approved for use after application has been made by the facility licensee, or (ii) a simulation facility consisting solely of a plant-referenced simulator which has been certified to the Commission by the facility licensee." In support of these regulations, the staff initiated a program to develop a procedure for its evaluation of selected certified simulation facilities. This procedure was subjected to a pilot test prior to being issued in draft form for comment. As a result of comments received, the procedure was revised and issued in final form as NUREG-1258[1084] in December 1987. Thus, the item was RESOLVED and new requirements were established.1098

ITEM I.A.4.3: FEASIBILITY STUDY OF PROCUREMENT OF NRC TRAINING SIMULATOR

DESCRIPTION

The description of this issue in NUREG-0660[48] was as follows:

"In addition to the increased use of industry simulators for training of NRC staff (notably, the work by OIE with the TVA training center simulators), a feasibility study of the lease or procurement of one or more simulators to be located in the NRC headquarters area will be performed. These simulators would be used in familiarizing the NRC staff with reactor operations, in assessing the effectiveness of operating and emergency procedures and in gathering data on operator performance. The study will include development of specifications, development of procurement and commissioning schedules, estimation of costs, and comparison with other methods of providing such training for NRC personnel."

The intent of this issue was to improve the NRC staff's familiarization with reactor operations. The study was an effort to establish the feasibility of procuring an NRC training simulator. The issue had no direct bearing on public risk reduction and, therefore, was considered to be a Licensing Issue.

CONCLUSION

Technical studies[262,263,264] of the issue performed by BNL indicated that existing simulators had significant modeling limitations. It was established that the capability of existing simulators was not acceptable at any but near-normal operating conditions, and that the lack of technical capability during two-phase conditions was significant. These results had an adverse effect on the feasibility of a training simulator for the NRC staff. Thus, this Licensing Issue was resolved.

ITEM I.A.4.4: FEASIBILITY STUDY OF NRC ENGINEERING COMPUTER

DESCRIPTION

The purpose[48] of this study was to fully evaluate the potential value of and, if warranted, propose development of an engineering computer that realistically modeled PWR and BWR plant behavior for small-break LOCA and other non-LOCA accidents and transients that may call for operator actions. Final development of the proposed engineering computer would depend on a number of research efforts.

Risk assessment tasks (interim reliability evaluation program, or IREP, for example) to define accident sequences covering severe core damage would also provide the guidelines for the experimental and analytical research programs needed to improve the diagnostics and general knowledge of nuclear power plant systems. The programs would assist the development and testing of fast running computer codes used to predict realistic system behavior for these multiple accident studies. These codes would provide the basic models for use in the improved engineering computer as well as the capability for NRC audit of NSSS analyses. This issue had no direct effect on public risk reduction and, therefore, was considered a Licensing Issue.

CONCLUSION

Reports[262,263] on the review of PWR and BWR simulators were completed by BNL while work on Plant Analyzers continued at BNL, INEL, and LASL. RES believed that Plant Analyzers (Engineering Computer) would be helpful in uncovering potential operational safety problems in LWRs, caused by operator errors or equipment malfunctions, which would lead to risk reductions through increased operator awareness, improved procedures, and equipment redundancy.

The Plant Analyzer is not a design tool but rather an aid to the NRC staff in performing an audit function in the licensing process. After the second year of research on the Engineering Computer (Nuclear Plant Analyzer), it was concluded that it was not feasible to develop a device that would be sufficiently accurate and function with sufficient speed (i.e., faster than real accident progression time) to give a plant operator information adequate to guide action he or she should take during an accident. It was found, however, that a Nuclear Plant Analyzer, which takes output from an NRC safety analysis code such as TRAC or RELAP and displays plant accident conditions in schematic form on a video screen, would considerably ease the burden of understanding the results of complex safety analysis calculations. The Plant Analyzer also would allow the safety analyst to interpose simulated operator actions into an accident calculation underway. Based on these findings, the objectives of the development program were reoriented toward assistance for plant safety analysis and away from operator accident assistance.

A Management Plan[968] for the Nuclear Plant Analyzer was prepared by the staff and included a listing of products expected to enter the regulatory arena in fiscal years 1985 through 1989. The staff concluded that it was not feasible to develop an Engineering Computer to provide input for operator actions during plant accidents; it was feasible to develop a device to give NRC an improved capability to audit NSSS analyses and this was being done in accordance with the Management Plan. Thus, this Licensing Issue was resolved.

REFERENCES

0048.	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.
0064.	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.
0262.	BNL/NUREG#28955, "PWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.
0263.	BNL/NUREG#29815, "BWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.
0264.	BNL/NUREG#30602, "A PWR Training Simulator Comparison with RETRAN for a Reactor Trip from Full Power," Brookhaven National Laboratory, 1981.
0299.	NUREG/CR#1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," U.S. Nuclear Regulatory Commission, August 1980.
0300.	NUREG/CR#2353, "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics," U.S. Nuclear Regulatory Commission, (Vol. 1) January 1982, (Vol. 2) May 1982.
0416.	NUREG/CR#1908, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, September 1981.
0417.	NUREG/CR#2598, "Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, July 1982.
0418.	NUREG/CR#2534, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulated Exercises," U.S. Nuclear Regulatory Commission, November 1982.
0419.	NUREG/CR#3092, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Simulator to Field Data Calibration," U.S. Nuclear Regulatory Commission, February 1983.

0439.	Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," U.S. Nuclear Regulatory Commission, April 1981 [8105220400], (Rev. 1) April 1987 [8704300503, 8601160291], (Rev. 2) April 1996 [9604170117].
0653.	NUREG/CR-3123, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: 1982 Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, June 1983.
0968.	Memorandum for J. Roe from R. Minogue, "Nuclear Plant Analyzer (NPA) Management Plan," December 12, 1985. [9909290129]
1045.	Memorandum for V. Stello from E. Beckjord, "Resolution of TMI Action Plan Items and Human Factors Issues," May 18, 1987. [8710280270]
1077.	*Federal Register* Notice 52 FR 9453, "10 CFR Parts 50 and 55, Operators' Licenses and Conforming Amendments," March 25, 1987.
1084.	NUREG-1258, "Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55," U.S. Nuclear Regulatory Commission, December 1987.

Task I.B: Support Personnel (Rev. 4) (Rev. 1) ( )

The objectives of this task were to:

(1) Improve licensee safety performance and ability to respond to accidents by upgrading the licensee groups responsible for radiation protection and plant operation in such areas as staff size; education and experience of staff members; plant operating and emergency procedures; management awareness of, and attention to, safety matters; and numbers and types of personnel available to respond to accidents.

(2) Improve licensee safety performance by establishing a full-time, dedicated, onsite safety engineering staff and providing, along with the concurrent dissemination of information to plant personnel, an integrated program for the systematic review of operating experience.

ITEM I.B.1.1: ORGANIZATION AND MANAGEMENT LONG-TERM IMPROVEMENTS

DESCRIPTION

Historical Background

This issue[48] dealt with implementation of long-term organization and management improvements, the overall objective of which was to improve licensee safety performance and ability to respond to accidents by upgrading licensee groups responsible for radiation protection and plant operation. The areas to be upgraded included: (1) staff size; (2) education and experience of staff members; (3) plant operating and emergency procedures; (4) management awareness of, and attention to, safety matters; and (5) numbers and types of personnel available to respond to accidents. The evaluation of this issue included consideration of Item II.J.3.1.

Safety Significance

The potential for accidents resulting from some measure of human error in operating a nuclear plant may be avoidable by the resolution of this issue.

Possible Solutions

Proper management and organization will improve administration, control, prevention, and coordination both within and among all key organizational components of a plant, including those located offsite. The management involved and their staff will be better qualified and trained and the staff will be increased. Thus, the management and organization will be better-prepared for both normal operations and emergency situations. Resolution of this safety issue was assumed to encompass the following:

(1) Each utility (licensee/applicant) would be required to submit a new proposed organization and management plan for review by the NRC, including a site review. No additional management staff would be required, but the qualifications and training of the management staff and the organization effectiveness would be improved substantially at most plants.

(2) Depending on the plant, up to 14 additional personnel would be required: maintenance (~9); health physics and chemistry (~3); and training (~2). Not included were staff to man a plant-specific simulator, if required by the NRC (this was considered under Item I.A.4.1). It was anticipated that 25% of the plants would require no staff additions, 50% would require only 8 people, and 25% would require all 14 people. Thus, on the average, a plant would require 7 additional staff members.

(3) OIE staff would perform annual assessments to ensure each utility satisfactorily met NRC management and organization requirements, as identified in the initial plant review.

(4) Regulatory Guides 1.33[225] and 1.8[226] would be revised and issued, along with other appropriate regulatory guidance, to define requirements in this area.

(5) Implementation of this issue was assumed to begin in FY 1984 at all operating plants and at those plants applying for an operating license, with all plants to be covered by mid-FY 1985; this included annual followup assessments underway in FY 1985.

PRIORITY DETERMINATION

To assess this issue, SPEB/NRR consulted with PNL as well as with NRR and RES personnel working on developing the management, organization, and staffing regulatory positions. The PNL personnel had expertise in general management, utility and nuclear plant management, reactor operations, reactor operation licensing, and general reactor safety areas. The technical analysis for this issue was provided by PNL.[64]

Assumptions

The major benefit from resolution of this issue would be a reduction in human errors (operators and maintenance personnel) resulting in lower public risk. This applied to the remaining operating life of all plants (142) operating and under construction, subsequent to implementation of the solution in 1985, which was approximately 26 years.

The PNL staff estimated that the proper actions could potentially result in a 20% reduction in human errors at a nuclear plant. However, many of the plants (assumed to be 25%) were already well-managed and organized; these would see no further improvement. Another 50% would obtain only half the benefit and the remaining 25% would obtain the full benefit. An average value of 10% for reduction of human errors was anticipated for the nuclear industry at large.

Frequency Estimate

All accident sequences, except an interfacing system LOCA, would be affected. Reducing the human error rate by 10% was calculated to decrease the frequency of core-melt in Oconee-3 by 5 x 10^{-6}/RY. The frequency of core-melt in Grand Gulf-1 was assumed to be reduced by the same ratio, or 2 x 10^{-6}/RY.

Consequence Estimate

All release categories were affected and the reduction in public risk was estimated to be 13 man-rem/RY for PWRs and 15 man-rem/RY for BWRs, based on the WASH-1600[16] release estimates and assuming a typical midwest-type meteorology and an average population density of 340 people per square-mile at U.S. reactor sites. Assuming 94 PWRs and 48 BWRs with an average remaining life of 26 years after implementation of the resolution in 1985, the total public risk reduction was 50,400 man-rem.

Cost Estimate

Industry Cost: The major cost of resolving this issue was that associated with possible additional staffing required at a plant; both BWRs and PWRs would be affected equally. Specifically, industry costs associated with this issue were expected to be as follows:

(1) An average of 7 people/plant would be required for operation and maintenance.

(2) Approximately 2 man-years of effort for "intermediate case" plants would be required for preparing the initial management plan and reviewing it with the NRC. (Triple that for "worst case" plants and half that for "best case" plants). An average of 2.75 man-years/ plant was used for implementation.

(3) Approximately 1 man-month of utility effort would be required at each plant in supporting the annual NRC management assessment of the solution.

The industry costs calculated by PNL[64] were $33M for implementation and $2.27M for operation and maintenance, for a total of $35.27M.

NRC Cost: NRC costs associated with resolving this issue were expected to be as follows:

(1) Approximately 22 man-years of effort by NRR and RES to develop the long-term regulatory position on management and organization after FY-1982.

(2) Approximately 2 man-years to write, obtain, and issue comments on revised and new regulatory guides. The major development effort behind these guides was included in (1) above.

(3) Approximately 5 man-months to review the initial management and organization plan proposed for each plant. This included time for the site visit and assessment report.

(4) Approximately 0.5 man-month to perform an annual assessment of the solution at each plant.

The total NRC cost calculated[64] by PNL was approximately $30.8M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(35.27 + 30.8)M or $66.07M.

Value/Impact Assessment

Based on an estimated public risk reduction of 50,400 man-rem and a cost of $66.07M for a possible solution, the value/impact score was given by:

$$S = \frac{50{,}400\text{man} - \text{rem}}{\$66.07\text{M}}$$
$$= 763\,\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

There would be some reduction in occupational risk primarily from lowering occupational exposure due to fewer unplanned outages caused by human error. Maintenance staffs would be primarily impacted; however, both operating and maintenance staffs would benefit from avoidance of major accidents.

The potential for exposure reduction was expected to be about 10% for the 25% "worst case" plants, 5% for the 50% "intermediate case" plants, and none for the 25% "best case" plants; an average value of 5% was used. It was estimated that 300 to 500 man-rem of occupational exposure occur annually at a typical facility. Assuming 400 man-rem as a best estimate, the 5% reduction resulted in an occupational dose reduction of 20 man-rem/RY. For 142 plants with an average remaining life of approximately 26 years, the total occupational risk reduction from this source was approximately 75,000 man-rem.

The industry accident avoidance cost was estimated by PNL[64] to be $26.2M.

CONCLUSION

The potential public risk reduction was relatively large (50,400 man-rem) and the potential for occupational risk reduction was also large (75,000 man-rem), if the estimate of the reduction in human error was correct. Since most of the costs were due to additional utility staff, this value/impact could be higher if a resolution were found that did not require added staff. Therefore, based on the large potential risk reduction, this issue was given a medium priority ranking (see Appendix C).

The NRC stated its movement toward performance-based rather than prescriptive regulation in its 1986 Policy and Planning Guidance (NUREG-0885, Issue 5).[210] The NRC was approaching Management and Organization by improving its responsiveness to licensee performance, e.g., systematic assessment of licensee performance (SALP) improvement. The Nuclear Utility Management and Resources Committee (NUMARC) was established by the industry in March 1984 with an objective of reviewing management and people-related issues in nuclear operations and developing industry-wide resolutions. The Commission responded to NUMARC initiatives in a positive manner. In a letter dated January 23, 1985, to J. H. Miller, Chairman of NUMARC, Chairman Palladino supported initiatives which include improvement in management and organization and stated, in part, "we strongly encourage industry efforts to enhance the performance of utility corporate management." Based upon industry initiatives and the Commission's policy guidance, work on Items I.B.1.1(1,2,3,4) was terminated.[956]

OIE routinely developed and issued inspection procedures which addressed new or revised regulations and requirements. Thus, Item I.B.1.1(5) was determined to be resolved.[441]

In July 1984, OIE noted[652] that the proposed revision to Regulatory Guide 1.33[225] was being addressed under Issue 75. This effort negated the need to pursue a separate resolution to those parts of Items I.B.1.1(6,7) that required a revision to Regulatory Guide 1.33.[225] The parts of Items I.B.1.1(6,7) that require a revision to Regulatory Guide 1.8[226] were addressed under Item I.A.2.6(1).[956]

ITEM I.B.1.1(1): PREPARE DRAFT CRITERIA

This item was evaluated in Item I.B.1.1 above and was determined to be RESOLVED[956] with no new requirements.

ITEM I.B.1.1(2): PREPARE COMMISSION PAPER

This item was evaluated in Item I.B.1.1 above and was determined to be RESOLVED[956] with no new requirements.

ITEM I.B.1.1(3): ISSUE REQUIREMENTS FOR THE UPGRADING OF MANAGEMENT AND TECHNICAL RESOURCES

This item was evaluated in Item I.B.1.1 above and was determined to be RESOLVED[956] with no new requirements.

ITEM I.B.1.1(4): REVIEW RESPONSES TO DETERMINE ACCEPTABILITY

This item was evaluated in Item I.B.1.1 above and was determined to be RESOLVED[956] with no new requirements.

ITEM I.B.1.1(5): REVIEW IMPLEMENTATION OF THE UPGRADING ACTIVITIES

This item was evaluated in Item I.B.1.1 above and was determined to be RESOLVED[441,956] with no new requirements.

ITEM I.B.1.1(6): PREPARE REVISIONS TO REGULATORY GUIDES 1.33 AND 1.8

This item was evaluated in Item I.B.1.1 above. The revision of Regulatory Guide 1.33[225] was covered in Issue 75[652,956] and the revision of Regulatory Guide 1.8[226] was covered in Issue I.A.2.6(1).

ITEM I.B.1.1(7): ISSUE REGULATORY GUIDES 1.33 AND 1.8

This item was evaluated in Item I.B.1.1 above. The revision of Regulatory Guide 1.33[225] was covered in Issue 75[652,956] and the revision of Regulatory Guide 1.8[226] was covered in Issue I.A.2.6(1).

ITEM I.B.1.2: EVALUATION OF ORGANIZATION AND MANAGEMENT IMPROVEMENTS OF NEAR-TERM OPERATING LICENSE APPLICANTS

DESCRIPTION

This NUREG-0660[48] item required the staff to evaluate organization and management capabilities of NTOL applicants before license issuance. NRR was to provide draft criteria and OIE was to manage an inter-office review team. The findings of the team was to be factored into the SER for each NTOL facility.

CONCLUSION

Between January and July 1980, 6 NTOLs (Sequoyah, North Anna-2, Salem-2, Diablo Canyon, McGuire, and Farley-2) were evaluated; Zion, Indian Point, and TMI-1 were also evaluated later. As part of its overall review responsibility, NRR was to manage similar reviews for other NTOL applicants.

ITEM I.B.1.2(1): PREPARE DRAFT CRITERIA

This item was evaluated in Item I.B.1.2 above and was determined to be RESOLVED[235] with no new requirements.

ITEM I.B.1.2(2): REVIEW NEAR-TERM OPERATING LICENSE FACILITIES

This item was evaluated in Item I.B.1.2 above and was determined to be RESOLVED with no new requirements.

ITEM I.B.1.2(3): INCLUDE FINDINGS IN THE SER FOR EACH NEAR-TERM OPERATING LICENSE FACILITY

This item was evaluated in Item I.B.1.2 above and was determined to be RESOLVED with no new requirements.

ITEM I.B.1.3: LOSS OF SAFETY FUNCTION

DESCRIPTION

This TMI Action Plan[48] item addressed regulatory action at an operating nuclear power plant in the event of human error leading to complete loss of a safety function required by the plant's TS. The following three options specified in the TMI Action Plan[48] were considered:

(1) Require licensees to immediately place the plant in the safest shutdown cooling condition following a total loss of a safety function due to personel error, if a total loss of a safety function had occurred within the previous year or two. Resumption of operation would require NRC approval based on a review of the licensee's program for corrective action.

(2) Use existing enforcement options (citations, fines, shutdowns).

(3) Use approaches such as a point system, licensee probations, and (in the extreme) license revocations.

Loss of a required safety function can lead to an increase in the probability that an event with an accident-initiating potential, should it occur, would lead to an actual major accident. This probability increase could be more or less substantial, depending on the specific function lost. The safety concern is heightened when the loss of safety function is caused by human error and this occurs more than once in a year or two. Such repeated personnel failures can bring into question whether the reliability of safety-related personnel actions at the plant involved are generally up to the standards expected and assumed in safety evaluations. This item was related to improving the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Option 2 was selected as the best option that would provide the latitude needed by NRC for determination whether a particular event falls under the definition of a "loss of safety function," the role of human error in causing the event, the acuteness of the risk, the urgency and nature of appropriate remedial action, conditions for resumption of operation, and such considerations as the public health-and-safety need for power at the time (see References *234*, *265*, *266*, *267*, *287*). With the selection of Option 2, Item I.B.1.3 was terminated, having become part of the Enforcement Policy issue (Item IV.A.2) which was completed.[288] Thus, this Licensing Issue was resolved.

ITEM I.B.1.3(1): REQUIRE LICENSEES TO PLACE PLANT IN SAFEST SHUTDOWN COOLING FOLLOWING A LOSS OF SAFETY FUNCTION DUE TO PERSONNEL ERROR

This Licensing Issue was evaluated in Item I.B.1.3 above and was determined to be resolved.

ITEM I.B.1.3(2): USE EXISTING ENFORCEMENT OPTIONS TO ACCOMPLISH SAFEST SHUTDOWN COOLING

This Licensing Issue evaluated in Item I.B.1.3 above and was determined to be resolved.

ITEM I.B.1.3(3): USE NON-FISCAL APPROACHES TO ACCOMPLISH SAFEST SHUTDOWN COOLING

This Licensing Issue was evaluated in Item I.B.1.3 above and was determined to be resolved.

REFERENCES

0016.	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.
0048.	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.
0064.	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.
0210.	NUREG#0885, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission, (Issue 1) January 1982, (Issue 2) January 1983, (Issue 3) January 1984, (Issue 4) February 1985, (Issue 5) February 1986, (Issue 6) September 1987.

0225.	Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, November 1972, (Rev. 1) February 1977, (Rev. 2) February 1978. [7907100144]
0226.	Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1971, (Rev. 1) September 1975 [8801130111], (Rev. 1#R) May 1977 [7907100073], (Rev. 2) April 1987. [8907180147]
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0288.	NEDO#10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," General Electric Company, October 1977, (Rev. 1) May 1980.
0441.	Memorandum for H. Denton from R. DeYoung, "Commission Paper on the Prioritization of Generic Safety Issues," April 20, 1983. [9705190224]
0652.	Memorandum for W. Dircks from R. DeYoung, "Elimination of Duplicative Tracking Requirements for Revision of Regulatory Guide 1.33," July 26, 1984. [9705190264]
0956.	Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987. [8701140115]

Task I.B.2: Inspection of Operating Reactors (Rev. 1) ( )

The objective of this task was to improve the safety of operations at nuclear power plants by increasing the effectiveness of the NRC inspection program as follows: (1) revise the existing inspection program; (2) implement the resident inspection program; and (3) systematically assess licensee performance so that the NRC could reapportion its inspection resources according to need.

ITEM I.B.2.1: REVISE OIE INSPECTION PROGRAMDESCRIPTION

This TMI Action Plan[48] item had the objective of enhancing the safety effectiveness of the operating reactor inspection program. It involved revision of the inspection program to provide more direct observation and independent verification of licensee activities and reduction of inspection documentation. Inspection program revisions focused on the following:

A. Inspections that included, on a sampling basis, such activities as:

(1) Verifying the adequacy of management and procedural controls and staff discipline for the conduct of day-to-day operational surveillance activities.

(2) Independently verifying that systems required to be operable were properly aligned.

(3) Following up on completed maintenance work orders to ensure proper testing and return to service.

(4) Observing surveillance tests to determine whether test instruments were properly calibrated and that approved procedures were followed, including taking equipment out of service during a test and returning it to service after the test.

(5) Verifying that each licensee was complying with the TS and operating parameters by daily control room observations.

(6) Observing routine maintenance to detect such things as the wrong lubricant, improper tightening of valve packing, substitution of unqualified parts, and lack of care in the protection of open systems.

(7) Inspecting the terminal boards, panels, and instrument racks for unauthorized jumpers and bypasses and checking locations against records to ascertain whether jumpers were removed as stated in the records.

B. Reactive efforts in response to operating events, allegations, or followup to previous findings.

C. Periodic Performance Appraisal Team inspections to supplement the resident inspector by an in-depth inspection of the overall plant operation.

D. Intensified inspection program at startup testing.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Procedures were issued and implemented to accomplish more direct observation and independent verification, emphasize reactor inspection efforts, and intensify the inspection program at startup testing. Periodic performance appraisal team inspections were functioning and documentation was streamlined.[239,247] (For specific procedures, see Chapters 2513, 2514, and 2515 and Part 9700 of the OIE Manual.) All required action on Item I.B.2.1 was completed[235,379,406] and this Licensing Issue was resolved with changes in the NRC procedures that addressed the operating reactor inspection program.

ITEM I.B.2.1(1): VERIFY THE ADEQUACY OF MANAGEMENT AND PROCEDURAL CONTROLS AND STAFF DISCIPLINE

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(2): VERIFY THAT SYSTEMS REQUIRED TO BE OPERABLE ARE PROPERLY ALIGNED

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(3): FOLLOW-UP ON COMPLETED MAINTENANCE WORK ORDERS TO ASSURE PROPER TESTING AND RETURN TO SERVICE

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(4): OBSERVE SURVEILLANCE TESTS TO DETERMINE WHETHER TEST INSTRUMENTS ARE PROPERLY CALIBRATED

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(5): VERIFY THAT LICENSEES ARE COMPLYING WITH TECHNICAL SPECIFICATIONS

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(6): OBSERVE ROUTINE MAINTENANCE

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.1(7): INSPECT TERMINAL BOARDS, PANELS, AND INSTRUMENT RACKS FOR UNAUTHORIZED JUMPERS AND BYPASSES

This Licensing Issue was evaluated in Item I.B.2.1 above and was determined to be resolved.

ITEM I.B.2.2: RESIDENT INSPECTOR AT OPERATING REACTORS

DESCRIPTION

This TMI Action Plan[48] item addressed implementation of the approved resident inspector program at operating reactors, as part of efforts to improve safety of operations at nuclear power plants by increasing the effectiveness of the NRC inspection program. Actions included recruiting, training, and assigning resident inspectors to provide a minimum of one resident inspector at each site. Additional resident inspectors were assigned to multi-unit sites. Resident inspectors were also assigned to construction sites with units that were 15% or more complete. This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

By June 1980, senior resident inspectors had been placed at all plants awaiting near-term operating licenses to ensure their presence by the time of fuel loading. By December 1980, manning reached 94% of the target level, approximately the highest level that could be expected to be maintained because of vacancies created by losses and the time lag in filling vacancies.[239] All required action was completed (see References *235*, *239*, *333*, *334*, *379*, and *406*) and this Licensing Issue was resolved[235] with changes to the NRC resident inspector program.

ITEM I.B.2.3: REGIONAL EVALUATIONS

DESCRIPTION

The TMI Action Plan[48] described this part of the program to enhance the safety effectiveness of inspections at operating reactors as follows:

The NRC will establish boards in each region to annually evaluate each licensee's performance. The Licensing Project Manager will participate on the board for the facilities he manages. The board will review the enforcement actions, licensee event reports, technical and management performance, significant personnel and organizational changes, licensee safety attitude, and observations by inspection supervisors and inspectors from all cognizant regional disciplines. The results of this evaluation will be documented and used to determine the adequacy of current enforcement sanctions and to redirect, as appropriate, the inspection effort and program plans. In addition, the evaluation will be used to provide a major input into the formal NRC review board discussed in Item I.B.2.4, Overview of Licensee Performance. Meetings with licensee management will be held to discuss board findings as appropriate.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Procedures incorporating regional evaluation of licensee performance were issued as part of the NRC Manual Chapter 0516 and all required action was completed.[235,239,379,406] Thus, this Licensing Issue was resolved with changes in NRC procedures that addressed regional evaluation of licensee performance.

ITEM I.B.2.4: OVERVIEW OF LICENSEE PERFORMANCE

DESCRIPTION

This item was part of the TMI Action Plan[48] program to improve the safety of operations at nuclear power plants by increasing the effectiveness of the NRC inspection program. The TMI Action Plan described this item as follows:

A formal NRC review group (composed of senior NRC personnel from OIE, NRR, NMSS, SD [now RES], as required) will be appointed to provide an overview function of the regional appraisals of the licensees' performance, to determine safety adequacy, and to assess corrective actions planned by regional offices. Based on the findings, the review group will be specifically charged to recommend major enforcement sanctions or license modifications to appropriate office directors. This review group, in addition to receiving inputs from regional evaluations [Item I.B.2.3], will receive inputs from NRR project managers, from NRR technical support program personnel, and from other NRC offices as appropriate. The findings from the board will be made public.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

The formal NRC review group was formed and performed an overview function of the initial set of regional appraisals of licensee performance.[298,336] Based on the results of the first set of appraisals, the Commission determined that future assessments should be made at a regional level, that involvement of the other NRC offices should continue as part of the assessment, and that headquarters activity should be redirected to evaluating policy, criteria, and methodology for these assessments.[364] Procedures for implementing the Commission's request were subsequently issued as part of NRC Manual Chapter 0516. All required action was completed[235,298,379,406] and this Licensing Issue was resolved with changes in NRC procedures to address the overview of licensee performance.

REFERENCES

0048.	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.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0239.	Memorandum for W. Dircks from V. Stello, "TMI Action Plan—Status Report," December 19, 1980. [8205260193]
0247.	NUREG/CR#5669, "Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators," U.S. Nuclear Regulatory Commission, July 1991.
0298.	Memorandum for W. Dircks from V. Stello, "TMI Action Plan—Status Report," April 17, 1981. [8205260194]
0336.	NUREG#0834, "NRC Licensee Assessments," U.S. Nuclear Regulatory Commission, August 1981.

0364.	Memorandum for W. Dircks from S. Chilk, "Systematic Assessment of Licensee Performance," October 20, 1981. [8210080207]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0406.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Status Report," March 4, 1982. [8204290601]

Task I.C: Operating Procedures (Rev. 4) ( )

The objective of this task was to improve the quality of procedures to provide greater assurance that operator and staff actions were technically correct, explicit, and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affected plant operation, administration, maintenance, testing, and surveillance were to be included.

ITEM I.C.1: SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURES REVISION

The four parts of this item were evaluated separately below.

ITEM I.C.1(1): SMALL-BREAK LOCAs

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.1(2): INADEQUATE CORE COOLING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-04 was established by DL/NRR for implementation purposes.

ITEM I.C.1(3): TRANSIENTS AND ACCIDENTS

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-05 was established by DL/NRR for implementation purposes.

ITEM I.C.1(4): CONFIRMATORY ANALYSES OF SELECTED TRANSIENTS

DESCRIPTION

Background

This NUREG-0660[48] item required confirmatory analyses of selected transients by NRR to provide the basis for comparisons with analytical methods that were being used by the reactor vendors. These comparisons were to ensure the adequacy of the analytical methods being used to generate emergency procedures. At the time this issue was initially evaluated, NRC had performed a limited number of confirmatory transient analyses and the remainder was being defined.

Safety Significance

The safety significance was the reduction in operator errors and upgrading of operating systems through confirmatory analyses of selected transients by NRC. These confirmatory analyses were expected to provide greater assurance that operator and staff actions were technically correct.

Possible Solution

Confirmatory analyses, using the best available computer codes, provided the basis for comparisons with the analytical methods that were being used by the reactor vendors. These comparisons, together with comparisons with other data, constituted the short-term verification effort to ensure the adequacy of the analytical methods being used to generate emergency procedures.

PRIORITY DETERMINATION

Frequency Estimate

To evaluate this issue, PNL assumed[64] improvements in two areas: the reduction in human error rate for operators, estimated to be 7%, and other operation improvements (set points for control systems, maintenance, hardware upgrade, etc.) estimated to be 4.5%. The total improvement percentages were applied to the base case frequencies and affected release categories for both PWRs and BWRs. The dominant accident sequences and base case frequencies for Oconee (B&W) were used for PWRs; for BWRs, Grand Gulf 1 was used as the model.

For PWRs, the base case core-melt frequency was determined to be 8.2 x 10^{-5}/RY. Considering the above improvements, the adjusted case core-melt frequency was determined to be 7.3 x 10^{-5}/RY with a resultant reduction in core-melt frequency of 9 x 10^{-6}/RY. For BWRs, the base case and adjusted case core-melt frequencies were determined to be 3.7×10^{-5}/RY and 3.3×10^{-5}/RY, respectively, with a resultant reduction in core-melt frequency of 4×10^{-6}/RY.

Consequence Estimate

Because of the multifactor influence of the estimated improvements, all seven of the PWR release categories and all four of the BWR release categories were assumed to be affected. The potential public risk reduction for PWRs was calculated to be 6.5×10^{4} man-rem, assuming 95 plants with an average remaining life of 28.5 years. The potential public risk reduction for BWRs was calculated to be 4×10^{4} man-rem, assuming 49 plants with an average remaining life of 27 years. In all cases, a population density of 340 persons per square-mile and typical meteorology were assumed. The total reduction in public risk, based on the above results, was about 1.05×10^{5} man-rem.

Cost Estimate

Industry Cost: The industry cost was estimated to be $61M based on the following assumptions: (1) a rate of $1,900/man-week; (2) 30 man-weeks to implement the resolution; (3) seven man-weeks/RY for operation and maintenance; and (4) 144 plants with an average remaining life of 28 years.

NRC Cost: The NRC cost, including implementation and reviews, was estimated to be $2.8M.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be approximately $64M.

Value/Impact Assessment

Based on an estimated public risk reduction of 105,000 man-rem and a cost of $64M for a possible solution, the value/impact score was given by:

$$S = \frac{105{,}000\text{man} - \text{rem}}{\$64\text{M}}$$

$$= 1{,}650\text{man} - \text{rem}/\$\text{M}$$

Other Considerations

Other factors considered were the accident avoidance costs and the potential occupational risk reductions. The accident avoidance cost was the product of the reduction in the probability of core-melt and industry cost factors, assuming cleanup, repair, refurbishment, and replacement power cost over a 10-year period.

The total accident avoidance cost for all 95 PWRs and 49 BWRs, which included existing operating plants and those plants expected to commence operation, was estimated to be approximately $49M. Therefore, the net industry cost for this issue, when reduced by the accident avoidance cost, would be approximately $12M.

The occupational dose incurred from accident recovery was estimated at 20,000 man-rem.[64] The total occupational dose reduction due to accident avoidance, considering all PWRs and BWRs, was 600 man-rem. Assuming a 5% reduction in annual operational doses due to imposed operating guidelines and upgraded control systems, the best estimate annual operational dose reduction would be 20 man-rem/RY. For all plants and all remaining plant life, the potential occupational dose reduction was 81,000 man-rem. These estimates indicated that the potential reduction in occupation doses during normal operation was significant and supported a high priority ranking for the issue.

CONCLUSION

Based on the value/impact score and the potential reduction in core-melt frequency, the issue would have been given a medium priority ranking. However, because of the potential public risk reduction of 105,000 man-rem, the issue was given a high priority ranking (see Appendix C). All required work was completed[382,383] and the issue was RESOLVED with no new requirements.

ITEM I.C.2: SHIFT AND RELIEF TURNOVER PROCEDURES

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.3: SHIFT SUPERVISOR RESPONSIBILITIES

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.4: CONTROL ROOM ACCESS

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.5: PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-06 was established by DL/NRR for implementation purposes.

ITEM I.C.6: PROCEDURES FOR VERIFICATION OF CORRECT PERFORMANCE OF OPERATING ACTIVITIES

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-07 was established by DL/NRR for implementation purposes.

ITEM I.C.7: NSSS VENDOR REVIEW OF PROCEDURES

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.8: PILOT-MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.C.9: LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

DESCRIPTION

Historical Background

The NRC effort on this TMI Action item[48] (to be led by NRR with involvement by OIE, SD, and RES) was to develop a long-term program plan for the upgrading of plant procedures. This plan would incorporate and expand on existing efforts associated with the development, review, and monitoring of procedures. Consideration of studies to ensure clear procedures were called for with particular emphasis on diagnostic aids for off-normal conditions. The interrelationships of administrative, operating, maintenance, test, and surveillance procedures were to be considered. The topics of emergency procedures, reliability analysis, human factors engineering, crisis management, and operator training were also to be addressed.

The part of Item I.C.9 that addressed emergency operating procedures (EOP) was implemented in accordance with Item I.C.1 of NUREG-0737.[98] SECY-82-111[151] requested Commission approval of a set of basic requirements for emergency response capability and approval for the staff to work with licensees to develop plant-specific implementation schedules. A significant amount of work on EOPs had been completed and all four NSSS vendors had submitted technical guidelines based on re-analysis of accidents and transients; these were in the final stages of review. In the area of human factors, a survey of existing practices, research on EOPs, and pilot monitoring of some NTOL plants had been completed and criteria for development of EOPs were published for public comment in NUREG-0799.[191] NUREG-0899[192] was published in final form in September 1982 and incorporated resolution of comments received on NUREG-0799.[191] The recommended requirements for EOPs,[151] which included some of these completed or nearly-completed tasks, had been conditionally approved.[190]

The part of Item 1.C.9[48] that pertained to other long-term procedures (which were not addressed in NUREG 0737[98]) required further staff effort. The priority ranking of this issue was based on this remaining staff effort.

Safety Significance

Resolution of this issue was expected to have a significant impact on plant procedures. The changes in procedures, in turn, were expected to improve the safety-related performance of all plant operations staff. This would apply to both routine and abnormal operating conditions.

Possible Solution

At the time this issue was initially evaluated, staff actions under Item I.C.9[48] which pertained to normal and abnormal operating procedures, maintenance, test, surveillance, and other safety-related procedures were ongoing and scheduled in three phases:

(1) Survey ongoing studies, existing procedures, and practices of related industries; assess problems; and prioritize solutions (FY 1982-1983).

(2) Prepare guidance (NUREGs, Regulatory Guides) for industry use (FY 1983-1984).

(3) Issue requirements, prepare inspection guidance, review or audit as necessary (FY 1985-1986).

PRIORITY DETERMINATION

Frequency Estimate

To estimate the change in core-melt frequency for this issue, PNL[64] assumed a human error rate reduction of 30% for operations staff. PNL also assumed that the dominant accident sequences for the Oconee-3 (B&W) plant were representative of all PWRs and that the fractional risk and core-melt frequency reductions were applicable to the representative BWR (Grand Gulf-1).

For PWRs, the base case core-melt frequency was determined to be 7.8 x 10^{-5}/RY. The adjusted case core-melt frequency, considering the above improvement, was determined to be 5.6 x 10^{-5} /RY. The result was a reduction in core-melt frequency of 2.2 x 10^{-5}/RY for PWRs. For BWRs, the base case core-melt frequency was determined to be 3.5 x 10^{-5}/RY and the reduction in core-melt frequency was 9.9 x 10^{-6}/RY.

Consequence Estimate

All seven of the PWR release categories and all four of the BWR release categories were affected by the improvement. The potential public risk reduction for PWRs was calculated to be 53 man-rem/RY, assuming WASH-1400[16] release categories, a population density of 340 persons per square-mile, and typical midwest meteorology. The public risk reduction for BWRs was calculated to be 64 man-rem/RY. Therefore, the total public risk reduction for all plants (90 PWRs and 44 PWRs) was 2.1 x 10^5 man-rem, assuming an average remaining life of 28 years.

Cost Estimate

Industry Cost: The industry cost was estimated to be $447M and included $67M to implement and upgrade and $380M for operation and maintenance.

NRC Cost: The NRC cost including implementation and reviews was estimated at $9M.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be approximately $(447 + 9)M or $456M.

Value/Impact Assessment

Based on an estimated public risk reduction of 210,000 man-rem and a cost of $456M for a possible solution, the value/impact score was given by:

$$S = \frac{210{,}000\text{man} - \text{rem}}{\$456\text{M}}$$

$$= 461\,\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

In the analysis of this issue, PNL[64] assumed a uniform 30% improvement in human error, including maintenance, through the dominant accident sequences. The 30% improvement was expected to overestimate reductions in maintenance outages. It was assumed that no significant reductions in maintenance outages would reduce the potential risk reduction calculated by PNL approximately 10%. These improvements transcended normal, abnormal, and emergency procedures during the event sequences as described in NUREG-0660,[48] Item I.C.9. However, the EOP concerns originally included in Item I.C.9 were separately addressed in NUREG-0737.[98]

It was believed that the results of the dominant accident sequences would be strongly influenced by the EOPs. This situation was expected to result in little or no change to the above value/impact score of 461 man-rem/$M since the smaller risk reduction that could be attributed to this issue, after the EOP effect was removed, was balanced by a lower implementation cost to complete the remaining part of the issue. The beneficial reduction in core-melt frequency and public risk calculated by PNL[64] was significantly less when dominant effects of the improvements in the EOPs were removed from the issue. Assuming that improved EOPs would contribute approximately 75% toward reducing the core-melt frequency and public risk, the benefit (risk reduction) attributed to improvements and upgrading of the other procedures was 25% of the total benefits previously calculated. This resulted in a total public risk reduction of $(0.9)(0.25)(2.1 \times 10^5)$ man-rem or 47,000 man-rem. These reductions were attributable to that part of Item I.C.9 not addressed in Item I.C.1 of NUREG-0737.[98]

CONCLUSION

The part of this issue that was clarified in Supplement 1 to NUREG-0737 (Generic Letter No. 82-33)[376] was resolved[805] with the publication of SRP[11] Section 13.5.2, Rev. 1, and Section 13.5.2, Appendix A, Rev. 0. With the exclusion of the EOPs (which were issued as requirements in NUREG-0737[98]), this issue was given a medium priority ranking (see Appendix C) and RESOLVED with no additional requirements.[955]

REFERENCES

0011.	*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.
0016.	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.
0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0151.	SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]
0190.	Memorandum for W. Dircks from S. Chilk, "Staff Requirements—Affirmative Session, 11:50 a.m., Friday July 16, 1982," July 20, 1982. [8208040248, 8209010068]
0191.	NUREG#0799, "Draft Criteria for Preparation of Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, June 1981.
0192.	NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures—Resolution of Comments on NUREG-0799," U.S. Nuclear Regulatory Commission, September 3, 1982.
0376.	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*]
0382.	Memorandum for W. Minners from R. Mattson, "Schedules for Resolving and Completing Generic Issues," January 21, 1983. [8301260532]
0383.	Memorandum for W. Dircks from R. Mattson, "Closeout of TMI Action Plan I.C.1(4), Confirmatory Analyses of Selected Transients," November 12, 1982. [8212080586]
0805.	Memorandum for T. Combs from H. Denton, "Revised SRP Section 13.5.2 and Appendix A to SRP Section 13.5.2 of *NUREG-0800*," July 17, 1985. [8508050283]
0955.	Memorandum for W. Dircks from H. Denton, "Close Out of Completed TMI Action Plan Item I.C.9, 'Long-Term Program Plan for Upgrading of Procedures,'" June 7, 1985. [8506200155]

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,[98] 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,[98] requirements were issued, and MPA F-09 was established by DL/NRR for implementation purposes. Generic Letter No. 82-33[376] transmitted Supplement 1 to NUREG-0737[98] 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,[98] 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.[98] Generic Letter 89-06[1205] (enclosing NUREG-1342[1206]) was issued to inform licensees of the staff's findings to aid in implementing SPDS requirements. NUREG-1342[1206] describes: (1) methods used by some 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-1342[1206] did not constitute new requirements; Supplement 1 to NUREG-0737[98] contains NRC's requirements for SPDS.

ITEM I.D.3: SAFETY SYSTEM STATUS MONITORING

DESCRIPTION

Historical Background

This TMI Action Plan item[48] recommended that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47[150] 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 1.47[150] (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.[150]

In an analysis of this issue performed by PNL,[64] 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.

Frequency Estimate

The reduction in core-melt frequency (F) for Oconee-3 was calculated to be 8.7×10^{-7}/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,[64] the frequency reduction (F) for Grand Gulf-1 was 3.9×10^{-7}/RY.

Consequence Estimate

Assuming WASH-1400[16] 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)M or $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:

$$S = \frac{1.2 \times 10^4 \text{ man-rem}}{\$48.6\text{M}}$$

$$= 240 \text{ man-rem/\$M}$$

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-82-111[151] and a letter[376] 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.[152,153] 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.[1536] In an RES evaluation,[1564] 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 Plan[48] 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 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.[64] 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 x 10^{-5} /RY and 6.1 x 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 x 10^{-5}/RY for PWRs and 5.95 x 10^{-6}/RY for BWRs. Thus, the reduction in core-melt frequency (due to issue resolution) was calculated to be 9 x 10^{-7}/RY for PWRs and 1.8 x 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.

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:

$$S = \frac{840 \text{ man} - \text{rem}}{\$1.39\text{M}}$$

$$= 600 \text{ man} - \text{rem} / \$\text{M}$$

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-0700[474] 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-0700[474] 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 SRP[11] Section 18.1 which addresses control room design and references NUREG-0700[474] 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-0700[474] 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.[1101] In NUREG/CR-5382,[1563] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

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

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

DESCRIPTION

This TMI Action Plan[48] 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.[244] Based on this report, RES issued RIL-124[245] which provided a recommendation for further action. Thus, this item was RESOLVED with no new requirements. In NUREG/CR-5382,[1563] 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 Plan[48] 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-1440[241] and BWR instrumentation requirements were analyzed in NUREG/CR-2100.[242] ESF Status Monitoring requirements were also studied in NUREG/CR-2278.[243] RIL No. 98[246] was issued in August 1980 and transmitted "the results of completed research describing an improved method for analyzing accident sequences." Revision 2 to Regulatory Guide 1.97[55] 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.[151,376] This item was RESOLVED and new requirements were established.

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

DESCRIPTION

This TMI Action Plan[48] 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.[1537] Thus, the issue was RESOLVED with no new requirements.[1538] 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 Plan[48] 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,[98] mandated 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,[1563] 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 Plan[48] 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.[64] 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.[307] 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:

PWR: (4.63 transients/RY)(0.80)(0.25)(0.75 day/shutdown) = 0.69 day/RY

BWR: (5.20 transients/RY)(0.80)(0.25)(0.75 day/shutdown) = 0.78 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 (T_2 and T_3 for PWRs and T_{23} 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 x 10^{-6} event/RY for PWRs and 2.6 x 10^{-6} 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 man-weeks/ RY)($2,270/man-week)(134 plants)(30 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 man-weeks/backfit plant)($2,270/man-week)(71 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)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:

$$S = \frac{53{,}000\,\text{man}-\text{rem}}{\$362.65\text{M}}$$

$$= 150\,\text{man}-\text{rem}/\$\text{M}$$

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/day)[(0.69 day/RY)(90 plants)(30 years) +

(0.78 day/RY)(44 plants)(30 years)] = $870M

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.[1099] 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.[1100]

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.[347] Thus, this Licensing Issue was resolved.[271]

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 report[306] 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.

REFERENCES

0011.	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.
0016.	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.
0048.	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.
0055.	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]
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0150.	Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]
0151.	SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]
0152.	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.
0153.	NUREG/CR-3621, "Safety System Status Monitoring," U.S. Nuclear Regulatory Commission, March 1984.
0241.	NUREG/CR#1440, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1980.
0242.	NUREG/CR#2100, "Boiling Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1981.
0243.	NUREG/CR#2278, "Light Water Reactor Engineered Safety Features Status Monitoring," U.S. Nuclear Regulatory Commission, October 1981.
0244.	NUREG/CR#2147, "Nuclear Control Room Annunciators," U.S. Nuclear Regulatory Commission, October 1981.
0245.	Memorandum for H. Denton et al. from R. Minogue, "Research Information Letter #RIL-124, `Control Room Alarms and Annunciators,'" October 20, 1981. [8111130045]
0246.	RIL#98, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, August 18, 1980. [8104230867]
0271.	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]
0306.	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.
0307.	EPRI NP#2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.

0347.	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.
0376.	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*]
0474.	*NUREG-0700*, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.
1099.	Memorandum for B. Morris from B. Sheron, "Updated GIMCS for GI I.D.5(5)," February 2, 1988. [9704150145]
1100.	Memorandum for V. Stello from E. Beckjord, "Redesignation of Generic Issue I.D.5(5), `Disturbance Analysis Systems,'" February 22, 1988. [8809190312]
1101.	Memorandum for V. Stello from E. Beckjord, "Closure of Generic Issue I.D.4 `Control Room Design Standard,'" March 28, 1988. [9704160014]
1205.	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*]
1206.	NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," U.S. Nuclear Regulatory Commission, April 1989.
1536.	Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-I.D.3, 'Safety System Status Monitoring,'" August 20, 1993. [9502070295]
1537.	Memorandum for T. Murley from E. Beckjord, "Research Information Letter Number 171,'Continuous On-Line Reactor Surveillance System,'" May 4, 1993. [9305100271]
1538.	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]
1563.	NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.
1564.	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]

Task I.E: Analysis and Dissemination of Operating Experience (Rev. 3) ( )

The objective of this task was to establish an integrated program which involved participation by the licensees, vendors, NSAC, INPO, and the NRC, and which included foreign operations experience for the systematic collection, review, analysis, and feedback of operating experience to NRC licensing, inspection, standards, and research activities, and to licensees for all NRC-licensed activities. Appropriate corrective action was expected to be taken in response to feedback.

ITEM I.E.1: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATADESCRIPTION

The purpose of this TMI Action Plan[48] item was to establish an NRC office which would be responsible for: (1) analysis and evaluation of operational data associated with all NRC-licensed activities; and (2) development of specific recommendations for action by other NRC offices.

Systematic evaluation of operating data can identify potential significant safety problems or their precursors. Dissemination to NRC and industry of evaluation results which identify such problems, along with recommendations for their resolutions, can avoid occurrence of these problems at other plants of similar design. This item was initiated to improve the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

The Commission approved the establishment of AEOD in July 1979 and an interim office was established in October 1979. As of June 1984, AEOD was staffed and functioning in accordance with the purpose and scope described in Chapter 0143 of the NRC Manual. Thus, this Licensing Issue was resolved.

ITEM I.E.2: PROGRAM OFFICE OPERATIONAL DATA EVALUATION

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to ensure that each NRC office conducted operational safety analyses. These analyses were to be coordinated and the results distributed as part of the integrated program on operating experience assessments. The work of each office was expected to complement the operational data evaluation activities conducted by AEOD under Item I.E.1.

Systematic evaluation of operational data can identify potential significant safety problems or their precursors. Dissemination to NRC offices and industry of such evaluations, along with their resolution, can avoid occurrence of these problems at other facilities of similar design that conduct similar operations. This item was initiated to improve the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

As of June 1984, each of the following NRC Offices had established responsibility and procedures for evaluating operational data:

(1) In OIE, the Events Analysis Branch had the lead responsibility for this activity using input from such sources as LERs, Preliminary Notices, 10 CFR 21 Reports, 10 CFR 50.55(e) Construction Deficiency Reports, and 10 CFR 50.72 Reports to the NRC Operations Center. Evaluations which identified potential significant safety problems were disseminated by means such as IE Notices or IE Bulletins.

(2) In NRR, the Operating Reactors Assessment Branch (ORAB) conferred daily with OIE on operational occurrences and made preliminary assessments of their safety significance as part of their functional responsibility described in the NRC Manual. Those occurrences which were considered to have potential safety significance were identified at weekly NRR management briefings on operational events conducted by ORAB with OIE participation. If deemed necessary, further evaluation was assigned to the appropriate NRR Division.

(3) In RES, the Reactor Risk Branch had lead responsibility for evaluating operational events and was responsible for issuance of periodic reports on Precursors to Potential Severe Core Damage Accidents,[76] which were derived from a systematic review of LERs. These reports provided operational experience data and were available for use in Event Tree Analyses and PRAs conducted by RES.

(4) NMSS had issued a procedure[305] for achieving a more formal review and evaluation of inspection and operational data and event reports to identify and correct generic problems. The procedure included criteria for identifying operational events that warranted detailed review and evaluation. Evaluation reports that identified safety significant operational events were distributed within NRC.

Based on the actions taken by the cognizant Offices, this Licensing Issue was resolved.

ITEM I.E.3: OPERATIONAL SAFETY DATA ANALYSIS

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to conduct special operational safety data analyses to determine equipment failure rates and to develop error data analysis for nuclear plant operations. As of June 1984, the Reactor Risk Branch of RES was performing studies to: (1) determine equipment failure rates using LERs; (2) develop and use common-cause/common-mode analysis of LERs; (3) analyze data from the NPRDS to distinguish order-of-magnitude differences in component failure rates between such factors as plants, sizes, service environment, status at time of failure, and manufacturer; (4) identify potential reliability problems evident in the LER data; and (5) identify potential accident precursors.

The information obtained from this item was used to: (1) provide more reliable equipment failure rate data, including common-cause/common-mode failure statistics to support PRAs of nuclear power plants (see Items II.C.1, "Interim Reliability Evaluation Program," and II.C.2, "Continuation of IREP"); and (2) to identify potentially serious equipment reliability problems evident from LER data and provide feedback to equipment maintenance/surveillance programs to reduce equipment failure rates (see Item II.C.4, "Reliability Engineering"). This item was initiated to improve the NRC capability to assess safety and, therefore, was considered a Licensing Issue.

CONCLUSION

At the time of this evaluation, implementation of the program had resulted in the following:

(1) Publication of data summaries of LERs on pumps, control rods and drive mechanisms, diesel generators, valves, primary containment penetrations, and instrumentation and control components (See References *344*, *345*, *346*, *348*, *349*, *350*, and *351*).

(2) Equipment unavailability data from nuclear plant log books obtained as part of the In-Plant Reliability Data System (IPRDS).[353,354]

(3) Publication of reports on common-cause/common-mode failures (See References *355*, *356*, *357*, *358*, *359*, *360*, and 361).

(4) Preparation of a computer program (FRANTIC) for use in upgrading equipment maintenance/surveillance programs (See References *124*, *138*, *362*, and *363*).

This item was an ongoing effort to collect and analyze data. While no quantified safety benefits could be directly assigned to it, the benefits occurred as the results of the equipment failure rate data and reliability analysis were used in assessing other specific related safety issues, including Items II.C.1, II.C.2, and II.C.4. The ongoing activities were to be conducted as described in Section 16.1 of the NRC Long Range Research Program.[133] Thus, this Licensing Issue was resolved.

ITEM I.E.4: COORDINATION OF LICENSEE, INDUSTRY, AND REGULATORY PROGRAMS

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to ensure coordination of licensee, industry, and NRC programs for evaluating plant operating experience. As part of the implementation of NUREG-0737,[98] licensees established the capability for evaluating plant operating experience and procedures for providing feedback of the information to operations personnel and for incorporating it into training programs, in accordance with Items I.A.1.1, I.B.1.2, and I.C.5. Industry evaluation programs were to be conducted by INPO. AEOD was responsible for coordinating the NRC programs for evaluation of operational data with those of licensees and industry.

Licensee evaluations of plant operating experience, coordinated with industry and NRC evaluations using common data bases, were to ensure that licensee, industry, and NRC corrective action recommendations were properly coordinated and applied. Effective feedback of prioritized and analyzed event descriptions to plant operating personnel and incorporation into training programs can avoid occurrence of these problems at other plants of similar design. This item was initiated to improve the NRC capability to assess safety and, therefore, was considered a Licensing Issue.

CONCLUSION

The results of industry and NRC operating experience evaluations are shared under an NRC-INPO Memorandum of Agreement[238] initially signed in June 1981 and revised in April 1982. Thus, this Licensing Issue was resolved.

ITEM I.E.5: NUCLEAR PLANT RELIABILITY DATA SYSTEM

This item was evaluated with Item I.E.6 below and resolved.

ITEM I.E.6: REPORTING REQUIREMENTS

Items I.E.5 and I.E.6 were combined and evaluated together.

DESCRIPTION

The objectives of these TMI Action Plan[48] items were to: (1) determine if there was a need to make licensee participation in NPRDS mandatory; and (2) establish improved reporting requirements for operating reactors.

NPRDS is a reliability-oriented data collection and reporting system for selected components and systems related to the safety of nuclear power plants. Periodic reports containing failure statistics are issued and licensee participation is voluntary. Improvements in reporting significant events of operating plants can identify potential significant safety problems or their precursors and can avoid occurrence of these problems at other plants of similar design. Improved reporting of system/component reliability data will increase the validity of operating experience assessments and PRA programs. This item was initiated to improve the NRC capability to assess safety and, therefore, was considered a Licensing Issue.

CONCLUSION

By affirmation of SECY-81-494,[260] the Commission endorsed the following staff actions to resolve these issues: (1) develop a proposed rule to modify and codify the existing LER requirements and to assure consistency with 10 CFR 50.72 which covers the immediate reporting of significant events; (2) endorse the INPO plan to assume responsibility for the management, funding, and technical direction of NPRDS; (3) coordinate with INPO to minimize duplication between the LER and NPRDS systems and between subsequent NRC and INPO analysis of NPRDS data; (4) obtain INPO assurance that NPRDS receives, processes, and disseminates the reliability data needed by industry and the NRC to support PRA programs; and (5) monitor (by AEOD) INPO's management of the NPRDS and provide the Commission with semi-annual reports on the effectiveness of INPO management of NPRDS.

As of January 1982, INPO had assumed responsibility for the NPRDS and the NRC was represented on the NPRDS Users' Group and participated in various NPRDS work groups. AEOD submits semi-annual reports to the Commission on the effectiveness of the INPO management of NPRDS. A proposed rule on LERs was published in the Federal Register (47 FR 19543) on May 6, 1982, and the final rule[597] was published in July 1983. Based on the actions described above, Items I.E.5 and I.E.6 are Licensing Issues that were resolved.

ITEM I.E.7: FOREIGN SOURCES

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to supplement domestic operating experience of safety significance by obtaining operating and design information from foreign reactors. To obtain foreign experience in a more systematic manner, the Office of International Programs (IP) participated with nuclear regulatory agencies of other nations in a centralized exchange of incident information with the Nuclear Energy Agency (NEA); this NEA exchange was initiated in 1980. Supplementing this effort was the upgrading of information exchange on significant incidents through direct contact and correspondence between the NRC and its bilateral partners, and by additional formal bilateral information exchange agreements which were concluded or renewed in 1981 and 1982.

AEOD also sponsors a program by which the Nuclear Operations Analysis Center at ORNL screens and stores for ready access reports of foreign reactor incidents and provides monthly summaries of these events that are potentially significant and relevant to U.S. LWRs. This item was initiated to improve the NRC capability to assess safety and, therefore, was considered a Licensing Issue.

CONCLUSION

As of June 1984, foreign reactor incident and operating experience reports were being routinely received and disseminated to NRC technical staffs. IP was also routinely sending these foreign reactor incident reports to INPO for use by industry in evaluating plant operating experience under Item I.E.4. Foreign reactor incident and operating experience reports were being assessed by AEOD and affected NRC Offices, as described in Items I.E.1 and I.E.2, respectively, to identify potential significant safety problems or their procedures which may be applicable to U.S. plants. Dissemination within NRC and to industry of such assessments, along with their resolutions, can avoid occurrence of these problems at other facilities of similar design. Thus, this Licensing Issue was resolved.

ITEM I.E.8: HUMAN ERROR RATE ANALYSIS

DESCRIPTION

The TMI-2 incident increased concern for the effect of human reliability on reactor safety. The lack of human reliability data applicable to nuclear power plants compared to hardware reliability data highlighted this concern in nuclear safety assessments and regulation. The purpose of this TMI Action Plan[48] item was to continue research to: (1) complete analysis of field-collected data for human reliability in maintenance and calibration activities at operating nuclear power plants; (2) review abnormal occurrence reports, licensee event reports, and compliance reports to identify areas where human performance reliability is low; (3) develop probability models to predict error rates for multiple human errors occurring as a function of coupling influences; and (4) identify patterns and basic associative factors for the human-error rates determined for basic test, maintenance, and operator actions.

The information obtained from this item was to be used to: (1) identify necessary improvements in operator training and training aids to reduce human error rates (see Items I.A.2.6, "Long-Term Upgrading of Training and Qualification of Operating Personnel," and I.A.4.2, "Long-Term Training Simulator Upgrade"); and (2) provide quantifiable human error data and models to support PRA of nuclear power plants [see Items II.C.1, "Interim Reliability Evaluation Program" (IREP), and II.C.3, "Continuation of IREP"]; and (3) provide human engineering criteria for evaluating the design of new or modified systems.

While no quantified safety benefit could be directly assigned to this item, the benefits were expected to occur as the results of the human error rate and reliability analyses were used in assessing other individual related safety issues, including TMI Action Plan Items I.A.2.6, I.A.4.2, II.C.1, II.C.2, and Task Action Plan Item B-17. Therefore, this item was considered a Licensing Issue.

CONCLUSION

The Human Factors Branch of RES implemented an expanded Human Reliability research program to accomplish the purpose of this item and provide the human error information for its end use as described above. Major reports issued included: (1) a human reliability data bank[338]; (2) a draft handbook for human reliability analysis[339]; (3) procedures for estimating human error probabilities[341,342]; and (4) a workbook for conducting human reliability analysis.[343] Future work included finalizing the handbooks, workbooks, and reliability models and maintaining the data bank; this work was described in Section 7.1 of the NRC Long Range Research Plan.[133] Thus, this Licensing Issue was resolved.

REFERENCES

0048.	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.
0076.	NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report," U.S. Nuclear Regulatory Commission, June 1982.

0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0133.	NUREG-0784, "Long Range Research Plan FY 1984–1988," U.S. Nuclear Regulatory Commission, August 1982.
0238.	Memorandum for Chairman Hendrie et al. from W. Dircks, "Memorandum of Agreement with INPO and NSAC on a Cooperative Relationship for the Collection and Feedback of Operational Data," June 16, 1981. [8106260511, 8106260514]
0260.	SECY-81-494, "Integrated Operational Experience Reporting System," U.S. Nuclear Regulatory Commission, August 18, 1981. [8109110483]
0305.	Memorandum for Distribution from J. Davis, "NMSS Procedure for Review of Routine Inspection Operational Data and Licensee Event Reports," March 9, 1982. [8312290164]
0338.	NUREG/CR#2744, "Human Reliability Data Bank for Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, November 1982.
0339.	NUREG/CR#1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, October 1983.
0341.	NUREG/CR#2255, "Expert Estimation of Human Error Probabilities in Nuclear Power Plant Operations: A Review of Probability Assessment and Scaling," U.S. Nuclear Regulatory Commission, May 1982.
0342.	NUREG/CR#2743, "Procedures for Using Expert Judgment to Estimate Human Error Probabilities in Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, February 1983.
0343.	NUREG/CR#2254, "Workbook for Conducting Human Reliability Analysis," U.S. Nuclear Regulatory Commission, February 1983.
0353.	NUREG/CR#2641, "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report," U.S. Nuclear Regulatory Commission, July 1982.
0354.	NUREG/CR-2886, "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Interim Data Report—The Pump Component," U.S. Nuclear Regulatory Commission, January 1983.
0597.	*Federal Register* Notice 48 FR 33850, "Licensee Event Report System," July 26, 1983.

Task I.F: Quality Assurance (Rev. 5) ( )

The objective of this task was to improve the quality assurance program (QA) for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety.

ITEM I.F.1: EXPAND QA LIST

DESCRIPTION

Historical Background

The Three Mile Island (TMI) Action Plan [48] identified that several systems important to the safety of Three Mile Island Unit 2 (TMI-2) were not designed, fabricated, and maintained at a level equivalent to their safety importance; i.e., they were not on the QA list for the plant. This condition existed at other plants and resulted primarily from the lack of clarity in U.S. Nuclear Regulatory Commission (NRC) guidance on graded protection. Evaluation of this issue included the consideration of Issue 5 (see Section 3).

Safety Significance

One of the difficulties in establishing a QA list based on safety importance was the absence of relative risk assignments to equipment. At the time this issue was initially evaluated, QA requirements were applied principally to structures, systems, and components that prevented or mitigated the consequences of postulated accidents that could cause undue risk to the health and safety of the public (Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants").

Possible Solution

The TMI Action Plan stated that the NRC would develop guidance for licensees to expand their QA lists to cover equipment important to safety (ITS) and rank the equipment in order of its importance to safety. Experience in the use of the revised Office of Nuclear Reactor Regulation review procedure for developing QA lists for individual operating license applicants was to be factored into the generic guidance to be developed and when determining backfit requirements.[48] At the time this issue was identified, there was a task underway to define the applicability of Appendix B to 10 CFR Part 50 to equipment that met the requirements of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

PRIORITY DETERMINATION

The principal benefit to be derived from an expanded QA list was the knowledge that adequate guidance provided to each licensee to establish QA programs and requirements that were commensurate with the safety importance of structures, systems, and components, as determined from completed risk assessments. This guidance would not only result in the inclusion or addition to each licensee’s QA list of other ITS systems that were previously excluded but would also aid in clarifying the QA level of effort deemed necessary.

The risk reduction was probably proportionate to the difference between what would normally be the level of effort expended and the level defined. At the time this issue was initially evaluated, there was no measure of risk variation as a function of the variance in QA level of effort. However, it appeared reasonable to assume that a significant reduction in public risk could be achieved at those plants where the QA levels were held to the previous minimum acceptable level. Important questions to which there were no answers were (1) the number of plants that would be designed, built, and maintained below the new quality acceptance level and (2) how far below the new level the QA programs of these plants would actually operate.

Cost Estimate

Industry Cost: It was estimated that (1) the plant user cost applied to 40 plants in the design phase or under construction, (2) an average of 0.5 man-year/plant was required to develop an expanded QA list, (3) an additional 0.25 man-year/plant over 4 years was required to ensure compliance with the added QA requirements, and (4) an additional 0.1 man-year/plant would be expended to ensure compliance with the expanded QA list during the 40-year operating life of each affected plant. These estimates totaled 220 man-years and, at a rate of $100,000/man-year, the total industry cost was estimated to be $22 million (M).

NRC Cost: The NRC cost was estimated in the TMI Action Plan[48] to be 2.5 man-years or $0.25M.

Total Cost: The total industry and NRC cost associated with the solution was $(22 + 0.25)M or $22.25M.

CONCLUSION

Although a value/impact score was not calculated, the staff believed that the assurance of safer operation justified a high priority ranking for the issue.

The original intent of this issue was to identify those systems, structures, and components beyond those labeled "safety-related," prioritize their importance to safety, and prepare a generic QA list. This was reflected in 10 CFR 50.34(f)(3)(ii), which states, "Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)." However, the staff's "Interim Reliability Evaluation Program [IREP] Procedures Guide," issued March 1983,[812] failed to identify either the need for a QA list for ITS structures, systems, and components or the basis for a generic list even if one should be needed. The first four IREP studies performed at nuclear plants were reported in NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One—Unit Once Nuclear Power Plant," issued June 1982;[366] NUREG/CR-2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," issued August 1982;[367] NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," issued April and July 1983;[810] and NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," issued May and October 1984.[811] The staff's resolution of the IREP issue is discussed in Item II.C.1.

In January 1984, the NRC issued Generic Letter 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related,'"[1177] to clarify agency use of the terms "important to safety" and "safety related." This letter summarized the NRC's intention to pursue QA requirements for ITS equipment on a case-by-case basis; further clarification was provided in the Commission's Memorandum and Order CLI-84-9[1178] in June 1984. The first proposed rule on ITS was presented in SECY-85-119, "Issuance of Proposed Rule on the Important-to-Safety Issue," dated April 5, 1985,[1179] and was later disapproved by the Commission, which concluded that a specific listing of ITS equipment was not required to be maintained.[1180] Thus, the issue of expansion of the QA list to cover ITS equipment was considered to be closed and was not addressed in the second staff submittal on the ITS rule in SECY-86-164, "Proposed Rule on the Important-to-Safety Issue," dated May 29, 1986.[1181] Therefore, this issue was RESOLVED with no new requirements.[1182]

ITEM I.F.2: DEVELOP MORE DETAILED QA CRITERIA

DESCRIPTION

Historical Background

The overall objective of this TMI Action Plan[48] item was the improvement of the QA program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety. Several systems important to the safety of TMI-2 were not designed, fabricated, and maintained at a level equivalent to their safety importance. This condition existed at other plants and resulted primarily from the lack of clarity in NRC guidance. This situation and other problems relating to the QA organization, authority, reporting, and inspection were identified by the various TMI accident investigations and inquiries.[48]

Safety Significance

The intent of this item was to provide more explicit and detailed criteria concerning the elements that, in general, were found in well-conducted QA programs. Providing these more detailed criteria was expected to result in the establishment of QA programs of the caliber desired. The NRC believed that such programs would result in the detection of deficiencies in design, construction, and operation.

Possible Solutions

The proposed more detailed QA criteria for design, construction, and operations included the following:[48]

(1)	Assure the independence of the organization performing the checking functions from the organization responsible for performing the tasks. For the construction phase, consider options for increasing the independence of the QA function. Include an option to require that licensees perform the entire quality assurance/quality control (QA/QC) function at construction sites. Consider using the third-party concept for accomplishing the NRC review and audit and making the QA/QC personnel agents of the NRC. Consider using the Institute of Nuclear Power Operations to enhance QA/QC independence.
(2)	Include the QA personnel in the review and approval of plant operational maintenance and surveillance procedures and quality-related procedures associated with design, construction, and installation.
(3)	Include the QA personnel in all activities involved in design, construction, installation, preoperational and startup testing, and operation.
(4)	Establish criteria for determining QA requirements for specific classes of equipment such as instrumentation, mechanical equipment, and electrical equipment.
(5)	Establish qualification requirements for QA and QC personnel.
(6)	Increase the size of the licensees' QA staff.
(7)	Clarify that the QA program is a condition of the construction permit and operating license and that substantive changes to an approved program must be submitted to the NRC for review.
(8)	Compare NRC QA requirements with those of other agencies (i.e., National Aeronautic and Space Administration, Federal Aviation Administration, U.S. Department of Defense) to improve NRC requirements.
(9)	Clarify organizational reporting levels for the QA organization.
(10)	Clarify requirements for maintenance of "as built" documentation.
(11)	Define the role of QA in design and analysis activities. Obtain views on prevention of design errors from licensees, architect-engineers, and vendors.

PRIORITY DETERMINATION

The NRC staff assumed that the above criteria would be adopted by the nuclear industry. The staff made a priority determination of the benefit of the above 11 items for improving QA. (The staff did not make a priority determination of the benefit of a QA program itself.)

To address this issue adequately, improvement in the QA program must be developed independent of the performing organization. Furthermore, the QA organization must have the confidence and the ear of higher management so that QA concerns can be heard and acted upon. The deficiency of the effort called for in this issue was that the effectiveness of the improvement program depended on the acceptance, attitudes, and emphasis given by plant management to the benefits to be derived from a QA program. Licensees that placed a high importance on QA efforts would probably be able to incorporate the intent of the QA enhancement program without making major changes to their organizational structure or in the way they performed their plant operations. However, for those licensees that continued to do business "as usual," the changes could be more cosmetic than real. They would probably seek ways to establish a QA organization that, on the surface, might appear reasonable but that, in reality, could be a "paper tiger." Enclosure 1 of SECY-82-352, "Assurance of Quality," dated August 20, 1982,[308] states the following: "In sum, the fundamental issues can best be characterized as a lack of total management commitment to quality and the uncertainty in industry's and NRC's ability to detect and correct the resulting deficiencies."

CONCLUSION

Although the QA improvement program could result in the establishment of an improved QA organizational structure at many plants, the results depended heavily on management acceptance. Lack of program implementation and management acceptance, rather than inadequate criteria as suggested by this issue, were the primary causes of deficiencies in QA. Increasing the detail of the QA criteria had little potential for improving the quality of design, construction, or operation and, therefore, reducing risk. Items I.F.2(2), I.F.2(3), I.F.2(6), and I.F.2(9), which addressed the concern stated above, were included in the July 1981 revision to Chapter 17 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (the SRP).[11]

The NRC believed that the issue of QA in nuclear power plants should be a high priority. However, the issue and solutions to QA deficiency as described herein (except for completed issues I.F.2(2), I.F.2(3), I.F.2(6), and I.F.2(9)) failed to address the problem of management acceptance of QA programs. Therefore, the residual items were given a LOW priority.

The NRC staff conducted a review[1964] of the seven LOW priority issues in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. The staff determined that the operating experience has not indicated a change in the safety significance of these issues. In addition, the staff verified that the current NRC regulatory requirements or guidance address these issues and identified the applicable regulatory framework as presented below. Because these items have been addressed by the existing regulations and the operating experience has not raised the significance of these issues, the NRC staff DROPPED these issues from further pursuit. The following section provides a discussion to demonstrate the application of the NRC regulatory framework for QA to each issue and to support their disposition.

ITEM I.F.2(1): ASSURE THE INDEPENDENCE OF THE ORGANIZATION PERFORMING THE CHECKING FUNCTION

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933, published in November 1983. In 1998, consideration of new information[1715] on the lack of independence in the checking function submitted by Region IV in April 1997 did not change this conclusion.[1716]

The staff conducted a review[1964] of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. According to 10 CFR 50.34(f)(3)(iii), "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions." In addition, Section 17.5 of the SRP[11] states that "the QA program requires independence between the organization performing checking functions from the organization responsible for performing the functions. (This provision applies to DC applicant, ESP, and construction QA programs. This provision is not applicable to design reviews/verifications.)"

The NRC staff concluded that this item has been adequately addressed by the NRC's regulations and DROPPED this item from further pursuit. [1964]

ITEM I.F.2(2): INCLUDE QA PERSONNEL IN REVIEW AND APPROVAL OF PLANT PROCEDURES

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(3): INCLUDE QA PERSONNEL IN ALL DESIGN, CONSTRUCTION, INSTALLATION, TESTING, AND OPERATION ACTIVITIES

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(4): ESTABLISH CRITERIA FOR DETERMINING QA REQUIREMENTS FOR SPECIFIC CLASSES OF EQUIPMENT

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50 states that "The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety." In addition, applicants or license holders commit to the following standards, which identify requirements for specific classes of equipment:

- Subpart 2.4, "Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities," American Society of Mechanical Engineers (ASME) NQA-1-1994
- Subpart 2.5, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants," ASME NQA-1-1994
- Subpart 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," ASME NQA-1-1994
- Subpart 2.8, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants," ASME NQA-1-1994

Based on the review of NRC regulations related to this issue, the staff concluded that Item I.F.2(4) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(4) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(5): ESTABLISH QUALIFICATION REQUIREMENTS FOR QA AND QC PERSONNEL

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion II of Appendix B to10 CFR Part 50 establishes requirements for the training of personnel: "The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained." In addition, Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants,"[226] Revision 3, provides guidance that is acceptable to the NRC staff on qualifications and training for nuclear power plant personnel. This regulatory guide endorses American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants,"[253] with certain clarifications, additions, and exceptions.

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "establish a quality assurance (QA) program based on consideration of…(E) establishing qualification requirements for QA and QC personnel." Finally, Section 17.5 of the SRP11 describes the SRP acceptance criteria for "Training and Qualification Criteria—Quality Assurance."

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(5) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(5) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(6): INCREASE THE SIZE OF LICENSEES' QA STAFF

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(7): CLARIFY THAT THE QA PROGRAM IS A CONDITION OF THE CONSTRUCTION PERMIT AND OPERATING LICENSE

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

The regulation at 10 CFR 50.54(a)(1) clearly states that implementation of the QA program is a condition in every nuclear power reactor operating license issued under 10 CFR Part 50: "Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in appendix B of this part shall implement, under § 50.34(b)(6)(ii) or § 52.79 of this chapter, the quality assurance program described or referenced in the safety analysis report, including changes to that report. However, a holder of a combined license under part 52 of this chapter shall implement the quality assurance program described or referenced in the safety analysis report applicable to operation 30 days prior to the scheduled date for the initial loading of fuel." In addition, 10 CFR 50.54(a)(1) is also a condition in every combined license issued under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Finally, 10 CFR 52.17(a)(1)(xi), 10 CFR 52.47(a)(19), and 10 CFR 52.79(a)(25) outline the QA program requirements for applicants for early site permits (ESPs), standard design certifications (DCs) and combined licenses, respectively. SRP[11] Section 17.5 outlines a standardized QA program for DC, ESP, construction permit, operating license, and combined license applicants and holders.

Moreover, this issue specifies that "substantive changes to an approved program must be submitted to NRC for review." This part of the issue is also addressed by 10 CFR 50.54(a)(4), which states that "Changes to the quality assurance program description that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation." The regulation at 10 CFR 50.54(a)(4)(i)–(iv) outlines the process to make these changes.

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(7) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(7) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(8): COMPARE NRC QA REQUIREMENTS WITH THOSE OF OTHER AGENCIES

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

On July 9, 2003, the results of the staff’s effort to review international quality assurance standards against the existing Appendix B to 10 CFR Part 50 framework were reported by issuance of SECY-03-0117, Approaches for Adopting More Widely Accepted International Quality Standards."[1965] In addition, approaches for adopting international quality standards for safety-related components in nuclear power plants into the existing regulatory framework were assessed. SECY-03-0117[1965] also reviewed existing NRC quality assurance requirements and efforts to improve their effectiveness and efficiency. The staff concluded in SECY-03-0117[1965] that considerable actions had already been taken or were in progress to reduce unnecessary regulatory burden on licensees resulting from compliance with Appendix B to 10 CFR Part 50 requirements. In addition, the proposed 10 CFR 50.69 risk-informed rulemaking would provide a more efficient and effective regulatory process while continuing to maintain safety. The staff evaluation of the differences between Appendix B to 10 CFR Part 50 and ISO 9001 is summarized in the attachment to SECY-03-0117.[1965]

The staff concluded that the analysis presented in SECY-03-0117[1965] addressed Item I.F.2(8) adequately and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(9): CLARIFY ORGANIZATIONAL REPORTING LEVELS FOR THE QA ORGANIZATION

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(10): CLARIFY REQUIREMENTS FOR MAINTENANCE OF "AS-BUILT" DOCUMENTATION

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion VI, "Document Control," and Criterion XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50 establish requirements for issuing, identifying, and retrieving QA records. In addition, NRC-accepted practices for the collection, storage, and maintenance of QA records for nuclear power plants, independent storage of spent nuclear fuel and high-level radioactive waste facilities, special nuclear materials, packaging and transportation of radioactive materials, and gaseous diffusion plants are described in ANSI/ASME NQA-1.[1966]

Criterion VI of Appendix B to 10 CFR Part 50 describes the requirements to control changes in documents: "Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization."

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of…(G) establishing procedures for maintenance of ‘as-built’ documentation." Finally, Section 17.5 of the SRP[11] states that "A program is required to be established to control the development, review, approval, issue, use, and revision of documents." This section includes as-built drawings as one of the examples of controlled documents: "Examples of controlled documents include design drawings, as-built drawings, engineering calculations."

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(10) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(10) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(11): DEFINE ROLE OF QA IN DESIGN AND ANALYSIS ACTIVITIES

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 describes the requirements of the program for the design of items. As explained in this criterion, measures should be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. In addition, these measures should include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. The design control measures provide for verifying or checking the adequacy of design and are applied to items such as the reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of…(H) providing a QA role in design and analysis activities." Finally, Section 17.5 of the SRP11 states that "The QA role in design and analysis activities is defined. Design documents are reviewed by individuals knowledgeable and qualified in QA to ensure the documents contain the necessary QA requirements. (This applies to DC applicants, ESP, and construction QA programs.)"

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(11) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(11) and DROPPED this item from further pursuit.[1964]

REFERENCES

0011.	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.
0048.	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.
0226.	Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1971, (Rev. 1) September 1975 [8801130111], (Rev. 1#R) May 1977 [7907100073], (Rev. 2) April 1987. [8907180147]
0253.	ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," American National Standards Institute, 1981.
0308.	SECY-82-352, "Assurance of Quality," U.S. Nuclear Regulatory Commission, August 20, 1982. [8209160068]
0366.	NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One—Unit One Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1982.
0367.	NUREG/CR#2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," U.S. Nuclear Regulatory Commission, August 1982, (Appendix A) August 1982, (Appendix B) August 1982, (Appendix C) August 1982.
0810.	NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983, (Vol. 3) July 1983, (Vol. 4) July 1983.
0811.	NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1984, (Vol. 2) October 1984.
0812.	NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," U.S. Nuclear Regulatory Commission, March 1983.
1175.	SECY-89-081, "Final Report on Chernobyl Implications," U.S. Nuclear Regulatory Commission, March 7, 1989. [8903200205]
1177.	Letter to All Holders of Operating Licenses, Applicants for Operating Licenses and Holders of Construction Permits for Power Reactors from U.S. Nuclear Regulatory Commission," NRC Use of the Terms, `Important to Safety' and `Safety Related' (Generic Letter 84-01)," January 5, 1984. [ML031150515]
1178.	Memorandum and Order CLI-84-9, U.S. Nuclear Regulatory Commission, June 6, 1984. [8406070146]
1179.	SECY-85-119, "Issuance of Proposed Rule on the Important-to-Safety Issue," U.S. Nuclear Regulatory Commission, April 5, 1985. [8505030656]
1180.	Memorandum for W. Dircks from S. Chilk, "Staff Requirements—SECY-85-119—`Issuance of Proposed Rule on the Important-to-Safety Issue,'" December 31, 1985. [8601160559]
1181.	SECY-86-164, "Proposed Rule on the Important-to-Safety Issue," U.S. Nuclear Regulatory Commission, May 29, 1986. [8607010004]
1182.	Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue I.F.1, `Expand QA List,'" January 12, 1989. [9704150147]
1716.	Memorandum for T. Gwynn from T. Martin, "Periodic Review of Low-Priority Generic Safety Issues," July 13, 1998. [9909290134]
1964.	Memorandum for B.W. Sheron from B.G. Beasley, "LOW Priority Generic Issues," March 17, 2011.[ML092520025]

1965.	SECY-03-0117, "Approaches for Adopting More Widely Accepted International Quality Standards," July 9, 2003. [*ML031490421*]
1966.	American Society of Mechanical Engineers, "Quality Assurance Program Requirements for Nuclear Facility Applications," ANSI/ASME Standard NQA -1, Washington, DC.

Task I.G: Preoperational and Low-Power Testing (Rev. 3) ( )

The objectives of this task were to: (1) increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events was conducted; and (2) review the comprehensiveness of test programs. Near-term operating license facilities were required to develop and implement intensified training exercises during the low-power testing programs

ITEM I.G.1: TRAINING REQUIREMENTS DESCRIPTION

This TMI Action Plan[48] item called for new OLs to conduct a set of low power tests to achieve the objectives of Task I.G. These tests were to be determined on a case-by-case basis.

CONCLUSION

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM I.G.2: SCOPE OF TEST PROGRAM

DESCRIPTION

Historical Background

The major thrust of TMI Action Plan[48] Task I.G was to use the preoperational and startup test programs as a training exercise for operating crews. In contrast to this, Item I.G.2 called for a more comprehensive test program to search for anomalies in a plant's response to a transient. This issue was suggested independently by the Kemeny Commission,[175] the Rogovin Commission,[161] the ACRS,[176] and the TMI Operations Team.[177]

Safety Significance

The safety significance of this issue was in the early discovery of anomalies or unanticipated plant behavior. The TMI-2 accident was the most well-known example, but other less severe examples, such as the core-annulus water level decoupling at Oyster Creek, have taken place.

When a plant responds to a transient in an anomalous or unanticipated manner, the result may be an accident caused directly by the new phenomena, or by the surprise or confusion on the part of the operators. The latter is probably the more likely of the two.

Possible Solution

The nature of the solution to this issue was implicit in its definition - an augmented test program. However, relatively little had been written concerning the nature and extent of this program. NUREG-O66O[48] merely called for the NRC to develop a program. Recommendations[177] made by an OIE team investigating TMI-2 were more specific: detailed review of all unscheduled transients during the first year as well as review of the preoperational and startup tests.

There was a spectrum of possible test programs ranging from the existing program to programs that would take years to implement. Morever, it might not have been necessary for each plant to perform each test. In addition, there was a large amount of operating experience data from which information could be gathered.

PRIORITY DETERMINATION

Frequency Estimate

At the time this issue was initially evaluated, transients occurred at an approximate rate of 10/RY; however, most of these were relatively routine (e.g., turbine trip) and were thus unlikely to produce unpleasant surprises. In any case, existing startup programs were expected to cover them adequately. Therefore, attention was focused on transients that were rare but nevertheless frequent enough to be considered "anticipated operational occurrences." EPRI NP-801[178] contains a history of the transients that were actually experienced during operation. Based on judgment, transients that were candidates for suspicion of anomalous behavior were selected.

PWR Transients	Frequency (RY^{-1})
Hi/Lo Pressurizer Pressure	0.10
Pressurizer Safety or Relief-Valve Opening	0.02
Inadvertent SIS	0.04
Loss of RCS Flow	0.04
Close All MSIVs	0.05
Sudden Opening of Secondary Relief Valves	0.06
Loss of Component Cooling	0.01
Loss of Service Water System	0.01
Total:	0.33

BWR Transients	Frequency (RY^{-1})
Pressure Regulator Fails Open	0.29
Pressure Regulator Fails Closed	0.14
Inadvertent Opening of S/RV	0.20
Trip One Recirculation Pump	0.02
Trip All Recirculation Pumps	0.06
Total:	0.71

At the time of this evaluation, reactor experience totaled 565 RY: 225 BWR-years and 340 PWR-years.[179] Thus, it was estimated that about 270 of the listed transients had occurred, some of which had indeed illustrated the need for corrective measures. Unfortunately, it was not practical to use the computerized data banks to search for "anomalous behavior." Once again, judgment was used.

At least four transients with anomalous response had occurred (Davis-Besse, Three Mile Island, Oyster Creek, Pilgrim) and were widely known. If a more thorough review of operating experience had been made, more would have been discovered. It was estimated that perhaps 10 transients had shown some sort of unanticipated phenomenon. However, the number of interest was the number of phenomena left to be discovered. With a history of about 270 transients of interest, anomalous events were not expected to be very common. Moreover, the discoveries that had been made led to measures intended to prevent future problems.

It was estimated that anomalous or unanticipated behavior could be expected at a rate of about 5 events in 565 RY (i.e., half the estimated historical rate) or about 10^{-2}/RY. This number was an "educated guess" that the actual number of events that had occurred was higher than the four events listed, but would be lower in the future because the experience had been used to correct the problems.

Consequence Estimate

Most anomalous transients have no consequences in the sense of releasing radioactivity. Based on the experience of TMI (one event in perhaps 10), it was assumed that one event in 10 will result in core damage (extensive cladding failure) and one event in 100 will result in a core-melt with a significant release. The former was approximated with a PWR-9 or BWR-5 Category event and the latter with a PWR-7 or BWR-4 event.

It was assumed that an augmented startup program would be 50% effective in discovering and correcting problems. The total risk reduction associated with this issue was 2.58×10^4 man-rem, based on 252 man-rem for 36 PWRs and 2.56×10^4 man-rem for 21 BWRs.

Cost Estimate

Industry Cost: As stated previously, there was a spectrum of possible test programs. It was assumed that the test program would average 2 weeks/plant. At $300,000/day for replacement power (which would dominate the cost), this was $4.2M/plant. The 2-week average estimate was based on the assumption that not every plant would perform every test. In many cases, the first product of a given product line would be subjected to a great deal of testing that would apply to all plants of the same design, or testing could be shared within a product line by some other plan. Therefore, the total industry cost was $239.4M.

NRC Cost: It was assumed that 5 staff-years would be required to develop guidelines and approve generic plans with one staff-month/plant of post-test review. With 57 OLs, (36 PWRs and 21 BWRs), this cost was about $1M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(239.4 + 1)M or approximately $240M.

Value/Impact Assessment

Based on an estimated public risk reduction of 2.58×10^4 man-rem and a cost of $240M for a possible solution, the value/impact score was given by:

$$S = \frac{2.58\text{x}10^4 \text{ man} - \text{rem}}{\$240\text{M}}$$

$$= 108\text{man} - \text{rem} / \$\text{M}$$

Uncertainties

The frequency estimates did not rest upon firm bases. This was not surprising because, like any other program where the goal is discovery, if good bases were available for estimates of effectiveness, the tests would not be necessary. Nevertheless, an attempt was made to put bounds on the estimates. The frequency of core damage was not likely to be uncertain by more than a factor of 10. If the true frequency were a factor of 10 higher, about 6 core-damage accidents would have occurred. If it were a factor of 10 lower, the TMI-2 accident would have a probability on the order of 0.05.

However, the core-melt frequency was subject to more uncertainty. It was assumed that the frequency of core-melt was one-tenth of that for core-damage. It was assumed that this figure could be either a factor of 5 higher (every second TMI-like event a core-melt) or a factor of 5 lower (one core-melt in 50 core-damage events).

Assuming that the public dose estimates were uncertain to a factor of 5 and the costs to a factor of 5, then the value/impact score would have a range from 3×10^0 to 4×10^3 man-rem/$M.

Other Considerations

The averted costs of cleanup were not considered in the value/impact score. If such costs ($0.25M/RY) were included, the value/impact score would be significantly higher, but not enough to justify a higher priority.

CONCLUSION

Based on the value/impact score and the associated public risk reduction, this issue was given a medium priority ranking. With revisions to SRP[11] Section 14 and the OIE Manual, the issue was RESOLVED with new requirements.[654]

REFERENCES

0011.	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.
0048.	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.

0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0161.	NUREG/CR-1250, "Three Mile Island: A Report to the Commission and to the Public," U.S. Nuclear Regulatory Commission, January 1980.
0175.	ZAR-791030-01, "Report of the President's Commission on the Accident at Three Mile Island," J. G. Kemeny et al., November 30, 1979.
0176.	Memorandum for J. Ahearne from M. Carbon, "Comments on the Pause in Licensing," December 11, 1979. [8001080218]
0177.	Memorandum for N. Moseley from J. Allan, "Operations Team Recommendations—IE/TMI Unit 2 Investigation," October 16, 1979. [8007160815]
0178.	EPRI NP-801, "ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients," Electric Power Research Institute, July 1978.
0179.	NUREG-0020, "Licensed Operating Reactors, Status Summary Report," U.S. Nuclear Regulatory Commission, (Vol. 6, No. 2) February 1982.
0654.	Memorandum for W. Dircks from H. Thompson, "Closeout of TMI Action Plan Task I.G.2, 'Scope of Test Program,'" October 5, 1984. [8410160524]

Task II.A: Siting (Rev. 2) ( )

The objective of this task was to provide an added contribution to safety through the development of siting criteria for new power plants and the reevaluation with regard to the new siting criteria of facilities either under construction or operating.

ITEM II.A.1: SITING POLICY REFORMULATION DESCRIPTION

In this TMI Action Plan item,[48] the staff was required to identify the principal criteria for evaluating proposed sites for nuclear power stations, recommend the adoption of these criteria in a Proposed Rule on Siting, and prepare an environmental assessment or environmental impact statement of the proposed revisions to meet NEPA requirements.

PRIORITY DETERMINATION

This issue was investigated by PNL but no risk or cost analyses were made.[64]

Frequency Estimate

Siting does not directly affect the frequency of radioactive releases. However, it should be noted that longer transmission lines will increase the frequency of load rejections and thus somewhat increase the probability of a release.

Consequence Estimate

WASH-1400[16] estimates of risk were used assuming a uniform population density of 340 people/square-mile, the average for U.S. sites. Multiplying frequency by consequences for each Release Category and then summing the products, the average risk was 70 man-rem/PWR-year and 150 man-rem/BWR-year. Thus, compared to an average site, the maximum difference remote siting could make would be 150 man-rem/RY, which corresponded to locating a BWR in a completely deserted area. This average population density was comparable to the existing criteria in SRP[11] Section 2.1.3 which limited the surrounding population density to about 500 people/square-mile.

Cost Estimate

Industry Cost: Remote siting involves a number of cost factors the most significant of which are: transmission line losses; lower plant availability due to longer transmission lines; cost of land for a major transmission line corridor and delays involved in acquiring the land; and recruiting and relocating personnel to staff the plant. The latter two factors, although widely recognized as significant, were difficult to quantify generically. However, assuming a 1% line loss (reasonable for a 100-mile line) and five additional load rejections per year, the first two factors totaled more than $100M for a 1,000 MWe plant over 40 years.

NRC Cost: NRC costs were insignificant in comparison to industry costs.

Total Cost: The total industry and NRC cost associated with the possible solution was $100M.

Value/Impact Assessment

Based on an estimated public risk reduction of 6,000 man-rem/reactor and a cost of $100M/reactor for a possible solution, the value/impact score was given by:

$$S \leq \frac{6{,}000 \text{ man} - \text{rem / reactor}}{\$100\text{M / reactor}}$$

$$\leq 60 \text{ man} - \text{rem / \$M}$$

Other Considerations

The relatively low value/impact score must be combined with consideration of the net risk of 70 man-rem/PWR-year and 150 man-rem/BWR-year. Over a 40-year plant life, this corresponded to 3,000 to 6,000 man-rem which would normally place the issue automatically in the high priority category, regardless of value/impact score or cost-effectiveness. However, this was the maximum risk reduction and most future sites would provide less. Specific sites may have better access to the grid and thus may be more cost-effective. At the time of this evaluation, there were no new plants being proposed.

CONCLUSION

Based on the above considerations and the need to address siting questions, this issue was given a medium priority ranking (see Appendix C). However, in 1984, the Commission decided to better define its safety objectives and better characterize radioactive source terms before proceeding with new siting regulations. As a result, it was decided that, before new siting efforts were undertaken, a new radioactive source term must be approved and the evaluation of the safety goal must be completed. Upon completion of these two tasks, the need for a revised siting rule was to be reassessed and, if necessary, a new generic safety issue was to be established to address siting rulemaking. Thus, this item was RESOLVED and no new requirements were established.[655]

ITEM II.A.2: SITE EVALUATION OF EXISTING FACILITIES

DESCRIPTION

In this TMI Action Plan item,[48] the staff was to "prepare an analysis for Commission decision of the NRC staff plans to reconsider, with regard to the revised siting policy, facilities either under construction or operating. The analysis would take, as a point of departure, the criteria expressed in the Proposed Rule on Siting (Item II.A.1) and would address a strategy for consideration of siting decisions of plants that have construction permits or operating licenses."

CONCLUSION

At the time of this evaluation, the basic purpose behind the issue was being addressed in the larger context of the Safety Goal[69] which was being developed under TMI Action Plan[48] Item V.A.1. Consequently, all NRC staff efforts on this issue were terminated in mid-1981.

REFERENCES

0011.	*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.
0016.	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.
0048.	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.
0064.	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.
0069.	NUREG#0880, "Safety Goals for Nuclear Power Plants: A Discussion Paper," U.S. Nuclear Regulatory Commission, February 1982, (Rev. 1) May 1983.
0655.	Memorandum for W. Dircks from H. Denton, "Generic Issue II.A.1, 'Siting Policy Reformulation,'" September 17, 1984. [8410090175]

Task II.B: Consideration of Degraded or Melted Cores in Safety Review (Rev. 5) ( )

The objective of this task was to enhance public safety and reduce individual and societal risk by developing and implementing a phased program to include, in safety reviews, consideration of core degradation and melting beyond the design basis.

ITEM II.B.1: REACTOR COOLANT SYSTEM VENTS

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-10 was established by DL/NRR for implementation purposes.

ITEM II.B.2: PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-11 was established by DL/NRR for implementation purposes.

ITEM II.B.3: POST-ACCIDENT SAMPLING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-12 was established by DL/NRR for implementation purposes.

ITEM II.B.4: TRAINING FOR MITIGATING CORE DAMAGE

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-13 was established by DL/NRR for implementation purposes.

ITEM II.B.5: RESEARCH ON PHENOMENA ASSOCIATED WITH CORE DEGRADATION AND FUEL MELTING

The three parts of this item are evaluated below.

ITEM II.B.5(1): BEHAVIOR OF SEVERELY DAMAGED FUEL

Items II.B.5(1) and II.B.5(2) were combined and evaluated together under Item II.B.5(1).

DESCRIPTION

Historical Background

For a number of key severe accident sequences, there are critical phenomenological unknowns or uncertainties that impact containment integrity assessments and judgments regarding the desirability of certain mitigating features. The phenomena fall into three broad categories: (1) the behavior of severely damaged fuel, including oxidation and H_2 generation; (2) the behavior of the core-melt in its interaction with water, concrete, and core-retention materials; and (3) the effect of potential H_2 burning and/or explosions on containment integrity. Steam explosions were also to be considered in this category. Previous work in these several areas received less attention since they related to accidents beyond the design basis. At the time this TMI Action Plan[48] item was raised, RES was conducting major programs to support the basis for rulemaking and to confirm certain licensing decisions. Complementary efforts conducted within NRR were to address specific licensing issues related to the subject research.

(1) **Behavior of Severely Damaged Fuel**

(a) **In-Pile Studies**: Fuel behavior research was to include in-pile testing to help evaluate the effects of conditions leading to severe fuel damage. Such tests were being performed in the INEL Power Burst Facility (PBF) in FY 1983 and later in the Annular Core Research Reactor (ACRR) at SNL and in the NRU reactor at Chalk River National Lab, Canada. In the PBF, RES was to perform a series of in-reactor fuel experiments to determine the effect of heating and cooling rates on damage to the bundle, rod fragmentation, distortion, and debris formation. Fission product release and H_2 generation were also to be measured during the testing. Separate effects studies were to be conducted on rubble beds in the ACRR at SNL.

(b) **Hydrogen Studies**: The objective of this work was to increase the understanding of the formation of H_2 in a reactor from metal-water reactions, radiolytic decomposition of coolant, and corrosion of metals, and to determine its consequences in terms of pressure-time histories and H_2 deflagration or detonation. This work was also to include: (1) the preparation of a compendium of information related to H_2 as it affects reactor safety; (2) analysis of radiolysis under accident conditions; (3) a review of H_2 sampling and analysis methods; (4) a study of the effects of H_2 embrittlement on reactor vessel materials; and (5) a review of means of handling accident-generated H_2 with recommendations on improving existing methods. Results of these studies were considered in the resolution of Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," and were not considered further in this issue.

(c) **Studies of Post-Accident Coolant Chemistry**: The RES objective in this area was the development of a relationship between fission product release and fuel failure and the improvement of post-accident sampling and analysis techniques. This was to be accomplished by the investigation of fission product release in a variety of fuel failure experiments.

(d) **Modeling of Severe Fuel Damage**: The effort in this area was the development of models for fuel rods operating beyond 2200°F that suffer a loss in geometry in order to compute extensive damage phenomena (such as eutectic liquid formation, fuel slumping, oxidation, and H_2 generation, fission product release and interaction with the coolant, rubble-bed particle size, extent of fuel and clad melting, and flow blockage).

(2) **Behavior of Core-Melt**

The RES fuel-melt research program was to develop a base and verified methodology for assessing the consequences and mitigation of fuel-melt accidents. The program addressed the range of severe reactor accident phenomena from the time when extensive fuel damage and major core geometry changes occur until the containment has failed and/or the molten core materials have attained a semi-permanent configuration and further movement is terminated. Studies of improvements in containment design to reduce the risk of core-melt accidents were also included.

The program was composed of integrated tasks that included scoping, phenomenological and separate effects tests, and demonstration experiments that provided results for the development and verification of analytical models and codes. These codes and supporting data were then used for the analysis of thermal, mechanical, and radiological consequences of accidents and for decisions related to requirements of design features for mitigation and performance confirmation. The technical scope of the program included work in the following areas: fuel debris behavior; fuel interactions with structure and soil; radiological source term; fuelcoolant interactions; systems analysis codes; and mitigation features.

Safety Significance

The results of the research programs described above were expected to find broad application in areas such as PRA, accident analysis, siting, evacuation planning, emergency procedures, code development, etc. Thus, these programs would have considerable value just as licensing improvement efforts. However, the programs had sufficiently well-defined scopes to permit some estimates of direct safety significance. These programs were directed at a better understanding of severely damaged and molten cores. Once a core is in this state, any safety significance has to be in the area of minimizing radioactive releases and consequent dose to the public.

Possible Solutions

It was assumed that means would be devised to reduce the probability of containment failure and release of activity to the environment. Completely different approaches could be suggested after the results of the research programs were known.

The "classical" engineering approaches to handling degraded or melted cores are filtered vents to prevent containment overpressure and core-retention devices (core catchers) to prevent containment basemat melt-through. These approaches were used for cost estimates, but the other priority parameters were not specific to these approaches.

PRIORITY DETERMINATION

Studies[64] of this issue by PNL considered only containment basemat melt-through. The approach presented here was expanded to include other aspects. The effect on a PWR with a dry containment was considered, based partly on the availability of information. It was not expected that the results for other containments or for BWRs would be greatly different, at least in the context of the uncertainty of such an analysis.

Frequency Estimate

Essentially, all core-melts are assumed to result in containment failure in WASH-1400.[16] To estimate the effect of being able to deal with a severely damaged core, this assumption was relaxed. The modes of containment failure for PWRs were defined as follows:

α - containment rupture due to a reactor vessel steam explosion

β - containment failure due to inadequate isolation of openings and penetrations

γ - containment failure due to H_2 burning

δ - containment failure due to overpressure

ε - containment vessel melt-through

Assuming that the research programs were successful in leading to engineering solutions, reductions in the frequency of the various failure modes were estimated as follows:

α - 10% (Little can be done about steam explosions.)

β - 0% (This does not affect isolation failure.)

γ,δ - 90% (Venting containment should be quite effective if methods are available for sizing the vent and determining what filtration is needed.)

ε - 90% (Should be achievable if a core catcher can be designed.)

Consequence Estimate

The consequences were straightforward in the sense that the consequences of each release category have been studied. However, the reduction in consequences was more difficult to assess since the release from a molten core in a tight containment is still not zero. Instead, it depends on the containment design leak rate, the efficiency of filtration of a containment relief vent, etc. To allow for this, it was assumed instead that the prevented releases corresponding to the α, γ, δ, and ε failure modes were similar to a PWR-9 release. The results of this calculation are summarized in Table II.B-1. For a new (forward-fit) plant (which was the most likely candidate for implementation), the public risk reduction was estimated to be 1,600 man-rem.

Cost Estimate

Industry Cost: PNL estimated[64] the cost of a core retention device to be $1.4M for a forward-fit. SNL estimated[312] the cost of a filtered containment vent to be on the order of a few million dollars. Thus, the industry cost was projected to be $10M/reactor.

NRC Cost: PNL estimated[64] the NRC cost to be $2.3M, assuming implementation at 134 reactors. In reality, implementation might take place at a far smaller number of plants due to considerations of containment type, backfit vs. forward-fit, etc. However, even if only 10 plants were affected, the NRC cost would be insignificant compared to licensee costs. Therefore, NRC costs were neglected.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $10M/reactor.

Release Category	Frequency* $(RY)^{-1}$	% Reduction**in Frequency	ΔF $(RY)^{-1}$	R (man-rem)	FR
Table II.B-1					
PWR-1	5.3×10^{-8}	10%	5.3×10^{-5}	4.9×10^{6}	2.6×10^{-2}
PWR-2	6.7×10^{-6}	90%	6.0×10^{-6}	4.8×10^{6}	2.9×10^{1}
PWR-3	2.6×10^{-6}	81%	2.1×10^{-6}	5.4×10^{6}	1.1×10^{1}

Release Category	Frequency* $(RY)^{-1}$	% Reduction**in Frequency	$\Delta F\ (RY)^{-1}$	R (man-rem)	FR
PWR-4	$2.1 \times l0^{-5}$	-	-	2.7×10^{6}	-
PWR-5	4.9×10^{-5}	-	-	1.0×10^{6}	-
PWR-6	6.3×10^{-5}	90%	5.7×10^{-7}	1.4×10^{5}	8.0×10^{-2}
PWR-7	3.4×10^{-5}	90%	3.1×10^{-5}	2.3×10^{3}	7.1×10^{-2}
PWR-8	8.0×10^{-5}	-	-	7.5×10^{4}	-
PWR-9	4.0×10^{-5}	-	-3.9×10^{-5}	1.2×10^{2}	-4.7×10^{-3}
			TOTAL:		4.0×10^{1}

* Because the specific containment failure mode was of interest here, the frequencies above were "unsmoothed." This is in contrast to the calculations in WASH-1400[16] which assumed a 10% contribution in frequency from adjacent release categories.

** Release Category PWR-1 is made up entirely of α failures and thus was assigned a 10% reduction in frequency. Categories PWR-2, PWR-6, and PWR-7 are made up of γ, δ, and ε failures and were thus assigned 90%. Category PWR-3 contains both α and δ failures which results in a net assignment of 81%.

Value/Impact Assessment

Based on a potential public risk reduction of 1,600 man-rem/reactor and a cost of \$10M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{1{,}600 man - rem / reactor}{\$10 M / reactor} = 160 man - rem / \$M$$

CONCLUSION

Based on the factors considered above, this issue was given a high priority ranking (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[138l] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issue. Because licensing and regulatory impact issue are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.5(2): BEHAVIOR OF CORE-MELT

This item was evaluated in Item II.B.5(1) above and determined to be a high priority (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[138l] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,1967 that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.5(3): EFFECT OF HYDROGEN BURNING AND EXPLOSIONS ON CONTAINMENT STRUCTURE

DESCRIPTION

Historical Background

TMI Action Plan[48] Item II.B.5 called for research into the phenomena associated with severe core damage and core melting. Item II.B.5(3) addressed the effect of H_2 burns and/or explosions on containment integrity.

Safety Significance

Whereas Items II.B.5(1) and II.B.5(2) dealt with (among other things) the generation of H_2 via radiolysis, metal-water interaction, interaction of a molten core with concrete, etc., Item II.B.5(3) was concerned with the effects on the containment of the burning and/or detonation of this H_2. If the containment retains its integrity, even a severe accident resulting in a damaged or molten core produces relatively low offsite consequences. Item II.B.5(3) also included the effect of steam explosions. Again, the emphasis here was not in preventing the explosion but, instead, in maintaining containment integrity.

Possible Solution

Most of the work on Item II.B.5(3) was couched in terms of a stronger containment.

PRIORITY DETERMINATION

Item II B.5(3) was, to a large extent, similar to Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." Issue A-48 was somewhat more general in that it included the effects of a H_2 burn or detonation on containment penetrations and on safety systems located within the containment, not just the structural response of the containment. In addition, Issue A-48 included measures for control of the H_2 burn and thus had preventive as well as mitigative aspects. However, even though Issue A-48 was expected to use the results of Item II.B.5(3), Item II.B.5(3) was not integrated into Issue A-48 because: (1) the scope of Issue A-48 was still under discussion; and (2) Item II.B.5(3) included steam explosions as well as H_2 burns.

Frequency/Consequence Estimate

In WASH-1400,[16] the PWR sequences refer to steam explosion-induced containment failures as "α" failures; containment failures induced by an H_2 burn are called "γ" failures. Sequences including these two failure modes can be found in Release Categories PWR-1, PWR-2, and PWR-3. It was assumed that the possible solution would result in a 90% reduction in the probabilities of the sequences involving these two failure modes. The results are tabulated in Table II.B-2 below.

Release Category (F)	$^{\alpha}$ Frequency (per RY)	$^{\gamma}$ Frequency(F) (per RY)	Consequences(R) (man-rem)	0.9FR (man-rem/RY)
TABLE II.B-2				
PWR-1	5.3×10^{-8}	-	4.9×10^{6}	0.23
PWR-2	-	7.0×10^{-7}	4.8×10^{6}	3.00
PWR-3	3.4×10^{-7}	-	5.4×10^{6}	1.70
PWR-7	-3.9×10^{-7}	-7.0×10^{-7}	2.3×10^{3}	- 0.002
			TOTAL:	4.9

The PWR-7 category has a negative contribution because a molten core still gives some release, even if containment failure is prevented. Thus, it was assumed that the events which would have been α or γ failures instead lead to PWR-7 releases.

Over a 40-year plant life, the risk reduction above corresponded to about 200 man-rem/reactor. This was calculated using WASH-1400[16] data for a PWR with a large, dry containment. BWR pressure-suppression containments and PWR ice-condenser containments have a much smaller free volume and thus are more susceptible to α and γ failures. Therefore, the risk for these plants could well be considerably higher.

Cost Estimate

Industry Cost: Without the results of research at the time of this evaluation, it was difficult to assess costs. A stronger containment could cost $15M, based on doubling the 3.5 foot wall thickness of a (150 ft x 200 ft) structure. (Such structures cost roughly $1,000/cubic yard of concrete.)

NRC Cost: NRC costs were considered to be negligible.

Total Cost: The total industry and NRC cost associated with the possible solution was $15M/reactor.

Value/Impact Assessment

Based on an estimated public risk reduction of 200 man-rem/reactor and a cost of $15M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{200\text{man} - \text{rem}/\text{reactor}}{\$15\text{M}/\text{reactor}} = 13\text{man} - \text{rem}/\$\text{M}$$

CONCLUSION

The public risk estimate for this issue was significant even for dry containments. Because of the difficulty in determining a cost-effective solution, the issue was given a medium priority ranking (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[138l] as part of SARP Issue L3, "Hydrogen Transport and Combustion," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.6: RISK REDUCTION FOR OPERATING REACTORS AT SITES WITH HIGH POPULATION DENSITIES

DESCRIPTION

Historical Background

This TMI Action Plan[48] item involved the review of operating reactors in areas of high population density to determine what additional measures and/or design changes could be implemented that would further reduce the probability of a severe reactor accident, and would reduce the consequences of such an accident by reducing the amount of radioactive releases and/or by delaying any radioactive releases, and thereby provide additional time for evacuation near the sites.

Risk studies were completed in 1981 for the Zion and Limerick sites and in 1982 for Indian Point. Although risk assessments of other sites were conducted by other NRC programs, e.g., National Reliability Evaluation Program (NREP), no further risk studies were envisioned as part of this issue. Further efforts directed towards this issue were review of the analyses and the possible implementation of site-specific fixes to reduce the risk at these sites. Special hearings were scheduled in FY 1982 to review possible design changes for Indian Point and follow-up work in connection with the accepted fixes was anticipated following these hearings.

Safety Significance

Concern existed over the potential for above-average societal risk due to accidents at reactor sites located near regions of high population densities.

Possible Solutions

As mentioned above, hearings were scheduled on possible fixes at the Indian Point site to reduce risk. The actual fixes that resulted from these hearings were unknown at the time of this evaluation. Nevertheless, it was assumed that fixes would be made to reduce the likelihood of the most dominant accident sequences contributing to the frequency of core-melt accidents.

PRIORITY DETERMINATION

Assumptions

Based on a review of similar Reactor Safety Study Methodology Application Program (RSSMAP) and Interim Reliability Evaluation Program (IREP) analyses, it was assumed that two sequences contributed to a large portion (50%) of the likelihood of a core-melt. It was further assumed that it was possible to reduce the frequency of each sequence by a factor of 10.

Frequency Estimate

Based on age and other related factors, it was believed that reactors in this category had an increased frequency of core-melt over the baseline plant (Oconee-3) by a factor of 5.5. Thus, the revised baseline core-melt frequency (F) was given by:

$F = (5.5)(8.2 \times 10^{-5}/RY) = 4.5 \times 10^{-4}/RY$

Assuming that the dominant sequences (50% of the frequency) could be reduced by a factor of 10, the revised core-melt frequency was $(0.55)(4.5 \times 10^{-4})/RY = 2.5 \times 10^{-4}/RY$.

Consequence Estimate

Considering the same factors used above to estimate the core-melt frequency, the affected plants would have an exposure increase over the mean population density (340 persons/square-mile) and release fractions by a factor of 3. Thus, this exposure increase (R) was given by:

$R = (3)(2.5 \times 10^{6}$ man-rem$) = (7.5 \times 10^{6})$ man-rem

The baseline public risk was $(4.5 \times 10^{-4}/RY)(7.5 \times 10^{6}$ man-rem) or 3,380 manrem/RY. The revised public risk was $(2.5 \times 10^{-4}/RY)(7.5 \times 10^{6}$ man-rem) or 1,880 man-rem/RY. The resulting change in public risk was then 1,500 man-rem/RY resulting from the reduction in core-melt frequency of $2 \times 10^{-4}/RY$. Over the estimated 27 years of remaining plant life, this would result in a total risk reduction of 40,500 man-rem/reactor.

Cost Estimate

Industry Cost: Licensee costs were estimated to be $4M/reactor to implement the changes required to reduce the two dominant sequences.

NRC Cost: NRC costs were estimated to be $22,000.

Total Cost: The total industry and NRC cost associated with the possible solution was $(4 + 0.02)M/reactor or $4.02M/reactor.

Value/Impact Assessment

Based on an estimated public risk reduction of 40,500 man-rem/reactor and a cost of $4.02M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{40{,}500\text{man} - \text{rem / reactor}}{\$4.02\text{M / reactor}}$$

$$= 10{,}000\text{man} - \text{rem / \$M}$$

Other Considerations

The accident avoidance cost was estimated to be approximately $11M which would result in a potential cost saving of $7M, considering the $4M implementation costs.

CONCLUSION

Based on the above value/impact score, this issue was given a high priority ranking (see Appendix C). A staff review of PRAs submitted by the affected licensees was used to identify the strengths and weaknesses of the various plants and to assess the risk associated with their operation. A special adjudicatory proceeding was held from 1982 to 1983 during which time the issues regarding continued operation and risk of the Indian Point plants were heard. Following these hearings, the Commission concluded that neither shutdown of Indian Point Units 2 or 3 nor imposition of additional remedial actions beyond those already implemented by the licensees were warranted.[806]

The staff also reviewed the Zion PRA and concluded that the risk posed by the Zion plants was small. The dominant contributors to severe accidents at the Zion plants were examined and the staff recommended that: (1) the integrity of the two motor-operated gate valves in the RHR suction line from the RCS be checked each refueling outage; and (2) the diesel-driven containment spray pump be modified so that it could be capable of operating without AC power.[806] Thus, this item was RESOLVED and new requirements were established. DL/NRR was responsible for managing the implementation of the above recommendations.[806]

ITEM II.B.7: ANALYSIS OF HYDROGEN CONTROL

DESCRIPTION

The TMI-2 accident resulted in a metal-water reaction which involved H_2 generation in excess of the amounts specified in 10 CFR 50.44. As a result, it became apparent to the NRC that additional H_2 control and mitigation measures would have to be considered for all nuclear power plants. The purpose of this TMI Action Plan[48] item was to establish the technical basis for the interim H_2 control measures on small containment structures and to establish the basis for continued operation and licensing of plants, pending long-term resolution of the H_2 control issue

CONCLUSION

The long-term resolution of this issue was accomplished by rulemaking as part of Item II.B.8. A final rule was published on December 2, 1981 requiring inerting of the small BWR MARK I and II containments. In addition, based on Commission guidance, interim H_2 control systems were required as a licensing condition for the intermediate volume Ice Condenser and MARK III containments. A proposed rule was published on December 23, 1981 (Federal Register 46 FR 62281) which required these systems for the intermediate volume containments. Except for pending construction permit (CP) and manufacturing license (ML) applications, no additional requirements for H_2 control or H_2 analyses were imposed at that time for large, dry containments. However, the proposed rule required that dry containments be analyzed to determine their ability to accommodate the release of large quantities of H_2 (75% metal-water reaction). Also, H_2 control requirements were established as part of the final Near-Term CP and ML Rule published on January 15, 1982.

Based on the accomplishments above, the basis for continued operation and licensing of plants with respect to the H_2 control issue was established. Future work related to finalizing the proposed rule dealing with intermediate volume containments (Ice Condenser and MARK III) and large, dry containments continued as part of Item II.B.8.

ITEM II.B.8: RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

DESCRIPTION

Historical Background

In the past, safety reviews concentrated on how to prevent a core from being damaged. Consequently, little attention was given to how a severely damaged core could be dealt with after damage occurred. Other subtasks within Task II.B were concerned with the study of the characteristics of degraded and melted cores (research programs) plus some immediate actions to be taken at plants in operation. Item II.B.8 envisioned both a short-term and a long-term rulemaking to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis.

Item II.B.8 included an Advance Notice of Proposed Rulemaking (ANPRM) and an Interim Rule. The ANPRM was issued on December 2, 1980 (45 FR 65474) and the Interim Rule was issued in two parts: the first was issued in effective form in October 1981 (46 FR 58484) and the second was issued as a proposed rule on December 23, 1981 (46 FR 62281).

On January 4, 1982, SECY-82-1[309] was forwarded to the Commission requesting reconsideration of the approach to long-term rulemaking. The events which prompted this request were as follows:

-	The Commission had required more protection from severe accidents in some licensing actions (e.g., Sequoyah) than was envisioned in the TMI Action Plan.[48]
-	A rule was developed to specify additional requirements for pending CP and ML applications. Again, these requirements were somewhat more extensive than that envisioned in the TMI Action Plan.[48]
-	New probabilistic risk assessments (PRAs) indicated lower risk than was previously estimated for large, dry PWR containments.
-	The safety of existing plants had been considerably improved by the modifications mandated by NUREG-0737.[98]
-	The industry initiated a program to study the costs and benefits of design features for mitigating severe accidents.
-	An extensive research program to study damaged and melted core behavior was underway.
-	A safety goal statement, based on PRA, was developed.

The substance of SECY-82-1[309] was that the uncertainty associated with long-term rulemaking was an inhibiting force on the industry. The paper then recommended that, since new applications were to be standardized, licensing could proceed on these standardized designs using the information available. PRAs and the safety goal would be used to assess plant safety. If plants needed safety features beyond the existing requirements to meet the safety goal, they could be included. This approach would not need rulemaking specifically directed at severe accident mitigation.

The Commission directed[310] the staff to make several changes recommended in SECY-82-1.[309] The staff then submitted revised papers SECY-82-1A[311] and SECY-82-1B[1405] that incorporated the changes directed by the Commission, including ACRS input. The evaluation of this item included consideration of Item II.B.7.

Safety Significance

Most of the engineered safety features at nuclear power plants of the existing generation were intended to prevent severe core damage. Relatively little attention was given in the past to dealing with a severely damaged or melted core. Once a core is damaged, the containment will still prevent the release of large amounts of radioactive material. However, once the core melts, the containment is likely to fail (although the hazard to the public varies widely, depending on the way in which the containment fails).

The degraded-core accident rulemaking was intended to require means for dealing with a damaged core. This translated into preventing the release of radioactivity and providing means for recovering from the accident. Specific items to be considered included the following: use of filtered, vented containment; H_2 control measures; core retention devices ("core catchers"); reexamination of design criteria for decay heat removal and other systems; post-accident recovery plans; criteria for locating highly radioactive systems; effects or accidents at multi-unit sites; and comprehensive review and evaluation of related guides and regulations.

PRIORITY DETERMINATION

The safety significance of this issue was essentially the same as that of the research programs described in the analyses of Items II.B.5(1) and II.B.5(2) above. Examination of the estimated frequency of core damage and/or core-melt, coupled with estimates of the potential effectiveness of engineering solutions (and their cost) led to the recommended high priority for Items II.B.5(1) and II.B.5(2). In the same manner, Item II.B.8 had the potential for a significant (and cost-effective) reduction in public risk. In addition, it should be noted that some of the plant modifications contemplated were far more expensive to backfit than to forward-fit. Unnecessary delay could have reduced the costeffectiveness of the resolution to this issue.

CONCLUSION

Based on the above evaluation, this item was given a high priority ranking (see Appendix C). Work performed by RES on the H_2 control aspect of the issue resulted in a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985.[807] The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070.[809] Thus, with the issuance of the rule on H_2 control, this item was RESOLVED and new requirements were established.[808]

REFERENCES

0016.	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.
0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0309.	SECY-82-1, "Severe Accident Rulemaking and Related Matters," U.S. Nuclear Regulatory Commission, January 4, 1982. [8201190416]
0310.	Memorandum for W. Dircks from S. Chilk, "Staff Requirements—Briefing on Status and Plan for Severe Accident Rulemaking (SECY-82-1)," January 29, 1982. [8202160202]
0311.	SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," U.S. Nuclear Regulatory Commission, July 16, 1982. [8208040432]
0312.	NUREG/CR#0165, "A Value-Impact Assessment of Alternate Containment topics," U.S. Nuclear Regulatory Commission, June 1978.
0806.	Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan, Task II.B.6, 'Risk Reduction for Operating Reactors at Sites With High Population Densities,'" September 25, 1985. [8510030342]
0807.	Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.B.8 `Rulemaking Proceeding on Degraded Core Accidents—Hydrogen Control,'" July 19, 1985. [8508010066]

0808.	Memorandum for W. Dircks from H. Denton, "Close Out of TMI Action Plan, Task II.B.8," August 12, 1985. [8508210316]
0809.	NUREG-1070, "NRC Policy on Future Reactor Designs," U.S. Nuclear Regulatory Commission, July 1985.
1102.	Memorandum for T. Speis from R. Houston, "Integration of Generic Issue Resolution," November 4, 1987. [9704150161]
1381.	Memorandum for W. Minners from B. Sheron, "Update of Generic Issue Management Control System (GIMCS)," July 5, 1991. [9312220300]
1382.	NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Rev. 1) December 1992.
1405.	SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," U.S. Nuclear Regulatory Commission, November 24, 1982. [8301120513]
1858.	Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.
1967.	SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [*ML111590814*]

Task II.C: Reliability Engineering and Risk Assessment (Rev. 3) ( )

The objective of this task was to develop and implement improved systems-oriented approaches to safety review. The NRC was to employ risk assessment methods to identify particularly high-risk accident sequences at individual plants and determine regulatory initiatives to reduce these high-risk sequences.

ITEM II.C.1: INTERIM RELIABILITY EVALUATION PROGRAM DESCRIPTION

Historical Background

The Interim Reliability Evaluation Program (IREP) was a planned multiplant reliability evaluation program to develop and standardize the reliability methodology involved in performing reliability and safety studies. This program was conceived in NUREG-0660[48] as a pilot study with a scaled-up study of an additional 6 plants.

At the time this issue was evaluated, the pilot study had been completed on the Crystal River plant and the results reported in NUREG/CR-2515.[365] Scaled-up analyses had been completed on 4 other plants and the results reported on 2 of these: ANO-1 (NUREG/CR-2787)[366] and Browns Ferry-1 (NUREG/CR-2802).[367] Remaining to be studied was one additional plant, probably a Mark II BWR plant, the analysis of which was to include other common cause initiators, e.g., fires, seismic events, and floods, that were considered in the other IREP analyses. Thus, this issue addressed the completion of a shortened 5-plant version of the IREP program.

Safety Significance

Based on published PRA studies of nuclear power plants, approximately one-third had predicted core-melt frequencies exceeding 10^{-4}/RY.

Possible Solution

The solution was to complete the planned analysis and report on the remaining plant.

PRIORITY DETERMINATION

Assumptions

It was assumed that those plants with core-melt frequencies exceeding 10^{-4}/RY had an average core-melt frequency of 3×10^{-4}/RY which could be reduced to 10^{-4}/RY.

Frequency Estimate

As stated in the assumptions above, it was assumed that there was one chance in three that the reactor to be analyzed would have a predicted core-melt frequency of 3×10^{-4}/RY and that this frequency would be reduced to 10^{-4}/RY, a frequency reduction of 2×10^{-4}/RY or a probable core-melt frequency reduction of 6.7×10^{-5} /RY.

Consequence Estimate

The total whole body man-rem dose was obtained using the CRAC Code[64] for the release fractions and categories of a BWR as given in WASH-1400.[16] The calculations assumed an average population density of 340 persons/square-mile (which was the average for U.S. domestic sites) from an exclusion area of one-half mile about the reactor to a 50-mile radius. A typical midwest plain meteorology was also assumed. It was further assumed that the reduction in public dose was in proportion to the reduction in accident frequency.

Assuming an average public risk exposure of 6.8×10^{6} man-rem/core-melt and an average remaining life of 27 years for BWRs, the reduction in core-melt frequency of 6.7×10^{-5}/RY resulted in a reduction in public risk of 455 man-rem/RY and a total public risk reduction of 12,150 man-rem for all affected plants.

Cost Estimate

Industry Cost: The contract cost for performing the analyses involved with the prior IREP-assessed plants averaged $900,000/plant. Since the staff could not predict what could be identified by the analysis as candidate modifications to reduce risk, the plant change cost could not be estimated. However, based upon a risk reduction of 12,000 man-rem, it was cost-effective for plants to spend up to $12M for this reduction in risk.

NRC Cost: Review of the analysis and preparation of findings were estimated to cost $200,000 plus 0.7 staff-year, or $270,000. As in the case with the initial IREP plant analysis, it was assumed that the analysis cost would be borne by the NRC. This resulted in a total NRC cost of $1.2M.

Total Cost: The total industry and NRC cost associated with the solution to this issue was $(12 + 1.2)M or $13.2M.

Value/Impact Assessment

Based on an estimated public risk reduction of 12,000 man-rem and a cost of $13.2M for a possible solution, the value/impact score was given by:

$$S = \frac{12{,}000 \text{man} - \text{rem}}{\$13.2\text{M}}$$

$$= 910 \text{man} - \text{rem} / \$\text{M}$$

Other Considerations

The findings from this analysis may have helped to identify generic safety issues for other reactors in the same class. An additional purpose of this evaluation was to demonstrate the suitability of newly developed methodology for the inclusion of external initiating events into PRA calculations. However, no credit for this benefit was considered or factored into the value/impact assessment.

CONCLUSION

Based on the value/impact score, this issue would have received a medium priority ranking. However, given the potential public risk reduction, it was given a high priority ranking (see Appendix C). Work completed by the staff in resolving the issue resulted in the publication of the following reports for the two remaining plants: NUREG/CR-3085[810] and NUREG/CR-3511[811] for Millstone-1 and Calvert Cliffs-1, respectively. A primary output of the IREP was NUREG/CR-2728[812] which was a guide that documented methods, codes, and data used in the IREP. This guide was intended to provide guidance for PRAs performed subsequent to the IREP. Thus, this item was RESOLVED with no new requirements.[813]

ITEM II.C.2: CONTINUATION OF INTERIM RELIABILITY EVALUATION PROGRAM

DESCRIPTION

Historical Background

IREP was a planned multiplant reliability evaluation to develop and standardize the reliability methodology involved in performing reliability and safety studies. It was conceived in NUREG-0660[48] that a National Reliability Evaluation Program (NREP) study, performed by licensees, should follow the IREP effort. This issue addressed the continuation of the IREP program to cover all the remaining operating reactors that were not covered in the initial IREP studies, to be performed either by the NRC or by licensees. Also, consideration was to be given to the inclusion of plants under design or construction.

Possible Solutions

Possible solutions ranged from the NRC sponsorship of an analysis of all plants, having the individual licensees perform an analysis on all or some plants, or reducing the effort to a limited study. The plan selected for this analysis consisted of three parts: (1) performance of an NREP by the licensees on the 4 plants without a risk/reliability analysis; (2) a careful review by the NRC of 7 other plants that had an existing PRA; and (3) an appraisal of the interim results of these reviews a year after implementation to consider the advisability of future extension of the NREP program to other plants. These 11 plants would be the same ones chosen for the first group of SEP Phase III plants.

PRIORITY DETERMINATION

Assumptions

At the time of this evaluation, there were 14 published PRA studies and the core-melt frequencies were predicted to be higher than 10^{-4}/RY in about one-third of these studies. Thus, it could be assumed that, of the 11 plants to be studied, about 4 might have some hardware or procedural fixes implemented to reduce the likelihood of the most dominant accident sequences with respect to core-melt. In addition, there was the potential that these analyses would result in generic resolutions of identified safety issues which could reduce risk at other plants without the expense of plant-specific PRAs being performed at these plants; but this assumption remained to be proven. Calculations were based partly on an analysis[64] of the issue by PNL.

Frequency Estimate

It was not unrealistic to postulate that 4 of the 11 reactors had an average core-melt frequency of 3 x 10^{-4}/RY and that changes were possible to reduce the core-melt frequency to 10^{-4}/RY. Therefore, a reduction in core-melt frequency of 2 x 10^{-4}/RY was postulated for these 4 plants (3 PWRs and 1 BWR).

Consequence Estimate

Assuming an average public exposure of 2.5 x 10^6 man-rem and 6.8 x 10^6 man-rem following a core-melt at a PWR and a BWR, respectively, the reduction in core-melt frequency resulted in a reduction in public risk of about 42,700 man-rem for the remaining life of the 3 PWRs and 36,700 man-rem for the remaining life of the BWR. This resulted in a total reduction in public risk of approximately 79,000 man-rem.

Cost Estimate

Industry Cost: Based on previous experience, the cost for each plant was expected to be between $1.5M to $2M to perform the NREP analysis (limited to analysis of core-melt from internal accident initiators), including a state-of-the-art systems interaction study of appropriate scope and depth. Using the upper bound licensee cost, it was assumed that licensee costs were $2M/reactor and, of this amount, $500,000 was the additional cost of performing the systems interaction in conjunction with the NREP. Thus, for the 4 plants to be analyzed, the cost for the NREP analysis would be $6M. For an effective cost-benefit ratio (based on a 79,000 man-rem risk reduction), the licensee backfit cost could be as high as $73M. Thus, the total industry cost was $(6 + 73) or $79M.

NRC Cost: The NRC cost was estimated to be $200,000 and 0.7 man-year/reactor. For the 11 reactors, this cost was $3.8M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(79 + 3.8)M or $82.8M.

Value/Impact Assessment

Based on an estimated public risk reduction of 79,000 man-rem and a cost of $82.8M for a possible solution, the value/impact score was given by:

$$S = \frac{79{,}000\text{man} - \text{rem}}{\$82.8\text{M}}$$

$$= 950\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

The value/impact score was strongly influenced by the uncertainty of the cost figures for licensees. Considerable risk reduction had been achieved by procedural changes that could be developed and implemented at much less cost than equipment changes. Therefore, the cost of licensee implementation could have been considerably less than the cost used in this assessment.

CONCLUSION

Although the value/impact score would only warrant a medium priority ranking, the large potential risk reduction (brought about by the reduction in core-melt frequency for those plants that were above 10^{-4}/RY) indicated a high priority ranking (see Appendix C).

Work completed by the staff in resolving the issue was closely related to the accomplishments under Item IV.E.5. Whereas Item II.C.2 called for the initiation of IREP studies (i.e., plant-specific PRAs) on all remaining operating reactors, Item IV.E.5 called for the development of a plan for the systematic assessment of the safety of all operating reactors. The Integrated Safety Assessment Program (ISAP), presented in SECY-84-133[814] and SECY-85-160,[815] provided for a comprehensive review of selected operating reactors to address all pertinent safety issues and to provide an integrated cost-effective implementation plan for making needed changes. Under ISAP, each plant would be subject to an integrated assessment of safety topics, a probabilistic safety assessment, and an evaluation of operating experience.

NRC guidance, as described in the Severe Accident Policy Statement (see Item II.B.8), stated that OLs were expected to perform plant-specific PRAs to find instances of particular vulnerability to a core-melt or poor containment performance, given a core-melt. Thus, this item was RESOLVED and no new requirements were established.[816]

ITEM II.C.3: SYSTEMS INTERACTION

DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering and scientific disciplines such as civil, electrical, mechanical, structural, chemical, hydraulic, nuclear, geological, seismological, and meteorological. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements are met. These reviews include failure mode analyses to assure that the single failure criterion is met.

The design and analyses by the plant designers and the subsequent review and evaluation by the NRC, take into consideration some interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, based upon operating experience, there was some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems. The objective of a systems interaction analysis was to provide assurance that the independent functioning of safety systems was not jeopardized by preconditions that cause faults to be dependent.

Concern over systems interactions was first documented explicitly by the ACRS in November 1974 when it was requested that the staff give "attention to the evaluation of ... potentially undesirable interactions between systems" from a multidisciplinary point of view. In October 1978, NUREG-0371[2] was published and included Issue A-17, "Systems Interactions in Nuclear Power Plants." In May 1980, NUREG-0660[48] provided for broadening the staff efforts in Item II.C.3. Efforts for the resolution of Item II.C.3 were included in activities for the resolution of Issue A-17.

CONCLUSION

This issue was not considered a separate issue since the safety concern was covered in Issue A-17.

ITEM II.C.4: RELIABILITY ENGINEERING

DESCRIPTION

Historical Background

At the time this TMI Action Plan[48] issue was evaluated, there was no requirement for licensees to develop and implement a reliability assurance program. In the absence of such a requirement, it was difficult to determine the nature and extent that was being exercised by licensees to implement a reliability assurance program.

Safety Significance

Typically, reliability assurance programs determine system availabilities, identify high component failure rates, determine basic causes for component failures, identify possible corrective actions, and perform other similar activities in what was generally called reliability engineering.

Possible Solution

A possible solution was to develop a Regulatory Guide that would define the elements and functions necessary for an applicant to plan and establish an acceptable reliability program. Applicants would further be required to implement the operation of a reliability program as a part of the requirements to obtain a CP or OL. The functioning of the reliability program would be inspected as a part of the ongoing inspection program.

PRIORITY DETERMINATION

Assumptions

Issues of this nature are difficult to quantify since the results are highly speculative depending upon such hard to quantify variables as management acceptance and backing. The approach used to estimate the effectiveness of this issue was to determine what might be a reasonable objective and evaluate the contribution to risk reduction that could be achieved and at what cost.

The defined objective for this evaluation was to maintain the reduction in core-melt frequency that was achievable by the NREP program. From previously published PRAs and IREP analyses, about one third of the plants had forecast accident frequencies that exceeded 10^{-4}/RY. It was assumed that, without a dedicated effort, the accident frequency for these plants would rise to 2 x 10^{-4}/RY at the end of their life. At a constant rate of increase in accident frequency over the remaining plant life, the average increase would be 5 x 10^{-5} /RY. Release fractions were based on the Oconee-3 and Grand Gulf-1 plants. Calculations used below were based partly on an analysis[64] of the issue by PNL.

Frequency Estimate

The reduction in core-melt frequency for 33% of the reactors was 5 x 10^{-5}/RY as previously described.

Consequence Estimate

The core-melt frequency reduction resulted in a risk reduction of 128.5 man-rem/RY for PWRs and 338 man-rem/RY for BWRs. Based upon 33% of all plants, 31 PWRs and 16 BWRs with average remaining lives of 28.5 years and 27 years, respecttively, the risk reduction was estimated to be 120,900 man-rem for PWRs and 146,200 man-rem for BWRs. Thus, the total risk reduction was 267,100 man-rem.

Cost Estimate

Industry Cost: The cost/plant, based on the estimates in NUREG-0660,[48] were 10 man-years to establish a program and 1 man-year/RY for operation for the remaining life of each plant. These costs amounted to $143M for implementation and $400M for operation. Thus, the total industry cost was estimated to be $543M.

NRC Cost: Implementation was estimated to require 3 man-years at a cost of $300,000. The cost for operation was estimated to be 2 man-weeks/RY or $15.4M for the remaining life of all the reactors. Thus, the total NRC cost was estimated to be $15.7M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(543 + 15.7)M or $558.7M.

Value/Impact Assessment

Based on an estimated public risk reduction of 267,100 man-rem and a cost of $558.7M for a possible solution, the value/impact score was given by:

$$S = \frac{267{,}100\text{man} - \text{rem}}{\$558.7\text{M}}$$

$$= 478\text{man} - \text{rem} / \$\text{M}$$

Other Considerations

One of the factors that drove up licensee costs was the annual cost associated with the maintenance of the program. However, given the cost of replacement power at $300,000/day, one day of increased productivity from increased plant reliability would cover three years of forecast reliability program operating costs. Thus, a reliability program had economic incentives for licensees in addition to the safety incentives.

The risk reduction was calculated only for those plants that were predicted to have a core-melt frequency exceeding 10^{-4}/RY. An additional reduction in risk would also be realized by maintaining the core-melt frequency at the calculated value on those plants that had a core-melt frequency less than 10^{-4}/RY.

CONCLUSION

Based solely on the value/impact score, this issue would have been assigned a medium priority ranking. However, it was given a high priority ranking (see Appendix C) based on the potential substantial change in core-melt/RY frequency and the large cost incentive that could be realized by licensees through increased availability.

The technical issue at the time the TMI Action Plan[48] was published was that the essential elements and process of a reliability program applicable to operational safety had not yet been identified. Although NRC requirements, such as Appendices A and B of 10 CFR 50, strongly reflected reliability principles (i.e., safety margins, redundancy, diversity, and corrective action), these principles had been applied to nuclear power plants primarily in the design phase and not in the operations phase. Reliability engineering practices at nuclear power plants had not yet resulted in strategies to help achieve and maintain the `designed-in' capability for reliable operation during the operating life of the plants.

The concept of an operational reliability program was based on a simple closed-loop strategy: monitoring and evaluating plant performance; identifying and prioritizing potential problems; diagnosing the causes; taking appropriate corrective actions; and verifying the effectiveness of these actions. The elements of a reliability program were summarized by the staff in RIL 158.[1130] These elements were included among NRC initiatives to improve maintenance and better manage the effects of aging, to improve TS, and to develop and use plant performance indicators. Also, an operational reliability program that was an acceptable means of meeting the Station Blackout Rule (10 CFR 50.63) was to be described in Revision 3 to Regulatory Guide 1.9 as part of the resolution of Issue B-56, "Diesel Generator Reliability."

Based on the above findings, the staff concluded that the safety concern of this issue was addressed in other NRC programs and the issue was considered RESOLVED with no new requirements.[1131]

REFERENCES

0002.	NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
0016.	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.
0048.	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.
0064.	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.
0365.	NUREG/CR#2515, "Crystal River 3 Safety Study," U.S. Nuclear Regulatory Commission, December 1981.
0366.	NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One—Unit One Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1982.
0367.	NUREG/CR#2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," U.S. Nuclear Regulatory Commission, August 1982, (Appendix A) August 1982, (Appendix B) August 1982, (Appendix C) August 1982.

0810.	NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983, (Vol. 3) July 1983, (Vol. 4) July 1983.
0811.	NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1984, (Vol. 2) October 1984.
0812.	NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," U.S. Nuclear Regulatory Commission, March 1983.
0813.	Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan, Task II.C.1, 'Interim Reliability Evaluation Program,'" July 9, 1985. [8507180593]
0814.	SECY-84-133, "Integrated Safety Assessment Program (ISAP)," U.S. Nuclear Regulatory Commission, March 23, 1984. [8404100072]
0815.	SECY-85-160, "Integrated Safety Assessment Program—Implementation Plan," U.S. Nuclear Regulatory Commission, May 6, 1985. [8505230571]
0816.	Memorandum for W. Dircks from H. Denton, "Close-out of Generic Issues II.C.2, 'Continuation of IREP' and IV.E.5, 'Assess Currently Operating Reactors,'" September 25, 1985. [9909290069]
1130.	RIL 158, "Operational Safety Reliability Program," U.S. Nuclear Regulatory Commission, October 31, 1988. [8811070111]
1131.	Memorandum for V. Stello from E. Beckjord, "Closure of Generic Issue II.C.4, `Reliability Engineering,'" October 31, 1988. [8811150124]

Task II.D: Reactor Coolant System Relief and Safety Valves (Rev. 3) ( )

The objective of this task was to demonstrate by testing and analysis that the relief and safety valves, block valves, and associated piping in the reactor coolant system (RCS) were qualified for the full range of operating and accident conditions; anticipated transients without scram (ATWS) could be considered in later phases of the testing. In addition, design changes or modifications that were necessary to provide positive indication of valve position were to be made.

ITEM II.D.1: TESTING REQUIREMENTS

DESCRIPTION

This TMI Action Plan[48] item called for applicants and licensees to conduct testing to qualify reactor coolant relief valves, safety valves, block valves, and associated discharge piping for all operating conditions and design basis accidents.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-14 was established by DL/NRR for implementation purposes.

ITEM II.D.2: RESEARCH ON RELIEF AND SAFETY VALVE TEST REQUIREMENTS

DESCRIPTION

Historical Background

This TMI Action Plan[48] item specified that RES contract with INEL to: (1) act as a systems integrator to technically monitor and analyze the planned industry valve test and analytical program at EPRI and collect, analyze, and compare information from foreign tests; (2) develop, improve, or verify available flow discharge and structural response models using the above information; (3) determine the need for a valve testing program by NRC, with the main focus to be on subcooled and two-phase discharge and on determining operability; and (4) conduct additional tests, as necessary, to ensure that the response to the full spectrum of fluid conditions that would be expected to result from anticipated operational occurrences and ATWS events had been adequately characterized. The above work, with the exception of the ATWS events, had been performed in conjunction with Item II.D.1 which was clarified in NUREG-0737.[98]

Safety Significance

The remaining concern under Item II.D.2 with respect to ATWS events was the capability to depressurize the reactor. Coupled with failure of the reactor protection system (RPS) following a transient, inadequate depressurization could result in rupture of the reactor coolant pressure boundary (RCPB) producing a loss-of-coolant accident (LOCA).

Possible Solution

To estimate the public risk associated with ATWS events, it was assumed[64] that a possible solution would be to increase the sizing of the relief and safety valves. This modification was assumed to decrease the likelihood of an ATWS-induced rupture of the RCPB by enhancing the depressurization capability of the system.

PRIORITY DETERMINATION

Assumptions

Using Oconee-3 as representative of PWRs, PNL assumed[64] that the dominant core-melt sequence representative of an ATWS event would involve a Power Conversion System (PCS) transient caused by events other than a loss-of-offsite power (LOOP) and failure of the RPS. The LOCA initiator was assumed to be a RCPB pipe rupture with an equivalent 4-inch diameter. Equipment failures included the containment spray recirculation system and emergency coolant injection and recirculation systems. The containment failure modes were assumed to be similar to other PWR Release Categories involving RCPB ruptures.

The Grand Gulf-1 reactor was assumed to be representative of BWRs. The dominant core-melt sequence used to model the ATWS event involved transients other than LOOP which require shutdown and a failure to achieve subcriticality. The LOCA initiator was assumed to be a RCPB rupture equivalent to an area of 1 ft^2. The equipment failure assumed was loss of the residual heat removal (RHR) system after the LOCA. The containment failure modes were similar to other BWR Release Categories involving a LOCA and subsequent loss of RHR.

Frequency Estimate

Based on the above assumptions, the reductions in core-melt frequency as a result of modifying the safety relief valves (SRVs) were calculated to be 3.8×10^{-7}/RY for PWRs and 7.1×10^{-9}/RY for BWRs.

Consequence Estimate

The reduction in public risk was calculated[64] to be 0.99 man-rem/RY for PWRs and 0.51 man-rem/RY for BWRs. Assuming at least one-half of the plants were affected (45 PWRs and 22 BWRs), with an average remaining life of 28.7 years for PWRs and 27.4 years for BWRs, the total public risk reduction was 1,300 man-rem.

Cost Estimate

Industry Cost: SRV modifications were assumed to require approximately 125 man-weeks/plant. At a rate of $2,270/man-week, the labor cost for this modification was estimated to be $284,000/plant. Equipment was estimated to be $100,000/plant. For backfit plants, the License Amendment Fee was $4,000. These costs resulted in a backfit cost of $388,000/plant and a forward-fit cost of $384,000/plant. For the forward-fit plants, it was assumed that only half of the plants scheduled to begin operation prior to 1986 would require modifications and, subsequent to that time, the modifications would be incorporated during initial installation. Based on these estimates, the total industry cost was $21M.

NRC Cost: Development and implementation costs were estimated to be $0.4M and $0.3M, respectively. The development cost was assumed to require 2 man-years of NRC effort and 2 man-years of contractor support. The implementation cost to monitor the hardware modifications at the affected plants was assumed to require 2 man-weeks/plant (36 backfit plants, 19 forward-fit plants). Based on these estimates, the total NRC cost was $0.7M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(21 + 0.7)M or $21.7M.

Value/Impact Assessment

Based on an estimated public risk reduction of 1,300 man-rem and a cost of $21.7M for a possible solution, the value/impact score was given by:

$$S = \frac{1{,}300 \text{ man} - \text{rem}}{\$21.7\text{M}}$$

$$= 60 \text{ man} - \text{rem} / \$\text{M}$$

CONCLUSION

With the exception of potential ATWS events, Item II.D.2 was integrated into Item II.D.1. Based on the above calculation, the part of Item II.D.2 that involved consideration of ATWS events was given a low priority ranking (see Appendix C) in November 1983. In NUREG/CR-5382,[1563] it was concluded that consideration of a 20-year license renewal period could change the ranking of the issue to medium priority. Further prioritization, using the conversion factor of $2,000/man-rem approved[1689] by the Commission in September 1995, resulted in an impact/value ratio (R) of $16,666/man-rem, which placed the issue in the DROP category. Consideration of new information[1715] on the phenomenon of "microbonding," submitted by Region IV in April 1997, did not change this conclusion.[1716]

ITEM II.D.3: RELIEF AND SAFETY VALVE POSITION INDICATION

DESCRIPTION

This TMI Action Plan[48] item called for all OLs and applicants for OLs to provide the RCS relief and safety valves with position indication in the control room.

CONCLUSION

This item was clarified in NUREG-0737[98] and requirements were issued.

REFERENCES

0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
1563.	NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.
1689.	Memorandum for J. Taylor from J. Hoyle, "COMSECY#95#033"Proposed Dollar per Person-Rem Conversion Factor; Response to SRM Concerning Issuance of Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission and SRM Concerning the Need for a Backfit Rule for Materials Licensees (RES-950225) (WITS-9100294)," September 18, 1995. [9803260148]
1715.	Memorandum for D. Morrison from T. Gwynn, "Periodic Review of Low-Priority Generic Safety Issues," April 16, 1997. [9909290132]
1716.	Memorandum for T. Gwynn from T. Martin, "Periodic Review of Low-Priority Generic Safety Issues," July 13, 1998. [9909290134]

Task II.E: System Design (Rev. 2) ( )

TASK II.E.1: AUXILIARY FEEDWATER SYSTEM

The objective of this task was to improve the reliability of the auxiliary feedwater (AFW) system.

ITEM II.E.1.1: AUXILIARY FEEDWATER SYSTEM EVALUATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-15 was established by DL/NRR for implementation purposes.

ITEM II.E.1.2: AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPAs F-16 and F-17 were established by DL/NRR for implementation purposes.

ITEM II.E.1.3: UPDATE STANDARD REVIEW PLAN AND DEVELOP REGULATORY GUIDE

DESCRIPTION

This TMI Action Plan[48] item called for NRR to update SRP[11] Section 10.4 and for RES was to revise Regulatory Guide 1.26 to cover AFW systems; the Regulatory Guide was to possibly endorse ANSI/ANS-51.00 and 52.00.

CONCLUSION

SRP[11] Section 10.4.9 was updated in July 1981 to include the requirements[98] of Items II.E.1.1 and II.E.1.2. Revision of Regulatory Guide 1.26[233] to endorse the ANSI/ANS Standards was not pursued because it was considered a low priority task by RES. No additional public and occupational risk reduction was identified to support the need for development of a Regulatory Guide. Thus, this item was RESOLVED and new requirements were established with changes to the SRP.[11]

REFERENCES

0011.	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.
0048.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0233.	Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1972, (Rev. 1) September 1974, (Rev. 2) June 1975, (Rev. 3) February 1976.

Task II.E.2: Emergency Core Cooling System (Rev. 3) ( )

The objectives of this task were to: (1) decrease reliance on the emergency core cooling system (ECCS) for events other than LOCAs; (2) ensure that the ECCS design basis reliability and performance were consistent with operational experience; (3) reach a better technical understanding of ECCS performance; and (4) ensure that the uncertainties associated with the prediction of ECCS performance were properly treated in small-break evaluations.

ITEM II.E.2.1: RELIANCE ON ECCS

DESCRIPTION

Historical Background

This TMI Action Plan[48] item called only for the collection of ECCS operating experience. Risk reduction would require that conclusions and recommendations be made and acted upon. Since the stated purpose was to decrease the reliance on ECCS for events other than LOCAs, it was assumed that this item would ultimately lead to the implementation of some hardware modifications.

Safety Significance

The ECCS of PWRs and BWRs was being actuated for events other than LOCAs. Reliance on the ECCS for events other than LOCAs should be evaluated to ensure that: (1) the ECCS design basis reliability and performance were consistent with operational experience; and (2) a better technical understanding of ECCS performance could be reached.

Possible Solution

In accordance with Item II.K.3(17),[98] licensees were requested to submit a report detailing dates and length of all ECCS outages for the previous 5 years of operation, including causes of the outages. This report would provide the staff with a quantification of historical unreliability due to test and maintenance outages, which was to be used to determine if a need existed for cumulative outage requirements in the TS. The requested report was to contain: (1) outage dates and duration of outages; (2) cause of each outage; (3) systems or components involved in each outage; and (4) corrective action taken. Test and maintenance outages were to be included in the above listings covering the 5-year period. The licensees were requested to propose changes to improve the availability of ECCS equipment, if needed.

CONCLUSION

This issue was covered under Item II.K.3(17) which was implemented as part of NUREG-O737.[98] Thirty out of 36 Technical Evaluation Reports (TERs) were expected from Franklin Institute by September 3O, 1982; at the time of this evaluation, 9 had been received. RRAB/DST/NRR was to issue SERs to DL/NRR for the 30 plants by November 15, 1982 and the task was to be closed out by DL/NRR by December 31, 1982. By December 31, 1982, Franklin Institute was expected to issue the remaining 25 TERs, and SERs were to be issued for these plants by RRAB/DST/NRR by February 15, 1983. The final 35 actions were to be closed out by DL/NRR by March 31, 1983.

ITEM II.E.2.2: RESEARCH ON SMALL BREAK LOCAs AND ANOMALOUS TRANSIENTS

DESCRIPTION

Historical Background

This TMI Action Plan[48] item was intended to focus research on small breaks and transients. It included experimental research in the loss-of-fluid test (LOFT) Semiscale, BWR full integral simulation test (FIST), and B&W Integral Systems Test facilities, systems engineering, and materials effects programs, as well as analytical methods development and assessments in the code development program. Most of the experimental work for small-break LOCAs (SBLOCAs) was completed in FY 1982 with data analysis to be conducted in FY 1983. Since October 1982, the LOFT project had been supported by an international consortium, of which NRC was a member.

Safety Significance

The primary goal of the small-break and transient research was to improve operator performance during off-normal events. The research on analytical methods development and assessment was directed toward improving existing computer codes, development and application of advanced computer codes for SBLOCA and other accident analysis, and development of a fast, easy to use, engineering analyzer capability.

Possible Solution

Part of the program was to examine SBLOCAs and anomalous transients; specifically, the ability of typical process instruments to provide accurate and sufficient information to operating personnel. Advanced control room and diagnostic instrumentation was used in LOFT as part of the augmented operator capabilities program to assess operator needs to mitigate the consequences of LOCA and transient sequences.

PRIORITY DETERMINATION

Assumptions

Only reduction in operator error during LOCA and transient sequences was assumed. It was also assumed[64] by PNL that SBLOCAs or transients leading to a LOCA, typically via a stuck-open pressure relief valve, represented the initiating events applicable to this issue. Using Oconee-3 as the representative PWR, these initiators were an S_3 LOCA and T_1, T_2, or T_3 transient coupled with relief valve closure failure (Q). This applied primarily to PWRs; however, the same approach was used for BWRs.

For PWRs, it was assumed that operator errors involved: (1) failure to align suction of high pressure recirculation system to the suction of the low pressure recirculation system; and (2) failure to open both containment sump suction valves in the low pressure containment spray recirculation system at the start of recirculation. For BWRs, it was assumed that the operator failed to manually initiate the automatic depressurization system (ADS). Operator error in such sequences was assumed to be reduced by one-third as a result of a combination of operator training and improved instrumentation.

Frequency Estimate

Based on the above assumptions and using the dominant accident sequences, the reductions in core-melt frequency were calculated[64] to be 5.2 x 10^{-6}/RY for PWRs and 1.8 x 10^{-7}/RY for BWRs.

Consequence Estimate

The reductions in public risk were calculated to be 15 man-rem/RY for PWRs and 0.5 man-rem/RY for BWRs. Assuming 90 PWRs with an average remaining life of 28.8 years and 44 BWRs with an average remaining life of 27.4 years, the total public risk reduction was 41,000 man-rem for all forward-fit and backfit plants.

Cost Estimate

Industry Cost: It was estimated that upgrading operator training and installing upgraded equipment would cost $0.5M/plant. It was assumed that equipment installation was primarily in the control room, with no increase in radiation exposure, and that only backfit plants were involved. Therefore, assuming 47 PWRs and 24 BWRs, the industry cost was estimated to be $36M. This cost was applied to backfit plants only since the changes resulting from this program would presumably be incorporated into the initial design and licensing of the forward-fit plants.

NRC Cost: This item was an ongoing program; therefore, sunk costs had already been taken in FYs 1980, 1981, and 1982. It was estimated that 20% of the FY 1983 LOFT budget was earmarked for the SBLOCA program. This represented approximately $3.1M. In addition, it was assumed that $0.2M would be required to establish new criteria for reactor instrumentation and operator training. NRC annual review was estimated to require an additional 1 man-day/RY. At a rate of $2,270/week and using the remaining plant life assumed above, this cost was about $1.7M. Therefore, the total NRC cost was estimated to be approximately $5M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(36 + 5)M or $41M.

Value/Impact Assessment

Based on an estimated public risk reduction of 41,000 man-rem and a cost of $41M for a possible solution, the value/impact score was given by:

$$S = \frac{41{,}000\text{man} - \text{rem}}{\$41\text{M}}$$

$$= 1{,}000\text{man} - \text{rem} / \$\text{M}$$

CONCLUSION

Based on a potential public risk reduction of 41,000 man-rem, a value/impact score of 1,000 man-rem/$M, and a reduction in core-melt frequency of less than 10^{-5}/RY, this issue was given a medium priority ranking (see Appendix C). The test program called for was completed by the staff and showed that the ECCS will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single failure criteria of 10 CFR 50, Appendix K. Ongoing thermal-hydraulic research was aimed at defining the degree of uncertainty in the ability of existing analytical models to simulate those transients on full-scale LWRs and not at proving capability. Thus, this item was RESOLVED and no new requirements were established.[817]

ITEM II.E.2.3: UNCERTAINTIES IN PERFORMANCE PREDICTIONS

DESCRIPTI0N

Historical Background

Small-break LOCA analyses performed by LWR vendors to develop operator guidelines had shown that large uncertainties may exist in system thermal-hydraulic response due to modeling assumptions or inaccuracies. It was necessary to establish that these assumptions or inaccuracies were properly accounted for in determining the acceptability of ECCS performance pursuant to Appendix K of 10 CFR 50.

The reason behind this TMI Action Plan[48] item was that, historically, the SBLOCA analyses were never reviewed by the NRC in the depth and detail with which the large-break analyses were reviewed. One of the obvious lessons of the TMI-2 accident was that SBLOCAs are much more likely to occur and, therefore, a highly detailed re-review of the small-break analyses might have been appropriate.

Safety Significance

SBLOCAs do not automatically result in rapid depressurization of the primary system. The more complicated blowdown makes it more difficult to predict ECC injection flow rates, water level, and many other parameters as a function of time. Moreover, there are many more possible locations for the break. In addition, the possibility of unexpected thermal-hydraulic phenomena cannot be ruled out. Since the SBLOCA analyses must conservatively bound a plant's response to all possible small breaks, all of these effects should be understood as well as possible.

Possible Solution

The proposed solution in NUREG-0660[48] called for NRR to issue instructions to holders of approved ECCS evaluation models to evaluate the uncertainty of small-break ECCS performance calculations; NRR was to evaluate these uncertainties. If changes were needed in the existing analysis methods to properly account for these uncertainties, recommendations were to be made to the Commission to adopt such changes. Ultimately, the adoption of these changes would result in changes to the analyses upon which plant TS were based. This could result in some restrictions on power level under certain circumstances.

PRIORITY DETERMINATION

Frequency Estimate

According to WASH-1400[16] estimates, small breaks (2 in. to 6 in. diameter) are expected to occur at a rate of 3×10^{-4} event/RY; very small breaks (0.5 in. to 2 in. diameter) are estimated to occur at a rate of 10^{-3} event/RY. Should such an event occur, it was estimated (based purely on judgment) that there may be a 10% chance of the actual peak cladding temperatures exceeding the temperatures predicted by the 10 CFR 50, Appendix K calculation due to the modeling uncertainties mentioned above.

However, in addition to the modeling conservatism, the 10 CFR 50, Appendix K calculations assumed the worst case single failure. Moreover, the small break analysis was very seldom limiting; usually the calculated small break peak cladding temperatures are about 400ºF below the 2200ºF Appendix K limit. Finally, a plant does not normally operate with the LOCA parameters (F_Q, MAPLHGR, etc.) at their limits.

Because the specific worst-case single failure varied for different plants, it was not practical to use fault trees to calculate the probability of such a failure. However, some perspective was gained by examining the following estimated failure rates from Appendix II of WASH-1400:[16]

PWR HPSI 1.2 x 10^{-2}/demand

PWR Emergency Power 10^{-5}/demand

BWR HPCI 9.8 x 10^{-2}/demand

BWR Emergency Power 10^{-6}/demand

The frequency of a system failure severe enough to approximate the Appendix K single failure assumptions was estimated to be, at most, 10^{-1}/demand. Given a small LOCA, a modeling uncertainty, and something approximating the worst-case single failure, the actual peak cladding temperature would be greater than that calculated by the analyses. However, there was still considerable margin to significant core damage because:

(1) The small-break analysis is rarely limiting. Usually there is about a 400ºF margin between the calculated small-break peak cladding temperature and the 2200ºF limit.

(2) Most plants operate well within their LOCA limits (i.e., are not "LOCA-limited").

(3) To get severe damage, a significant amount of cladding must achieve a temperature significantly higher than 2200ºF. The case of the hottest point of the core barely exceeding the temperature limit does not automatically imply severe damage.

These three considerations were summed by assuming that there was, at most, a 5% chance of significant core damage given a small LOCA, a model problem, and a near-worst-case single failure. Putting all this together, the frequency of events with significant core damage was estimated to be, at most, about 7 x 10^{-7} /RY.

Consequence Estimate

If cladding temperatures rise significantly above 2200ºF in a large portion of the core, the likely result would be a bed of debris. It was assumed that there was a 10% chance of a core-melt and a 90% chance of widespread cladding failure but no fuel melting. Neither of these fit readily into the WASH-1400[16] Release Categories. The core-melt case was approximated with 5 x 10^{6} man-rem (which is greater than or approximately equal to the consquences of PWR-1 through PWR-7 and BWR-1 through BWR-4), and the non-core-melt case by 120 man-rem (which bounds PWR-9 and BWR-5).

Cost Estimate

Industry Cost: It was estimated[48] that 15 staff-years and $1M of computer time would be required to perform the studies. In addition, 3 staff-months per operating plant were needed to implement procedural and TS changes. Since there were 70 plants operating, the estimated total direct industry cost was $4.25M (The 57 plants under construction at the time of this evaluation would not require implementation costs since the new analyses would displace analyses which would have been required in any case.)

In addition to the direct cost, there was an indirect cost due to the effect of further restricting operating parameters. Using the earlier assumptions that there was a 10% chance of finding a non-conservatism and a 5% chance of being SBLOCA-limited, and assuming further that at least a 1% power reduction resulted under such circumstances, the indirect costs averaged at least $5,500/RY. There were 43 operating PWRs with a cumulative experience of 350 RY and 27 BWRs with a cumulative experience of 260 RY. Adding the 36 PWRs and 21 BWRs that were under construction and assuming a plant life of 40 years, there were 4,470 RY remaining. Thus, the indirect cost was $24.6M and the total industry cost was $(4.25 + 24.6)M or $28.85M.

NRC Cost: It was estimated that 15 staff-years and $100,000 would be necessary for the staff to review the studies; in addition, the 70 backfit plants would require one staff-month each. (Again, the 57 plants under construction would not need a significant amount of extra review effort since the new reviews would displace the reviews of other analyses that would have been submitted.) Thus, NRC costs were estimated to be about $1.2M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(28.85 + 1.2)M or 30.05M.

Value/Impact Assessment

Based on an estimated public risk reduction of 1,565 man-rem and a cost of $30.05M for a possible solution, the value/impact score was given by:

$$S = \frac{1{,}565 \text{man} - \text{rem}}{\$30.05\text{M}}$$
$$= 52 \text{man} - \text{rem} / \$\text{M}$$

CONCLUSION

Based on the safety importance and value/impact score above, this issue had a low priority ranking. In addition, RSB/DSI/NRR had noted that much of the technical concern of the issue was automatically being investigated[98] in the implementation of Item II.K.3(30) which was in progress at the time of the initial evaluation of the issue in November 1983. In order to prevent duplication of effort and because the work on Item II.K.3(30) was in progress, the issue was given a low priority ranking (see Appendix C). In NUREG/CR-5382,[1563] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue. Further prioritization, using the conversion factor of $2,000/man-rem approved by the Commission in September 1995, resulted in an impact/value ratio (R) of $19,230/man-rem which placed the issue in the DROP category.

REFERENCES

0016.	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.
0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0817.	Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.E.2.2, ‘Research on Small Break LOCAs and Anomalous Transients,'" July 25, 1985. [9909290072]
1563.	NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

Task II.E.3: Decay Heat Removal (Rev. 2) ( )

The objective of this task was to improve the reliability and capability of nuclear power plant systems for removing decay heat and achieving safe shutdown conditions following transients and under post-accident conditions.

ITEM II.E.3.1: RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION

DESCRIPTION

This TMI Action Plan[48] item resulted in the issuance of requirements for: (1) upgrading the pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions; and (2) establishing new procedures and training for maintaining the RCS at hot standby conditions with only onsite power available.

CONCLUSION

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM II.E.3.2: SYSTEMS RELIABILITY

DESCRIPTION

One of the basic tenets of reactor safety is that there must always be a means of removing decay heat. The shutdown heat removal systems are designed to removed this heat at a rate that will enable the reactor to be brought to, and maintained in, a state of cold shutdown. This TMI Action Plan[48] item was intended to focus on shutdown heat removal system reliability.

Shutdown heat removal systems generally consist of two independent trains, each of which is quite reliable. Moreover, other systems can be used to prevent a core-melt under many circumstances. Nevertheless, given the importance of decay heat removal, the reliability of the shutdown heat removal systems remained in question.

The issue called for NRR to conduct a generic study to assess the capability and reliability of shutdown heat removal systems under various transients and degraded plant conditions, including complete loss of all feedwater. Deterministic and probabilistic methods were to be used to identify design weaknesses and possible system modifications that could be made to improve the capability and reliability of these systems under all shutdown conditions (i.e., startup, hot standby, shutdown, etc.).

CONCLUSION

This item was integrated[68] into the resolution of Issue A-45.

ITEM II.E.3.3: COORDINATED STUDY OF SHUTDOWN HEAT REMOVAL REQUIREMENTS

DESCRIPTION

A shutdown heat removal system is necessary in a nuclear reactor to establish and maintain a safe shutdown condition during normal and accident conditions. If the normal shutdown heat removal system does not perform its intended safety function, then an alternate method must be used. Therefore, this TMI Action Plan[48] item called for a coordinated effort to evaluate shutdown heat removal requirements in a comprehensive manner, thereby permitting a judgment of adequacy in terms of overall system requirements. As part of this effort, a study was to be conducted to assess the desirability of, and possible requirements for, a diverse heat removal path, such as feed-and-bleed, particularly if all secondary side cooling were unavailable. The need for alternate shutdown heat removal systems for PWRs and BWRs was to be evaluated based on value/impact or cost/benefit analyses. If such systems appeared to provide a significant safety benefit, alternative concepts were to be studied and recommendations made.

CONCLUSION

This item was reviewed and considered in the resolution of Issue A-45.

ITEM II.E.3.4: ALTERNATE CONCEPTS RESEARCH

DESCRIPTION

This TMI Action Plan[48] item involved a specific study related to the usefulness of installing an add-on decay heat removal system in existing nuclear power plants to improve the overall operational reliability of decay heat removal. The study entailed a review of the detailed design of a decay heat removal system (to be designed under DOE auspices) and was expected to result in suggested systems performance and safety design criteria as well as a value/impact analysis. In addition, scoping studies were to be performed to develop further information regarding the usefulness of other alternate concepts proposed for decay heat removal systems.

CONCLUSION

This item was RESOLVED with the publication of NUREG/CR-2883,[760] the results of which were considered in the resolution of Issue A-45.

ITEM II.E.3.5: REGULATORY GUIDE

DESCRIPTION

This TMI Action Plan item[48] called for the issuance of Revision 1 of Regulatory Guide 1.139[929] which includes requirements for reaching cold shutdown using safety-grade equipment.

CONCLUSION

In accordance with NUREG/CR-2883,[760] this issue was integrated into the resolution of Issue A-45.

REFERENCES

0048.	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.
0068.	Memorandum for R. Fraley from K. Kniel, "Draft Task Action Plan for TASK A#45, Shutdown Decay Heat Removal Requirements," May 22, 1981. [8106010652]
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0760.	NUREG/CR-2883, "Study of the Value and Impact of Alternative Decay Heat Removal topics for Light Water Reactors," U.S. Nuclear Regulatory Commission, (Vol. 1) June 1983, (Vol. 2) June 1983, (Vol. 3) June 1983.
0929.	Regulatory Guide 1.139, "Guidance for Residual Heat Removal," U.S. Nuclear Regulatory Commission, May 1978.

Task II.E.4: Containment Design (Rev. 2) ( )

The objective of this task was to improve the reliability and capability of nuclear power plant containment structures to reduce the radiological consequences and risk to the public from design basis events and degraded-core and core-melt accidents.

ITEM II.E.4.1: DEDICATED PENETRATIONS

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-18 was established by DL/NRR for implementation purposes.

ITEM II.E.4.2: ISOLATION DEPENDABILITY

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-19 was established by DL/NRR for implementation purposes.

ITEM II.E.4.3: INTEGRITY CHECKDESCRIPTION

Historical Background

In this TMI Action Plan[48] item, a requirement was proposed for the performance of a feasibility study to evaluate the need and possible methods for performing a periodic or continuous test to detect unknown gross openings in the containment structure. A prime example of the type of operational error this issue was directed at was the incident at Palisades where the reactor was operated for about 1.5 years while the containment isolation valves in a purge system bypass line were unknowingly locked in the open position.

Safety Significance

Should a LOCA resulting in major fuel damage occur in a plant that has an undetected breach in the containment building, severe offsite exposure would be expected.

Possible Solutions

Systems which can continuously monitor containment pressure, temperature, in-flow or outflow of fluids, and alarm upon abnormal conditions could be provided for in some containment designs such as inerted BWR MARK I and II containments, sub-atmospheric containments, and possibly some PWR dry containments which operate with a small positive differential containment pressure with respect to atmospheric pressure. Most PWR dry containments might require a system which can produce a small positive pressure in the containment periodically, perhaps quarterly, and perform a gross containment leak rate test to assure the plant is not operated for an extended time period with an undetected breach of containment integrity.

PRIORITY DETERMINATION

Frequency Estimate

Using known incidents in which breaches in containment integrity were revealed (mostly during the containment integrated leak rate testing required by 10 CFR 50, Appendix J), estimates of the duration of the breached condition, and the average number of plants in operation, an estimate of the expected frequency of an undetected breach in containment integrity was derived. The Palisades incident and three other incidents (in the five years prior to this evaluation) in which holes were detected in the containment liner were considered. The estimated frequency of an undetected breach in containment integrity was determined to be 1.1 x 10^{-2}/RY. The unavailability of containment due to a breach of containment integrity was also estimated to be about 10^{-2} year/RY, assuming in two instances the breach remained undetected for about 1.5 years, in another instance the breach was undetected for one year, and the remaining one was detected immediately.

From WASH-1400,[16] the dominant risk sequences which are affected by containment isolation (or integrity) failure are those which result in Category 4, 5, and 8 releases for PWRs and a Category 4 release for BWRs. These are all scenarios in which the containment failure mode is containment isolation failure. Since the WASH-1400[16] containment isolation failure frequency did not include contribution from undetected breach of containment integrity, the frequencies of the dominant scenarios from these categories were adjusted to include the additional probability of undetected breach of containment integrity. The base case risk was then calculated using the adjusted frequencies and the dose equivalent factors from Table D.1 of NUREG/CR-2800[64] for the affected scenarios for both PWRs and BWRs.

An estimate was then made of the potential effects of the above possible solutions in reducing the expected extent of containment unavailability as a result of undetected breach of containment integrity. Breaching of containment integrity is almost always found during containment integrated leakage rate tests which are performed about every 3.5 years. Continuous or quarterly testing will assure early detection of operational error resulting in breach of containment integrity. It was estimated that systems like those assumed could reduce the expected unavailability of containment due to breach of containment integrity events to 1.6×10^{-3}/demand. It was assumed that the frequency of unknown containment integrity violations was 1.1×10^{-2}/RY, as determined above, and that the average duration of such events (including detection and correction time) was 3 days for plants having continuous detection means and 1.5 months for plants utilizing periodic detection means. It was conservatively assumed that all breaches of containment integrity will be found by periodic testing. Using this new unavailability, the base case risk was adjusted to represent the expected risk at PWRs and BWRs following implementation of the solution.

Consequence Estimate

The difference between the base case risk and the adjusted risk represented the potential risk reduction which that be gained by the resolution of the issue. The potential risk reduction was found to be 10.1 man-rem/RY for PWRs and 6.1 man-rem/RY for BWRs. With an expected population of 95 PWRs and 48 BWRs and an expected average remaining life of 28 years/plant (Table C.1, NUREG/CR-2800),[64] the total expected public risk reduction from resolution of this issue was calculated to be 3.5×10^4 man-rem. The average public risk reduction was estimated to be 3.5×10^2 man-rem/plant. Resolution of this issue was not expected to affect the frequency of core-melt events.

Cost Estimate

Industry Cost: The population of 143 reactors was divided into two groups. One group represented those plants which, because of specific containment design features, may find it relatively easy to develop and install a continuous monitoring system. Plants with BWR MARK I and II inerted containments, subatmospheric containments, and PWR dry containments which normally operate with a small positive containment pressure would be expected to fall into this group. It was found that about 56 plants might fit into this group. It was expected that these plants might require a control room alarm, some containment pressure and/or temperature instruments to augment existing capacity, a flow measuring device, and a software routine which may be suitable for operation on the plant computer. This equipment was not expected to be safety grade as it would have no post-accident function. We estimated this equipment, installed, to cost about $80,000/plant. Operation, maintenance, and repair costs for the system were estimated at $20,000/RY. This resulted in a total cost for the 56 plants of $36M.

The remaining plants (87) were felt to be more suitable for a periodic test system which might pressurize the containment to a small positive pressure and check containment integrity by performing a low pressure leak rate test. These plants would be expected to require quite a bit more special pressure and temperature instrumentation than needed for a continuous monitoring type system. In addition, a high volume compressor would be needed. A program suitable for operation on the plant computer would also be required for data reduction and analysis. We estimate that such a system, installed, would cost about $250,000. Maintenance and operation of this system was estimated at $40,000/RY. This resulted in a total plant cost for these 87 plants of $121M. Thus, the total expected industry cost was $157M.

NRC Cost: Resolution of the issue and implementation of the solution were expected to require the following:

(a) Data collection, analysis, and definition of the expected frequency of breach of containment - 1 man-year

(b) Preliminary design of containment integrity test methods, systems, and equipment - 3 man-years

(c) Cost analysis - 0.5 man-year

(d) Development of NRC requirements, review and approval, issuance of order to licensees - 2.5 man-years

(e) Review of licensee implementation - 0.05 man-year/plant

(f) Surveillance of test results of all operating plants - 0.5 man-year/year.

At the rate of $100,000/man-year for all NRC and consultant manpower, the estimated cost was about $2.84M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(157 + 2.84)M or $159.84M.

Value/Impact Assessment

Based on an estimated public risk reduction of 35,000 man-rem and a cost of $159.84M for a possible solution, the value/impact score was given by:

$$S = \frac{35,000 \text{man} - \text{rem}}{\$159.84\text{M}}$$
$$= 220 \text{man} - \text{rem} / \$\text{M}$$

CONCLUSION

A value/impact score of 220 man-rem/$M was indicative of a medium priority ranking (see Appendix C). However, the evaluation of expected frequency for undetected breach of containment integrity performed for this effort indicated an unexpectedly high frequency (1.1 x 10^{-2}/RY). This exceeded the safety goal maximum probability for loss of a layer of "defense-in-depth" (i.e., the containment). For this reason, the issue was pursued on a high priority basis with the first order of business to be the establishment (as accurately as possible) of the expected frequency of undetected breach of containment integrity and the expected unavailability of containment and their uncertainty bounds.

The staff concluded its review of the issue and the results were presented in NUREG-1273[1104] which included a review of relevant LERs, the sensitivity of offsite dose to the containment leakage rate, and an assessment of potential methods for continuous monitoring of containment integrity.

All relevant LERs from April 1965 through May 1983 were reviewed to evaluate occurrences of suspected containment isolation failure; LERs are required to be submitted when the measured leakage exceeds the TS limits (0.6 of allowable containment leakage). This study indicated that reportable occurrences were divided about equally for BWRs and PWRs (~2/RY), and that only 16% of the reportable events were for components (mainly valves) located in systems that could provide a direct air path outside of containment (assuming failure of the second isolation valve). In addition, less than 5% of the events could be characterized as large or very large leaks (more than ten times allowable) within direct air pathways, and only a few could be considered as extended undetected breaches in the containment building. The probability of an undetected direct open air path in a BWR containment was estimated to be about 0.1 for small leaks to 0.001 for large leaks. For PWRs, the comparable probabilities were about 0.3 and 0.07.

A study of the potential risk as a function of containment leakage rate was provided in NUREG/CR-4330.[971] These analyses indicate that containment leakage provides only a small contribution (1 to 2 man-rem/RY) to the total exposure from postulated design basis accidents. Therefore, increasing the containment leakage up to a factor of 10 results in only a very small increase in total risk. Thus, containment leakage rate was not found to be an important contributor to the total risk on a probabilistic basis.

Item II.E.4.3 deals with containment leakage during postulated (i.e., design basis) accidents and does not address the issue of containment integrity and associated radiation consequences during severe accidents. This last issue is being addressed as part of implementation of the Commission's policy on severe accidents and, more specifically, in the Individual Plant Examination (IPE) and Containment Performance Improvement (CPI) programs. Thus, this issue was RESOLVED with no new requirements.[1103]

ITEM II.E.4.4: PURGING

The primary purpose of this item is to reevaluate the acceptability of purging/ venting nuclear power plant containments during the reactor operating modes of startup, power operation, hot standby, and hot shutdown. The five parts of this item are listed below.

ITEM II.E.4.4(1): ISSUE LETTER TO LICENSEES REQUESTING LIMITED PURGING

DESCRIPTION

A number of events occurred over a span of several years during and prior to 1978 that were directly related to containment purging during normal plant operation. Some of these events raised questions relating to automatic isolation of the purge penetrations which are used during power operation. Instances occurred at Millstone-2 where intermittent containment purge operations were conducted with the safety actuation isolation signals to both in-board and out-board containment isolation valves in the purge system inlet and outlet lines manually overridden and inoperable. Other instances occurred at Salem-1 where venting of the containment through the containment ventilation system valves to reduce pressure was conducted. In certain instances, this venting occurred with the containment high particulate radiation monitor isolation signal to the purge and pressure vacuum relief valves overridden.

These events raised concerns relative to potential failures affecting the purge penetration valves which could lead to a degradation in containment integrity and, for PWRs, a degradation in ECCS performance because of insufficient containment back pressure. In order to reduce the probability of these potential accident scenarios, the NRC was to issue letters to licensees of operating plants requesting limited purging of containment and justification for additional purging.

CONCLUSION

NRR issued a letter[142] to all licensees of operating plants on November 28, 1978 (Docket No. 50-348) requiring compliance with specific requests enclosed with that letter. This issue was RESOLVED with the issuance of the letter to the licensees.

ITEM II.E.4.4(2): ISSUE LETTER TO LICENSEES REQUESTING INFORMATION ON ISOLATION VALVE

DESCRIPTION

By letter dated November 28, 1978,[142] [see Item II.E.4.4(1)] the NRC requested all licensees of operating reactors to respond to generic concerns about containment purging or venting during normal plant operation. The generic concerns were two-fold:

(1) Events occurred where licensees overrode or bypassed the safety actuation isolation signals to the containment isolation valves. These events were determined to be abnormal occurrences and reported to Congress in January 1979.

(2) Licensing reviews required tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a design basis LOCA.

The staff visited several plants, met with some licensees, and held telephone conferences with many other licensees and valve manufacturers. As a result of these meetings and conferences and in light of the new information gained, the NRC determined that an interim commitment from all licensees of operating plants was warranted.

CONCLUSION

NRR issued a letter[143] with an interim position to all licensees of operating reactors requesting compliance with the specific items of the position. Thus, the issue was RESOLVED and requirements were issued.

ITEM II.E.4.4(3): ISSUE LETTER TO LICENSEES ON VALVE OPERABILITY

DESCRIPTION

By letter dated November 28, 1978,[142] NRC requested all licensees of operating reactors to respond to generic concerns about containment purging and venting during normal plant operation. As a result of the review of licensee responses to this letter, NRC learned that at least three valve vendors reported that their valves may not close against ascending differential pressure and the resulting dynamic loading of the design basis LOCA. For plants utilizing valves from these manufacturers, it was determined that the containment integrity could be sufficiently assured by maintaining the valves in the closed position or by restricting the angular opening of the valves whenever primary containment integrity is required. NRC is to issue guidelines to all affected licensees in order to ensure operability of purge and vent valves.

CONCLUSION

NRR issued a letter[162] to all licensees of operating plants requesting compliance with the specific guidelines enclosed with that letter. All licensees that utilized valves identified by the three manufacturers as having potential closure problems were required to either maintain the valves closed or install devices to limit the opening angle at all times when containment integrity is required, until such time that full opening was justified to the NRC. This issue was RESOLVED with the issuance of the letter to the licensees.

ITEM II.E.4.4(4): EVALUATE PURGING AND VENTING DURING NORMAL OPERATION

Items II.E.4.4(4) and II.E.4.4(5) were combined and evaluated together.

DESCRIPTION

Historical Background

This TMI Action Plan[48] item required NRR to generically evaluate the radiological consequences of containment purging of nuclear power plants while in the power operation mode. Item II.E.4.4(5) established a requirement for NRR to utilize the results of the radiological evolution from Item II.E.4.4(4) and other efforts already completed to reevaluate current NRC requirements established in SRP[11] Section 6.2.4 and the associated BTP CSB/6-4. Item II.E.4.4(5) anticipated a need to require modification of the current requirements on the use of purge systems of nuclear power plants. Therefore, Items II.E.4.4(4) and II.E.4.4(5) were combined and evaluated together.

Safety Significance

Should a LOCA occur during a period in which the containment building is being purged while the plant is operating at power, radiation releases will occur. If the purge system containment isolation valves meet the closure requirements of BTP/CSB 6-4, the containment purge system should be closed prior to any LOCA-induced fuel damage and releases to the public would be small. However, if the LOCA resulted in major fuel damage and the containment purge system is not isolated (due to isolation valve or signal failures), releases and, therefore, public exposure would be large.

Possible Solution

A possible solution to further reduce the probability of failure to isolate the purge system was to limit the use of the purge system when RCS temperatures are greater than 200^0F. The imposition of limits on the use of purge systems which have containment isolation valves meeting the staff's operability requirements for active valves (BTP/CSB 6-4) has been considered from time to time but as yet has not been implemented. A few of the older operating plants require either very frequent or even continuous purging to control containment temperature and/or pressure. If containment purge system use were limited to some small fraction of the time (1% to 10%) that the plant is in operating modes 1-4, these plants would either have to shut down to purge or modify the plant to add larger containment cooling or pressure control systems. In addition, a few plants which require frequent entry by operators to perform safety-related surveillance and maintenance would find it necessary to add containment air filtration systems to reduce operator exposures in order that plant shutdowns not be incurred to purge the containment prior to an entry, if use of the purge system is drastically limited.

PRIORITY DETERMINATION

Assumptions

It was assumed that the solution would entail some limit on the use of purge systems. Using existing knowledge of operating practices at the time of this evaluation, it was estimated that, of the 72 operating plants, 25 inerted BWRs and 8 PWRs with sub-atmospheric containments did not purge during plant operation. There were about 20 to 22 newer PWRs with dry containments that purged very little (~1% to 5% or less). This leaves 17 PWRs which we assumed purge continuously. Of these 17 plants, we assumed that 7 (about 10% of all operating plants) need to purge continuously for containment temperature or pressure control (violation of current requirements). We assumed that the remaining 10 plants purge continuously because they have no containment air filtration systems and thus purge frequently or continuously for the purpose of maintaining operating personnel exposure as low as possible. If low percentage use limits are placed on containment purge systems, it was assumed that the group of 7 plants would be required to purchase and install containment pressure and temperature control systems and suffer replacement power costs during plant shutdowns to purge until these systems are installed. The group of 10 plants was assumed to have to purchase and install filtration systems for containment air, but were not assumed to encounter plant shutdown and replacement power costs prior to installation. It was instead assumed that higher in-plant personnel exposures were incurred until the modifications were completed.

Frequency Estimate

Use of the containment purge system during plant power operation will result in two distinct scenarios by which significant radiation release to the environment would be expected. The two scenarios are: (1) LOCA with core-melt and the containment purge system fails to isolate; and (2) successfully mitigated LOCA but the containment purge system fails to isolate. In early 1981, SPEB/DST/NRR evaluated[206] three different positions regarding the use of containment purge systems during plant operation. This report developed best estimates of the frequency of accident scenarios which might result while the containment purge system is in use. This study showed the expected frequency of the two scenarios above to be 4×10^{-9}/RY (Scenario 1) and 7×10^{-8}/RY (Scenario 2). These frequencies were for an assumed purge usage of 20% of plant operating time. In this analysis, the above values were adjusted to determine the expected frequency of the scenarios as a function of purge limit (from 0% to 100%).

Consequence Estimate

WASH-1400[16] PWR Release Category 4 represents the offsite consequences of coremelt events in which the containment is not isolated. In this scenario, a 4-inch penetration was assumed to be open resulting in atmospheric releases. Most PWR purge system penetrations are large (24" to 60" in diameter). We assumed a 40" diameter purge line. We ratioed the releases by the square of the ratio of the diameter of the purge line to the diameter of the unisolated line in the WASH-1400[16] PWR-4 event. In this case, the ratioed consequence would have exceeded the consequence of the PWR-1 event (early overpressure failure of containment with energetic release of the greatest fission product inventory). We, therefore, limited the release for core-melt scenarios in which the containment is not isolated to that for the PWR-1 event. This resulted in a calculated dose of 5.4×10^{6} man-rem/event (Table D.1, NUREG/CR-2800),[64] assuming a core-melt LOCA in which the purge system fails to isolate (Scenario 1), midwest-type meteorology, and a uniform population density of 340 people/square-mile.

The same ratioing technique of the dose resulting from a PWR-8 release was used to determine an expected dose for the mitigated LOCA in which the containment is not isolated (Scenario 2). For this event, the offsite dose was found to be 2.3×10^{5} man-rem/event.

The expected frequencies of the two scenarios were multiplied by the dose consequence of the appropriate scenario and summed. This resulted in an averted public risk, assuming the base case in which there is no limit on purge system use (100% limit), of 0.106 man-rem/RY for the case in which there is no use of the purge system allowed (0% limit). When applied to the 17 plants for their average expected remaining life (25 yrs), this results in a maximum total averted public risk of 46 man-rem for a 0% limit on the use of purge systems during plant power operation. Averted total public risk varies linearly from nothing, when 100% use of purge systems is allowed, to the maximum (46 man-rem), when no purging is allowed. The maximum potential total risk reduction afforded by a complete ban on the use of purge systems while the plant is in PWR operating modes 1 through 4 (about 46 man-rem) represents less than 0.02% of the total plant risk as determined by WASH-1400.[16] The average public risk averted per plant if a 0% purge limit is imposed is 0.32 man-rem/reactor.

Cost Estimate

Industry Cost: Costs were limited to the 17 plants that were expected to purge frequently or continuously and were estimated to cover both the cost of containment pressure and temperature control systems or filtration systems, as appropriate. The cost of replacement power at $300,000/day was also estimated for the 7 plants which were assumed to require pressure and temperature control system additions. In the analysis, we assumed that the affected plants could purge for 1 day and then operate for 3 days before containment purging would be required again. We estimated the cost of a pressure/temperature control system addition to be $2.5M and a filtration system addition to be $1M. Industry costs were calculated as a function of containment purge limit. Due to the above assumption on the amount of purge versus non-purge operation attainable, there are no industry costs between 25% and 100% of the purge limit. Different ratios of purge to non-purge time would alter the purge limit at which negligible industry cost would be reached.

NRC Cost: We estimated a total of 19.5 man-years of staff and consultant effort to do the following: study purge system use, operational data, and designs; prepare preliminary design of potential plant modifications; perform cost analysis; develop, review, and approve new requirements and issue orders; review licensee responses to orders, including plant modifications when proposed; and perform yearly surveillance of plant purge system usage. At $100,000/man-year, these efforts were estimated to be about $2M.

Value/Impact Assessment

The value/impact score as a function of containment purge limit increases slightly to about 0.4 man-rem/$M in the purge limit range of zero to 25%. At 25%, a maximum value/impact score of 17 man-rem/$M was found. The value/impact score decreases as the purge limit is increased from 25% to 100%.

Other Considerations

The value/impact score as a function of purge limit varied from low category to the drop category. The value/impact score calculated is a direct function of the probability of the failure of the containment purge system isolation valve (large butterfly valve) to close. The best estimate value for failure to close (which was used in the prior SPEB study) was conservatively chosen to be 3 x 10^{-3} /demand. WASH-1400[16] found the mean failure rate of all qualified safety system valves (including butterfly valves) to be 3 x 10^{-4}/demand. If the failure rate of containment purge system isolation valves were found to be much greater than the value assumed in these studies (i.e., on the order of 10^{-1}/demand), the public risk associated with containment purging during power operations would be greatly increased. The public risk due to containment purging during plant operations, instead of being less than 1% of total plant risk, could be large enough to become a dominant risk factor. In that case, action to reduce the public risk from purging of plants during power operation would probably be warranted. The resolution of the issue might take the form of increased reliability requirements for active purge system isolation valves, strict limits on the use of purge system during normal plant operation, or a combination of both approaches. This analysis indicates that, if the isolation valve failure rate is high at all plants, the more attractive means to reduce risk would be to improve the valve reliability.

CONCLUSION

The value/impact score indicated a low priority ranking for Items II.E.4.4(4) and II.E.4.4(5) (see Appendix C). The key to a better risk/benefit insight to the value of further changes in criteria for the use of containment purge systems centered around the failure rate of the large butterfly valves utilized as containment isolation valves.

At the time prioritization of Items II.E.4.4(4) and II.E.4.4(5) was initiated, work was not yet completed on these items. Since that time, AEB/DSI/NRR[229] and CSB/DSI/NRR[230] reported that the efforts called for by these items were completed and the EDO was informed.[231] Thus, these issues were RESOLVED.[382]

ITEM II.E.4.4(5): ISSUE MODIFIED PURGING AND VENTING REQUIREMENT

This item was evaluated in Item II.E.4.4(4) above and was determined to have a low priority ranking. However, all required action was completed as described in Item II.E.4.4(4) above. Thus, this issue was RESOLVED.[231]

REFERENCES

0011.	*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.
0016.	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.
0048.	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.
0064.	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.

0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0142.	Letter to Alabama Power Company from U.S. Nuclear Regulatory Commission, "Containment Purging During Normal Plant Operation," November 28, 1978. [7812140364]
0143.	Letter to Nebraska Public Power District from U.S. Nuclear Regulatory Commission, "Containment Purging and Venting during Normal Operation," October 22, 1979. [7911190034]
0162.	Letter to All Light Water Reactors from U.S. Nuclear Regulatory Commission, "Containment Purging and Venting During Normal Operation—Guidelines for Valve Operability," September 27, 1979. [9705190209]
0206.	Memorandum for L. Rubenstein from M. Ernst, "Proposed Position Regarding Containment Purge/Vent Systems," April 17, 1981. [8105260251]
0229.	Memorandum for T. Speis from R. Houston, "Containment Venting and Purging—Completion of TMI Action Plan Item II.E.4.4(4)," March 3, 1982. [8401170023, 8203240149]
0230.	Memorandum for R. Mattson from T. Speis, "Containment Purge and Venting—Completion of TMI Action Plan Item II.E.4.4(5)," April 9, 1982. [8204260021]
0231.	Memorandum for W. Dircks from R. Mattson, "Status Report on Containment Purge Evaluations," May 13, 1982. [8401170021]
0382.	Memorandum for W. Minners from R. Mattson, "Schedules for Resolving and Completing Generic Issues," January 21, 1983. [8301260532]
0971.	NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1986, (Vol. 2) June 1986.
1103.	Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Safety Issue II.E.4.3, `Containment Integrity Check,'" March 22, 1988. [8809150125]
1104.	NUREG-1273, "Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3, `Containment Integrity Check,'" U.S. Nuclear Regulatory Commission, April 1988.

Task II.E.5: Design Sensitivity of B&W Reactors (Rev. 2) ( )

The objective of this task was to reduce the sensitivity of B&W plants to feedwater transients, with emphasis on the overcooling transients that had been observed at B&W operating plants.

ITEM II.E.5.1: DESIGN EVALUATION

DESCRIPTION

Historical Background

The NRC staff concluded that B&W reactors exhibited unique sensitivity to secondary system transients (both undercooling and overcooling events). Therefore, B&W plants under construction were required to propose recommendations on hardware and procedure changes relative to the need for methods for damping primary system sensitivity to perturbations in the once-through steam generator (OTSG). This issue also considered the backfitting of the recommendations on operating plants.

Safety Significance

The safety significance of this TMI Action Plan item[48] was the same as that for Item II.E.5.2, i.e., the perception of what constitutes acceptable response to transients.

Possible Solution

All B&W plants under construction were required [10 CFR 50.54(f)] to provide recommendations to reduce plant sensitivity.[154] The recommendations (with proposed modifications) were submitted for NRC review. The staff also evaluated the modifications proposed by the applicants for possible backfit to operating plants.[159, 160, 443] The staff concluded that the portion of this issue that dealt with plants under construction was completed with the issuance of the Midland-1&2 SER which evaluated the modifications.[159, 160, 443] The other B&W plants under construction were to be evaluated as part of the normal licensing review.

The portion of the issue which dealt with backfit considerations was also completed.[159, 160, 443] Specifically, the staff concluded that the Midland modifications would be effective in reducing both the frequency and severity of overcooling transients and recommended that similar modifications be made at operating B&W plants. The staff also concluded that the following related activities were underway:

(1) Operating B&W plants were implementing upgrades to meet NUREG-0737.[98]

(2) Issue A-47, "Safety Implications of Control Systems," was addressing steam generator overcooling/overfilling as it related to control system failures.

(3) The staff was also pursuing resolution of overcooling events (steam bubble formation/natural circulation interruption) on a generic basis with the B&W Owners' Group [NUREG-0737,[98] Item II.K.3(30)].

(4) Consideration of pressurized thermal shock (PTS) concerns relating to overcooling were being addressed by the staff as part of the resolution of Issue A-49, "Pressurized Thermal Shock."

CONCLUSION

Based on the above, the staff concluded that the B&W-designed operating reactors had responded to staff concerns regarding the frequency of overcooling and steam generator overfill events by implementing plant modifications. The adequacy of these modifications were to be confirmed by other ongoing programs. Thus, this item was RESOLVED and requirements were established.

ITEM II.E.5.2: B&W REACTOR TRANSIENT RESPONSE TASK FORCE

DESCRIPTION

Historical Background

After TMI-2, the NRC staff investigated[155] the response of B&W reactors to transients and determined that, in their opinion, they were overly responsive to certain transients. This responsiveness or sensitivity was attributed to a number of design and operational features including the small secondary water inventory in the steam generator, the small pressurizer volume, the pilot-operated relief valve (PORV) set-point, and the high pressure injection (HPI) set-points. As a result of the investigation, a number of recommendations were made for improving the plant response.[155]

The recommendations covered a number of design changes and operational considerations. DST/NRR provided a prioritization for the recommendations[158] in August 1980. A number of these recommendations (referred to as Category A items) had already been implemented (or were being implemented) for the B&W operating plants.[156, 157] The other recommendations (referred to as Category B items) had not been issued as requirements, although a number of them had been implemented by some licensees with B&W plants as part of their own investigations.

Safety Significance

The safety significance of this TMI Action Plan[48] item depended on the perception of what constitutes acceptable response to transients. NRC requirements were outlined in the SRP[11] and all plants were required to meet these, as a minimum. It was suggested[159] by DSI/NRR that additional performance criteria were necessary to more restrict the plants' response to transients and, as a result, limit the potential for plant damage.

Possible Solution

The technical resolution to this issue was defined in NUREG-0667.[155] It was suggested[159] that implementing the resolution required additional specification of the staff's performance criteria for transient response. (Existing criteria were contained in the SRP.[11]) Therefore, DSI/NRR proposed[159] that a uniform requirement in the form of criteria be issued by the NRC to ensure that adequate steps were taken by all B&W plants. Specifically, the recommended criteria were:

(1) ECCS actuation or loss of pressurizer level indication should not normally occur following a reactor trip or main feedwater control failure.

(2) Credit for operator action to mitigate overcooling events should be consistent with the guidelines of ANSI/ ANS-58.8.[45]

(3) Steam generators should be protected from overfill from main or auxiliary feedwater flow to limit overcooling. This equipment should be safety grade if flooding of the steam lines is an unanalyzed event.

CONCLUSION

Based on a DST/NRR evaluation[160] of the issue, it was recommended that implementation would be best accomplished by issuance of a statement of NRC's performance criteria for transients. It was also recommended that the first two criteria and accompanying value/impact statements be submitted to CRGR for review. The third criterion was included in Issues A-47 and A-49. Thus, the issue was RESOLVED and requirements were established.[656, 657]

REFERENCES

0011.	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.
0045.	ANSI/ANS-58.8, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," American Nuclear Society, 1984.
0048.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

0154.	Letter to Construction Permit Holders of B&W Designed Facilities from U.S. Nuclear Regulatory Commission, October 25, 1979.
0155.	NUREG-0667, "Transient Response of Babcock & Wilcox Designed Reactors," U.S. Nuclear Regulatory Commission, May 1980.
0156.	Memorandum for H. Denton from D. Eisenhut, "NUREG-0667, Transient Response of Babcock & Wilcox Designed Reactors, Implementation Plan," June 3, 1981. [8510070181]
0157.	Memorandum for D. Eisenhut from G. Lainas, "Status Report on Implementation of NUREG-0667 Category A Recommendations," December 15, 1981. [8201190550]
0158.	Memorandum for H. Denton from R. Mattson, "Review of Final Report of the B&W Reactor Transient Response Task Force (NUREG-0667)," August 8, 1980. [8010270109, 8010240413]
0159.	Memorandum for S. Hanauer from R. Mattson, "Design Sensitivity of B&W Reactors, Item II.E.5.1 of NUREG-0660," February 26, 1982. [8203170235]
0160.	Memorandum for R. Mattson from S. Hanauer, "Design Sensitivity of B&W Reactors," June 21, 1982. [8207150195]
0443.	Memorandum for W. Dircks from R. Mattson, "Closeout of NUREG#0660 Item II.E.5.1, Design Sensitivity of B&W Plants for Operating Plants," March 15, 1983. [8304080415]
0656.	Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan Task II.E.5.2, Transient Response of B&W Designed Reactors," September 28, 1984. [8410110596]
0657.	Memorandum for D. Crutchfield from D. Eisenhut, "TMI Action Plan Task II.E.5.2," November 6, 1984. [8411270129]

Task II.E.6: In Situ Testing of Valves (Rev. 2) ( )

The objective of this task was to evaluate whether existing requirements for valve testing provided adequate assurance of performance under design conditions.

ITEM II.E.6.1: TEST ADEQUACY STUDY

DESCRIPTION

Historical Background

The purpose of this TMI Action Plan[48] item was to establish the adequacy of existing requirements for safety-related valve testing. It recommended a study which would result in recommendations for alternate means of verifying performance requirements.

Safety Significance

Valve performance is critical to the successful functioning of a large number of a plant's safety systems.

Possible Solution

It could be assumed that a study would be conducted for both PWRs and BWRs and that it could result in recommendations for additional testing and/or maintenance on all safety-related valves. A program to implement the recommendations would then be required at all plants.

PRIORITY DETERMINATION

Assumptions

In an analysis of this issue by PNL,[64] it was assumed that all safety-related valves would be affected by resolution of the issue. Then, since all the dominant accident sequences (of Oconee-3 and Grand Gulf-1, the representative plants) involved failures of such valves, the sequences themselves were assumed to be directly affected. It was assumed that the new program would produce a reduction of 5% in the frequencies of the affected accident sequences (those that involved safety-related valves).

Frequency Estimate

It was determined[64] that all accident sequences for Oconee-3, except the following, involved safety-related valves and were thus assumed to be affected: T_2MLUO, T_2KMO, T_1(B_3)MLU, T_1MLUO, and T_3MLUO. For Grand Gulf-1, the only exception was T_{23}C.

For all the affected parameters, the base case frequency was taken as the original value. The adjusted case frequency was then calculated by the 5% reduction. The core-melt frequency reduction was then calculated to be 3×10^{-6} /RY and 10^{-6}/RY for Oconee-3 and Grand Gulf-1, respectively.

Consequence Estimate

Based on the 5% reduction, the public risk reduction was calculated to be 7.1 man-rem/RY and 7.8 man-rem/RY for Oconee-3 and Grand Gulf-1, respectively. The average remaining lives of the 95 affected PWRs and the 49 affected BWRs were calculated to be 28.2 years and 26.2 years, respectively. This resulted in a potential risk reduction of 1.9×10^4 man-rem for PWRs and 10^4 man-rem for BWRs. Thus, the total risk reduction associated with this issue was approximately 3×10^4 man-rem.

Cost Estimate

Industry Cost: It was estimated that the implementation effort for engineering, etc., would be about 10 man-weeks/plant for PWRs and 8 man-weeks/plant for BWRs. (The difference was due to the fewer number of affected valves in a BWR.) The cost was then calculated as follows:

PWRs: (10 man-weeks/plant)($2,000/man-week) = $20,000/plant

BWRs: (8 man-weeks/plant)($2,000/man-week) = $16,000/plant

For the 95 PWRs and 49 BWRs, this cost amounted to $2.7M.

The annual industry effort for operations and maintenance was estimated to be 16 man-weeks/RY for PWRs and 12 man-weeks/RY for BWRs with resultant costs of $16,000/RY and $12,000/RY for PWRs and BWRs, respectively. For the 95 PWRs with an average remaining life of 28.2 years, the cost was approximately $42.9M. For the 49 BWRs with an average remaining life of 26.2 years, the cost was approximately $15.4M.

Thus, the total industry cost to implement the possible solution to this issue was $(2.7 + 42.9 + 15.4)M or $61M.

NRC Cost: NRC labor for development of the solution for PWRs was estimated to be 1 man-year. Implementation of the solution was estimated to take 1 man-week/plant. Development of the solution for BWRs was estimated to be 0.5 man-year. Implementation time expended was estimated to be the same as for PWRs. Therefore, the estimated NRC costs were $0.43M.

It was also estimated that NRC labor for periodic review of operation and maintenance of the solution would be 1 man-week/RY for PWRs and 0.5 man-week/RY for BWRs. This translated into $2,000/RY and $1,000/RY, respectively, for all plants for a cost of $6.7M. Thus, the total NRC cost was $(0.43 + 6.7)M or $7.1M.

Total Cost: The total industry and NRC cost to resolve this issue was estimated to be $(61 + 7.1)M or $68.1M.

Value/Impact Assessment

Based on a potential risk reduction of 3 x 10^4 man-rem and an estimated implementation cost of $68.1M, the value/impact score was given by:

$$S = \frac{3x10^4 \text{ man-rem}}{\$68.1M}$$

$$= 440 \text{man-rem}/\$M$$

Uncertainty

The value/impact score was significantly influenced by the assumption that a 5% frequency reduction could be obtained; this number was highly judgmental.

Other Considerations

(1) Occupational dose would lower (significantly) this value/impact score because the labor required in a radiation zone would be significant. The estimated occupational dose from performing this periodic testing was about 24 man-rem/RY for PWRs and 18 man-rem/RY for BWRs. Over the life of a plant, the overall (total) occupational dose was estimated to be 8.9 x 10^4 man-rem.

(2) Occupational risk reduction due to accident avoidance was concluded to be small and accident avoidance costs, although large when considered in relation to the other costs, would not significantly change the score.

CONCLUSION

Based on the value/impact score and the additional considerations, this issue was given a medium priority ranking and was later divided into four parts during resolution: (1) pressure isolation valves; (2) check valves; (3) reevaluation of thermal-overload protection provisions of Regulatory Guide 1.106[1215] for MOVs; and (4) in-situ testing of MOVs.

The investigation of alternatives to leak rate testing of pressure isolation valves, including check valves, was integrated into the resolution of Issue 105, "Interfacing Systems LOCA." These alternatives included non-intrusive methods to detect check valve disk position and motion, as well as surveillance of internal parts by various means. Any new issue regarding testing of check valves that may be identified in the future will be prioritized as a new generic issue. The results of the staff's study of MOV thermal overload protection were published in NUREG-1296.[1216] The staff concluded that, although misinterpreted by the industry at times, the guidelines in Regulatory Guide 1.106[1215] were adequate. Several suggestions for improving MOV thermal overload protection were outlined in NUREG-1296.[1216] In addition, letters were sent to the pertinent IEEE and ASME subcommittees encouraging the development of standards for MOV thermal overload protection. In-situ testing and surveillance of check valves was being addressed by an industry effort; in-situ testing of MOVs was resolved with the issuance of Generic Letter 89-10.[1217] Thus, this issue was RESOLVED and requirements were established.[1218]

REFERENCES

0048.	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.
0064.	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.
1215.	Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," U.S. Nuclear Regulatory Commission, November 1975, (Rev. 1) March 1977. [7907100392]
1216.	NUREG-1296, "Thermal Overload Protection for Electric Motors on Safety-Related Motor-Operated Valves—Generic Issue II.E.6.1," U.S. Nuclear Regulatory Commission, June 1988.
1217.	Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants from U.S. Nuclear Regulatory Commission, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10)—10 CFR 50.54(f)," June 28, 1989 [*ML031150300*], (Supplement 1) June 13, 1990 [*ML031130421*], (Supplement 2) August 3, 1990 [*ML031150307*], (Supplement 3) October 25, 1990 [*ML031150326*], (Supplement 4) February 12, 1992 [*ML031150330*], (Supplement 5) June 28, 1993 [*ML031140103*], (Supplement 6) March 8, 1994 [*ML031140111*].
1218.	Memorandum for V. Stello from E. Beckjord, "Close-out of Generic Issue II.E.6.1, `In Situ Testing of Valves,'" June 30, 1989. [8907100275]

Task II.F: Instrumentation and Controls (Rev. 3) ( )

The objective of this task was to provide instrumentation to monitor plant variables and systems during and following an accident. Indications of plant variables and status of systems important to safety are required by the plant operator (licensee) during accident situations to:

(1) provide information needed to permit the operator to take pre-planned manual actions to accomplish safe plant shutdown;

(2) determine whether the reactor trip, engineered safety features systems, an manually-initiated systems are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity);

(3) provide information to the operator that will enable him to determine the potential for a breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and if a barrier has been breached;

(4) furnish data for deciding on the need to take unplanned action if an automatic or manually-initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation;

(5) allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of the impending threat;

(6) improve requirements and guidance for classifying nuclear power plant instrumentation control and electrical equipment important to safety.

ITEM II.F.1: ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPAs F-20, F-21, F-22, F-23, F-24, and F-25 were established by DL for implementation purposes.

ITEM II.F.2: IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-26 was established by DL for implementation purposes.

ITEM II.F.3: INSTRUMENTS FOR MONITORING ACCIDENT CONDITIONS

DESCRIPTION

Prior to the TMI-2 event, the August 1977 version of Regulatory Guide 1.97[55] had been used as guidance during licensing reviews. Item II.F.3 called[48] for this regulatory guide to be updated to include the TMI-2 concerns.

CONCLUSION

After the TMI-2 event, Task II.F of the TMI Action Plan[48] addressed several concerns regarding the availability and adequacy of instrumentation to monitor plant variables and systems during and following an accident. Revision 2 to Regulatory Guide 1.97[55] was published in December of 1980 and implementation was carried out as discussed in SECY-82-111[151] and a letter[376] issued to all licensees of operating reactors. Thus, this item was RESOLVED and new requirements were established.

ITEM II.F.4: STUDY OF CONTROL AND PROTECTIVE ACTION DESIGN REQUIREMENTS

DESCRIPTION

Historical Background

After the TMI-2 event, the Special Inquiry Group made recommendations[161] for the staff to study three items in the area of control and protection systems. These were: (1) automatic reactor protection actions should be derived, to the degree possible, from independent process variables; (2) automatic actions through coincidence of independent process variables should be limited, to the degree possible, for non-reactor protection functions;

(3) control circuit components should be designed and periodically tested at expected degraded power supply conditions to ensure that they are capable of performing their intended function.

Safety Significance

The report[161] concluded that improvements in these areas may help prevent specific occurrences which were noted upon evaluation of the TMI-2 event.

Possible Solutions

This TMI Action Plan[48] item addressed the performance of a study that could indicate potential deficiencies and identify possible fixes which could be incorporated as design criteria in the SRP.[11] Industry would then be required to meet these criteria.

PRIORITY DETERMINATION

No attempt was made to estimate a value/impact score for this issue. It appeared that the non-specific nature of the recommendations (i.e., use of words like "to the degree possible") would require a large amount of additional study prior to defining any specific implementation requirements. Therefore, neither potential risk reduction or costs could be estimated. The following considerations were taken into account.

(1) The first criterion, to a large degree, was typically addressed by existing protection systems. The use of a number of different plant parameters to initiate the protection system was an indication of the application of this criterion. There may have been instances in different plant designs where, for certain events, this criterion had not been adequately addressed; however, it was believed that these were isolated instances. Furthermore, the ATWS rule, which included NUREG-0460[704] requirements, addressed monitoring of independent process variables. As another consideration, protection system design requirements were expected to undergo another review as a result of preparation of a Regulatory Guide to endorse IEEE Std. 603-1977.[200]

(2) The second criterion addressed non-protection systems. At the time this issue was initially evaluated, the staff did not have detailed design criteria for these systems (typically referred to as "control systems") in the SRP.[11] It was believed that, if any criteria were to be included, they would be the result of a comprehensive program such as the existing program addressing Issue A-47, "Safety Implications of Control Systems."

(3) One part of the third criterion was addressed in SRP[11] Section 3.11, "Environmental Qualification of Equipment." Specifically, safety-related components are designed for performance at varying power supply conditions. Typically, they are initially tested to these conditions as part of their qualification program. The other part of the third criterion was not required at the time this issue was evaluated. Under conditions with offsite power feeding all plant components, it could be postulated that redundant components could experience some degraded power supply conditions; however, this concern was addressed through various plant fixes as part of their degraded grid analysis. Under conditions with onsite power feeding the components, the independence of the systems would prevent redundant components from experiencing degraded power.

CONCLUSION

Based on the considerations listed above, this issue was placed in the DROP category.

ITEM II.F.5: CLASSIFICATION OF INSTRUMENTATION, CONTROL, AND ELECTRICAL EQUIPMENT

DESCRIPTION

Historical Background

After the TMI-2 event, the staff recommended[48] that the existing method of classifying instrumentation, control, and electrical equipment needed revision to allow graded criteria that would more closely correspond to the equipment's importance to safety.

Safety Significance

Such a grading could place emphasis on improvements in the non-class 1E systems which could affect core-melt frequency. It could also allow more design flexibility and result in potentially more cost-effective electrical, instrumentation, and control system designs.

Possible Solution

It was recommended that the NRC, in conjunction with IEEE, develop a standard which would provide a classification approach based on the level of importance to safety of equipment. The standard would then be endorsed by a Regulatory Guide. Utility conformance to important criteria such as redundancy, reliability, etc., for selected systems would be mandated.

PRIORITY DETERMINATION

Assumptions

A program to classify and upgrade non-1E instrumentation, controls, and electrical systems was assumed to improve balance-of-plant system reliability and thus reduce transient frequencies. Based on EPRI transient data,[307] a number of transient categories and frequencies of interest were identified.

In a PNL assessment[64] of this issue, it was assumed that 50% of all these transients were attributable to instrumentation, control, and electrical system failures. Then it was assumed that resolution of this issue would result in about a 10% reduction in such failures.

Frequency Estimate

The reduction assumed above translates into about a 6% reduction in transients (other than loss of offsite power) for PWRs and a 4% reduction in transients for BWRs. Therefore, the 6% reduction was divided between the T_2 and T_3 transients for PWRs in the Oconee-3 risk equations. The 4% reduction was applied to the T_{23} transients for BWRs in the Grand Gulf-1 equations. This resulted in reductions in core-melt frequency of 2.1 x 10^{-6}/RY for PWRs and 9 x 10^{-7}/RY for BWRs.

Consequence Estimate

The above data translated (assuming a population density at 340 people/square-mile) to a per plant reduction in public risk of 5.6 man-rem/RY for PWRs and 7 man-rem/RY for BWRs. Assuming 90 PWRs with an average remaining life of 28.8 yrs and 44 BWRs with an average remaining life of 27.4 yrs, the total public risk reduction was estimated to be 23,000 man-rem.

Cost Estimate

Industry Cost: An estimate of costs for implementing improved non-1E systems was based on the installation cost ($1M) of a safety parameter display system (SPDS) at Yankee Rowe. The SPDS is considered a non-1E system which includes certain design features beyond those of a typical non-1E system. It was assumed that classification and upgrading of all remaining non-1E systems would represent a similar cost of $1M/plant, divided evenly between equipment costs and manpower costs for backfit plants. Forward-fit plants should only require additional equipment costs. Total industry cost would then be (based on 47 backfit and 43 forward-fit PWRs and 24 backfit and 20 forward-fit BWRs) about $100M.

NRC Cost: Since the IEEE Trial Use Guide P-827,[233] "A Method for Determining Requirements for Instrumentation, Control and Electrical Systems Important to Safety," had been released, the NRC cost for development was considered minimal (i.e., on the order of 0.5 man-year). The cost for support of the resolution was believed to be potentially significant and was assumed to be 1 man-year/plant with a resultant cost of $13.4M.

Total Cost: The total industry and NRC cost associated with the possible solution to this issue was estimated to be $(100 + 13.4)M or $113.4M.

Value/Impact Assessment

Based on a potential public risk reduction of 23,000 man-rem and an estimated cost of $113.4M for a possible solution, the value/impact score was given by:

$$S = \frac{23{,}000 \text{ man} - \text{rem}}{\$113.4M}$$
$$= 200 \text{ man} - \text{rem} / \$M$$

Uncertainties

(1) The estimates of the transient frequency reductions were subject to many assumptions which themselves are uncertain.

(2) Cost estimates were extremely hard to calculate without a clearer fix in mind.

(3) NRC review time would also vary based on the actual fix involved.

Other Considerations

(1) A significant industry cost saving (which would outweigh the industry cost) could be calculated based on a saving in plant outage time resulting from improved non-1E system reliability. For example, if it were assumed that non-loss of offsite power transients would be reduced from 7 to 6.58/RY with a loss of one day of power generation per transient, then unscheduled outages would be reduced by 0.42 day/RY. Based on a replacement power cost of $300,000/day, the cost savings would be (0.42 day/RY)($300,000/day) or $130,000/RY. For 134 plants with a remaining lifetime of 30 years, the total cost savings would be (134 plants)(30 years)($130,000/RY) or $523M.

(2) A draft of IEEE P-827, "A Method for Determining Requirements for Instrumentation, Control and Electrical Systems Important to Safety," was issued.

(3) RES was in the process of developing a draft regulatory guide for the classification of systems important to safety that would provide for a Class 2E instrumentation, control, and electrical power system and equipment. This effort was proceeding independently of the IEEE/ANS efforts.

CONCLUSION

Based on the favorable value/impact score, the effort expended up to the time of the above analysis, and the potential risk reduction and cost saving, this issue was given a medium priority ranking. However, after further evaluation, it was reclassified as a Licensing Issue based on the continuation of the staff's support of the IEEE efforts to develop a standard to define requirements for equipment and systems that are not safety-related, but are sufficiently important to safety to warrant special consideration.[1105]

The Draft Trial Use Guide P-827 was developed by IEEE but was never published; the project was withdrawn in 1983. Under a separate activity, BNL, under contract with the NRC, attempted to develop a methodology to address the classification issue. In both instances, these activities were terminated due to a lack of agreement on the scope and content of the issue.

In 1989, the IEEE/NPEC Working Group SC 6.2 continued to develop a Position Paper on this issue that would only address the possible benefits of establishing a graduated classification program and would provide a list of attributes that would be prudent to incorporate into such a program. However, the Position Paper was not expected to establish any specific guidelines for an acceptable program.

Based on the lack of new plants being constructed, the industry's reluctance to change their existing classification documentation, and the previous efforts both by the NRC staff and the industry to develop a classification methodology, the staff concluded that no additional NRC action should be taken. Thus, the issue was resolved.[1187]

REFERENCES

0011.	*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.

0048.	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.
0055.	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]
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0151.	SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]
0161.	NUREG/CR-1250, "Three Mile Island: A Report to the Commission and to the Public," U.S. Nuclear Regulatory Commission, January 1980.
0200.	IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc., 1980.
0233.	Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1972, (Rev. 1) September 1974, (Rev. 2) June 1975, (Rev. 3) February 1976.
0307.	EPRI NP#2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.
0376.	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*]
0704.	NUREG-0460, "Anticipated Transients without Scram for Light Water Reactors," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1978, (Vol. 2) April 1978, (Vol. 3) December 1978, (Vol. 4) March 1980.
1105.	Memorandum for T. Speis from G. Arlotto, "Generic Issues Program," January 14, 1988. [9704160053]
1187.	Memorandum for V. Stello from E. Beckjord, "Closeout of Generic Issue II.F.5, `Classification of Instrumentation, Control and Electrical Equipment,'" May 5, 1989. [8906270390]

Task II.G: Electrical Power (Rev. 1) ( )

The objective of this task was to increase the reliability and diversification of the electrical power supplies for certain safety-related equipment.

ITEM II.G.1: POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATORS

DESCRIPTION

This TMI Action Plan[48] item called for licensees to develop and implement procedures and modifications to upgrade motive and control components to safety-grade criteria. Motive and control components of PORVs and PORV block valves were to have the capability of being supplied either from the offsite power source or from the emergency power source when offsite power was not available. Motive and control power connections to the emergency buses for the PORVs and their associated block valves were to be through devices that had been qualified in accordance with safety-grade requirements. The pressurizer level indication instrument channels were to be powered from the vital instrument buses that had the capability of being supplied either from the offsite power source or from the emergency power source when offsite power was not available.

CONCLUSION

This item was clarified in NUREG-0737[98] and requirements were issued.

REFERENCES

0048.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

Task II.H: TMI-2 Cleanup and Examination (Rev. 3) ( )

The objectives of this task were to: (1) maintain safety and minimize environmental impact of post-accident operation and cleanup of TMI-2; and (2) obtain and factor into regulatory programs safety-related and environmental information from the TMI-2 cleanup.

ITEM II.H.1: MAINTAIN SAFETY OF TMI-2 AND MINIMIZE ENVIRONMENTAL IMPACT

DESCRIPTION

This TMI Action Plan[48] item covered the efforts by NRC to monitor, review, and assess the safety and environmental impact of the post-accident operation, cleanup, and possible recovery operations at TMI-2 to ensure that: (1) reactor safety and reactor building integrity was maintained; (2) environmental impacts were minimized and radiation exposure to workers, the public, and the environment was within regulatory limits and was as low as reasonably achievable (ALARA); and (3) storage and/or disposal of radioactive wastes from cleanup operations were safe. The TMI Program Office (TMIPO) within NRR directed the NRC activities under this task.

NUREG-0698,[198] Rev. 1, was issued in February 1982 and provided an updated chronology of TMI-2 cleanup activity, major milestones, and accomplishments summarized as follows:

(1) In March 1981, the NRC issued NUREG-0683,[199] a Final Programmatic Environmental Impact Statement (PEIS) related to the decontamination and disposal of radioactive wastes resulting from the accident.

(2) In conjunction with the issuance of the PEIS, the NRC also issued a Policy Statement[211] in April 1981 which stated that the cleanup should be expedited consistent with maintaining public health and safety.

(3) In July 1981, a Memorandum of Understanding (Appendix A to NUREG-0698[198]) concerning the removal and disposition of radioactive solid wastes from the cleanup operations was signed by representatives of NRC and DOE.

(4) Cleanup operations were implemented according to the plan. Decontamination of accident-generated water in the auxiliary and fuel handling buildings was completed by mid-1981. Decontamination of accident water located in the reactor building sump and reactor coolant system was initiated in September 1981. Visual examination of the top of the damaged reactor core was performed with the use of a remote miniature TV introduced through control rod drive housing.

This issue was identified in NUREG-0885[210] as one of NRC's highest safety priorities.

CONCLUSION

The cleanup operation was implemented[377] and the issue was programmatically RESOLVED with appropriate management resources and priorities assigned; no new requirements were established. In NUREG/CR-5382,[1563] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

ITEM II.H.2: OBTAIN TECHNICAL DATA ON THE CONDITIONS INSIDE THE TMI-2 CONTAINMENT STRUCTURE

DESCRIPTION

Pertinent technical information was to be obtained on the conditions of the TMI-2 facility as cleanup operations proceeded. The information to be gathered and disseminated (Item II.H.3) was divided into two distinct categories: (1) data to be obtained prior to gaining access to the primary system; and (2) data to be obtained after access to the primary system. In the first category, information was to be obtained on: (1) instrumentation and electrical equipment survivability under the accident conditions; (2) environmental conditions in the containment and auxiliary buildings; (3) fission-product release, transport, and deposition; (4) decontamination, dose reduction, and waste handling; and (5) debris in the containment building, in particular the containment sump.

After access to the primary system was obtained, the primary system pressure boundary was to be characterized including the steam generators, pumps, and other mechanical and structural components. Techniques were to be developed for a non-destructive assay of fuel distribution in the primary system for assessing criticality control during examination and cleanup operations and for fuel removal, packaging, shipment, and disposal. Detailed pre-access reactor and core damage assessments were to be made followed by careful in situ and away-from-site fuel and reactor internals examinations.

The societal risk from the operation of nuclear power plants would not be reduced by just obtaining, preserving, and disseminating information as outlined above. However, the potential for risk reduction due to proper use of increased knowledge obtained by studying the TMI-2 facility cannot be denied. The information that could be obtained through this item was to be used in the pursuance of other safety issues such as:

A-45 Shutdown Decay Heat Removal Requirements

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

II.B.5 Research on Phenomena Associated with Core Degradation on Fuel Melting

II.B.7 Analysis of Hydrogen Control

II.B.8 Rulemaking Proceedings on Degraded Core Accidents

II.E.3.4 Alternate Decay Heat Removal Concepts

Insights gained from the above TMI-2 information were assumed in a qualitative sense in the development of the potential risk reduction for the 6 issues outlined above. The total risk reduction estimated for the resolution of these 6 issues was 610,000 man-rem of public exposure and 650,000 man-rem of occupational exposure; to include further potential risk reduction under Item II.H.2 (and II.H.3) would result in double-counting.

At the time this issue was evaluated, it was assumed that the TMI-2 cleanup was about 40% complete, about 60% of the $1.2 Billion licensee estimated cost remained to be expended, and about 10% of the licensee's costs was consumed in the preservation and recording of technical data. It was estimated that there was $72M of licensee funding yet to be expended on this effort. Using the TMI Action Plan[48] cost and manpower estimates and extrapolating through FY-1985, it was determined that the NRC cost would be about $36M, of which, about 60% or $22M had not yet been expended. It was also assumed that a DOE commitment of approximately $22M had yet to be expended. This resulted in a total future cost of about $116M for the completion of Items II.H.2 and II.H.3.

Table II.H.2-1 shows the estimated risk reduction, cost, and recommended priority for each of the above 6 issues. The total future costs estimated for all 6 issues was approximately $2 Billion. The total future cost for completion of Items II.H.2 and II.H.3, although large ($116M), was a reasonably small portion (~6%) of total future costs expected for the resolution of those safety issues that will utilize information obtained from the TMI-2 facility. If the cost associated with these items was compared only with the estimated total cost for resolving Issues A-45 and A-48, the cost of the TMI information retrieval program would represent only about 15% of the cost of these two issues. Compared to Items II.B.5 and II.B.8, the cost of the TMI information retrieval program represented less than 10% of the estimated cost for the completion and implementation of Items II.H.2 and II.H.3.

Table II.H.2-1

Issue	Recommended Priority	Risk Reduction (Man-Rem)	Total Cost ($M)
A-45*	High	4.7×10^5	500
A-48*	High	5.2×10^5	208
II.B.5	High	2.2×10^5	1,300
II.B.7	(Subsumed in A-48)	-	-
II.B.8	(Subsumed in II.B.5)	-	-
II.E.3.4	(Subsumed in A-45)	-	-

Issue	Recommended Priority	Risk Reduction (Man-Rem)	Total Cost ($M)
TOTAL		1.2×10^6	2,008

* Unreleased Draft Analyses

CONCLUSION

This issue addressed the collection (Item II.H.2) and dissemination (Item II.H.3) of information that was to be used in the completion of other specific safety issues and thus was not analyzed separately. However, examination of the recommended priority for those issues that depended in part on input from the TMI information to be obtained via this issue indicated that this issue supported other high priority issues. Thus, this issue was given a high priority (See Appendix C).

Core examinations indicated that a large flow of molten material (about 19 metric tons) relocated into the lower plenum after the accident had been in progress for about 225 minutes. All vessel steel, nozzle, and guide tube samples extracted from TMI-2 were tested and analyses of the potential reactor vessel failure modes were conducted. The staff's findings were forwarded to the Commission in SECY-93-119.[1539] Thus, this issue was RESOLVED with no new requirements.[1540] Consideration of a 20-year license renewal period would not affect this resolution.

ITEM II.H 3: EVALUATE AND FEEDBACK INFORMATION OBTAINED FROM TMI-2

DESCRIPTION

This TMI Action Plan[48] item involved the analysis of data obtained during the examination of systems inside the containment building at TMI-2, the subsequent decontamination and restoration of the facility, and the feedback of the information obtained into other appropriate regulatory programs. Item II.H.2 was devoted to the efforts necessary to acquire and record information during the cleanup of the TMI-2 facility.

CONCLUSION

Since the acquisition of the TMI-2 data had to be accomplished before the data could be evaluated, no changes in requirements could be ascertained until those data were evaluated. Therefore, Items II.H.2 and II.H.3 were inextricable and were combined and evaluated together under Item II.H.2.

ITEM II.H.4: DETERMINE IMPACT OF TMI ON SOCIOECONOMIC AND REAL PROPERTY VALUES

DESCRIPTION

Studies were to be conducted on: (1) the effect of the TMI accident on the value of real property in the Harrisburg, Pennsylvania, area; and (2) the socioeconomic impact of the TMI accident on the region in south-central Pennsylvania which surrounds TMI. This item was initiated to increase the NRC knowledge in assessing levels of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

A Pennsylvania State University study[313] of the effects of the accident on property values in the vicinity of the TMI-2 site was accepted by the staff and published in March 1981. A study of the socioeconomic effects of the accident in the region surrounding the plants was performed by Mountain West Research Incorporated. This report[314] was accepted by the staff and published in July 1982. Thus, this Licensing Issue was resolved.

REFERENCES

0048.	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.
0198.	NUREG#0698, "NRC Plans for Cleanup Operations at Three Mile Island Unit 2," U.S. Nuclear Regulatory Commission, July 1980.
0199.	NUREG#0683, "Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979

	Accident at Three Mile Island Nuclear Station, Unit 2," U.S. Nuclear Regulatory Commission, March 1981.
0210.	NUREG#0885, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission, (Issue 1) January 1982, (Issue 2) January 1983, (Issue 3) January 1984, (Issue 4) February 1985, (Issue 5) February 1986, (Issue 6) September 1987.
0211.	*Federal Register* Notice 46 FR 764, "NRC Policy Statement on Cleanup of the Three Mile Island Plant," May 1, 1981.
0313.	NUREG/CR#2063, "Effects of the Accident of Three Mile Island on Property Values and Sales," U.S. Nuclear Regulatory Commission, March 1981.
0314.	NUREG/CR-2749, "Socioeconomic Impacts of Nuclear Generating Stations—Three Mile Island Case Study," U.S. Nuclear Regulatory Commission, (Vol. 12) July 1982.
0377.	Memorandum for W. Minners from B. Snyder, "Schedule for Resolving and Completing Generic Issues," December 16, 1982. [8312290162]
1539.	SECY-93-119, "TMI-2 Vessel Investigation Project," U.S. Nuclear Regulatory Commission, May 5, 1993. [9305100253]
1540.	Memorandum for J. Taylor from E. Beckjord, "Closure of Generic Issue II.H.2, 'Obtain Data on Conditions Inside TMI-2 Containment,'" February 9, 1994. [9502070304]
1563.	NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

Task II.J: General Implications of TMI for Design and Construction Activities (Rev. 1) ( )

TASK II.J.1: VENDOR INSPECTION PROGRAM

The objective of this task was to improve vendor-supplied components and services through a modified and more effective vendor inspection program.

ITEM II.J.1.1: ESTABLISH A PRIORITY SYSTEM FOR CONDUCTING VENDOR INSPECTIONS

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to develop an integrated information system to establish priorities for selecting vendors for inspection in order to permit optimum utilization of available resources. Priorities were to be based on the relative safety significance of products and services provided by the vendors. The information necessary to establish the priorities was to be collected and integrated from LERs, deficiency reports from holders of construction permits and non-licensees, and other relevant information. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

A contract study, "Development of the Automated Vendor Selection System," was completed by Gasser Associates, Inc. on June 30, 1980, and was reviewed by OIE. Changes in the vendor selection and inspection procedures that were considered appropriate were incorporated into the OIE Manual, Chapter 2700, in July 1981. Thus, all required action on this item was completed[235, 248, 379, 406] and the issue was resolved with changes in NRC procedures that address vendor selection and inspection.

ITEM II.J.1.2: MODIFY EXISTING VENDOR INSPECTION PROGRAM

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to improve existing vendor inspection procedures by including more routine technical assessments of products, by expanding the scope to reflect operational and construction feedback experience, and by placing greater emphasis on design control and the use of independent measurements. Full implementation of the expanded scope of this program required an increase in vendor inspection staff. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Chapter 2700 of the OIE Manual, which described the overall licensee contractor and vendor inspection program, was revised to incorporate the fundamental changes defined by this item.[259, 297] With respect to staffing, additional positions for the vendor inspection program were authorized and, by November 1983, 26 people were performing vendor inspection functions. The program changes required by this item were incorporated into the routine ongoing vendor inspection program. Detailed inspection procedures covering these program activities were prepared as the needs of the program were identified. Thus, all required action on this item was completed[297, 379] and the issue was resolved with changes to NRC procedures that address licensee vendor inspection programs.

ITEM II.J.1.3: INCREASE REGULATORY CONTROL OVER PRESENT NON-LICENSEES

DESCRIPTION

This TMI Action Plan[48] item required the NRC to study the need to extend its licensing authority over vendors who supply components and services to licensees. Nuclear steam system suppliers, architect/engineers, constructors, and designated vendors were to be included in the study. Upon completion of the study, the staff was to present a paper to the Commission for a decision on the subject. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

As part of the resolution of Item II.J.4.1, OIE submitted to RES recommended changes to 10 CFR 21 that would revise deficiency reporting requirements for NSSS vendors, A/E firms, and others. These revised deficiency reporting requirements would provide increased information on component failures that affect safety, so that prompt and effective corrective action could be taken. OIE stated[235] that further extension of NRC authority over non-licensees with licensing requirements was not warranted and would not be cost-effective. In light of the proposed rule change, all required action was completed[379] and the issue was resolved.

ITEM II.J.1.4: ASSIGN RESIDENT INSPECTORS TO REACTOR VENDORS AND ARCHITECT ENGINEERS

DESCRIPTION

This TMI Action Plan[48] item required the NRC to evaluate the desirability of assigning resident inspectors to NSSS vendors and A/Es. The staff was to prepare a Commission Paper describing a proposed trial program to be applied to selected NSSS vendors and A/Es. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

The proposal to assign resident inspectors to NSSS vendors and A/Es as a part of the vendor inspection program was reviewed by the staff who concluded[235, 268] that such a program should not be initiated. It was further recommended[379] that the item be deleted for the following reasons[268]: (1) more effective utilization of existing vendor inspection resources could be obtained by retaining inspectors in the regional offices; (2) the absence of new orders resulted in significant changes in NSSS and A/E work activity, in that more sub-contracting to numerous small firms was occurring; (3) to provide inspection coverage of the activities required greater mobility and flexibility from the vendor inspection staff; and (4) the trial program would require resources that were not available. Based on these recommendations, the issue was resolved March 14, 1982.

TASK II.J.2: CONSTRUCTION INSPECTION PROGRAM (REV. 1)

The objective of this task was to provide greater assurance that nuclear plants are properly constructed by improving construction inspection programs.

ITEM II.J.2.1: REORIENT CONSTRUCTION INSPECTION PROGRAM

DESCRIPTION

This TMI Action Plan[48] item called for OIE to change its reactor construction inspection program and its Inspection Manual to require increased observation of work activities, more attention to the involvement of licensees in construction activities, independent verification that as-built conditions met design requirements, and followup of reported incident information, as applicable, from operating reactors. This item addressed the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Chapter 2512 of the Inspection Manual was revised on August 1, 1980, as part of the OIE program to incorporate increased observation of work activities and to increase inspection of licensees' involvement in the overall construction of plants. In addition, program changes to ensure earlier and continuing inspection of construction QA activities were made. A trial program involving team inspections was also completed. Thus, this issue was resolved with changes in the NRC procedures that address construction inspection.[235, 239, 379, 406]

II.J.2.2: INCREASE EMPHASIS ON INDEPENDENT MEASUREMENT IN CONSTRUCTION INSPECTION PROGRAM

DESCRIPTION

This TMI Action Plan[48] item called for OIE to evaluate trial programs involving independent measurements (non-destructive examination) at construction sites. NRC was to buy a van to be fitted with equipment to conduct ultrasonic, liquid penetrant, and magnetic particle non-destructive examinations. If the evaluations were successfully made from the equipment-fitted van, additional vans were to be purchased for use at each Regional Office. In addition, a contract was awarded to the Franklin Research Center to provide services involving independent assessment (destructive testing) of material samples. Data from these assessments were to supplement the testing to further verify conformance with licensee commitments, specifications and/or codes and standards requirements. Five uniquely qualified inspectors were to be assigned full-time to each van to ensure maximum use of the vans. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

A contractor for destructive testing was hired and tests were performed on an ongoing basis. An NRC mobile van was purchased, equipped, and staffed with contractor assistance. The original plan called[235] for the staff to evaluate a trial program involving independent measurements at construction sites and then, based upon the results of the trial program, equip each region with the capability and equipment necessary to conduct independent measurements on a routine basis. The trial program was a success; however, based on budgetary constraints, a cutback in the effort was necessitated. OIE recommended a modified scope of the item so that the effort was limited to purchasing one van which would be available to all five regions. Personnel to utilize van equipment were supplied by an NRC contractor. This eliminated the need to hire additional full-time personnel and to provide a training program necessary to maintain personnel competency in NDE disciplines.

This issue was resolved when the scope of the action plan was revised and the program of independent measurements was incorporated into routine NRC operations.[379] Followup was to be performed via routine programmatic action, and further expansion was to be based on continuing OIE appraisal of the program's effectiveness.

ITEM II.J.2.3: ASSIGN RESIDENT INSPECTORS TO ALL CONSTRUCTION SITES

DESCRIPTION

This TMI Action Plan[48] item called for OIE to expand the resident inspector program to include one inspector at each power plant construction site. Previous experience had shown the need for inspection at all stages of construction. This conclusion contradicted earlier criteria that delayed the assignment of resident inspectors to a plant site until 50% of the construction was completed. Schedules and resources for assigning resident inspectors to construction sites were to be developed in connection with routine agency budgetary processes. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

OIE assigned resident inspectors to all active construction sites that were greater than 15% complete.[235] In November 1983, there were 23 resident inspectors at various construction sites. This item was developed as part of the routine program for NRC operators and was resolved when it was decided that future specific allocation of resources in this inspection program would be reevaluated as part of the annual budget process.[379]

TASK II.J.3: MANAGEMENT FOR DESIGN AND CONSTRUCTION (REV. 1)

The objective of this task was to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design, construction, and modification activities.

ITEM II.J.3.1: ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to require "license applicants and licensees to improve the oversight of design, construction, and modification activities so that they will gain the critical expertise necessary for the safe operation of the plant."

CONCLUSION

The criteria and regulatory guidelines for this issue were addressed and developed by DHFS/NRR as a part of Item I.B.1.1. Therefore, this issue was covered in Item I.B.1.1.

ITEM II.J.3.2: ISSUE REGULATORY GUIDE

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to issue a Regulatory Guide to codify the criteria relating to organization and staffing to oversee design and construction (Item II.J.3.1).

CONCLUSION

This item required the utilization of criteria developed from Item II.J.3.1. Therefore, this item was evaluated together with Item II.J.3.1 under Item I.B.1.1.

TASK II.J.4: REVISE DEFICIENCY REPORTING REQUIREMENTS (REV. 3)

The objective of this task was to clarify deficiency report requirements to obtain uniform reporting and earlier identification and correction of problems.

ITEM II.J.4.1: REVISE DEFICIENCY REPORTING REQUIREMENTS

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to revise, as necessary, the event-reporting requirements of 10 CFR 21 to assure that all reportable items are reported promptly and that the information submitted is complete. Improvements were to be implemented by rule changes, as appropriate, and coordinated with those made under TMI Action Plan Item I.E.6. The reports received as a result of these rule changes were to provide increased information on component failures that affect safety so that prompt and effective corrective action could be taken. The information was also to be used as input to an augmented role of the NRC's vendor and construction inspection program.

CONCLUSION

This issue was originally classified as nearly-resolved, based on changes to 10 CFR 21 and 10 CFR 50.55(e) proposed by OIE,[291, 292] and was later RESOLVED with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued.[1396] The staff's changes were presented to the Commission in SECY-91-150.[1397] In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

REFERENCES

0048.	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.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0239.	Memorandum for W. Dircks from V. Stello, "TMI Action Plan—Status Report," December 19, 1980. [8205260193]
0248.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," December 28, 1981. [8205260197]
0259.	Memorandum for J. Sniezek from J. Taylor, "TMI Action Plan Item II.J.1.2, Modification of Vendor Inspection Program," October 13, 1982. [8301050485]
0268.	Memorandum for W. Dircks from V. Stello, "Assignment of Resident Inspectors to Nuclear Steam System Suppliers and Architect-Engineers," September 14, 1981. [8111030559]
0291.	Memorandum for E. Jordan et al. from R. Bernero, "Proposed Rule Review Request—10 CFR Part 21, `Reporting of Defects and Noncompliance,'" September 28, 1982. [8210150634]
0292.	Memorandum for R. Minogue from R. DeYoung, "Proposed Rule Amending 10 CFR Parts 50.55(e) and 21: RES Task Numbers RA 128#1 and RA 808#1," July 13, 1982.
0297.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Item," October 29, 1982. [8401170104]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]

0406.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Status Report," March 4, 1982. [8204290601]
1396.	*Federal Register* Notice 56 FR 36081, "10 CFR Parts 21 and 50, Criteria and Procedures for the Reporting of Defects and Conditions of Construction Permits," July 31, 1991.
1397.	SECY-91-150, "Proposed Amendments to 10 CFR Part 21, `Reporting of Defects and Noncompliance' and 10 CFR 50.55(e), `Conditions of Construction Permits,'" U.S. Nuclear Regulatory Commission, May 22, 1991. [9106040262]
1564.	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]

Task II.J.2: Construction Inspection Program (Rev. 1) ( )

The objective of this task was to provide greater assurance that nuclear plants are properly constructed by improving construction inspection programs.

ITEM II.J.2.1: REORIENT CONSTRUCTION INSPECTION PROGRAM

DESCRIPTION

This TMI Action Plan[48] item called for OIE to change its reactor construction inspection program and its Inspection Manual to require increased observation of work activities, more attention to the involvement of licensees in construction activities, independent verification that as-built conditions met design requirements, and followup of reported incident information, as applicable, from operating reactors. This item addressed the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

Chapter 2512 of the Inspection Manual was revised on August 1, 1980, as part of the OIE program to incorporate increased observation of work activities and to increase inspection of licensees' involvement in the overall construction of plants. In addition, program changes to ensure earlier and continuing inspection of construction QA activities were made. A trial program involving team inspections was also completed. Thus, this issue was resolved with changes in the NRC procedures that address construction inspection.[235, 239, 379, 406]

II.J.2.2: INCREASE EMPHASIS ON INDEPENDENT MEASUREMENT IN CONSTRUCTION INSPECTION PROGRAM

DESCRIPTION

This TMI Action Plan[48] item called for OIE to evaluate trial programs involving independent measurements (non-destructive examination) at construction sites. NRC was to buy a van to be fitted with equipment to conduct ultrasonic, liquid penetrant, and magnetic particle non-destructive examinations. If the evaluations were successfully made from the equipment-fitted van, additional vans were to be purchased for use at each Regional Office. In addition, a contract was awarded to the Franklin Research Center to provide services involving independent assessment (destructive testing) of material samples. Data from these assessments were to supplement the testing to further verify conformance with licensee commitments, specifications and/or codes and standards requirements. Five uniquely qualified inspectors were to be assigned full-time to each van to ensure maximum use of the vans. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

A contractor for destructive testing was hired and tests were performed on an ongoing basis. An NRC mobile van was purchased, equipped, and staffed with contractor assistance. The original plan called[235] for the staff to evaluate a trial program involving independent measurements at construction sites and then, based upon the results of the trial program, equip each region with the capability and equipment necessary to conduct independent measurements on a routine basis. The trial program was a success; however, based on budgetary constraints, a cutback in the effort was necessitated. OIE recommended a modified scope of the item so that the effort was limited to purchasing one van which would be available to all five regions. Personnel to utilize van equipment were supplied by an NRC contractor. This eliminated the need to hire additional full-time personnel and to provide a training program necessary to maintain personnel competency in NDE disciplines.

This issue was resolved when the scope of the action plan was revised and the program of independent measurements was incorporated into routine NRC operations.[379] Followup was to be performed via routine programmatic action, and further expansion was to be based on continuing OIE appraisal of the program's effectiveness.

ITEM II.J.2.3: ASSIGN RESIDENT INSPECTORS TO ALL CONSTRUCTION SITES

DESCRIPTION

This TMI Action Plan[48] item called for OIE to expand the resident inspector program to include one inspector at each power plant construction site. Previous experience had shown the need for inspection at all stages of construction. This conclusion contradicted earlier criteria that delayed the assignment of resident inspectors to a plant site until 50% of the construction was completed. Schedules and resources for assigning resident inspectors to construction sites were to be developed in connection with routine agency budgetary processes. This item addressed improvement in the NRC capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.

CONCLUSION

OIE assigned resident inspectors to all active construction sites that were greater than 15% complete.[235] In November 1983, there were 23 resident inspectors at various construction sites. This item was developed as part of the routine program for NRC operators and was resolved when it was decided that future specific allocation of resources in this inspection program would be reevaluated as part of the annual budget process.[379]

REFERENCES

0048.	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.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0239.	Memorandum for W. Dircks from V. Stello, "TMI Action Plan—Status Report," December 19, 1980. [8205260193]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0406.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Status Report," March 4, 1982. [8204290601]

Task II.J.3: Management for Design and Construction (Rev. 1) ( )

The objective of this task was to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design, construction, and modification activities.

ITEM II.J.3.1: ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to require "license applicants and licensees to improve the oversight of design, construction, and modification activities so that they will gain the critical expertise necessary for the safe operation of the plant."

CONCLUSION

The criteria and regulatory guidelines for this issue were addressed and developed by DHFS/NRR as a part of Item I.B.1.1. Therefore, this issue was covered in Item I.B.1.1.

ITEM II.J.3.2: ISSUE REGULATORY GUIDE

DESCRIPTION

The purpose of this TMI Action Plan[48] item was to issue a Regulatory Guide to codify the criteria relating to organization and staffing to oversee design and construction (Item II.J.3.1).

CONCLUSION

This item required the utilization of criteria developed from Item II.J.3.1. Therefore, this item was evaluated together with Item II.J.3.1 under Item I.B.1.1.

REFERENCES

0048.	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.

Task II.J.4: Revise Deficiency Reporting Requirements (Rev. 3) ( )

The objective of this task was to clarify deficiency report requirements to obtain uniform reporting and earlier identification and correction of problems.

ITEM II.J.4.1: REVISE DEFICIENCY REPORTING REQUIREMENTS

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to revise, as necessary, the event-reporting requirements of 10 CFR 21 to assure that all reportable items are reported promptly and that the information submitted is complete. Improvements were to be implemented by rule changes, as appropriate, and coordinated with those made under TMI Action Plan Item I.E.6. The reports received as a result of these rule changes were to provide increased information on component failures that affect safety so that prompt and effective corrective action could be taken. The information was also to be used as input to an augmented role of the NRC's vendor and construction inspection program.

CONCLUSION

This issue was originally classified as nearly-resolved, based on changes to 10 CFR 21 and 10 CFR 50.55(e) proposed by OIE,[291, 292] and was later RESOLVED with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued.[1396] The staff's changes were presented to the Commission in SECY-91-150.[1397] In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

REFERENCES

0048.	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.
0291.	Memorandum for E. Jordan et al. from R. Bernero, "Proposed Rule Review Request—10 CFR Part 21, `Reporting of Defects and Noncompliance,'" September 28, 1982. [8210150634]
0292.	Memorandum for R. Minogue from R. DeYoung, "Proposed Rule Amending 10 CFR Parts 50.55(e) and 21: RES Task Numbers RA 128#1 and RA 808#1," July 13, 1982.
1396.	*Federal Register* Notice 56 FR 36081, "10 CFR Parts 21 and 50, Criteria and Procedures for the Reporting of Defects and Conditions of Construction Permits," July 31, 1991.
1397.	SECY-91-150, "Proposed Amendments to 10 CFR Part 21, `Reporting of Defects and Noncompliance' and 10 CFR 50.55(e), `Conditions of Construction Permits,'" U.S. Nuclear Regulatory Commission, May 22, 1991. [9106040262]
1564.	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]

Task II.K: Measures to Mitigate Small-Break Loss-Of-Coolant Accidents and Loss-Of-Feedwater Accidents (2)

The objectives of this task were to perform systems reliability analyses and to effect changes in emergency operating procedures and operator training to improve the capability of plants to mitigate the consequences of the small-break LOCAs and loss-of-feedwater events.

ITEM II.K.1: IE BULLETINS

Between April 1, 1979 and July 26, 1979, OIE issued 9 bulletins to various operating plants, depending on their reactor design, and a review of the affected licensee responses was conducted by the NRR Bulletins and Orders Task Force (BOTF). The responses were determined to be acceptable and separate evaluation reports were prepared and issued to some licensees. Thus, prior to the publication of NUREG-0660,[48] several parts of this item were either com-pleted or found to be covered in other TMI Action Plan items. This status was reported in Table C.1 of NUREG-0660.[48] The following is a summary of the 28 parts of this item.

ITEM II.K.1(1): REVIEW TMI-2 PNS AND DETAILED CHRONOLOGY OF THE TMI-2 ACCIDENT

DESCRIPTION

This NUREG-066048 item affected all OLs and was originated to effect short-term changes in emergency operating procedures and operator training in order to improve the capability of plants to mitigate the likelihood and consequences of SBLOCAs and loss-of-feedwater events. For all OL applicants, this item was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(2): REVIEW TRANSIENTS SIMILAR TO TMI-2 THAT HAVE OCCURRED AT OTHER FACILITIES AND NRC EVALUATION OF DAVIS-BESSE EVENT

DESCRIPTION

This NUREG-066048 item affected all B&W operating plants. For OL applicants with B&W reactors, this item was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(3): REVIEW OPERATING PROCEDURES FOR RECOGNIZING, PREVENTING, AND MITIGATING VOID FORMATION IN TRANSIENTS AND ACCIDENTS

DESCRIPTION

This NUREG-066048 item affected all operating PWRs. For OL applicants with PWRs, it was determined that the issue was covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(4): REVIEW OPERATING PROCEDURES AND TRAINING INSTRUCTIONS

DESCRIPTION

This NUREG-066048 item was divided into 4 parts to ensure: (a) that operators do not override ESF actions unless continued operation is unsafe; (b) HPI system operation; (c) RCP operation; and (d) that operators are instructed not to rely on level indication alone in evaluating plant conditions.

- Part (a) affected all operating plants. However, for all OL applicants it was determined that this part was covered by Items I.C.1, I.C.7, I.G.1, and I.C.8.

- Part (b) affected all **W**, CE and B&W operating plants with specific requirements issued to ANO-1; Davis-Besse 1; Oconee 1, 2, and 3; Crystal River 3; and Rancho Seco. For OL applicants with **W**, CE, or B&W reactors, it was determined that this part was covered by Item I.C.1.
- Part (c) affected all PWRs and was completed by OLs prior to the publication of NUREG-0660.48 For OL applicants with PWRs, it was determined that this part was covered by Items I.C.1 and I.A.1.3.
- Part (d) affected all plants and was completed by OLs prior to the publication of NUREG-0660.48 For all OL applicants, it was determined that this part was covered by Items I.C.1, I.A.3.1, and II.F.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(5): SAFETY-RELATED VALVE POSITION DESCRIPTION

DESCRIPTION

This NUREG-066048 item was divided into 2 parts and required plants to: (a) review all valve positions and positioning requirements and positive controls along with all related test and and maintenance procedures to assure proper ESF functioning, if required; and (b) verify that AFW valves are in the open position.

- Part (a) affected all operating plants. For all OL applicants, it was determined that this part was covered by Items I.C.2 and I.C.6.
- Part (b) affected all B&W operating plants. For OL applicants with B&W reactors, this part was also determined to be covered by Items I.C.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(6): REVIEW CONTAINMENT ISOLATION INITIATION DESIGN AND PROCEDURES

DESCRIPTION

This NUREG-066048 item affected all operating plants and was initiated to assure isolation of all lines that do not degrade safety features or cooling capability upon automatic initiation of SI. For all OL applicants, it was determined that this issue was covered by Item II.E.4.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(7): IMPLEMENT POSITIVE POSITION CONTROLS ON VALVES THAT COULD COMPROMISE OR DEFEAT AFW FLOW

DESCRIPTION

This NUREG-0660[48] item affected all B&W operating plants. For OL applicants with B&W reactors, this issue was determined to be covered by Item II.E.1.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(8): IMPLEMENT PROCEDURES THAT ASSURE TWO INDEPENDENT 100% AFW FLOW PATHS

DESCRIPTION

This NUREG-0660[48] item required all operating B&W plants to immediately implement procedures that assure two independent 100% AFW flow paths or specify explicitly LCO with reduced AFW capacity. For OL applicants with B&W reactors, this issue was determined to be covered by Item II.E.1.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(9): REVIEW PROCEDURES TO ASSURE THAT RADIOACTIVE LIQUIDS AND GASES ARE NOT TRANSFERRED OUT OF CONTAINMENT INADVERTENTLY

DESCRIPTION

This NUREG-0660[48] item required all operating plants to review their procedures to assure that radioactive liquids and gases are not transferred out of containment inadvertently, especially upon ESF reset. All applicable systems and interlocks were required to be listed. For OL applicants, this item was determined to be covered by Items II.E.4.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(10): REVIEW AND MODIFY PROCEDURES FOR REMOVING SAFETY-RELATED SYSTEMS FROM SERVICE

DESCRIPTION

This NUREG-0660[48] item required all operating plants to review and modify (as required) their procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. For OL applicants, the issue was determined to be covered by Items I.C.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(11): MAKE ALL OPERATING AND MAINTENANCE PERSONNEL AWARE OF THE SERIOUSNESS AND CONSEQUENCES OF THE ERRONEOUS ACTIONS LEADING UP TO, AND IN EARLY PHASES OF, THE TMI-2 ACCIDENT

DESCRIPTION

This NUREG-0660[48] item affected all operating plants. For OL applicants, the issue was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(12): ONE HOUR NOTIFICATION REQUIREMENT AND CONTINUOUS COMMUNICATIONS CHANNELS

DESCRIPTION

This NUREG-0660[48] item affected all operating plants. For OL applicants, the issue was determined to be covered by Items I.E.6 and III.A.3.3.

CONCLUSION

This item was RESOLVED and requirements were issued.

Item II.K.1(13): Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items

DESCRIPTION

This NUREG-0660[48] item required all operating plants to propose TS changes reflecting implementation of all Bulletin items, as required.

CONCLUSION

This item was RESOLVED and requirements were issued.

Item II.K.1(14): Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen

DESCRIPTION

This NUREG-0660[48] item affected all operating plants with **W**, CE, and GE reactors.

For OL applicants with **W**, CE and GE reactors, it was determined that the issue was covered by Items II.B.4, II.B.7, II.E.4.1, and II.F.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

Item II.K.1(15): For Facilities with Non-automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW

DESCRIPTION

This NUREG-066048 item affected all operating plants with **W** and CE reactors. However, prior to the publication of NUREG-0660,48 all necessary action was completed by the affected OLs. For OL applicants with **W** and CE reactors, it was determined that the issue was covered by Item II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(16): IMPLEMENT PROCEDURES THAT IDENTIFY PRZPORV "OPEN" INDICATIONS AND THAT DIRECT OPERATOR TO CLOSE MANUALLY AT "RESET" SETPOINT

DESCRIPTION

This NUREG-066048 item affected all operating plants with **W** and CE reactors. However, prior to the publication of NUREG-0660,48 all necessary action was completed by the affected OLs. For OL applicants with **W** and CE reactors, it was determined that the issue was covered by Items I.C.1 and II.D.3.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(17): TRIP PZR LEVEL BISTABLE SO THAT PZR LOW PRESSURE WILL INITIATE SAFETY INJECTION

DESCRIPTION

This NUREG-066048 item required all OLs and OL applicants with **W** reactors to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection. For testing, the plants were required to reset the low level bistable. However, prior to the publication of NUREG-0660,48 all necessary action was completed by the affected OLs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(18): DEVELOP PROCEDURES AND TRAIN OPERATORS ON METHODS OF ESTABLISHING AND MAINTAINING NATURAL CIRCULATION

DESCRIPTION

This NUREG-066048 item affected all operating B&W plants. However, prior to the publication of NUREG-0660,48 all necessary action was completed by the affected plants. For OL applicants with B&W reactors, it was determined that the issue was covered by Items I.C.1 and I.G.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(19): DESCRIBE DESIGN AND PROCEDURE MODIFICATIONS TO REDUCE LIKELIHOOD OF AUTOMATIC PZR PORV ACTUATION IN TRANSIENTS

DESCRIPTION

This NUREG-066048 item required all operating B&W plants to describe their design and procedure modifications (based on analysis) to reduce the likelihood of automatic pressurizer PORV actuation in transients. For OL applicants with B&W reactors, it was determined that the issue was covered by Item II.E.5.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(20): PROVIDE PROCEDURES AND TRAINING TO OPERATORS FOR PROMPT MANUAL REACTOR TRIP FOR LOFW, TT, MSIV CLOSURE, LOOP, LOSG LEVEL, AND LO PZR LEVEL

DESCRIPTION

This NUREG-066048 item affected all OLs and OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(21): PROVIDE AUTOMATIC SAFETY-GRADE ANTICIPATORY REACTOR TRIP FOR LOFW, TT, OR SIGNIFICANT DECREASE IN SG LEVEL

DESCRIPTION

This NUREG-066048 item affected all OLs and OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(22): DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM NOT OPERABLE

DESCRIPTION

This NUREG-066048 item affected all OLs and OL applicants with BWRs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(23): DESCRIBE USES AND TYPES OF RV LEVEL INDICATION FOR AUTOMATIC AND MANUAL INITIATION SAFETY SYSTEMS

DESCRIPTION

This NUREG-066048 item required all OLs and OL applicants with BWRs to describe their uses and types of reactor vessel indication for automatic and manual initiation safety systems. The affected plants were also required to describe their alternative instrumentation.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(24): PERFORM LOCA ANALYSES FOR A RANGE OF SMALL-BREAK SIZES AND A RANGE OF TIME LAPSES BETWEEN REACTOR TRIP AND RCP TRIP

DESCRIPTION

This NUREG-066048 item affected all operating PWRs. However, prior to the publication of NUREG-0660,48 all necessary action was completed by the affected OLs. For OL applicants with PWRs, the issue was determined to be covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(25): DEVELOP OPERATOR ACTION GUIDELINES

DESCRIPTION

This NUREG-066048 item required all operating PWRs to develop operator action guidelines, based on the analyses performed in response to Item II.K.1(24). However, prior to the publication of NUREG-0660,48 it was determined that all necessary action was completed by the affected plants. For OL applicants with PWRs, the issue was determined to be covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(26): REVISE EMERGENCY PROCEDURES AND TRAIN ROs AND SROs

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to revise their emergency procedures and train ROs and SROs, based on guidelines developed in response to Item II.K.1(25). However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with PWRs, it was determined that the issue was covered by Items I.A.3.1, I.C.1, and I.G.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(27): PROVIDE ANALYSES AND DEVELOP GUIDELINES AND PROCEDURES FOR INADEQUATE CORE COOLING CONDITIONS

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to provide analyses and develop guidelines and procedures for inadequate core cooling conditions. The affected plants were also required to define their RCP restart criteria. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with PWRs, it was determined that the issue was covered by Items I.C.1 and II.F.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(28): PROVIDE DESIGN THAT WILL ASSURE AUTOMATIC RCP TRIP FOR ALL CIRCUMSTANCES WHERE REQUIRED

DESCRIPTION

This NUREG-0660[48] item affected all operating PWRs. For OL applicants with PWRs, it was determined that the issue was covered by Item II.K.3(5).

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2: COMMISSION ORDERS ON BABCOCK AND WILCOX PLANTS

This item contained 21 requirements for 7 operating plants with B&W reactors that were issued confirmatory shutdown orders shortly after the TMI-2 accident. Some of these requirements were also applicable to OL applicants with B&W reactors. These requirements were divided into two groups: short-term actions and long-term actions. The short-term actions were essentially those that were listed in Table 2-1 of NUREG-0645[580] while the long-term actions were also dilineated in NUREG-0645.[580]

However, prior to the publication of NUREG-0660,[48] 10 of these requirements were either completed or found to be covered by other TMI Action Plan items. This status was reported in Table C.2 of NUREG-0660.[48] Since that time, some of the remaining items have been clarified in NUREG-0737[98] and others have been completed. The status of the MPAs established for implementation can be found in NUREG-0748.[578] The following is a summary of the 21 parts of this item.

ITEM II.K.2(1): UPGRADE TIMELINESS AND RELIABILITY OF AFW SYSTEM

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(2): PROCEDURES AND TRAINING TO INITIATE AND CONTROL AFW INDEPENDENT OF INTEGRATED CONTROL SYSTEM

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made prior to the publication of NUREG-0660.[48] This requirement was also applicable to OL applicants with B&W reactors and was clarified in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(3): HARD-WIRED CONTROL-GRADE ANTICIPATORY REACTOR TRIPS

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660.[48] This requirement was not applicable to OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(4): SMALL-BREAK LOCA ANALYSIS, PROCEDURES AND OPERATOR TRAINING

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in September 1979, prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items I.A.3.1 and I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(5): COMPLETE TMI-2 SIMULATOR TRAINING FOR ALL OPERATORS

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.A.2.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(6): REEVALUATE ANALYSIS FOR DUAL-LEVEL SETPOINT CONTROL

DESCRIPTION

Prior to the TMI-2 accident, Toledo Edison Company (TECO) was authorized by the NRC (pending incorporation of permanent design modifications to provide automatic dual setpoint steam generator level control) to manually control steam generator level at 35 in. for all events requiring auxiliary feedwater, unless a safety feature actuation system Level 2 signal occurred. Following the TMI-2 accident, the staff required additional information to verify that the effects of manually controlling steam generator level at 35 in. was adequate for the Davis-Besse plant, in light of the revised small-break LOCA analyses that were performed by B&W after the TMI-2 accident.

The only operating plant affected by this item, Davis-Besse 1, completed this short-term NUREG-0660[48] action before it was permitted to restart. This accomplishment was made in July 1979, prior to the publication of NUREG-0660.[48] This requirement was not applicable to OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(7): REEVALUATE TRANSIENT OF SEPTEMBER 24, 1977

DESCRIPTION

In September 1977, Davis-Besse 1 experienced an event which started out very similar to the one that occurred at TMI-2. In light of the information gained from the TMI-2 accident, the staff felt it was necessary to review the previous evaluation prepared by Toledo Edison Company for the Davis-Besse 1 event which involved equipment problems and depressurization of the primary system.

The only plant affected by this item, Davis-Besse 1, completed this short-term NUREG-066048 action before it was permitted to restart. This accomplishment was made in July 1979, prior to the publication of NUREG-0660.48 This require-ment was not applicable to OL applicants with B&W reactors.

CONCLUSION

The item was RESOLVED and requirements were issued.

ITEM II.K.2(8): CONTINUED UPGRADING OF AFW SYSTEM

DESCRIPTION

All 7 B&W plants with OLs were initially required to complete this long-term NUREG-066048 action. However, a clarification was issued in NUREG-073798 which superseded this item with Items II.E.1.1 and II.E.1.2. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item is covered in Items II.E.1.1 and II.E.1.2.

ITEM II.K.2(9): ANALYSIS AND UPGRADING OF INTEGRATED CONTROL SYSTEM

DESCRIPTION

This NUREG-066048 item called for licensees with B&W reactors to provide a failure mode effects analysis on the integrated control system. All 7 B&W plants with OLs as well as OL applicants with B&W reactors were required to complete this long-term action. A clarification that affected both groups of plants was issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-27 was established by DL for implementation purposes.

ITEM II.K.2(10): HARD-WIRED SAFETY-GRADE ANTICIPATORY REACTOR TRIPS

DESCRIPTION

This NUREG-066048 item called for licensees with B&W reactors to provide a design and schedule for implementation of a safety-grade reactor trip upon loss of feedwater, turbine trip, and significant reduction in steam generator level. These requirements were listed as Item 5 of IE Bulletin 79-05B which was issued on April 21, 1979. All 7 B&W plants with OLs as well as OL applicants with B&W reactors were required to complete this long-term action. Clarifications that affected both groups of plants were issued in NUREG-073798 and OL applicants with B&W reactors were given the option of complying with Item II.K.1 (Part C.1.21) of NUREG-0694579 to satisfy this requirement.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-28 was established by DL for implementation purposes.

ITEM II.K.2(11): OPERATOR TRAINING AND DRILLING

DESCRIPTION

All 7 B&W plants with OLs were required to complete this long-term NUREG-066048 item which called for continued operator training and drilling to assure a high state of preparedness. For the affected OLs, a clarification to the requirement was issued in NUREG-0737.98 For OL applicants with B&W reac-tors, this item was determined to be covered Items I.A.2.2, I.A.2.5, I.A.3.1, and I.G.1 prior to the publication of NUREG-0660.48

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-29 was established by DL for implementation purposes.

ITEM II.K.2(12): TRANSIENT ANALYSIS AND PROCEDURES FOR MANAGEMENT OF SMALL BREAKS

DESCRIPTION

The only operating B&W plant affected by this NUREG-066048 item was Davis-Besse 1. However, prior to the publication of NUREG-0660,48 it was determined that the issue was covered by Item I.C.1 for Davis-Besse 1 and all OL applicants with B&W reactors.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.2(13): THERMAL-MECHANICAL REPORT ON EFFECT OF HPI ON VESSEL INTEGRITY FOR SMALL-BREAK LOCA WITH NO AFW

DESCRIPTION

This item required the affected plants to demonstrate that sufficient mixing of the high pressure injection water would occur with the reactor coolant so that significant thermal shock effects to the reactor vessel would be precluded. All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-066048 item. A clarification was issued in NUREG-073798 to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issed, and MPA F-30 was established by DL for implementation purposes.

ITEM II.K.2(14): DEMONSTRATE THAT PREDICTED LIFT FREQUENCY OF PORVs AND SVs IS ACCEPTABLE

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-066048 item. A clarification affecting both groups of plants was issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-31 was established by DL for implementation purposes.

ITEM II.K.2(15): ANALYSIS OF EFFECTS OF SLUG FLOW ON ONCE-THROUGH STEAM GENERATOR TUBES AFTER PRIMARY SYSTEM VOIDING

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-066048 item which called for the affected plants to assess the loading on steam generator tube sheets induced from slug flow during natural circulation cooldown. A clarification affecting both groups of plants was issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(16): IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-066048 item which called for the investigation of the consequences of losing coolant to the seals of the reactor coolant pumps during loss of offsite power. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-32 was established by DL for implementation purposes.

ITEM II.K.2(17): ANALYSIS OF POTENTIAL VOIDING IN RCS DURING ANTICIPATED TRANSIENTS

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-066048 item which called for the plants to determine the consequence of voiding in the reactor vessel and the hot legs during normal anticipated transients. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-073798 to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-33 was established by DL for implementation purposes.

ITEM II.K.2(18): ANALYSIS OF LOSS OF FEEDWATER AND OTHER ANTICIPATED TRANSIENTS

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors plants were affected by this NUREG-066048 item. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in by Item I.C.1.

ITEM II.K.2(19): BENCHMARK ANALYSIS OF SEQUENTIAL AFW FLOW TO ONCE-THROUGH STEAM GENERATOR

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-066048 item which called for the evaluation of the steam generator model in the small-break licensing code (CRAFT-2) by predicting the Crystal River asymmetric cooldown start-up test. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-073798 to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-34 was established by DL for implementation purposes.

ITEM II.K.2(20): ANALYSIS OF STEAM RESPONSE TO SMALL-BREAK LOCA THAT CAUSES SYSTEM PRESSURE TO EXCEED PORV SETPOINT

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-066048 item which called for the assessment of small-break LOCAs which result in pressurization of the primary system to the PORV setpoint. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1.

A clarification affecting the 7 OLs was issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-35 was established by DL for implementation purposes.

ITEM II.K.2(21): LOFT 3-1 PREDICTIONS DESCRIPTION

DESCRIPTION

The adequacy of B&W's small-break LOCA model needed to be benchmarked against integral systems test data. By performing this pretest prediction of LOFT L3-1, the staff was able to determine this information. All 7 B&W plants were affected by this NUREG-066048 item which was completed in December 1979, prior to the publication of NUREG-0660.48 OL applicants with B&W reactors were not affected by this item.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3: FINAL RECOMMENDATIONS OF BULLETINS AND ORDERS TASK FORCE

This item contained 57 requirements that affected OLs and OL applicants. These requirements were based on recommendations that were developed by the staff and issued in the following reports: NUREG-056596 (B&W reactors), NUREG-061193 (**W** reactors), NUREG-062694 (GE reactors), NUREG-063595 (CE reactors), and NUREG-0623.97 However, prior to the publication of NUREG-0660,48 some of these requirements were superseded by other TMI Action Plan items. This status was reported in Table C.3 of NUREG-0660.48 Since that time, some of the remaining items have been clarified in NUREG-073798 and others have been completed. The status of the MPAs established for implementation can be found in NUREG-0748.578 The following is a summary of the 57 parts of this item.

ITEM II.K.3(1): INSTALL AUTOMATIC PORV ISOLATION SYSTEM AND PERFORM OPERATIONAL TEST

DESCRIPTION

This NUREG-066048 item required all operating PWRs to provide a system that uses the PORV block valve to protect against a small-break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-36 was established by DL for implementation purposes.

ITEM II.K.3(2): REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

DESCRIPTION

This NUREG-066048 item required all operating PWRs to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-37 was established by DL for implementation purposes.

ITEM II.K.3(3): REPORT SAFETY AND RELIEF VALVE FAILURES PROMPTLY AND CHALLENGES ANNUALLY

DESCRIPTION

This NUREG-066048 item required all operating plants to report safety and relief valve failures promptly and challenges annually. All OL applicants were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-38 was established by DL for implementation purposes.

ITEM II.K.3(4): REVIEW AND UPGRADE RELIABILITY AND REDUNDANCY OF NON-SAFETY EQUIPMENT FOR SMALL-BREAK LOCA MITIGATION

DESCRIPTION

This NUREG-066048 item only affected OL applicants. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items II.C.1, II.C.2, and II.C.3.

CONCLUSION

This item is covered in Items II.C.1, II.C.2, and II.C.3.

ITEM II.K.3(5): AUTOMATIC TRIP OF REACTOR COOLANT PUMPS

DESCRIPTION

This NUREG-066048 item required all PWR operating plants to study the need for automatic trip of RCPs and to modify procedures or designs, as appropriate. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-39 was established by DL for implementation purposes.

ITEM II.K.3(6): INSTRUMENTATION TO VERIFY NATURAL CIRCULATION

DESCRIPTION

This NUREG-066048 item affected all PWRs (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.C.1, II.F.2, and II.F.3.

CONCLUSION

This item is covered in Items I.C.1, II.F.2, and II.F.3.

ITEM II.K.3(7): EVALUATION OF PORV OPENING PROBABILITY DURING OVERPRESSURE TRANSIENT

DESCRIPTION

This NUREG-066048 item required all B&W operating plants (OLs and OL applicants)

to document that their PORVs would open in less than 5% of all anticipated overpressure transients. Clarifications were issued in NUREG-073798 to include all PWRs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(8): FURTHER STAFF CONSIDERATION OF NEED FOR DIVERSE DECAY HEAT REMOVAL METHOD INDEPENDENT OF SGs

DESCRIPTION

This NUREG-066048 item affected all PWRs (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items II.C.1 and II.E.3.3.

CONCLUSION

This item is covered in Items II.C.1 and II.E.3.3.

ITEM II.K.3(9): PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION

DESCRIPTION

This NUREG-066048 item required all **W** plants (OLs and OL applicants) to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-40 was established by DL for implementation purposes.

ITEM II.K.3(10): ANTICIPATORY TRIP MODIFICATION PROPOSED BY SOME LICENSEES TO CONFINE RANGE OF USE TO HIGH POWER LEVELS

DESCRIPTION

This NUREG-066048 item required that the anticipatory trip modification proposed

by some licensees to confine the range of use of high-power levels not be made until it could be shown that the probability of a small-break LOCA resulting from a stuck-open PORV was substantially unaffected by the modification. The applicability of the item to **W** operating plants and OL applicants with **W** reactors was to be determined on a plant-by-plant basis. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-41 was established by DL for implementation purposes.

ITEM II.K.3(11): CONTROL USE OF PORV SUPPLIED BY CONTROL COMPONENTS, INC. UNTIL FURTHER REVIEW COMPLETE

DESCRIPTION

This NUREG-066048 item required plants to justify the use of PORVs that had failed during testing. The applicability of the item to all operating plants and OL applicants was to be determined on a case-by-case basis. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(12): CONFIRM EXISTENCE OF ANTICIPATORY TRIP UPON TURBINE TRIP

DESCRIPTION

This NUREG-066048 item required all **W** plants (OLs and OL applicants) to confirm that their plants have an anticipatory reactor trip upon turbine trip. Plants that did not have this trip were required to provide a conceptual design and evaluation for the installation of the trip. Clarifications affecting both groups of plants were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-42 was established by DL for implementation purposes.

ITEM II.K.3(13): SEPARATION OF HPCI AND RCIC SYSTEM INITIATION LEVELS

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to analyze the benefits to be gained from separating HPCI and RCIC initiation levels and providing auto-start of RCIC on low-low level. Clarifications were issued in NUREG-073798 to include all operating BWRs and OL applicants with RCIC and HPCI systems.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-43 was established by DL for implementation purposes.

ITEM II.K.3(14): ISOLATION OF ISOLATION CONDENSERS ON HIGH RADIATION

DESCRIPTION

This NUREG-066048 item required all operating GE plants with isolation conden-sers to increase the availability of the isolation condensers as heat sinks by providing high radiation isolation signals at the vent rather than at the steam lines. Clarifications affecting all operating BWRs with isolation condensers were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-44 was established by DL for implementation purposes.

ITEM II.K.3(15): MODIFY BREAK DETECTION LOGIC TO PREVENT SPURIOUS ISOLATION OF HPCI AND RCIC SYSTEMS

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to modify their pipe break detection circuitry to prevent isolation of system(s) due to startup pressure transient. Clarifications were issued in NUREG-073798 to address all BWRs (OLs and OL applicants) with HPCI and RCIC systems.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-45 was established by DL for implementation purposes.

ITEM II.K.3(16): REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES-FEASIBILITY STUDY AND SYSTEM MODIFICATION

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to study the reduction in challenge and failure rates of relief valves to minimize the most possible cause of a small-break LOCA. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-46 was established by DL for implementation purposes.

ITEM II.K.3(17): REPORT ON OUTAGE OF ECC SYSTEMS - LICENSEE REPORT AND TECHNICAL SPECIFICATION CHANGES

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in technical specifications. Clarifications were issued in NUREG-073798 to include all operating reactors and OL applicants.

CONCLUSION

This item was RESOLVED, requirements were issued and MPA F-47 was established by DL for implementation purposes.

ITEM II.K.3(18): MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to modify their ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. A feasibility study and risk assessment study were required to determine the optimum approach. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-48 was established by DL for implementation purposes.

ITEM II.K.3(19): INTERLOCK ON RECIRCULATION PUMP LOOPS

DESCRIPTION

This NUREG-066048 item required all GE operating plants with non-jet pumps to install interlocks to assure that level measurements are representative of the level in the core. Clarifications were issued in NUREG-073798 to address all operating BWRs with non-jet pumps, except Humboldt Bay.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-49 was established by DL for implementation purposes.

ITEM II.K.3(20): LOSS OF SERVICE WATER FOR BIG ROCK POINT

DESCRIPTION

This NUREG-066048 item required Big Rock Point to evaluate the acceptability or the consequences of a loss of service water. A clarification to this requirement was issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(21): RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to modify their core spray and LPCI system logic so that these systems would restart, if required, to assure adequate core cooling. It was believed that the core spray and LPCI system flow may be stopped by the operator. These systems could not start automatically on loss of water level if an initiation signal were still present. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-50 was established by DL for implementation purposes.

ITEM II.K.3(22): AUTOMATIC SWITCHOVER OF RCIC SYSTEM SUCTION - VERIFY PROCEDURES AND MODIFY DESIGN

DESCRIPTION

This NUREG-066048 item affected all GE plants (OLs and OL applicants). The RCIC system takes suction from the condensate storage tank with manual switch-over to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool. Clarifications affecting all operating BWRs and OL applicants with RCIC systems were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-51 was established by DL for implementation purposes.

ITEM II.K.3(23): CENTRAL WATER LEVEL RECORDING

DESCRIPTION

This NUREG-066048 item was originated to address GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.D.2, III.A.1.2, and III.A.3.4.

CONCLUSION

This item is covered in Items I.D.2, III.A.1.2, and III.A.3.4.

ITEM II.K.3(24): CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI AND RCIC SYSTEMS

DESCRIPTION

This NUREG-066048 item required all operating GE plants (OLs and OL applicants) to verify that HPCI and RCIC are designed to withstand loss of offsite power for at least 2 hours. Clarifications affecting all BWRs (OLs and OL applicants) with HPCI and RCIC systems were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-52 was established by DL for implementation purposes.

ITEM II.K.3(25): EFFECT OF LOSS OF AC POWER ON PUMP SEALS

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to verify the adequacy of pump seals to withstand loss of cooling water due to loss of AC power for at least 2 hours. Clarifications were issued in NUREG-073798 to include all BWRs, **W**, and CE operating reactors, and all OL applicants.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-53 was established by DL for implementation purposes.

ITEM II.K.3(26): STUDY EFFECT ON RHR RELIABILITY OF ITS USE FOR FUEL POOL COOLING

DESCRIPTION

This NUREG-066048 item was originated to affect GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.E.2.1.

CONCLUSION

This item is covered in Item II.E.2.1.

ITEM II.K.3(27): PROVIDE COMMON REFERENCE LEVEL FOR VESSEL LEVEL INSTRUMENTATION

DESCRIPTION

This NUREG-066048 item affected all GE plants (OLs and OL applicants) and required all reactor vessel level instruments to be referenced to the same point. It was believed that different reference points of the various reactor vessel water level instruments could cause operator confusion. Either the bottom of the vessel or the active fuel were considered to be reasonable reference points. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-54 was established by DL for implementation purposes.

ITEM II.K.3(28): STUDY AND VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

DESCRIPTION

This NUREG-066048 item affected all plants with GE reactors (OLs and OL applicants). These plants were required to assure that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. However, clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-55 was established by DL for implementation purposes.

ITEM II.K.3(29): STUDY TO DEMONSTRATE PERFORMANCE OF ISOLATION CONDENSERS WITH NON-CONDENSIBLES

DESCRIPTION

This NUREG-066048 item affected all operating plants with GE isolation condensers. These plants were required to demonstrate the adequacy of isolation condensers with non-condensibles. Clarifications affecting all operating BWRs with isolation condensers were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-56 was established by DL for implementation purposes.

ITEM II.K.3(30): REVISED SMALL-BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10 CFR 50, APPENDIX K

DESCRIPTION

This NUREG-066048 item required all OLs and OL applicants to revise and submit for NRC approval the analyses used by NSSS vendors and/or fuel suppliers for SBLOCA analysis in compliance with 10 CFR 50, Appendix K. The revised analyses were to account for comparisons with experimental data, including data from the LOFT and semiscale test facilities. Clarifications were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-57 was established by DL for implementation purposes.

ITEM II.K.3(31): PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46

DESCRIPTION

This NUREG-066048 item required all OLs and OL applicants to submit for NRC approval plant-specific calculations using NRC-approved models for SBLOCA, to show compliance with 10 CFR 50.46. Clarifications were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-58 was established by DL for implementation purposes.

ITEM II.K.3(32): PROVIDE EXPERIMENTAL VERIFICATION OF TWO-PHASE NATURAL CIRCULATION MODELS

DESCRIPTION

This NUREG-066048 item was originated to require PWRs to provide experimental verification of two-phase natural circulation models. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.E.2.2.

CONCLUSION

This item is covered in Item II.E.2.2.

ITEM II.K.3(33): EVALUATE ELIMINATION OF PORV FUNCTION

DESCRIPTION

This NUREG-066048 item was originated to require PWRs (OLs and OL applicants) to evaluate elimination of the PORV function. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.C.1.

CONCLUSION

This item is covered in Item II.C.1.

ITEM II.K.3(34): RELAP-4 MODEL DEVELOPMENT

DESCRIPTION

This NUREG-066048 item was originated to address RELAP-4 model development in PWRs. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.E.2.2.

CONCLUSION

This item is covered in Item II.E.2.2.

ITEM II.K.3(35): EVALUATION OF EFFECTS OF CORE FLOOD TANK INJECTION ON SMALL BREAK LOCAs-

DESCRIPTION

This NUREG-066048 item was originated to evaluate the effects of core flood tank injection on SBLOCAs in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(36): ADDITIONAL STAFF AUDIT CALCULATIONS OF B&W SMALL-BREAK LOCA ANALYSES

DESCRIPTION

This NUREG-066048 item was originated to address B&W plants, but was determined to be covered by Item I.C.1 prior to the publication of NUREG-0660.48

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(37): ANALYSIS OF B&W RESPONSE TO ISOLATED SMALL-BREAK LOCA

DESCRIPTION

This NUREG-066048 item was originated to analyze the response of B&W plants (OLs and OL applicants) to isolated SBLOCAs. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(38): ANALYSIS OF PLANT RESPONSE TO A SMALL-BREAK LOCA IN THE PRESSURIZER SPRAY LINE

DESCRIPTION

This NUREG-066048 item was originated to analyze the reponse of B&W plants (OLs and OL applicants) to a SBLOCA in the pressurizer spray line. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(39): EVALUATION OF EFFECTS OF WATER SLUGS IN PIPING CAUSED BY HPI AND CFT FLOWS

DESCRIPTION

This NUREG-066048 item was originated to evaluate the effects of water slugs caused by HPI and CFT flows in the piping of B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(40): EVALUATION OF RCP SEAL DAMAGE AND LEAKAGE DURING A SMALL- BREAK LOCA

DESCRIPTION

This NUREG-066048 item was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.K.2(16).

CONCLUSION

This item is covered in Item II.K.2(16).

ITEM II.K.3(41): SUBMIT PREDICTIONS FOR LOFT TEST L3-6 WITH RCPs RUNNING

DESCRIPTION

This NUREG-066048 item was originated to require B&W plants (OLs and OL applicants) to submit to the NRC predictions for LOFT Test L3-6 with RCPs running. However, prior to the publication of NUREG-0660,48 it was determined that the issued was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(42): SUBMIT REQUESTED INFORMATION ON THE EFFECTS OF NON-CONDENSIBLE GASES

DESCRIPTION

This NUREG-066048 item was originated to require B&W plants (OLs and OL applicants) to submit to the NRC requested information on the affects of non-condensible gases. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

The item is covered in Item I.C.1.

ITEM II.K.3(43): EVALUATION OF MECHANICAL EFFECTS OF SLUG FLOW ON STEAM GENERATOR TUBES

DESCRIPTION

This NUREG-066048 item was originated to evaluate the mechanical affects of slug flow on the steam generator tubes of B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item II.K.2(15).

CONCLUSION

This item is covered in Item II.K.2(15).

ITEM II.K.3(44): EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO SIGNIFICANT FUEL FAILURE

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to show that transients combined with the worst single failure would not result in significant fuel damage. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-59 was established by DL for implementation purposes.

ITEM II.K.3(45): EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to analyze depressurization modes other than full ADS for possible inclusion in emergency procedures. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-60 was established by DL for implementation purposes.

ITEM II.K.3(46): RESPONSE TO LIST OF CONCERNS FROM ACRS CONSULTANT

DESCRIPTION

This NUREG-066048 item required all GE plants (OLs and OL applicants) to respond to concerns raised by ACRS consultants. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.98

CONCLUSION

This item is RESOLVED, requirements were issued, and MPA F-61 was established by DL for implementation purposes.

ITEM II.K.3(47): TEST PROGRAM FOR SMALL-BREAK LOCA MODEL VERIFICATION PRETEST PREDICTION, TEST PROGRAM, AND MODEL VERIFICATION

DESCRIPTION

This NUREG-066048 item was originated to require GE plants (OLs and OL applicants) to complete a test program for SBLOCA model verification. However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.C.1 and II.E.2.2.

CONCLUSION

This item is covered in Items I.C.1 and II.E.2.2.

ITEM II.K.3(48): ASSESS CHANGE IN SAFETY RELIABILITY AS A RESULT OF IMPLEMENTING B&OTF RECOMMENDATIONS

DESCRIPTION

This NUREG-066048 item was originated to require GE plants (OLs and OL applicants) to assess the change in safety reliability as a result of implementing the recommendations of the Bulletins and Orders Task Force (B&OTF). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items II.C.1 and II.C.2.

CONCLUSION

This item is covered in Items II.C.1 and II.C.2.

ITEM II.K.3(49): REVIEW OF PROCEDURES (NRC)

DESCRIPTION

This NUREG-066048 item was originated to address all **W** and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.9 for OLs and by Items I.C.8 and I.C.9 for OL applicants.

CONCLUSION

This item is covered in Items I.C.8 and I.C.9.

ITEM II.K.3(50): REVIEW OF PROCEDURES (NSSS VENDORS)

DESCRIPTION

This NUREG-066048 item was originated to address all **W** and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.9 for OLs and by Items I.C.7 and I.C.9 for OL applicants.

CONCLUSION

This item is covered in Items I.C.7 and I.C.9.

ITEM II.K.3(51): SYMPTOM-BASED EMERGENCY PROCEDURES

DESCRIPTION

This NUREG-066048 item was originated to address all **W**, CE, and GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.C.9.

CONCLUSION

This item is covered in Item I.C.9.

ITEM II.K.3(52): OPERATOR AWARENESS OF REVISED EMERGENCY PROCEDURES

DESCRIPTION

This NUREG-066048 item was originated to address all GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.B.1.1, I.C.2, and I.C.5.

CONCLUSION

This item is covered in Items I.B.1.1, I.C.2, and I.C.5.

ITEM II.K.3(53): TWO OPERATORS IN CONTROL ROOM

DESCRIPTION

This NUREG-066048 item was originated to address all GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.A.1.3.

CONCLUSION

This item is covered in Item I.A.1.3.

ITEM II.K.3(54): SIMULATOR UPGRADE FOR SMALL-BREAK LOCAs

DESCRIPTION

This NUREG-066048 item was originated to address all plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Item I.A.4.1.

CONCLUSION

This item is covered in Item I.A.4.1.

ITEM II.K.3(55): OPERATOR MONITORING OF CONTROL BOARD

DESCRIPTION

This NUREG-066048 item was originated to address all W and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.C.1, I.D.2, and I.D.3.

CONCLUSION

This item is covered in Items I.C.1, I.D.2, and I.D.3.

ITEM II.K.3(56): SIMULATOR TRAINING REQUIREMENTS

DESCRIPTION

This NUREG-066048 item was originated to address all **W** and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,48 it was determined that the issue was being addressed by Items I.A.2.6 and I.A.3.1.

CONCLUSION

This item is covered in Items I.A.2.6 and I.A.3.1.

ITEM II.K.3(57): IDENTIFY WATER SOURCES PRIOR TO MANUAL ACTIVATION OF ADS

DESCRIPTION

This NUREG-066048 item required all operating GE plants to revise their emer-gency procedures to include verification that low pressure cooling systems are available prior to manual ADS. For OL applicants, the issue was determined to be covered by Item I.C.1. Clarifications affecting all operating BWRs were issued in NUREG-0737.98

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-62 was established by DL for implementation purposes.

REFERENCES

0048.	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.

Task III.A: Emergency Preparedness and Radiation Effects ( )

TASK III.A.1: IMPROVE LICENSEE EMERGENCY PREPAREDNESS - SHORT-TERM

The objectives of this task were to improve and upgrade licensee emergency preparedness by requiring improvements in facilities, plans, procedures, offsite support, technical assistance, equipment, and supplies required to adequately respond to and manage an accident.

ITEM III.A.1.1: UPGRADE EMERGENCY PREPAREDNESS

The two parts of this item are evaluately separately below.

ITEM III.A.1.1(1): IMPLEMENT ACTION PLAN REQUIREMENTS FOR PROMPTLY IMPROVING LICENSEE EMERGENCY PREPAREDNESS

DESCRIPTION

This TMI Action Plan[48] item called for licensees to promptly upgrade their overall state of emergency preparedness for accidents, including the integration of onsite and offsite emergency preparedness. The plan for staff review of licensee actions was documented in SECY-79-450.

In the short-term, the staff was directed to make an integrated assessment of licensee, local, and state capabilities and interfaces based on: (1) a review of existing plans and a meeting in each site area to communicate upgraded criteria and to identify to licensees the areas requiring improvements; and (2) a review of upgraded licensee, local, and state plans submitted by each licensee, after the site visit was summarized in an SER. A status report on this item was issued in December 1981.[248]

CONCLUSION

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM III.A.1.1(2): PERFORM AN INTEGRATED ASSESSMENT OF THE IMPLEMENTATION

DESCRIPTION

This TMI Action Plan[48] item called for the staff to perform a long-term integrated assessment of the implementation of the actions required by Item III.A.1.1(1). This assessment consisted of: (1) a review of implementation procedures, including onsite and offsite personnel and equipment; (2) observation and critique of exercise involving licensee, local, and state capabilities; and (3) observation and critique of exercises involving licensee, local, state, and federal capabilities.

CONCLUSION

Procedures for routine, periodic inspection of licensees' emergency preparedness programs were developed by the staff and used for subsequent routine inspections; observation of exercises is an ongoing function of the regions. Thus, this item was RESOLVED and no new requirements were established.

ITEM III.A.1.2: UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

The three parts of this item are evaluated separately below.

ITEM III.A.1.2(1): TECHNICAL SUPPORT CENTER

DESCRIPTION

This TMI Action Plan[48] item called for a dedicated Technical Support Center (TSC) to provide a place for management and technical personnel to support reactor control functions, to evaluate and diagnose plant conditions, and for a more orderly conduct of emergency operations. The TSC was required to be separate from but near the control room and was expected to have the capability to dis-play and transmit plant status to those individuals knowledgable of and responsible for engineering and management support of reactor operations, in the event of an accident.

CONCLUSION

This item was clarified in both NUREG-0737[98] and Generic Letter No. 82-33[376] and requirements were issued.

ITEM III.A.1.2(2): ON-SITE OPERATIONAL SUPPORT CENTER

DESCRIPTION

This TMI Action Plan[48] item called for the establishment of an Operational Support Center (OSC) separate from the control room as a place in which operations support personnel could assemble in an emergency situation to receive instructions from the operating staff. The OSC was to be provided with communication capability with the plant control room, TSC, and the near-site Emergency Operations Facility (EOF).

CONCLUSION

This item was clarified in both NUREG-0737[98] and Generic Letter No. 82-33[376] and requirements were issued.

ITEM III.A.1.2(3): NEAR-SITE EMERGENCY OPERATIONS FACILITY

DESCRIPTION

This TMI Action Plan[48] item called for a near-site EOF to provide a planned, organized, central focal point for coordination of onsite and offsite activities for reactor emergency situations. The EOF was required to be operated by licensees and sized and equipped to function as a center for: (1) licensee command and control functions of onsite operations and evaluation and coordination of all onsite and offsite licensee activities related to an emergency having actual or potential environmental consequences; and (2) analysis of plant efflu-ent monitors, meteorological conditions, and offsite radiation measurements, and for offsite dose projections.

CONCLUSION

This item was clarified in both NUREG-0737[98] and Generic Letter No. 82-33[376] and requirements were issued.

ITEM III.A.1.3: MAINTAIN SUPPLIES OF THYROID BLOCKING AGENT

Both parts of this item were combined and evaluated together.

DESCRIPTION

Historical Background

This TMI Action Plan item[48] addressed the issue of providing potassium iodide (KI) as a thyroid blocking agent for nuclear power plant onsite personnel, off-site emergency response personnel, and the general population near nuclear power plants. NUREG-0654[224] required licensees to have adequate supplies of KI available for onsite personnel and for offsite emergency response support personnel, including offsite agencies. The item also called for an evaluation by the Department of Health and Human Services (HHS) and the Environmental Protection Agency (EPA) regarding use of KI by the general public.

In accordance with SECY-82-396A,[369] RES was expected to complete a technical paper which evaluated the cost/benefit of the use of KI by the general public. These results were to be sent to the other federal agencies involved with the final decision.

Safety Significance

It is possible that a nuclear power reactor accident could release radionuclides, including isotopes of radioiodine, into the environment. The radioactive iodine, if taken up by the thyroid gland, could induce nodules of cancer in the thyroid.[64]

Possible Solution

If stockpiles of KI are made available for public use, the KI could help prevent radiation injury to the thyroid gland by saturating the gland with non-radioactive iodine.[64] This would block the thyroid from taking up the radioactive iodine.

CONCLUSION

The licensees are already required to maintain supplies of the thyroid blocking agent (KI) as a protective measure for emergency workers and other individuals onsite during an emergency.[48,224] Therefore, Item III.A.1.3(1) was resolved.

Work completed by the staff on the subject of stockpiling KI for public use resulted in a cost/benefit study which was published in NUREG/CR-1433.[831] HHS completed its recommendations on the methods for administration of KI to the general public (130 milligrams/day at projected thyroid doses of 25 rem or greater) and published them in the Federal Register in 1982 (47 FR 28158). NUREG/CR-1433[831] showed that the use of KI by the general public has a very low cost/benefit ratio. FEMA, through a special subcommittee of the Federal Radiological Preparedness Coordinating Committee (FRPCC), developed a draft federal policy statement in July 1982 on the use of KI for thyroid blocking by the gen-eral public. This draft policy statement left the decision on distribution and use of KI for thyroid blocking by the general public to the state and local authorities on a site-specific basis. The HHS guidance on KI use was addressed in the statement as well as many of the problems and difficulties in distribu-tion and administration of the drug (e.g., timeliness, interference with other protective actions, and limited protection). The NRC staff did not agree with the draft federal policy statement because it believed that the statement should recommend that KI not be distributed for use by the general public. A new cost/benefit study was prepared using an uncertainty analysis of the information in NUREG/CR-1433[831] and showed that KI offered an extremely small benefit in relation to its cost over the uncertainty range.

The new cost/benefit study and prepared changes to the draft federal policy statement were reviewed by the ACRS and forwarded to the Commission for consideration in SECY-83-362.[832] While the Commission was considering the staff posi-tion, FEMA decided to revise the draft federal policy statement because of the lack of concurrence by NRC and several other member agencies of the FRPCC. The Commission decided to review this new policy statement before responding to FEMA.

The new draft federal policy statement was completed by the FRPCC on March 26, 1985 and was sent to the Commission for review on May 13, 1985 (SECY-85-167).[833]

This new policy statement recommended against a nationwide requirement for the distribution or stockpiling of KI for use by the general public and left the final decision for its use to state and local authorities on a site-specific basis. On June 11, 1985, the Commission concurred with the new policy statement. FEMA published the policy statement in the Federal Register on July 24, 1985 (50 FR 30258). With the publication of the federal policy statement on the distribution and stockpiling of KI for use in the event of a nuclear power reactor accident, this item was RESOLVED and no new requirements were established.[818]

ITEM III.A.1.3(1): WORKERS

This item was evaluated in Item III.A.1.3 above and was determined to be RESOLVED. No new requirements were established.[818]

ITEM III.A.1.3(2): PUBLIC

This item was evaluated in Item III.A.1.3 above and was determined to be RESOLVED. No new requirements were established.[818]

REFERENCES

0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

0224.	NUREG#0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.
0248.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," December 28, 1981. [8205260197]
0369.	SECY-82-396A, "Withdrawal of SECY-82-396 (Federal Policy Statement on Use of Potassium Iodide)," U.S. Nuclear Regulatory Commission, October 15, 1982. [8211040047]
0376.	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*]
0818.	Memorandum for W. Dircks from J. Taylor, "TMI Action Plan—Completed Item," August 15, 1985. [8508200726]
0831.	NUREG/CR-1433, "Examination of the Use of Potassium Iodide (KI) as an Emergency Protective Measure for Nuclear Reactor Accidents," U.S. Nuclear Regulatory Commission, October 1980.
0832.	SECY-83-362, "Emergency Planning—Predistribution/Stockpiling of Potassium Iodide for the General Public," U.S. Nuclear Regulatory Commission, August 30, 1983. [8309080120]
0833.	SECY-85-167, "Federal Policy Statement on the Distribution and Use of Potassium Iodide," U.S. Nuclear Regulatory Commission, May 13, 1985. [8505310621]

Task III.A.2: Improving Licensee Emergency Preparedness - Long-Term ( )

The objective of this task was to upgrade the emergency preparedness of nuclear power plants. Specific criteria to meet this objective were delineated in NUREG-0654.[224]

ITEM III.A.2.1: AMEND 10 CFR 50 AND 10 CFR 50, APPENDIX E

The four parts of this item were evaluated separately below.

ITEM III.A.2.1(1): PUBLISH PROPOSED AMENDMENTS TO THE RULES

DESCRIPTION

This TMI Action Plan[48] called for the staff to revise 10 CFR 50 and 10 CFR 50, Appendix E to require licensees to: (1) revise their emergency plans to meet new requirements; and (2) conduct extensive coordination and planning with state and local officials. These rule changes were considered an upgrade of NRC emergency planning regulations that would provide prompt clarification and expansion in areas that were perceived to be deficient as a result of previous experiences.

CONCLUSION

In June 1979, the NRC began a formal reconsideration of the role of emergency planning in ensuring the continued protection of the public health and safety in the areas around nuclear power plants. Proposed amendments to the rules were published for public comment on December 19, 1979 (44 FR 75167). The Commission approved the rule changes that were recommended by the staff in SECY-80-275.[1566] Thus, this item was RESOLVED and requirements were issued.

ITEM III.A.2.1(2): CONDUCT PUBLIC REGIONAL MEETINGS

DESCRIPTION

This TMI Action Plan[48] item called for the staff to conduct regional public meetings with state and local authorities in the formulation of recommendations for the rules outlined in Item III.A.2.1(1).

CONCLUSION

The staff conducted four regional workshops in New York, San Francisco, Chicago, and Atlanta on January 15, 17, 22, and 24, 1980, respectively, to present proposed rule changes and solicit comments. In developing the final rule changes, the staff considered all information received at the workshops as well as all letters with public comments. The comments generated from these meetings were documented in NUREG/CP-0011.[1567] Thus, this issue was RESOLVED.

ITEM III.A.2.1(3): PREPARE FINAL COMMISSION PAPER RECOMMENDING ADOPTION OF RULES

DESCRIPTION

This TMI Action Plan[48] item called for the staff to prepare a final Commission Paper recommending the adoption of effective rules. The final rule was to consider experience gained in Item III.A.1.1, comments on the proposed rule, input obtained at the regional meetings [Item III.A.2.1(2)], and recommendations of the President's Commission[175] and the NRC Special Inquiry Group.

CONCLUSION

The final Commission Paper recommending adoption of the rules was issued as SECY-80-275[1566] on June 3, 1980. Thus, this item was RESOLVED.

ITEM III.A.2.1(4): REVISE INSPECTION PROGRAM TO COVER UPGRADED REQUIREMENTS

DESCRIPTION

This TMI Action Plan[48] called for the staff to revise its inspection program to cover upgraded requirements mandated by the rule changes of Item III.A.2.1(1).

CONCLUSION

OIE revised its inspection program to cover upgraded requirements. The routine inspection program for emergency preparedness was suspended and procedures governing team reviews were issued to ensure compliance with the new requirements (Item III.A.1.1).[248] This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-67 was established by NRR/DL for implementation purposes.

ITEM III.A.2.2: DEVELOPMENT OF GUIDANCE AND CRITERIA

DESCRIPTION

This TMI Action Plan[48] item called for the NRC and FEMA to use NUREG-0654[224] as interim guidance and upgraded criteria in judging the adequacy of licensee, state, and local government emergency plans and preparedness until the final NRC requirements and guidance were promulgated.

CONCLUSION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-68 was established by NRR/DL for implementation purposes. Inspection programs[1565] were put in place to ensure the initial and continuing adequacy of emergency response facilities of all plants in meeting the requirements of Supplement 1 to NUREG-0737.[98]

REFERENCES

0048.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0175.	ZAR-791030-01, "Report of the President's Commission on the Accident at Three Mile Island," J. G. Kemeny et al., November 30, 1979.
0224.	NUREG#0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.
0248.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," December 28, 1981. [8205260197]
1565.	Memorandum for T. Murley from W. Russell and J. Partlow, "Closeout of TMI Action Plan Items III.A.1.2 and III.A.2.2 (Multi-Plant Actions F-63, F-64, F-65, and F-68)," October 2, 1990. [9010160111]
1566.	SECY-80-275, "Final Rulemaking on Emergency Preparedness," U.S. Nuclear Regulatory Commission, June 3, 1980. [8007090015]
1567.	NUREG/CP-0011, "Proceedings to Workshops Held on Proposed Rulemaking on Emergency Planning for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 1980.

Task III.A.3: Improving NRC Emergency Preparedness ( )

The objective of this task is to enable NRC, in the event of a nuclear accident at a licensed reactor facility, to: (1) monitor and evaluate the situation and potential hazards, (2) advise the licensee's operating staff as needed, and (3) in an extreme case, be able to issue orders governing such operations.

Item III.A.3.1: NRC Role in Responding to Nuclear Emergencies

The five parts of this item have been combined and evaluated together.

Description

This TMI Action Plan[48] item was to define the NRC role in emergency situations involving NRC licensees. The definition of the NRC emergency response role will be used by OIE in revising and upgrading plans and procedures for the NRC emergency operations center. OIE, with input from other NRC offices, will revise NRC Manual Chapter 0502 and NUREG-0610[488] to describe and implement the NRC emergency response program.

NUREG-0610[488] was revised as Appendix I to NUREG-0654[224] in November 1980. NUREG-0728,[257] published in September of 1980, provided the basis for continued upgrading of the NRC Incident Response Program and information to be included in the revised NRC Manual Chapter 0502. In the interim, until NRC Manual Chapter 0502 was approved by the Commission, NUREG-0845[(258)] written for trial use in March 1982, provided detailed procedures for the NRC Incident Response Plan. When the Commission approves the proposed revisions to NRC Manual Chapter 0502, NUREG-0845[258] will be issued for final publication.

The proposed revision to NRC Manual Chapter 0502 was approved by the Commission on January 5, 1983. Resolution of Item III.A.3.1 also resolved Item B-71, "Incident Response," which was essentially superseded by Item III.A.3.1. All required action on this item has been completed.[408,548]

Conclusion

This item has been RESOLVED.

Item III.A.3.1(1): Define NRC Role in Emergency Situations

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.1(2): Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.1(3): Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.1(4): Prepare Commission Paper

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.1(5): Revise Implementing Procedures and Instructions for Regional Offices

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.2: Improve Operations Centers

Description

This TMI Action Plan[48] item called for the NRC Operations Center (OC) in Bethesda, Maryland to be upgraded to support activities in response to a major accident.

Near-term improvements[235] made to the OC during 1979-1981 included improved physical space, rearrangement, better facilities (such as status systems and weather information), and improved telecommunications equipment including the possible use of HF radios. A study has recently been completed on long-term improvements in the OC. This study addressed a complete redesign of the OC taking into account specifically human factors considerations and improved communications.

OIE considers implementation of this item important and is working toward its completion. Taking into account the problems in logistics of construction relocation, equipment purchase, and budget constraints, implementation should be completed by December 1983.[235,379]

Conclusion

This item was RESOLVED and no new requirements were established.

Item III.A.3.3: Communications

Both parts of this item have been combined and evaluated together.

Description

The TMI Action Plan[48] included communications in the required improvements for NRC emergency preparedness. Included in communications are two items: (1) direct and dedicated telephone lines (OPX) between the licensee facilities and NRC; and (2) the use of the dedicated short-range radio communication system (FIRS).

OPX and HPN telephone systems were installed at all operating reactors by August 1980 and are being installed at newer plants prior to operation. FIRS has been obtained for use by NRC field personnel during emergencies. All required action on this item has been completed (see References 235, 248, 379, and 406).

Conclusion

This item has been RESOLVED.

Item III.A.3.3(1): Install Direct Dedicated Telephone Lines

This item was evaluated in Item III.A.3.3 above and was determined to be RESOLVED. New requirements were established.

Item III.A.3.3(2): Obtain Dedicated, Short-Range Radio Communication Systems

This item was evaluated in Item III.A.3.3 above and was determined to be RESOLVED. New requirements were established.

Item III.A.3.4: Nuclear Data Link

Description

Historical Background

After the TMI event, the NRC concluded that the NRC Operations Center (Incident Response Center) should be upgraded to allow NRC personnel to analyze and evaluate plant conditions based on directly transmitted information, as opposed to a voice link. The term "Nuclear Data Link" (NDL) was given to a conceptual system that would access plant data and directly transmit the information to the OC.

Safety Significance

It was believed that, with more current and reliable plant data available to the NRC, the staff could help develop and evaluate accident mitigating actions.

Possible Solution

It was determined that a phased approach would be utilized. The first phase was to have Sandia study the available options and to report their findings. Sandia completed their report which will not be published. The Sandia options were evaluated and it was determined that an elaborate NDL configuration which was interactive with the licensees' system was inappropriate to the NRC's role. The second phase is to be implementation of a prototype which will be evaluated to help the Commission decide whether an NDL is needed and, if so, what it should look like.[240]

Priority Determination

Assumptions

PNL did an assessment of this issue.[64] To assess the impact of this issue, we needed to consider all the other related issues which are involved with the OC. (See Items III.A.3.1, III.A.3.2, III.A.3.3, III.A.3.5, and III.A.3.6.) Many of these issues have been completed or almost completed. In addition, we considered the utilities' emergency response facilities (ERFs). The ERFs are planned to be completed [along with the Safety Parameter Display System (SPDS) and the Data Acquisition System (DAS)] according to requirements outlined in SECY-82-111[151] and a letter[376] issued to all licensees of operating reactors.

Frequency/Consequence Estimate

We constructed an event tree which assumed a base case core-melt based on the Oconee and Grand Gulf risk studies. We then analyzed certain event tree branches based on a risk reduction with results from the possibility that NRC personnel at the OC could: (1) detect and correct an error by the plant operators during accident recovery; or (2) provide optimum approaches to the operators for the mitigation of particular evolving sequences.

It was first assumed that the base case core-melt frequencies are 8.15×10^{-5}/RY for PWRs and 3.67×10^{-5}/RY for BWRs. We assumed that 90% of the core-melt scenarios would proceed slowly enough to allow input from observers at the OC or ERFs. Next, it was assumed that the operator's judgment was not optimum in about 50% of the cases. This includes consideration of the fact that he is not able to take a step back and completely evaluate the accident sequence or evaluate and/or anticipate ahead in the scenario. We then assumed that, given the above, the utilities' ERFs would be manned and available in 90% of the cases and that the utilities' ERF personnel could provide successful input in 75% of the cases.

Of the remaining 25% of the cases, we assumed that the OC would be available 90% of the time and that the NRC personnel could provide the successful input about 50% of the time. This number was assumed smaller than the utility's ERF success rate because of the data available at the OC, i.e., it is not complete and available only by voice communications. This would somewhat hinder the NRC staff's performance.

For this calculation, we ignored the smaller contribution of the event tree branch which is due to the 10% unavailability of the utility's ERF and the success of the OC staff.

Therefore, with the assumption that Items III.A.3.1, III.A.3.2, III.A.3.3, III.A.3.5, and III.A.3.6 are completed and the ERFs are in place, we estimated a potential core-melt frequency reduction for the present OC of about 4.5%.

This was then considered the base-case value for the overall OC as it is completed to date. We then estimated that the incorporation of an NDL could improve the success of the OC staff by about 50% due to the availability of more complete, more accurate, and more timely information. This would then equal an additional core-melt frequency reduction of about 2%.

From the reduction in core-melt frequency, the per plant reduction in public risk was then calculated (based on a population density of 340 people per square mile) to be 4.5 man-rem/RY for PWRs and 5.5 man-rem/RY for BWRs. With 95 PWRs, 49 BWRs, and an average remaining life of 28.5 years for PWRs and 22 years for BWRs, the total public risk reduction is then 18,000 man-rem.

Cost Estimate

Industry Cost: Licensees are not implementing standard data sets, formats, or equipment and the NRC will have to electronically process each of the data outputs that it receives from licensees. Relatively simple equipment at each site, costing perhaps $20,000 for hardware and $15,000 for labor to install, will transmit data in the licensee format to the NRC. There are 50 sites with operating reactors (counting Indian Point as two sites because of the mixed ownership) and 35 addi-tional sites with reactors under construction. New reactors at six existing sites might also be built with new (separate) DAS. Rounding off to be conservative, an estimated 100 sites will require data-transmitting equipment at a total initial cost of $3.5M.

NRC Cost: It would be expected that the NRC would incur the majority of the cost of the overall data link. It was assumed that the OC will have been im-proved (Item III.A.3.2) before the NDL is implemented. With respect to

NRC equipment costs, it was assumed that the ERF at individual utilities would be completed. Based on this, the DAS necessary for support of the facility will already be implemented.

At the OC, the NRC will need a unit for receiving and processing the data. The unit may cost up to $500,000 and software as much as $30,000 for each site, since processing instructions will be different for each different licensee output. Therefore, the estimated initial cost at the OC is $3.5M. System main-tenance is estimated at 2% of equipment costs per year for 30 years, or $1.5M.

The total estimated NDL system cost, regardless of who pays it, is $8.5M for concepts currently envisioned. The planned Prototype Program will develop more refined evaluations and cost estimates to permit the Commission to decide what is really needed.

Value/Impact Assessment

Based on the estimated public risk reduction of 18,000 man-rem, the value/impact score is given by:

$$S = \frac{18{,}000\text{man} - \text{rem}}{\$8.5\text{M}}$$

$$= 2{,}100\text{man} - \text{rem} / \$\text{M}$$

Other Considerations:

(1)	Present plans are to implement a prototype system.[254,255]
(2)	More accurate cost estimates are difficult without clearer system definition which is to be provided by evaluation of the prototypes.
(3)	The estimate of the potential reduction in core-melt frequency is subject to large uncertainty because of the sequences of assumptions which went into the event tree.
(4)	OIE believes that this issue should receive high priority.

Conclusion

Based on the value/impact score and the total risk reduction potential, this issue was given a medium priority ranking. However, in June 1985, it was determined by the staff that the design that met NRC requirements was one that utilized electronic data transmission systems that were already being developed by licensees for their own ERFs. This concept, Emergency Response Data System (ERDS), was approved by the Commission in March 1985.[779] Licensees will not be required to backfit their systems to include additional parameters to provide data on NRC's parameter list. Data that is not available from the electronic data stream can be provided by voice over existing phone lines. Thus, this item was RESOLVED and no new requirements were established.

Item III.A.3.5: Training, Drills, and Tests

Description

The TMI Action Plan[48] identified a need to improve the capability to respond to emergencies by continuing the headquarters and regional drills and exercises. The scope is envisioned to be slowly expanded to include joint exercises with State and local agencies and other Federal response capabilities. A schedule involving various levels of participation by the various parties is to be prepared.

Exercises, scheduling, and training are being conducted with gradually increas-ing scope and continuing programs related to this item have been incorporated into routine ongoing NRC operations. (See References 235, 248, 379, and 406.)

Conclusion

This item was RESOLVED and no new requirements were established.

Item III.A.3.6: Interaction of NRC and Other Agencies

The three parts of this item have been combined and evaluated together.

Description

The TMI Action Plan[48] identified the requirement to establish interaction agreements between NRC and other agencies for cooperation, communication, and assistance during emergency situations. Agencies involved include other international governments, i.e., Mexico and Canada, other Federal agencies, and State and local governmental bodies.

In September 1980, the NRC published NUREG-0728[257] which described in general the NRC's responsibilities and plans for responding to emergencies at nuclear power reactors. This report further described the coordination/liaison with other agencies and organizations. In March 1982, the NRC published NUREG-0845,[258] which contains detailed agency procedures for the NRC incident response plan. It also includes the details for providing the interaction between NRC and other involved Federal agencies and other organizations.

All work required by this item has been completed and the NRC Incident Response Plan is being implemented.[235,256,379]

Conclusion

This item has been RESOLVED with changes in the NRC procedures that address the interaction with other agencies during emergency situations.

Item III.A.3.6(1): International

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.6(2): Federal

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

Item III.A.3.6(3): State And Local

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

REFERENCES

0048.	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.
0064.	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.
0151.	SECY-82-111, "Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, March 11, 1982. [8203180409]
0224.	NUREG#0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.

0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0240.	SECY-81-153, "Nuclear Data Link," U.S. Nuclear Regulatory Commission, March 11, 1981. [8103240155]
0256.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," June 2, 1982. [8401170114]
0257.	NUREG-0728, "Report to Congress—NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, September 1980.
0258.	NUREG#0845, "Agency Procedure for the NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, March 1982.
0376.	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*]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0408.	NUREG#1509, "Radiation Effects on Reactor Pressure Vessel Supports," U.S. Nuclear Regulatory Commission, May 1996.
0488.	NUREG#0610, "Draft Emergency Action Level Guidelines for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1979.
0548.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan Completed Items," January 26, 1983. [8303090323]
0779.	Memorandum for W. Dircks from J. Taylor, "TMI Action Plan—Completed Item," June 26, 1985. [8507080034]

Task III.B: Emergency Preparedness of State and Local Governments ( )

The objective of this task is to upgrade the state of emergency preparedness of State and local governments affected by nuclear facilities. The Federal Emer-gency Management Agency was given the lead on this effort by the President on December 7, 1979.

ITEM III.B.1: TRANSFER OF RESPONSIBILITIES TO FEMA

Items III.B.1 and III.B.2 have been combined and evaluated together.

DESCRIPTION

TMI Action Plan48 Item III.B.1 called for the NRC is to transfer lead responsibility with regard to state and local government radiological emergency preparedness to the Federal Emergency Management Agency (FEMA) in accordance with the December 7, 1979 Presidential Order. This issue is limited to dividing the responsibility for the review and approval of state and local government emergency preparedness between NRC and FEMA and to the timely imple-mentation of their respective review and approval responsibilities. Public and occupational risk benefits resulting from the development of upgraded and additional emergency planning and preparedness requirements and the implementation of those requirements have been assessed in the evaluation of the items under Task III.A "Emergency Preparedness and Radiation Effects."

Memoranda of Understanding between the NRC and FEMA were signed on October 22, 1980315 regarding incident response and November 4, 1980316 regarding radiological emergency planning and preparedness by the state and local governments.

As delineated in these memoranda, NRC will make decisions with regard to the overall state of emergency preparedness for the issuance of operating licenses or the shutdown of operating reactors. Onsite emergency preparedness will be determined by NRC, offsite radiological emergency preparedness will be determined by FEMA with review by NRC, and the integration of the two determinations will be made by NRC.

Coordinated NRC and FEMA efforts in the review of state and local emergency plans, evaluation of exercises to test plans, preparation of emergency preparedness guidance, training of state and local officials, and development of a public information program concerning emergency preparedness have been completed.

FEMA and NRC have completed the evaluation of the first round of joint exercises at all operating plants. All work required by Items III.B.1 and III.B.2 has been completed.256,379

CONCLUSION

Items III.B.1 and III.B.2 have been RESOLVED.

ITEM III.B.2: IMPLEMENTATION OF NRC AND FEMA RESPONSIBILITIES

ITEM III.B.2(1): THE LICENSING PROCESS

This item was evaluated in Item III.B.1 above and was determined to be RESOLVED.

ITEM III.B.2(2): FEDERAL GUIDANCE

This item was evaluated in Item III.B.1 above and was determined to be RESOLVED.

Task III.C: Public Information ( )

The objectives of this task are: (1) to have information available for the news media and the public describing how nuclear plants operate, radiation and its health effects, and protective actions against radiation; and (2) to provide training for members of the technical staff on how to interface with the news media and other interested parties.

ITEM III.C.1: HAVE INFORMATION AVAILABLE FOR THE NEWS MEDIA AND THE PUBLIC

The three parts of this item have been combined and evaluated together.

DESCRIPTION

The objective of this TMI Action Plan[48] item is to have information available to the news media and the public describing how nuclear plants operate, radia-tion and its health effects, and protective actions against radiation.

PA has completed a review of publicly available documents and found the list wanting. FEMA, partially in response to the above, is working on a booklet about radiation facts and how power reactors work. Several drafts of the booklet have been distributed to appropriate agencies for review but the timing of actual publication is not known. Additionally, information about radiation is now disseminated as part of the off-site emergency plan for each site. The first series of news media seminars about how reactors work and facts on radiation has been completed (one in each region). A second round is now underway.[340] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM III.C.1(1): REVIEW PUBLICLY AVAILABLE DOCUMENTS

This Licensing Issue was evaluated in Item III.C.1 above and was determined to be resolved.

ITEM III.C.1(2): RECOMMEND PUBLICATION OF ADDITIONAL INFORMATION

This Licensing Issue was evaluated in Item III.C.1 above and was determined to be resolved.

ITEM III.C.1(3): PROGRAM OF SEMINARS FOR NEWS MEDIA PERSONNEL

This Licensing Issue was evaluated in Item III.C.1 above and was determined to be resolved.

ITEM III.C.2: DEVELOP POLICY AND PROVIDE TRAINING FOR INTERFACING WITH THE NEWS MEDIA

Both parts of this item have been combined and evaluated together.

DESCRIPTION

The TMI Action Plan48 identified a need to develop policy and procedures for dealing with briefing requests from State and local officials, Congress, other Federal officials, the media, and others during emergencies. A plan for prompt but accurate notification of the news media is also included.

Technical staff from the NRC have been designated to assist PA in the event of an accident. A briefing book consisting of detailed visual materials covering reactor facilities manufactured by all four vendors has been completed and distributed to the Regional offices, the Operations Center and the Commissioners.[340] This item is not directly related to safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM III.C.2(1): DEVELOP POLICY AND PROCEDURES FOR DEALING WITH BRIEFING REQUESTS

This Licensing Issue was evaluated in Item III.C.2 above and was determined to be resolved.

ITEM III.C.2(2): PROVIDE TRAINING FOR MEMBERS OF THE TECHNICAL STAFF

This Licensing Issue was evaluated in Item III.C.2 above and was determined to be resolved.

REFERENCES

0048.	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.
0340.	Memorandum for H. Denton from J. Fouchard, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 17, 1983. [8302030055]

Task III.D: Radiation Protection ( )

TASK III.D.1: RADIATION SOURCE CONTROL

The objective of this task is to perform evaluations to establish additional design features that should be included in the rulemaking proceeding of Item II.B.8. The purpose of these evaluations is to identify design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident.

ITEM III.D.1.1: PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

The three parts of this item are evaluated separately below.

ITEM III.D.1.1(1): REVIEW INFORMATION SUBMITTED BY LICENSEES PERTAINING TO REDUCING LEAKAGE FROM OPERATING SYSTEMS

This item was clarified in NUREG-0737[98] and requirements were issued.

ITEM III.D.1.1(2): REVIEW INFORMATION ON PROVISIONS FOR LEAK DETECTION

DESCRIPTION

This TMI Action Plan[48] item called for evaluations to identify design features that will reduce the potential for radiation exposure to workers at nuclear power plants and to the offsite population following an accident. Item III.D.1.1 called for the staff to evaluate the likelihood of worker exposure and of releases of radioactivity due to potential sources of radiation and airborne radioactivity from primary coolant that may be in **systems outside the containment structure** following an accident. The adequacy of the existing acceptance criteria for the design of vent-gas and other systems outside the containment structure were to be evaluated and the need for leak detection systems determined. Item III.D.1.1(2) called for NRR to select a contractor to review information on : (1) provisions for leak detection, equipment arrangement drawings, piping drawings, and fabrication criteria (specifications) for systems (e.g., makeup and purification, RHR, RCIC, vent gas, etc.)[57] that may contain substantial amounts of radioactivity after an accident; and (2) primary-to-secondary steam generator leakage. The review was to be performed on selected operating reactors and for plants that were in the OL review stage at the time the TMI-2 accident occurred. Plants were to be selected to provide those typical of each NSSS vendor.

CONCLUSION

Radiation protection of workers has been and continues to be addressed at nuclear power plants by various means, including area radiation monitors, health physics surveys, personnel dosimetry and administrative controls (locked doors to radiation areas, HP procedures, etc.). These provisions have the capability to protect workers from excess exposure both during routine operation and after an accident. In addition, plant systems outside of containment which have the potential to be contaminated with radioactive material (either by leakage from the primary system or by misoperation) are provided with process radiation monitors. These include the following:

PWRs

Containment Atmosphere Vent Condenser Air Ejector Steam System

RHR

Containment Cooling Service Water Service Water Waste Processing Effluent (wich includes monitoring for the Chemical and Volume Control System)

BWRs

Reactor Building Ventilation Condenser Off-Gas RHR Service Water Reactor Building Closed Cooling Water Service Water Radiaoactive Waste Reactor Water Cleanup

These radiation monitors have the capability to alert the operators if radioactive material is present in the system.

Additional radiation monitoring provisions, beyond those discussed above, were addressed in October 1980 when TMI Action Plan[48] Item II.F.1, "Additional Accident-Monitoring Instrumentation," was clarified in NUREG-0737[98] and requirements were issued to licensees. These requirements consisted of the following: (1) noble gas effluent radiological monitors; (2) provisions for continuous sampling of plant effluents for post-accident releases of radioactive iodines and particulates and on-site laboratory capabilities; (3) containment high-range radiation monitor; (4) containment pressure monitor; (5) containment water level monitor; and (6) containment hydrogen concentration monitor. As a result of these requirements, displays and controls were to be added to control rooms for use by operators during normal and abnormal plant conditions. MPAs F-20, F-21, F-22, F-23, F-24, and F-25 were established by the staff to follow licensee implementation of these requirements. Items (1) and (2) are directed toward monitoring releases from the plant and would provide input for taking measures to protect the offsite population, if such measures were necessary.

The advent of Leak-Before-Break (LBB) technology has also focused attention on leakage detection systems and methodologies. On August 28, 1987, the Broad Scope amendment to GDC-4 was published for public comment. This amendment will allow licensees to apply LBB technology to many systems both inside and outside of containment. Stringent acceptance criteria apply, including requirements to have leakage detection systems and/or methodologies in place for both radioactive and non-radioactive systems; the leak detection capability must be equivalent to that specified in Regulatory Guide 1.45.[603] Where LBB is used in the design, such leak detection provisions will provide additional assurance that the release of radioactive material is detected.

Other ongoing activities directly related to the concerns of this item include Issue 66, "Steam Generator Requirements," which addresses primary-to-secondary leakage limits, and Issue 119.5, "Leak Detection Requirements," which addresses leak detection in BWR reactor coolant pressure boundary stainless steel piping (4" diameter or larger) inside or outside of containment.

The safety concerns raised in this issue are similar to those that have been addressed in reactor designs in other issues and in operating practices directed toward worker protection, as outlined above. No additional radiation monitoring, protection, or leak detection provisions have been identified as needed. Therefore, this item was DROPPED from further consideration.

ITEM III.D.1.1(3): DEVELOP PROPOSED SYSTEM ACCEPTANCE CRITERIA

DESCRIPTION

This TMI Action Plan[48] item called for NRR to review the findings of Item III.D.1.1(2), determine the need for requiring leak-detection systems, and develop proposed acceptance criteria for these systems. The proposed acceptance criteria were to be factored into the resolution of Item II.B.8, "Rulemaking Proceeding on Degraded Core Accidents."

CONCLUSION

The need for requiring leak-detection systems and the development of new acceptance criteria for these systems were pursued by the staff in other issues [see Item III.D.1.1(2)]. As a result, Item III.D.1.1(3) did not provide any data for consideration in Item II.B.8 which was resolved in August 1985. Therefore, Item III.D.1.1(3) was DROPPED from further consideration.

ITEM III.D.1.2: RADIOACTIVE GAS MANAGEMENT

DESCRIPTION

Historical Background

The concern expressed in this TMI Action Plan[48] item is that an accident at any operating nuclear plant could result in the release of significant quantities of radioactive noble gases to the containment atmosphere. Since there are no noble gas recovery systems installed in nuclear plants that could process these large volumes of noble gases, there is no viable alternative to eventual discharge of long-lived noble gases to the environment. It was recommended that a study be initiated to determine the applicability and desirability of the use of available technology to minimize the release of radioactive noble gases during and following various postulated accidents. This study should include an investigation of viable alternatives for storage or disposal of the gases, an assessment of the potential pathways for gaseous releases, and consideration of accelerated rates of treatment of large gas volumes present in large containment structures.

Safety Significance

Discharge of large volumes of long-lived noble gases to the environment following an accident can increase the exposure to personnel on site and to the population in close proximity to the site.

Possible Solution

For the purpose of this analysis it is assumed that the study described above will result in increased capacity of the radioactive gas management systems at all plants.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to increasing the capacity of radioactive gas management systems is not certain, but it is estimated to range from zero to 5%. From calculations for Item III.D.2.1 in NUREG/CR-2800,[64] a 1% decrease in the dose factors for PWR Release Categories 1 through 7 and BWR Release Categories 1 through 4 results in an estimated total public risk reduction of 8,500 man-rem for all plants (144). Assuming a decrease in the dose factors of 0.5% for this issue, the estimated public risk reduction is 4,250 man-rem.

Cost Estimate

It will be assumed that space is available at all plants for increasing the retention capacity of radioactive gas management systems by installing pressure vessels such as tanks. The hardware cost per plant is estimated to be about $100,000. Engineering and design costs are estimated to be $50,000 and installation costs would be about $100,000. Therefore, the total industry cost for equipment development, installation, support facilities, and construction labor is estimated to be $0.25M per plant and the total industry cost for implementing the possible solution in 144 plants is $36M.

Industry costs for labor and material associated with operation and maintenance of the possible solution are estimated to be similar to those for Item III.D.2.1, i.e., $16M, based on a cost of $4,000/RY.

The NRC cost is assumed to be limited to implementation costs for development and plant installation of increased-capacity gas management systems. It is esti-mated that 1.5 years (~75 man-weeks) of NRC time would be required for research, criteria development, and engineering development and 0.3 man-week/plant (43 man-weeks) would be required for NRC administrative and technical effort for the review and approval of licensee submittals. Thus, the total NRC cost associated with the possible solution is (75 + 43) man-weeks x $2,000/man-week or $0.24M.

Therefore, the total of all costs related to resolution of this issue is $[36 + 16 + 0.24]M or $52.2M.

Value/Impact Assessment

Based on a total risk reduction of 4,250 man-rem, the value/impact score is given by:

$$S = \frac{4{,}250\ man-rem}{\$52.2\,M}$$
$$= 82\ man-rem/\$M$$

CONCLUSION

An evaluation of this issue was performed by the staff in SECY-81-450[228] in which a portable emergency unit was considered as a viable solution to this issue. This unit could be transported to the site of a serious reactor accident and used to selectively absorb and contain the noble gases from the containment atmosphere. It was concluded in SECY-81-450[228] that the cost of developing and maintaining such a system would be high for a relatively small dose reduction. The evaluation in SECY-81-450[228] supported the value/impact assessment above and the issue was DROPPED from further consideration.

ITEM III.D.1.3: VENTILATION SYSTEM AND RADIOIODINE ADSORBER CRITERIA

The four parts of this item have been combined and analyzed together.

DESCRIPTION

This TMI Action Plan[48] item required the NRC to make provisions to ensure that there is adequate filtration of radioactivity in ventilation exhausts and that acceptable collection efficiencies of radioiodine adsorbers are maintained during accident conditions. Items III.D.1.3(1), (2), and (3) called for various studies and possible modifications/upgrades as well as revisions to several SRP Sections and Regulatory Guides. These items were defined in May 1980 when the TMI Action Plan[48] was published and are now no longer valid. Subsequent to the publication of the TMI Action Plan, the NRC developed the Severe Accident Research Program (SARP) and the Source Term Program Plan. The objective of Items III.D.1.3(1), (2), and (3) are covered by various programs within the SARP. The Source Term Program provides the mechanism that assures the results of the SARP are incorporated into the licensing process.[149] Item III.D.1.3(4), associated with the evaluation of charcoal adsorber radioiodine collection performance under accident conditions, was completed149 in July 1982 and documented in NUREG/CR-2550.[473]

CONCLUSION

Items III.D.1.3(1), (2), and (3) are covered in the SARP and the Source Term Program Plan and were dropped from further consideration as separate issues; Item III.D.1.3(4) was resolved with no new requirements.

ITEM III.D.1.3(1): DECIDE WHETHER LICENSEES SHOULD PERFORM STUDIES AND MAKE MODIFICATIONS

This item was evaluted in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(2): REVIEW AND REVISE SRP

This item was evaluated in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(3): REQUIRE LICENSEES TO UPGRADE FILTRATION SYSTEMS

This item was evaluated in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(4): SPONSOR STUDIES TO EVALUATE CHARCOAL ADSORBER

This item was evaluated in Item III.D.1.3 above and was determined to be RESOLVED with no new requirements.

ITEM III.D.1.4: RADWASTE SYSTEM DESIGN FEATURES TO AID IN ACCIDENT RECOVERY AND DECONTAMINATION

DESCRIPTION

Historical Background

This TMI Action Plan48 Item required a study to investigate the improvements that could be made to radwaste system design features to provide the capability to process accident-related liquids and gases and to achieve decontamination effectively.

Safety Significance

The resolution of this issue would have no effect on reducing public risk. Any improvement in radwaste system design features would not reduce the core melt frequency or public dose. However, there is some occupational risk reduction associated with the radwaste system design improvements.

Possible Solutions

For the purpose of this evaluation, it will be assumed that the study will result in the following recommended changes to the radwaste systems of all plants in operation and under construction: (1) piping and connections installed for attaching a portable demineralization system; (2) additional spray nozzles in containment directed for wash-down of major surfaces; and (3) addition of shielding on stairways inside containment.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

An analysis of this issue was performed64 by PNL and it was found that there is no public risk reduction associated with this issue.

Other Considerations

There are several factors to be considered before a conclusion can be drawn on this issue:

(1) **Accident Occupational Risk Reduction**

The implementation of the possible solution is expected to reduce occupational radiation dose from cleanup, repair, and refurbishment. From studies conducted by PNL in NUREG/CR-2800,64 the total accident occupational risk reduction is 510 man-rem for all 134 affected plants.

(2) **Implementation Occupational Risk Increase**

Implementation of the solution would require work to be performed in radiation zones while the reactor is in a shut-down mode. Based on an average of 300 man-weeks/plant in a field of 2.5 millirem/hr, the total occupation dose due to implementation back-fit for 71 operating plants was calculated by PNL64 to be 1,630 man-rem.

(3) **Operation and Maintenance Occupational Risk Increase**

Based on an average time of 1 man-week/plant-yr in a field of 2.5 millirem/hr, the total occupational dose due to operation and maintenance of the possible solution was calculated by PNL64 to be 284 man-rem for all 134 affected plants.

(4) **Industry Cost Estimate**

PNL has estimated that the total cost for implementation of the solution on all plants is $375M, with operation and maintenance costs amounting to an additional $8.6M. In the event of accidents, industry is expected to save approximately $12M for cleanup, repair, and refurbishment based on a 10% reduction in the occupational dose associated with these accidents. Thus, the net industry cost for this issue is $(375 + 8.6 - 12)M or $372M.

(5) **NRC Cost Estimate**

NRC costs associated with this issue are insignificant in comparison to industry costs and have been estimated by PNL 64 to be approximately $3M for all plants.

CONCLUSION

A consideration of the risk associated with this issue shows that the occupational dose increases (for implementation and operation and maintenance) of the possible solution far outweigh the occupational risk reduction during accident conditions. In addition, the cost for implementing the solution is very high with no resultant reduction in public dose. As a result, this issue was placed in the DROP category.

REFERENCES

0048.	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.
0057.	NUREG#0578, "TMI#2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

0149.	Memorandum for J. Funches from R. Mattson, "Comments on Prioritization of Licensing Improvement Issues," February 2, 1983. [8401170099]
0228.	SECY-81-450, "Development of a Selective Absorption System Emergency Unit," U.S. Nuclear Regulatory Commission, July 27, 1981. [8108140094]
0473.	NUREG/CR#2550, "Charcoal Performance Under Simulated Accident Conditions," U.S. Nuclear Regulatory Commission, July 1982.
0603.	Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," U.S. Nuclear Regulatory Commission, May 1973. [7907100185]

Task III.D.2: Public Radiation Protection Improvement (Rev. 4) ( )

The objective of this task was to improve public radiation protection in the event of a nuclear power plant accident by improving (1) radioactive effluent monitoring, (2) the dose analysis for accidental releases of radioiodine, tritium, and carbon-14, (3) the control of radioactivity released into the liquid pathway, (4) the measurement of offsite radiation doses, and (5) the ability to rapidly determine offsite doses from radioactivity release by meteorological and hydrological measurements so that population-protection decisions can be made appropriately.

ITEM III.D.2.1: RADIOLOGICAL MONITORING OF EFFLUENTS

The three parts of this item were combined and evaluated together.

DESCRIPTION

Historical Background

This Three Mile Island (TMI) Action Plan[48] item required development and implementation of acceptance criteria for monitors used to evaluate effluent releases under accident and postaccident conditions. Criteria were to be developed for pathways to be monitored (stack, plant vent, steam dump vents) as well as for monitoring instrumentation. This was seen to encompass the requirements in Recommendation 2.1.8-b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," issued July 1979,[57] and Appendix 2 to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."[224]

Liquid effluents were not envisioned as posing a major release pathway because licensees typically had installed, or were installing, adequate storage capacity to prevent discharges. Consequently, existing liquid effluent monitoring systems were considered to be adequate.

Safety Significance

This issue had no impact on core-melt accident frequency.

Possible Solution

The envisioned monitoring system would provide automatic online analysis of airborne effluents, including isotopic analyses of particulate, radioiodine, and gas samples. To prevent saturation of detectors, an automatic sample cartridge changeout feature would be included. The system would include microprocessor control and real-time readouts and would be located in a low postaccident background area. The sampling system would be designed to provide a representative sample under anticipated accident release conditions.

A pressurized-water reactor (PWR) steam-dump sampling and monitoring system would be provided for PWR safety relief and vent valves. Such a system might consist of a noble gas monitor and a radioiodine sampling and monitoring system. The features of such a system would be similar to the above-described airborne effluent monitor with two notable differences: (1) the system would be required to function in a very high humidity (steam-air mixture) environment, and (2) operation would only be required during actual steam venting. Because such venting is usually of a short-term or intermittent duration, the monitoring system activation could be keyed to the opening of the vents.

PRIORITY DETERMINATION

Assumptions

It was assumed that improved radiological monitoring of airborne effluent would result in a reduction of public risk. The following section presents the U.S. Nuclear Regulatory Commission (NRC) staff analysis for prioritizing this issue, which was performed in 1998. This analysis, which includes frequency, consequence, and cost estimates and a value/impact assessment, has not been updated in the 2011 revision of this issue.

Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to improved radiological monitoring of airborne effluents was not certain, but it was estimated by Pacific Northwest Laboratory (PNL)[64] to range from 0 to 1 percent, based on the following logic.

Existing radiological monitoring requirements, as contained in NUREG-0737, "Clarification of TMI Acton Plan Requirements,"[98] require real-time noble gas monitoring with sampling and laboratory analysis capabilities for radioiodines and particulates. Design-basis conditions defined in NUREG-0737[98] (100 microcuries per cubic centimeter radioiodines and particulates, 30-minute sample time) indicated that sample collection devices would pose special handling and analysis problems due to very high radioactivity buildup. Consequently, licensees typically provided alternate sample collection and analysis procedures. Execution of those procedures was estimated to require between 2 and 3 hours. During this time, radioiodine and particulate releases would be estimated based on computer-modeled interpretation of noble gas monitor readings, or on previous postaccident containment atmosphere analysis results, if such results were available. Public protective action recommendations would be made based on modeled estimates rather than actual effluent data. It was assumed that these recommendations would err on the conservative side (e.g., evacuate when not really required), due to the conservatism built into the modeled source terms for radioiodine and particulate releases.

Requiring licensees to have more sophisticated airborne effluent monitors would reduce the time required to obtain actual radioiodine and particulate release data to 15 minutes and essentially eliminate reliance on conservative theoretical release models extrapolated from noble gas monitor readings. As projected by the possible solution, real-time isotopic monitoring would save nearly 2 hours in arriving at realistic protective action recommendations based on actual releases.

Under these circumstances, the public risk reduction would be directly attributed to the decrease in public radiation exposure that would result from a more rapid assessment of the radioactive releases (about a 2-hour savings in analysis time). In addition, public risk may be reduced as a result of nonevacuation. The need for evacuation (presumed to exist if release knowledge was based only on noble gas monitor data) could be eliminated as a result of better knowledge of the isotopic releases. Nonevacuation would result in fewer evacuation-related risks (e.g., traffic accidents), the avoidance of which may outweigh the radiation exposure received. However, it was assumed that the public risk reduction would result primarily from the first effect (decrease in exposure due to more rapid assessment).

While protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario was such that actions would be recommended based on anticipated releases, before the actual releases themselves. Under this assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification. This possible solution would not impact core-melt accident frequency.

At the time of this evaluation, there were 134 plants affected by the issue: 71 operating (47 PWRs and 24 boiling-water reactors (BWRs)) and 63 planned (43 PWRs and 20 BWRs). It was assumed that the average remaining plant life was 27.4 years for the 44 BWRs and 28.8 years for the 90 PWRs. The dose factors for PWR Release Categories 1 through 7 and BWR Release Categories 1 through 4 were assumed to be affected by the possible solution. From NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development,"[64] a 1-percent decrease in the dose factors resulted in an estimated total public risk reduction of 8,500 man-rem for all plants. Assuming a decrease in the dose factors of 0.5 percent for this issue, the estimated public risk reduction was 4,250 man-rem.

Cost Estimate

Industry Cost: The industry cost for equipment development, installation, support facilities, and construction was estimated to be $600,000/plant. Development of procedures, software, and calibration for the equipment was estimated to require 16 man-weeks of effort, with an additional 4 man-weeks for the initial training of all licensee operators and health physics personnel. This was estimated to add $45,400/plant to the implementation cost. Based on an estimated cost of $645,000/plant for labor and equipment, the industry cost for implementing the possible solution was (134 plants)($645,000/plant) or $86.5 million (M).

The recurring industry operation and maintenance costs were estimated at 2 man-weeks/plant-year for retraining, 1 man-week/plant-year for calibration, and a reduction of 1 man-week/plant-year (reduced laboratory analyses due to a fully automated system) for a net increase of 2 man-weeks/plant-year, or an increased cost of

$4,540/plant-year. As a result, industry costs for labor and material associated with operation and maintenance of the possible solution were estimated to be $17.2M.

Thus, the total industry cost associated with this issue was $(86.5 + 17.2)M or $103.7M.

NRC Cost: The NRC cost was assumed to be limited to implementation costs for development and plant installation. Because it was assumed that the new radiological monitoring systems would require no periodic inspection effort beyond that required for current systems, no additional NRC operation cost was envisioned. The NRC development cost included 1.5 man-years and $200,000 for research, criteria development, and engineering development, for a total cost of $350,000. The NRC administrative and technical effort associated with the review and approval of licensee submittals was estimated at 0.3 man-week/plant for a total cost of $91,000 for all plants. Therefore, the total NRC cost associated with this issue was $441,000.

Total Cost: The total industry and NRC cost associated with the possible solution was $(103.7 + 0.441)M or $104.1M.

Value/Impact Assessment

Based on an estimated public risk reduction of 4,250 man-rem and a cost of $104.1M for a possible solution, the value/impact score was given by the following:

$$S = \frac{4{,}250 \text{ man-rem}}{\$104.1\text{M}} = 41 \text{ man-rem/\$M}$$

Other Considerations

It was anticipated that improvement of radiological monitoring of airborne effluents would have no significant impact on occupational risk. The dose required to install equipment would probably not exceed 0.5 man-rem, which was negligible compared to the typical 600 man-rem/year required to operate a plant. Minor man-rem savings might occur under accident conditions due to better direction of field survey teams; however, such savings would be negligible compared to the 19,900 man-rem total associated with response and cleanup following an accident.

Based on an estimated occupational dose of 0.5 man-rem/plant for implementation of the possible solution in 71 operating plants, the total risk increase was 36 man-rem for all plants. Inclusion of this factor into the above calculation would reduce the value/impact score.

There was no accident avoidance cost for the resolution of this issue because improved radiological effluent monitoring systems would have no impact on accident frequency or cleanup and refurbishing costs.

CONCLUSION

Based on the risk reduction potential and value/impact score, the issue was given a LOW priority ranking (see Appendix C) in November 1983. NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," issued December 1991,[1563] concluded that consideration of a 20-year license renewal period could change the ranking of the issue to medium priority. Further prioritization in 1995, using the conversion factor of $2,000/man-rem approved[1689] by the Commission in September 1995, resulted in an impact/value ratio (*R*) of $24,390/man-rem, which did not change the priority ranking. In 2010, the staff reviewed three parts of this issue in accordance with the SRM 871021A, "Staff Requirements—Briefing on Status of Unresolved Safety/Generic Issues," dated November 6, 1987,[1980] which directed the staff to conduct periodic reviews of existing LOW-priority issues to determine whether any new information was available that would necessitate reassessment of the original prioritization evaluations.[1964] Based on this review, the status of these issues was changed as described below.

ITEM III.D.2.1(1): EVALUATE THE FEASIBILITY AND PERFORM A VALUE/IMPACT ANALYSIS OF MODIFYING EFFLUENT-MONITORING DESIGN CRITERIA

The overall objective of this issue, which "is to provide assurance that all possible accident effluent-release pathways are monitored and that monitors will perform properly under accident conditions," is covered by

General Design Criterion (GDC) 64, "Monitoring Radioactivity Releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." GDC 64 states that "Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents." Moreover, 10 CFR 50.34(f)(2)(xvii)(E) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "Provide instrumentation to measure, record and readout in the control room:…(E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples." Finally, 10 CFR 50.34(f)(2)(xxvii) and (2)(xxviii) establish requirements for monitoring of inplant radiation and airborne radioactivity for a broad range of routine and accident conditions and for evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions.

In addition to the regulations stated above, Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (the SRP),[11] states that "Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems." Table 1 of Section 11.5 of the SRP[11] specifies the gaseous streams or effluent release points that should be monitored and sampled. In addition, for monitoring the effluents during a postulated event, Section 11.5 of the SRP[11] states that "Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident."

As explained earlier, implementation of the proposed solutions has no impact on the core-melt accident frequency. Moreover, "while protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario was such that actions would be recommended based on anticipated releases prior to the actual release themselves. Under this assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification."

Specific requirements related to some of the factors in the proposed design criteria mentioned in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," have not been established; however, based on the review of the NRC’s regulations presented above, the staff concluded that the overall objectives of Item III.D.2.1(1) are met by the existing regulations. Moreover, the low safety significance of the issue does not warrant further actions to evaluate and implement the proposed solutions. Therefore, the staff changed the status of Item III.D.2.1(1) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.1(2): STUDY THE FEASIBILITY OF REQUIRING THE DEVELOPMENT OF EFFECTIVE MEANS FOR MONITORING AND SAMPLING NOBLE GASES AND RADIOIODINE RELEASED TO THE ATMOSPHERE

In addition to Criterion 64 of Appendix A to 10 CFR Part 50, the regulation at 10 CFR 50.34(f)(2)(xvii) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, "each applicant for a light-water reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "Provide instrumentation to measure, record and readout in the control room: …(E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

Based on the review of the NRC regulations related to this issue presented above and the low safety significance of this issue, the staff concluded that Item III.D.2.1(2) is adequately addressed by the existing regulations. Therefore, the staff changed the status of Item III.D.2.1(2) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.1(3): REVISE REGULATORY GUIDES

NUREG-0660[48] called for this issue to "revise Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Standard Review Plan Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems, and further revise Regulatory Guide 1.97, as necessary." All of these documents have been updated since the issuance of NUREG-0660.[48] Some specific factors of the design criteria mentioned in NUREG-0660[48] have not been included in these updates. However, the overall objective of the issue has been thoroughly addressed in these updates. As of April 2010, the latest revision of each document is available as follows: Regulatory Guide (RG) 1.21, Revision 2, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," issued June 2009[1968]; SRP[11] Section 11.5, issued March 2007; and RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, issued June 2006.[55]

Because of the revisions made to RG 1.21,[1968] SRP[11] Section 11.5, and RG 1.97,[55] the staff changed the status of Item III.D.2.1(3) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.2: RADIOIODINE, CARBON-14, AND TRITIUM PATHWAY DOSE ANALYSIS

The four parts of this item were combined and evaluated together.

DESCRIPTION

Historical Background

This TMI Action Plan[48] item addressed the issue of further research for improving the understanding of radioiodine partitioning in nuclear power reactors and of the environmental behavior of radioiodine, carbon-14, and tritium following an accident and during normal operation.

Iodine isotopes are considered to be major contributors to the occupational and public dose during a loss-of-coolant accident, along with noble gases and fission products. A study in these areas was documented in NUREG-0772, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents," issued June 1981,[212] with the following major conclusions: (1) uncertainties in predicting atmospheric release source terms were very large (at least a factor of 10), (2) source terms for certain accident sequences may have been overestimated in past studies; e.g., WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," issued October 1975,[16] and (3) cesium iodide should be the predominant chemical form of iodine under severe accident conditions.

Safety Significance

The above conclusions indicated that the methodology and assumptions used for evaluating radioiodine release could result in unrealistic estimates (e.g., RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,"[213] and RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"[214]). Also indicated was that more research in aerosol behavior and fission product chemistry was needed in order to improve and support the calculation methodology concerned with radioiodine partitioning, fission product behavior, and related topics.

Possible Solution

The NRC assumed that further study would improve the understanding of this issue and result in more realistic assumptions and methods for evaluating source terms, releases, and the environmental behavior of radioiodine, carbon-14, and tritium following an accident. This research would not affect accident frequencies at nuclear power plants. However, the NRC assumed that the results of these studies would be used to revise the SRP[11] and RGs.

The NRC also assumed that the RG revisions could result in reducing the size of existing emergency planning zones from a 10-mile radius to a 2-mile radius. This assumption was based on a reduction of source terms in a core-melt accident by a factor of 10. This would result in reducing dose concentration at a particular distance from the nuclear reactor also by a factor of 10. Assuming neutral weather conditions with a 30-meter-high plume, the offsite dose predicted at 2 miles from the accident scene, using the reduced source term assumption, would be the same as that predicted at 10 miles from the reactor.

CONCLUSION

The study of radioiodine, carbon-14, and tritium behavior at Three Mile Island Unit 2 (TMI-2) called for in Item III.D.2.2(1) was completed in June 1981 and documented in NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions," issued June 1981,[455] and NUREG-0772.[212] Items III.D.2.2(2), (3), and (4) called for a series of studies and evaluations of various radionuclide pathways and models followed, if necessary, by revisions to several SRP[11] sections and RGs. As part of the staff’s task to prepare and publish a manual (referred to as the "Offsite Dose Calculation Manual"[598]) to be used by the NRC and industry to estimate individual and population doses during normal and accident conditions, Items III.D.2.2(2), (3), and (4) were assessed. This Offsite Dose Calculation Manual was prepared under Item III.D.2.5 and fully described each of the theoretical models used to predict radionuclide transport.[149] Thus, Items III.D.2.2(2), (3), and (4) were covered under Item III.D.2.5.

ITEM III.D.2.2(1): PERFORM STUDY OF RADIOIODINE, CARBON-14, AND TRITIUM BEHAVIOR

This item was evaluated in Item III.D.2.2 above and was RESOLVED with no new requirements.

ITEM III.D.2.2(2): EVALUATE DATA COLLECTED AT QUAD CITIES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2.(3): DETERMINE THE DISTRIBUTION OF THE CHEMICAL SPECIES OF RADIOIODINE IN AIR-WATER-STEAM MIXTURES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2.(4): REVISE SRP AND REGULATORY GUIDES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.3: LIQUID PATHWAY RADIOLOGICAL CONTROL

The four parts of this item were combined and evaluated together.

DESCRIPTION

This TMI Action Plan[48] item was concerned with improving public radiation protection in the event of a nuclear power plant accident by improving the control of radioactivity released into the liquid pathway. This control could be accomplished by the application of various interdictive measures at the source of the release and/or along the liquid pathway. Techniques were developed and were being used to evaluate the liquid pathway effects of an accident for each reactor site. Sites that might require interdictive measures related to liquid pathway releases were to be determined. Interdictive measures were to be assessed as to their effectiveness in improving public radiation protection.

CONCLUSION

A liquid pathway analysis for Zion Nuclear Power Station was completed by the Office of Nuclear Reactor Regulation’s Division of Engineering in 1980.[391] In addition, a liquid pathway analysis was performed for the Indian Point nuclear power plant. Both analyses were used in NUREG-0850, "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," issued November 1981.[390] A simplified analysis for liquid pathway studies (NUREG-1054, "Simplified Analysis for Liquid Pathway Studies,")[658] was published in August 1984, and Section 7.1.1 of NUREG-0555, "Environmental Standard Review Plans for the Environmental Review of Construction Permit Applications for Nuclear Power Plants" (the ESRP), issued May 1979,464 was drafted with no new requirements for licensees or applicants.[659, 660] ESRP Section 7.1.1 was finally published as NUREG-1165, "Environmental Standard Review

Plan for ES Section 7.1.1,"[838] in November 1985. Thus, this item was RESOLVED and no new requirements were established.[799]

ITEM III.D.2.3(1):DEVELOP PROCEDURES TO DISCRIMINATE BETWEEN SITES/PLANTS

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(2): DISCRIMINATE BETWEEN SITES AND PLANTS THAT REQUIRE CONSIDERATION OF LIQUID PATHWAY INTERDICTION TECHNIQUES

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(3): ESTABLISH FEASIBLE METHOD OF PATHWAY INTERDICTION

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(4): PREPARE A SUMMARY ASSESSMENT

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.4: OFFSITE DOSE MEASUREMENTS

ITEM III.D.2.4(1):STUDY FEASIBILITY OF ENVIRONMENTAL MONITORS

DESCRIPTION

This TMI Action Plan[48] item called for the staff to study the feasibility of environmental monitors capable of measuring real-time rates of exposure to noble gases and radioiodines. Monitors or samplers capable of measuring respirable concentrations of radionuclides and particulates were also considered. This activity supported proposed revisions to RG 1.97[55] (see Item II.F.3).

CONCLUSION

The establishment of guidance in RG 1.97[55] for fixed monitors to detect unidentified releases was postponed pending the outcome of a feasibility study that was completed in April 1982.[188] Using this study as a basis, the staff concluded that environmental monitors of this nature were not practical and that proposed requirements for these monitors should be dropped from consideration.[189] Thus, all required action on this item was completed[382] and the issue was RESOLVED with no new requirements.

ITEM III.D.2.4(2): PLACE 50 TLDs AROUND EACH SITE

DESCRIPTION

This TMI Action Plan[48] item called for Office of Inspection and Enforcement (OIE) to place 50 thermo-luminescent dosimeters (TLDs) around each site in coordination with States and utilities. During normal operation, OIE quarterly reports from these dosimeters were to be provided to NRC, State, and Federal organizations. In the event of an accident, the dosimeters could then be read at a frequency appropriate to the needs of the situation.

The specific objectives of this program were to (1) establish preoperational, historical, baseline radiation dose levels, whenever possible, for each monitored facility, (2) provide ongoing radiation dosimetry data during routine operations, (3) provide postaccident radiation dosimetry to aid in assessment of population exposures and radiological impact, (4) allow for independent verification of the adequacy of NRC licensees' environmental radiation monitoring programs, (5) provide uniform treatment of dosimeters with respect to handling, shipping, calibrating, reading, and data processing for all monitored facilities in the United States, and (6) provide uniform, consistent environmental radiation monitoring data for use by the Congress, Federal and State agencies, monitored facilities, and the public.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered to be a licensing issue.

CONCLUSION

OIE completed installation of TLDs at all operating reactors in August 1980 in accordance with the TMI Action Plan schedule. A direct radiation monitoring network was established and a program for routine reporting began. The completion of these activities was described in an OIE memorandum.[236] With the establishment of the NRC TLD direct radiation monitoring network, the installation of TLDs at all operating reactor sites, and the routine reporting of the TLD measurements, all work required by this item was completed.[236, 379] Thus, this licensing issue was resolved.

ITEM III.D.2.5: OFFSITE DOSE CALCULATION MANUAL

DESCRIPTION

Historical Background

This TMI Action Plan[48] item called for the Office of Nuclear Reactor Regulation to prepare a manual to be used by the NRC and plant personnel to estimate the maximum individual doses and population doses during an accident.

Safety Significance

This issue did not affect core-melt frequency or the amount of radioactivity released. Instead, it was intended to reduce the consequences of a major release by assuring that licensees have a rapid and sufficiently accurate method of estimating dose, and that communication between licensees and the NRC be expedited by a common standard calculation method used by both.

Possible Solution

The proposed manual was expected to include formulations with which to combine source term and meteorological measurements. This would determine offsite dose rates in a manner that would be standard among all parties making decisions on public protection and emergency response. Appendix 2 to NUREG-0654[224] established criteria for automated assessment of radiation doses in the event of an accident.

PRIORITY DETERMINATION

Frequency Estimate

Because the proposed solution to the issue did not affect core-melt accident frequency, the frequencies for the various release categories given for Oconee Nuclear Station, Unit 3, and Grand Gulf Nuclear Station, Unit 1, were used unchanged in the value/impact calculation.

Consequence Estimate

In an assessment[64] of this issue, PNL experts judged that a 1-percent reduction in public dose (man-rem) might be expected as a result of having an offsite dose calculation manual available. It was estimated that the changes in consequences would be much less (0.01 percent to 0.1 percent). Because all sequences would be affected and the risk from both PWRs and BWRs was about 210 to 250 man-rem/reactor-year (RY), the risk reduction was estimated to be 0.02 to 0.2 man-rem/RY.

At the time of the evaluation of this issue in November 1983, there were 43 PWRs and 27 BWRs operating, with cumulative experience of 350 RY and 260 RY, respectively. Considering the 36 PWRs and 21 BWRs that were under construction and assuming a plant life of 40 years, there were 2,810 PWR-years and 1,660 BWR-years in the future, for a total of 4,470 RY. Therefore, the total risk reduction associated with this issue was (0.2)(4,470)man-rem or 894 man-rem.

Cost Estimate

Industry Cost: For licensees, 4 man-weeks of training for implementation were assumed, since operators were being retrained periodically and this retraining could include dose calculation methods. This different method would not incur additional recurring costs. Thus, the total industry cost was estimated to be $7,700/plant or $0.98M for 127 plants.

NRC Cost: The NRC had already completed work on development of a portable computerized system for dose calculations to be used by the NRC regional offices. This was part of the program for NUREG-0654.[224] This program was developed to the point of field trials for the computerized system. Based on the development costs, an additional $125,000 to develop this package into a manual form for use by utilities was assumed. It was estimated that NRC site representatives could spend a minimal amount of time (about 2 days) to evaluate initial utility performance with the package. This was estimated to be $600/plant. Thus, the total NRC cost was approximately $200,000 for all plants.

Total Cost: The total industry and NRC cost associated with the possible solution was $(0.98 + 0.2)M or approximately $1.2M.

Value/Impact Assessment

Based on an estimated public risk reduction of 894 man-rem and a cost of $1.2M for a possible solution, the value/impact score was given by the following:

$$S = \frac{894 \text{ man-rem}}{\$1.2M} = 758 \text{ man-rem}/\$M$$

CONCLUSION

Based on the above value/impact score, the issue would have had a MEDIUM priority ranking (see Appendix C). However, before approval of the prioritization evaluation in November 1983, the Offsite Dose Calculation Manual was published as NUREG/CR-3332, "Radiological Assessment—A Textbook on Environmental Dose Analysis,"[599] in September 1983. Thus, the issue was RESOLVED and no new requirements were issued.[598]

ITEM III.D.2.6: INDEPENDENT RADIOLOGICAL MEASUREMENTS

DESCRIPTION

This TMI Action Plan[48] item dealt with independent radiological measurements; i.e., means of collecting data independent of licensees' programs. An OIE task force developed a plan and requirements for upgrading the capability of regional offices to perform independent radiological measurements during routine inspections and emergency response operations. The objective of the upgrade was to achieve consistent capability among the regional offices, including standardization in major equipment items such as mobile laboratory vans, gamma spectrum analysis equipment, radiation survey instrumentation, and air-sampling and monitoring devices.

Based on the recommendations of the task force, each region was equipped with complete mobile laboratories.[235] In some cases, this represented upgrading certain equipment or purchasing new equipment. This action item required that revisions be made to the inspection program to include the upgrading of the independent radiological measurements. The program was included in the routine OIE program for review and revision of the inspection program. As new equipment needs were identified, the program was to be revised and the equipment acquired.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered to be a licensing issue.

CONCLUSION

With the upgrading of independent radiological measurements and the implementation of other recommendations made by the task force, all work required by this item was completed.[235, 379] Thus, this licensing issue was resolved.

REFERENCES

0011.	*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.
0016.	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.

0048.	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.
0055.	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]
0057.	NUREG#0578, "TMI#2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0149.	Memorandum for J. Funches from R. Mattson, "Comments on Prioritization of Licensing Improvement Issues," February 2, 1983. [8401170099]
0188.	NUREG/CR#2644, "An Assessment of Offsite, Real-Time Dose Measurements for Emergency Situations," U.S. Nuclear Regulatory Commission, April 1982.
0189.	Memorandum for K. Goller from R. Mattson, "Proposed Changes to Regulatory Guide 1.97," July 29, 1982. [8208060339]
0212.	NUREG-0772, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents," U.S. Nuclear Regulatory Commission, June 1981.
0213.	Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974. [7907100054]
0214.	Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974. [7907100058]
0224.	NUREG#0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0236.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," June 30, 1982. [8208110023]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0382.	Memorandum for W. Minners from R. Mattson, "Schedules for Resolving and Completing Generic Issues," January 21, 1983. [8301260532]
0390.	NUREG#0850, "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," U.S. Nuclear Regulatory Commission, November 1981.
0391.	Memorandum for E. Reeves from J. Knight, "Zion Liquid Pathway Analysis," August 8, 1980. [8008210647]

0455.	NUREG#0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions," U.S. Nuclear Regulatory Commission, June 1981.
0598.	Memorandum for W. Dircks from R. Mattson, "Closeout of TMI Action Plan Task III.D.2.5, 'Offsite Dose Calculation Manual,'" January 17, 1984. [8402020114]
0599.	NUREG/CR-3332, "Radiological Assessment—A Textbook on Environmental Dose Analysis," U.S. Nuclear Regulatory Commission, September 1983.
0658.	NUREG-1054, "Simplified Analysis for Liquid Pathway Studies," U.S. Nuclear Regulatory Commission, August 1984.
0659.	Memorandum for H. Denton from R. Vollmer, "ESRP 7.1.1 `Environmental Impacts of Postulated Accidents Involving Radioactive Materials—Releases to Groundwater,'" September 25, 1984. [8410100758]
0660.	Memorandum for W. Dircks from H. Denton, "Generic Issue III.D.2.3 'Liquid Pathway Radiological Control,'" October 29, 1984. [8411190057]
0799.	Memorandum for W. Dircks from H. Denton, "Resolution of Generic Issue III.D.2.3—Liquid Pathway Studies," August 28, 1985. [8509050212]
0838.	NUREG-1165, "Environmental Standard Review Plan for ES Section 7.1.1," U.S. Nuclear Regulatory Commission, November 1985.
1563.	NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.
1689.	Memorandum for J. Taylor from J. Hoyle, "COMSECY#95#033"Proposed Dollar per Person-Rem Conversion Factor; Response to SRM Concerning Issuance of Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission and SRM Concerning the Need for a Backfit Rule for Materials Licensees (RES-950225) (WITS-9100294)," September 18, 1995. [9803260148]
1964.	Memorandum for B.W. Sheron from B.G. Beasley, "LOW Priority Generic Issues," March 17, 2011.[*ML092520025*]
1968.	Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," U.S. Nuclear Regulatory Commission, Washington, DC, December 1971, (Rev. 1) June 1974, (Rev. 2) June 2009.
1980.	Memorandum for V. Stello from S. Chilk, "Staff Requirements—Briefing on Status of Unresolved Safety/Generic Issues," November 6, 1987. [8711100418]

Task III.D.3: Worker Radiation Protection Improvement (Rev. 3) ( )

The objective of this task is to improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents, by improving radiation protection plans, health physics, inplant radiation monitoring, control room habitability, and radiation worker exposure data base.

ITEM III.D.3.1: RADIATION PROTECTION PLANS

DESCRIPTION

Historical Background

The purpose of this TMI Action Plan[48] item is to improve nuclear power plant worker radiation protection programs by better defining the criteria and responsibility for such programs. Detailed appraisals of health physics programs at all operating nuclear power plants were performed in 1980 and 1981. These appraisals, summarized in NUREG-0855,[204] indicated that certain generic deficiencies existed at many plants due in part to lack of specific performance criteria and/or assigned responsibility for programs. The establishment of a radiation protection plan as a guiding document for implementing procedures has been proposed as a method for formalizing commitment to specific performance criteria contained in Regulatory Guides and SRP Section 12.[11] Proposed guidance and acceptance criteria for radiation protection plans have been published in draft form as NUREG-0761.[205] A proposed amendment to 10 CFR 50 has been drafted.[206]

Safety Significance

The development of radiation protection plans has no impact on public safety. Instead, the safety significance lies in the reduction of occupational exposure.

Possible Solutions

As currently envisioned, radiation protection plans would tie together specific implementing procedures, many of which currently exist at licensed plants. Additional procedures may be required at many plants to fully implement the plan; however, extensive revision of procedures should not generally be required. Administrative and technical manpower would be required to develop the plan, revise and write procedures as necessary, and some additional equipment (such as additional survey equipment) may be required. Installation of such equipment should not require any significant work in radiation areas. The benefit of radiation protection plans would be in two primary areas: (1) reduction of individual and collective dose due to improved planning and controls for work in radiation areas, and (2) improved confidence in results of radiation protection programs.

PRIORITY DETERMINATION

The assessment of this issue and its resolution was first performed[64] by consensus opinions of four PNL health physicists who were extensively involved in the Health Physics Appraisal Program. These personnel included expertise from both industry and regulatory sides of the issue. Estimates of routine cost and probable man-rem reductions were discussed and agreed upon. For core-melt accident recovery and refurbishing, the panel assumed man-rem savings comparable on a percentage basis to those for routine operations. The cost impact of these man-rem savings was then estimated by a PNL expert involved in estimating accident recovery costs.

Frequency/Consequence Estimate

There are three terms in the estimation of occupational dose change due to this safety issue. These are the change due to accidents, the change due to issue resolution implementation, and the change due to resolution operation.

The estimated change due to accidents (the first term) is the change in the product of accident frequency and occupational dose associated with the recovery from an accident. As previously stated, no change in accident frequency is expected to occur due to this issue. However, a small change in occupational accident recovery dose is expected. Radiation protection plans are primarily oriented toward routine plant operation. In the event of a major core-melt accident, specialized procedures would have to be developed. Having the upgraded radiation protection plan for normal operation in place, however, is expected to result in improved specialized procedures if required. The resulting reduction in occupational dose for plant recovery is estimated to be slightly less than 5%. Using the estimates of total occupational dose resulting from recovery from an accident, as listed in Appendix D of NUREG/CR-2800,[64] this works out to 3.3 x 10^{-2} man-rem/RY for BWRs and 7.4 x 10^{-2} man-rem/RY for PWRs.

The implementation of radiation protection plans (the second term) would be an administrative effort. Therefore, there is zero exposure associated with implementation.

The establishment of radiation protection plans is estimated to result in a reduction of occupational risk during operation (the third term). This reduction would be due to improved controls on personnel dose and an improved ALARA Program. PNL's experts estimated the occupational dose reduction to be on the order of 5%.[64] However, the Occupational Radiation Protection Branch (ORPBR) of RES has been investigating the costs and benefits associated with radiation protection plans. Based on a comparison of plants with and without major radiation protection plans, it was estimated that occupational doses could be reduced by at least 10%. Savings of 25% or more appear achievable.[207] The 1980 average occupational dose was about 800 man-rem. Therefore, we will assume that radiation protection plans could avert 200 man-rem/RY.

Cost Estimate

PNL estimated that 35 man-weeks at a cost of $35,000 and equipment worth $50,000 would be required per plant to implement the radiation protection plans.[64] In order to operate under the new radiation protection plans, it was felt that most plants would have to add personnel. It was estimated that one professional and one technical staff member would be needed. At 52 weeks per year, this gives an additional 104 man-weeks per year for each plant, or $104,000 plant cost per year.

However, ORPBR has noted that the licensees' cost will vary widely depending on the adequacy of the present program.[206] In addition, since radiation protection plans have the effect of reducing the time workers are exposed, individual tasks are often speeded up. Some licensees have found that the savings resulting from reduced downtime have compensated for the cost of the program.

Currently, there are 43 operating PWRs with a cumulative experience of 350 RY and 27 BWRs with a cumulative experience of 260 RY. If we add to these the 36 PWRs and 21 BWRs under construction and assume a plant lifetime of 30 years, there are 3,200 RY remaining: 1,180 RY for BWRs and 2,020 RY for PWRs.

ORPBR has estimated that 5 NRC staff-years will be required.[206] Thus, NRC costs are estimated to be $500,000.

The total cost associated with the solution to this issue is $340.5M.

$$S = \frac{6.4x10^5 \text{man} - \text{rem}}{\$340.5M}$$

$$= 1.880 \text{man} - \text{rem} / \$M$$

Value/Impact Assessment

The total risk reduction associated with this issue is 6.4 x 10^5 man-rem. Therefore, the value/impact score is given by:

Uncertainties

The dominant parameters in the evaluation of this issue are the percent saving in occupational dose during normal operation, which is unlikely to be incorrect by more than a factor of ten, and the cost to the licensee, which is expected to be within a factor of 5. This implies a range in S from 100 to 30,000 man-rem/$M and a range in total man-rem saved of 6 x 10^4 to 6 x 10^6.

CONCLUSION

Based on the value/impact score and potential reduction in occupational dose, this issue was give a high priority ranking. In resolving this issue, the staff agreed to support alternative regulatory concepts which recognize the contributions of industry self-policing programs to the extent that such programs are effective and consistent with NRC regulatory responsibilities. As a result, the staff entered into a "Coordination Plan for Radiological Protection Activities" with INPO under a "Memorandum of Agreement Between INPO and the USNRC." Under this agreement, over the two-year period outlined in the Coordination Plan, NRR staff developed a method for evaluating industry performance in radiation protection programs incorporating ALARA concepts at power reactors and observed the INPO evaluation and assistance process at a number of operating facilities.

The staff performed analyses of a number of radiological data trends as part of the effort to determine if the power reactor industry has achieved successful ALARA-integrated radiation protection programs. An analysis of these trends and portions of the supporting data bases were documented in the report, "Summary Analysis of Selected Radiological Trends at Power Reactors."[912]

Following the staff's compilation of data and evaluation of a number of trends in radiological protection at power reactors, the staff concluded that most power reactor radiation protection programs are adequately incorporating ALARA concepts and can satisfactorily perform at a level which meets the objectives of Item III.D.3.1 Thus, this issue was RESOLVED and no new requirements were established.[913]

ITEM III.D.3.2: HEALTH PHYSICS IMPROVEMENTS

The four parts of this item have been combined and evaluated together.

DESCRIPTION

Historical Background

In this TMI Action Plan[48] item, four specific items were identified for resolution: (1) Requirement for Use of Certified Personnel Dosimeter Processors; (2) Audible Alarm Dosimeter Regulatory Guide; (3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments; and (4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria. Item (2) will not be considered further since Regulatory Guide 8.28 was issued in final form prior to this evaluation. Thus, Item (2) is considered resolved.

Safety Significance

(1) Requirement for Use of Certified Personnel Dosimetry Processors.

The proposed resolution would amend 10 CFR 20 to require that only nationally certified dosimetry processors be used by NRC licensees for personnel radiation dosimetry. Processors would be required to meet ANSI N13.11 (or its replacement standard) criteria for testing. Certification of processors would be performed by the National Voluntary Laboratory Accreditation Program (NVLAP) administered under the auspices of the U.S. Department of Commerce (DOC).

This issue does not specifically address core-melt accidents nor the public risk, occupational dose, or accident avoidance costs associated with such accidents. It is related to the worker's right to accurate measurements of occupational dose. The proposed resolution would require accurate and precise determinations of individual worker doses using dosimeters, readout systems, and processing procedures certified to be capable of meeting minimum criteria defined in a national standard. The administrative and regulatory limits for occupational dose would be unaffected by this work.

A draft ANSI standard (ANSI N13.11) for dosimeter testing was issued for trial use in 1978. This standard has undergone substantial testing and remains only to be finalized for issuance as a new ANSI standard. Once issued, it will form the basis for amending 10 CFR 20. Testing and certification of dosimeter processors for criteria contained in this standard will be performed by NVLAP under DOC.

(2) This item has been resolved as discussed before.

(3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments

Testing of radiation survey and monitoring instruments will provide a high degree of quality assurance that instruments are capable of performing intended functions under specified conditions. This will allow consistent utilization of workers without impacting current individual or collective occupational dose. A draft standard for health physics instrumentation testing (ANSI N42.17-D2) has been developed.

This standard will undergo a field trial period, using off-the-shelf instruments, to determine its adequacy. This trial period is presently estimated to continue through FY-1984 and is jointly funded by NRC and the Department of Energy (DOE) at $400,000 each. Following the trial period, a final standard will be adopted by NRC and only those instruments meeting this standard would be acceptable for use in NRC licensed facilities.

At this time, a plan for implementing the testing program has not been developed. It is anticipated, however, that independent testing laboratories would, for a fee, test instruments submitted by vendors or reactor licensees.

The testing laboratories would be certified by NVLAP under DOC. Costs associated with NVLAP certification and instrument testing fees would be passed on to industry in the form of higher instrument prices.

(4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria

Air purifying respirators are not currently acceptable for radioiodine protection due to the lack of accepted test procedures for certifying cartridge filtering efficiency. The result is that bulky self-contained breathing apparatus (SCBA) must be worn by workers in radioiodine environments. Such environments are expected during and after core-melt accidents. The results of wearing SCBA is to substantially reduce worker efficiency due to physical stress and the relatively short working time limited by air tank capacity. Use of air purifying respirators would reduce worker stress and improve worker efficiency.

It is expected that operator dose would be unaffected by the availability of respirators. Immediately after an accident, SCBA would still be used due to immediate hazards. During long-term recovery activities respirators could be used. However, reduced external dose due to efficient use of time

in radiation zones is expected to be offset by the reduced effectiveness of the respirators, compared to SCBA, in avoiding internal exposures. Criteria and test procedures for radioiodine cartridges have been under development by LASL using NRC funds. The technology has been developed and is in the process of being transferred to NIOSH. When transfer is complete, it is anticipated that NIOSH will amend 30 CFR 11 to incorporate the testing methods and criteria into respirator test and certification schedules. Respirator and cartridge manufacturers would submit products for certification testing and periodic quality control checks would be performed.

Following establishment of certification programs, NRC evaluation is anticipated regarding the need to specify the quantity and types of respirators necessary for normal and emergency use at a typical power reactor.

This issue will have no impact on public risk associated with core-melt accidents. The occupational dose impact is also considered to be zero, the benefit to workers being reduced stress, improved comfort and, consequently, better worker performance.

CONCLUSION

The above issues and their proposed resolutions do not impact public risk nor are they expected to increase or decrease occupational dose. They relate to the rights of workers to be assured of adequate radiation protection and would reduce stress during the performance of work in radiation zones. Therefore, this item is considered to be a Licensing Issue. The disposition of the four parts of this item is listed below.

ITEM III.D.3.2(1): AMEND 10 CFR 20

This Licensing Issue was evaluated in Item III.D.3.2 and was later resolved in February 1987 with the publication of a final rule on the requirement for the use of NBS-accredited personnel dosimetry processors.[1046]

ITEM III.D.3.2(2): ISSUE A REGULATORY GUIDE

This Licensing Issue was evaluated in Item III.D.3.2 above and was determined to be resolved.

ITEM III.D.3.2(3): DEVELOP STANDARD PERFORMANCE CRITERIA

The NRC/DOE project has produced several procedure manuals for future performance testing of radiation survey instruments and airborne radioactivity monitoring systems, after a certification program is established. These manuals are based on laboratory tests of sample instruments and monitoring systems using a draft of ANSI 42.17, "Performance Specifications for Health Physics Instrumentation." The IEEE Standard development working group is now using the results of the NRC/DOE project to finalize the standard for use in the accreditation program.

No further NRC action will be taken unless the instrument manufacturing industry fails to establish a satisfactory certification program within a reasonable period of time following final publication of ANSI 42.17. The final draft of this standard is under review by ANSI participants; some manufacturers' concerns still need to be resolved.

The NRC staff has taken the position that the industry should establish its own certification program and that the NRC would intervene only if the industry failed to do so, or if its program proved to be unsatisfactory. Thus, this Licensing Issue has been resolved.[954]

Item III.D.3.2(4): Develop Method For Testing and Certifying Air-Purifying Respirators

A research project has been completed that provides experimental data and recommendations for establishing a standard test procedure and acceptance criteria for air purifying respirator cartridges and canisters used to protect workers, and simultaneously measure penetrations of radioiodine and normal iodine vapor species through beds of various charcoals. The effects of various conditions of use (bed depth, contact time, concentration, relative humidity, temperature, flowrate, and flow cycling) were studied to identify testing requirements. Recommendations for testing and approval were based on consideration of the effects of these parameters. An apparatus designed and built for testing has been delivered to NIOSH, the responsible institute for testing and certifying respiratory protection equipment. Such certification is required in 10 CFR Part 20. In 1983, the staff published NUREG/CR-3403.[969]

NIOSH certification is now available. Licensees who wish to take credit for such equipment may do so after obtaining individual authorization from NRC. Thus, this Licensing Issue has been resolved.[954]

ITEM III.D.3.3: IN-PLANT RADIATION MONITORING

The four parts of this item are listed separately below.

ITEM III.D.3.3(1): ISSUE LETTER REQUIRING IMPROVED RADIATION SAMPLING INSTRUMENTATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-69 was established by DL for implementation purposes.

ITEM III.D.3.3(2): SET CRITERIA REQUIRING LICENSEES TO EVALUATE NEED FOR ADDITIONAL SURVEY EQUIPMENT

DESCRIPTION

This NUREG-0660[48] item required NRR to set criteria requiring licensees to evaluate in their plants the need for additional survey equipment and radiation monitors in vital areas and requiring, as necessary, installation of area monitors with remote readout. NRR was to evaluate the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. Operating reactors were to be reviewed for conformance with SRP[11] Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation." NRR was to revise SRP Sections 12.5 and 12.3.4 to incorporate additional monitor requirement criteria.

CONCLUSION

In December 1980, the requirements for high range area and portable monitors were incorporated into Regulatory Guide 1.97, Revision 2. In July 1981, SRP[11] Sections 12.3 and 12.5 were revised to incorporate the requirements for in-plant radiation monitoring. Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.3(3): ISSUE A RULE CHANGE PROVIDING ACCEPTABLE METHODS FOR CALIBRATION OF RADIATION-MONITORING INSTRUMENTS

DESCRIPTION

This NUREG-0660[48] item required RES to issue a rule change providing acceptable methods for calibration of radiation-monitoring instruments.

CONCLUSION

The required change was covered in the overall revision to 10 CFR 20, Paragraph 20.501(c). Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.3(4): ISSUE A REGULATORY GUIDE

DESCRIPTION

This NUREG-0660[48] item required RES to issue a Regulatory Guide providing acceptable methods for calibration of air-sampling instruments.

CONCLUSION

Regulatory Guide 8.25 was issued in August 1980. Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.4: CONTROL ROOM HABITABILITY

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-70 was established by DL for implementation purposes.

ITEM III.D.3.5: RADIATION WORKER EXPOSURE

The three parts of this item have been combined and evaluated together.

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to continue its efforts to improve and expand the data base on industry employees in order to facilitate possible future epidemiological studies on worker health. The three parts of this item are as follows:

(1)	"Improve and expand the data base on industry employees." This item is considered important in improving a data base used by the NRC in judging the adequacy of its radiation protection standards. Meetings have been held with DOE, ORM, NCI, AIF, and officials of Canadian and British national dose registries and health statistics organizations to discuss issues related to this item. Although these meetings have resolved certain generic issues, this item is a long-term goal requiring on-going cooperation between nuclear regulators, industries, and workers.[409]
(2)	"Investigate non-legislative means of obtaining employee health data." This item was completed in September 1982 following discussions about worker health data with DOE, AIF, EPRI, and officials of British and Canadian national dose registries and health statistics organizations.[409]
(3)	"Include as part of the overall rewrite of 10 CFR Part 20 consideration of a requirement for licensees to collect worker medical data." This item was completed in February 1981 following a decision by the Part 20 task force not to require the collection of worker medical data.[409]

The value of this item does not lie in the reduction of public or occupational risk. Instead, it will provide data on which future regulatory decisions will be based. Therefore, this item is not directly related to public safety and is considered a licensing issue.

CONCLUSION

The disposition of the three parts of this Licensing Issue is listed below.

ITEM III.D.3.5(1): DEVELOP FORMAT FOR DATA TO BE COLLECTED BY UTILITIES REGARDING TOTAL RADIATION EXPOSURE TO WORKERS

10 CFR 20.408 requires utilities that operate nuclear power plants to submit to the NRC a report that provides identification and exposure information for each monitored individual at the time of completion of the individual's assignment or employment at a particular plant. In order to improve the processing of this worker dose data, the NRC staff developed NRC Form 439, "Report of Terminating Individual's Occupational Exposure." This new form improved and expanded the dose data base that would be needed to support possible future epidemiological studies. The NRC staff, in cooperation with HHS, plans to recommend that the Committee for Interagency Radiation Research Policy Coordination (CIRRPC) review the issue of a worker registry and epidemiologic studies and formulate recommendations. The staff concluded[954] that the NRC does not have the authority or the resources to support a worker registry or epidemiological health effects studies. Thus, this Licensing Issue has been resolved.

ITEM III.D.3.5(2): INVESTIGATIVE METHODS OF OBTAINING EMPLOYEE HEALTH DATA BY NON-LEGISLATIVE MEANS

This Licensing Issue was evaluated in Item III.D.3.5 above and was determined to be resolved.

ITEM III.D.3.5(3): REVISE 10 CFR 20

This Licensing Issue was evaluated in Item III.D.3.5 above and was determined to be resolved.

REFERENCES

0011.	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.
0048.	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.
0064.	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.
0098.	NUREG#0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
0204.	NUREG#0855, "Health Physics Appraisal Program," U.S. Nuclear Regulatory Commission, March 1982.
0205.	NUREG#0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees," U.S. Nuclear Regulatory Commission, March 1981.
0206.	Memorandum for L. Rubenstein from M. Ernst, "Proposed Position Regarding Containment Purge/Vent Systems," April 17, 1981. [8105260251]
0207.	IE Bulletin 81-03, "Flow Blockage of Cooling Water to Safety System Components by **CORBICULA** SP. (Asiatic Clam) and MYTILUS SP. (Mussel)," U.S. Nuclear Regulatory Commission, April 10, 1981. [ML031210703]
0912.	Memorandum to T. Murley et al. from H. Denton, "Evaluation of Industry Success in Achieving ALARA-Integrated Radiation Protection Plans—Data Trend Assessments," May 19, 1986.
0913.	Memorandum for V. Stello from H. Denton, "Resolution of Generic Issue III.D.3.1, 'Radiation Protection Plans,'" May 19, 1986.
0954.	Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Items," November 13, 1986.

0969.	NUREG/CR-3403, "Criteria and Test Method for Certifying Air-Purifying Respirator Cartridges and Canisters Against Radioiodine," U.S. Nuclear Regulatory Commission, November 1983.
1046.	Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Item," February 27, 1987. [9704150146]

Task IV.A: Strengthen Enforcement Process ( )

The objective of this task is to substantially improve licensee awareness of and attitude toward safety by vigorous enforcement of NRC rules. The two major aspects of this objective are as follows: (1) assess substantial penalties for licensee failure to report safety-related information or for violations of rules defining safety practices or conditions; (2) adopt criteria for revocation of licenses, sanctions short of revocation, such as probation, and safety violation that would require immediate plant shutdown or other operational safeguards.

ITEM IV.A.1: SEEK LEGISLATIVE AUTHORITY DESCRIPTION

At the time NUREG-0660[48] was being prepared, the NRC had requested Congressional approval to increase the civil penalty limit to $100,000 for each licensee violation of the rules defining safety practices, with no upper limit on the number of violations. Obtaining approval of authority to increase the civil penalty was included in the TMI Action Plan[48] along with a request for staff considera-tion of the desirability of seeking further legislative modifications to: (1) permit civil penalties for a category of actions relating to safety; (2) pro-vide order authority against non-licensees and authority for enforcement sanction (including assessment of civil penalties) against an individual not employed by a licensee; and (3) extend criminal penalties to willful violation of a license condition.In 1980, approval to increase the civil penalty limit was granted by Congress in Public Law 96-295 and is being implemented by the NRC. In that same year, the office of the General Counsel submitted possible legislative proposals to the Commission in SECY-80-366.[237] That paper included the first two proposals described above. SECY-80-366[237] was withdrawn after Commissioner Offices advised the Secretariat that these legislative proposals were "overtaken by events and no longer served a useful purpose."[442] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM IV.A 2: REVISE ENFORCEMENT POLICY DESCRIPTION

At the time NUREG-0660[48] was being prepared, the NRC was in the process of revising its enforcement policy and guidance for the imposition of civil penalties, orders, and other sanctions and consideration was being given to the use of probation as an enforcement action. As a result, the revision to the NRC enforcement policy was included in the TMI Action Plan.48Methods of informing the public, such as forums near plant sites, were to be included in the revised NRC policy. The public and licensees were to be informed of the new policy through information releases and regional meetings. A revised General Statement of Policy was published in the Federal Register[234] in March 1982 and all work required by this item has been completed.[235, 379, 407] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.
0234.	*Federal Register* Notice 47 FR 9987, "10 CFR Part 2, General Statement of Policy and Procedure for Enforcement Actions," March 9, 1982.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0237.	SECY-80-366, "NRC Legislative Program for 97th Congress," U.S. Nuclear Regulatory Commission, August 6, 1980. [8101050634]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]

0407.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Item," May 11, 1982. [8401170108]
0442.	Memorandum for R. Emrit from T. Rothschild, "Establishing Priorities for Generic Safety Issues," April 21, 1983. [8312290167]

Task IV.B: Issuance of Instructions and Information to Licensees ( )

The objective of this task is to develop a more efficient and effective managment method for issuing information and requirements to licensees to elimination the duplication of staff effort for NRC and licensees. Provide an NRC-wide system for tracking safety issues.

ITEM IV.B.1: REVISE PRACTICES FOR ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES

DESCRIPTION

In addition to providing information to licensees in circulars, notices, and letters, the NRC requests actions from licensees lin various forms such as generic letters and bulletins. NSSS vendors also issue instructions that are periodically referenced in NRC bulletins. However, at the time NUREG-0660[48] was being prepared, it was believed that coordination between NRC offices was not always effective and inefficiency or duplication resulted. Necessary information was not being received promptly by cognizant supervisors and this adversely affected licensee actions and the understanding of safety issues. As a result, the technical resources of both the NRC and licensees were being diluted.

In order to address this problem, the TMI Action Plan[48] called for the establishment of an NRC staff-level task force to review overall NRC practices concerning issuance of information to licensees, requests for information from licensees, and issuance of various requirements for licensees (including staff issuance of Technical Specifications without request by licensees). The objectives of this review were: (1) to identify for further study other practices which detract from the application of resources that should be applied to improvement of safety; and (2) to evaluate related matters such as systems to track resolution of safety issues.

After the task force was established and recommendations made, a significant reorganization that addressed the issue was approved by Chairman Palladino for implementation in November 1981. This reorganization established the new posi-tion of Deputy Executive Director for Regional Operations and Generic Requirements (DEDROGR) to support the EDO's management, control, and tracking of requirements placed on licensees.

In accordance with the NRC Manual Chapter NRC-0103, issued on September 2, 1982, the DEDROGR is responsible for "...establishing and directing the implementation of procedures for controlling and tracking generic communications with, and generic requirements placed on, licensees; serving as chairman of a committee to review generic requirements; overseeing and implementing procedures to reduce and control the backlog of licensing actions;...."

As a result of the above accomplishments, all action required by this item has been completed.[248,379,406] This item is not directly related to safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.
0248.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Completed Items," December 28, 1981. [8205260197]
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0406.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Status Report," March 4, 1982. [8204290601]

Task IV.C: Extend Lessons Learned to Licensed Activities Other than Power Reactors ( )

The objective of this task is to assure that the lessons learned from TMI are applied to other NRC programs.

ITEM IV.C.1: EXTEND LESSONS LEARNED FROM TMI TO OTHER NRC PROGRAMS

DESCRIPTION

This TMI Action Plan Item[48] required that lessons learned from TMI be extended to other key NRC programs where a potential exists for nuclear accidents including, but not restricted to, the transportation of nuclear materials, waste management, research reactors, fuel facilities, and Category I materials licensees. An NRC study was to be performed to identify the lessons learned from TMI and the resulting agency actions to determine if NRC policies and practices related to key programs, other than light-water power reactor safety, should be revised and upgraded.

Studies performed by NMSS resulted in the issuance of a Draft BTP on Waste Form on October 30, 1981 which incorporated the resin degradation experience gained from the EPICOR-II system design used at TMI-2. This BTP[249] was issued in February 1983. As a result, all necessary work on this item has been completed.[410]

CONCLUSION

This item has been RESOLVED.

REFERENCES

0048.	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.
0249.	NUREG/CR#6210, "Computer Codes for Evaluation of Control Room Habitability (HABIT)," U.S. Nuclear Regulatory Commission, June 1996.
0410.	Memorandum for T. Speis from R. Browning, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," April 1, 1983. [8304200629, 9705190233]

Task IV.D: NRC Staff Training ( )

The objectives of this task are: (1) to improve and expand the NRC training program for the technical staff and resident inspectors, including, where appropriate, hands-on training; and (2) to establish a program to provide technically qualified entry-level professionals to counter recruiting difficulties resulting from increased industry demands and reduced university output.

ITEM IV.D.1: NRC STAFF TRAINING

DESCRIPTION

In order to fulfill the TMI Action Plan[48] requirements of improving and expanding the NRC training program for the technical staff and resident inspectors, OIE had to conduct a needs analysis of technical training requirements and then reexamine its training program in reference to this analysis. Inspector training and reactor technology training were to be modified accordingly. Consideration was to be given to: (1) a determination of the skills required to perform professional duties; (2) a comparison of the skills of newly-hired and incumbents with job skill requirements and an identification of deficient areas which can be improved through change in OIE training; and (3) the development or modification of courses to meet identified requirements. As a result of this analysis, simulator training was increased. The control of this training program and the periodic reexamination of the curriculum offered are part of the routine opera-tion of OIE. These responsibilities are outlined in an OIE memorandum[235] to NRR in June 1982.

At the same time that NUREG-0660[48] was being prepared, the following actions related to NRC staff training were underway and were included in the scope of Item IV.D.1: (1) Simulator training was being increased; (2) relevant graduate-level education in the areas of Safety, Safety Management, Systems Management, and Engineering Systems Analysis and Management had been identified and were being funded as Master's degree programs; (3) alternatives were being developed to obtain qualified technical employees and inspectors in a climate of heavy competition for nuclear engineers and nuclear-trained individuals created by post-TMI industry requirements and shrinking university output.

As stated above, simulator training for NRC personnel was increased by OIE. However, ADM was responsible for the identification of relevant graduate-level education for NRC personnel and the funding of such a program. At the end of 1981, it was reported that multidisciplinary training had been provided to approximately 120 professional members of the NRC staff. This program is currently being implemented under the direction of ADM. In addition to this, a plan for obtaining qualified technical employees in a climate of heavy competition and limited university output was presented by ADM in SECY-80-331.[280] This plan is based on the "grow-our-own" concept and is estimated to cost $3.7M for training 100 college graduates per year (200 graduates in the program at any one time) to fill future positions in the NRC.

All necessary staff work to implement the "Grow-Our-Own" program has been completed and there are no outstanding issues to be resolved. The proposed program is periodically reviewed.[301] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.
0235.	Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]
0280.	SECY-80-331, "NRC Training Program," U.S. Nuclear Regulatory Commission, July 14, 1980. [8009100166]
0301.	Memorandum for R. Emrit from P. Goldman, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," December 29, 1982. [8312290171]

Task IV.E: Safety Decision-Making (Rev. 3) ( )

The objective of this task is to develop plans for an integrated program of safety decision-making. These plans include: (1) an expanded program of regulatory research covering methodologies for making safety decisions and safety-cost tradeoffs, with application both to decisions regarding the overall risk of nuclear power plants and the nuclear fuel cycle and to specific licensing and inspection decisions; (2) early resolution of safety issues after they are identified, including application of the decisions to operating reactors, reactors under construction, and standard designs; (3) elimination of repetitive consideration of identical issues at several stages of the licensing process; (4) expanded use of rulemaking to implement safety criteria developed as a result of the various Task Action Plans; and (5) improved and expanded systematic assessments of operating reactors.

ITEM IV.E.1: EXPAND RESEARCH ON QUANTIFICATION OF SAFETY DECISION-MAKING

DESCRIPTION

This issue is described in NUREG-0660[48] as follows:

"The purpose of this task is to proceed toward better quantification of safety objectives, including safety-cost tradeoffs. The concept will use ongoing research that one might quantify risk and possible application of formal decision-making techniques to the regulatory environment. Future programs will build on the risk assessment and systems reliability work currently underway and incorporate a better assessment of common-mode and human failures. Safety objectives will be developed for components and systems, and eventually these might be amalgamated into a more tightly bounded, quantitative safety standard, as opposed to a safety objective having fairly large inherent uncertainties."

The approach to the resolution of this item is also outlined in NUREG-0660[48] as follows:

(1)	RES will assemble a research task force from a wide variety of professional disciplines. The task force will formulate several possible sets of numerical criteria using different technical approaches. The formation of the research task force and the conduct of its meetings are being coordinated through IEEE with cooperation from other professional societies.
(2)	BNL has been contracted to independently formulate criteria to investigate the implications of safety criteria and to determine the impact of attempting to satisfy such criteria.
(3)	Decision theory and survey methods for obtaining criteria are being investigated as extensions of previous projects on risk analysis. These methods can provide a separate approach to obtain acceptable risk criteria.
(4)	Negotiations are underway with various governmental and private agencies for information on proposed criteria. In addition, letters have been sent to several hundred individuals announcing the project and requesting their contributions.
(5)	To assure that the criteria receive rigorous peer review, negotiations are underway with the National Science Foundation, the National Academy of Sciences, and the American Statistical Association.

The current accomplishments include completion of NUREG/CR-1614,[275] NUREG/CR-1539,[276] NUREG/CR-1930,[277] NUREG/CR-1916,[278] and NUREG/CR-2040.[279] The current status is such that PNL, ORNL, BNL, ANL, IEEE, NRC (Office of Policy Evaluation), and the ACRS are completing various elements of the overall program. These activities will develop and exhibit approaches with which to better factor risk evaluation into NRC decision-making regarding reactor plant safety. This issue does not appear to have a direct effect on public risk reduction or to have any industry cost directly associated with its resolution. Therefore, it is a licensing issue.

CONCLUSION

A value/impact handbook (NUREG/CR-3568)[970] was developed by the staff to support specific cost/benefit analyses of proposed rules. In November 1986, RES determined that all other staff work required by this issue was being pursued in the ongoing work related to the Commission's Safety Goal.[954] Thus, this Licensing Issue has been resolved.

ITEM IV.E.2: PLAN FOR EARLY RESOLUTION OF SAFETY ISSUES

DESCRIPTION

This TMI Action Plan[48] item required NRR, in consultation with other appropriate offices, to develop a plan for the early identification, assessment, and resolution of safety issues. This item is related to the establishment and implementation of an NRC program to identify and resolve safety issues and, therefore, is considered a licensing issue.

CONCLUSION

The plan was presented in SECY-81-513[1] on August 25, 1981 and is currently being implemented by SPEB. Thus, this Licensing Issue has been resolved.

ITEM IV.E.3: PLAN FOR RESOLVING ISSUES AT THE CP STAGE

DESCRIPTION

According to NUREG-0660,[48] NRR and ELD transmitted a consent calendar item to the Commission on February 14, 1980, entitled "Response to Staff Requirements Memorandum (Affirmation Session 79-40) With Respect to Post-CP Design and Other Changes," SECY-80-90. This paper discussed five options regarding the establishment of construction requirements. The recommendation of this consent paper is to publish an advance notice of public rulemaking to obtain comments on these options. After receipt of public comment on the above, the staff will prepare a plan to implement methods to resolve as many issues as possible at the construction permit stage before major financial commitments in construction occur.

An advanced notice of rulemaking was published in the Federal Register in December 1980 with a public comment period ending on February 9, 1981. On August 18, 1981, the Director of the Division of Risk Analysis sent a memo to distribution proposing an approach to the Rule and requested examples of the types of characteristic alterations representing post-CP changes. The draft of the Rule is currently being reviewed.

In view of the intent of this item, it is concluded that its resolution does not have a direct effect on public risk reduction and is, therefore, considered to be a licensing issue.

CONCLUSION

Staff stated in the Supplement to this report published in 1986 that the resolution of this Licensing Issue was available. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM IV.E.4: RESOLVE GENERIC ISSUES BY RULEMAKING

DESCRIPTION

This TMI Action Plan[48] item states that the NRC will undertake the additional task of developing a program for reviewing new criteria before their promulgation to determine whether rulemaking would be the desirable means of implementation. The intent will be to implement new NRC criteria by rule, wherever feasible and timely, instead of by license changes, orders, or changes in regulatory guides.

This item does not have a direct effect on public risk reduction nor is there any industry cost associated with the completion or implementation of the issue resolution. Thus, it is considered a licensing issue.

CONCLUSION

In November 1986, RES concluded that ongoing NRC activities such as the Safety Goal Program, RES independent review of rulemaking, and the Commission policy on backfitting had effectively addressed the concerns of this issue.[954] Thus, this Licensing Issue has been resolved.

ITEM IV.E.5: ASSESS CURRENTLY OPERATING REACTORS

DESCRIPTION

Historical Background

As part of developing plans for an integrated program of safety decision making, NRR, in consultation with other appropriate offices, will develop a plan for approval by the Commission for the systematic assessment of the safety of all operating reactors. Development of such a plan will take into account the SEP, the ACRS comments on the program, the IREP plan, and ongoing TMI lessons-learned activities. This value/impact assessment of Item IV.E.5 deals with the work under the SEP. Value/impact assessments of IREP and NREP are presented in Items II.C.1 and II.C.2, respectively.

SEP is now reviewing the 10 oldest plants against current licensing review safety criteria, including the SRP, to provide the basis for integrated and balanced backfit decisions. This review is nearly complete and, therefore, is not part of this assessment. The next SEP phase involves evaluation of 11 additional plants. In this next phase, PRA evaluations will be coordinated with the deterministic review method (review against current licensing safety criteria). The PRA will be done as part of NREP (TMI Action Plan Item II.C.2).

Possible Solutions

As safety-related problems are identified for each plant, resolutions are developed using procedural and administrative changes, possible credit for non-safety systems where justified, and hardware backfits as necessary to reduce risk levels. The process used to decide appropriate corrective actions employs the judgment of a team of NRC staff familiar with each plant.

PRIORITY DETERMINATION

This priority determination uses potential risk reduction analyses and cost estimate information provided by PNL.[64]

Frequency/Consequence Estimate

This public risk reduction analysis for SEP considers only the 11 additional plants currently proposed to be reviewed in the first group of Phase III plants, since much of the review of the first 10 plants in Phase II has been performed. The 11 plants consist of 7 PWRs and 4 BWRs with estimated average remaining lives of 24 and 22 years, respectively. In Item II.C.2 (NREP), it is estimated that an overall core-melt frequency reduction of 2 x 10^{-4}/RY could be achieved for one-third of the plants to be reviewed under NREP. Although the NREP evaluation of these plants will identify some areas of potentially high risk, the NREP methods do not address areas such as external events and structural design which are included in the SEP deterministic review. For this issue, it is assumed that the risk reduction estimated for NREP could be achieved by the SEP considering only the benefit resulting from using the deterministic review method for external events and other issues outside the scope of PRAs (e.g., adequacy of design, structural issues, and design errors).

Using the base case core-melt frequency and the base case public risk for each type plant, and assuming a population of 340 persons per square mile over an area having a 50 mile radius, the average risk per core-melt is 2.5 x 10^6 man-rem for PWRs and 6.8 x 10^6 man-rem for BWRs.

Using the average risk value and the assumption stated above that the deterministic review method can achieve the core-melt frequency reduction estimated for NREP for one-third of the plants reviewed, we can estimate the potential reduction for the SEP Phase III as follows:

PWRs: Risk Reduction = (2.5 x 10^6 man-rem/core-melt)(2 x 10^{-4} core-melt/RY)

= 500 man-rem/RY

BWRs: Risk Reduction = (6.8 x 10^6 man-rem/core-melt)(2 x 10^{-4} core-melt/RY)

= 1,360 man-rem/RY

Summed over the average remaining plant life for the 11 plants proposed, the total public risk reduction is calculated to be approximately 80,000 man-rem.

Cost Estimate

Industry Cost: Based on SEP studies completed to date, the following costs per plant are estimated: up to $2M for engineering studies to identify areas of plant modification and $2M to $20M to design and install modifications.

For purposes of this analysis, assume a conservative implementation cost per plant of $2M for engineering studies at each of the 11 plants plus $10M average design and installation (including capital equipment cost) at one-third of the plants. For 11 plants, the total industry cost is $[(11)(2) + (1/3)(11)(10)]M or $55M.

NRC Cost: Based on past studies, NRC staff effort has totaled 10 man-yr/plant plus $700,000 contract technical support per plant. Thus, total development and implementation cost, at $100,000/man-year, is:

(10 man-years/plant)($100,000/man-yr) + ($700,000/plant)(11 plants) = $19M.

Assuming NRC staff effort for review and inspection of plant modifications at one-third of the plants is 0.5 man-wk/RY and the average remaining life of these plants is 23 years, then the total plant review cost is:

(0.5 man-wk/RY)($2,000/man-wk)[(1/3)(11)(23)RY] = $0.1M.

Value/Impact Assessment

Based on a public risk reduction of 80,000 man-rem, the value/impact score is given by:

$$S = \frac{80{,}000\text{man} - \text{rem}}{\$(55+19)\text{M}}$$

$$= 1{,}000\text{man} - \text{rem}/\$\text{M}$$

Other Considerations

If the cleanup of an accident is assumed to require 19,900 man-rem and the same assumption on accident frequency reduction is retained, the total reduction in occupational exposure would be 170 man-rem. An estimate of the occupational exposure to implement any changes cannot be made without identifying the specific changes. However, there would likely be some increase in occupational exposure, but it would be small compared to the public risk reduction.

An additional consideration is that plant damage is estimated to be $1,650M per plant for core-melt. Thus, total averted plant damage for one-third of the plants with a reduced core-melt frequency could be

($1,650M)(2 x 10^{-4}/RY)[(1/3)(11)(23)RY] = $28.9M

Uncertainties

Since the 11 plants considered are older plants, it is possible that the assumed 10^{-4}/RY risk reduction may be achieved for more than one-third of the 11 plants as assumed, thus resulting in greater risk reduction with an associated increase in implementation cost. However, the value/impact score would not change appreciably.

CONCLUSION

The value/impact score indicates a medium priority. However, the potentially large, though uncertain, risk reduction of nearly 80,000 man-rem justified a high priority ranking.

Work completed by the staff on this item was closely related to the accomplishments under Item II.C.2. Whereas Item II.C.2 called for the initiation of IREP studies (i.e. plant-specific PRAs) on all remaining operating reactors, Item IV.E.5 called for the development of a plan for the systematic assessment of the safety of all operating reactors. The Integrated Safety Assessment Program (ISAP), presented in SECY-84-133[814] and SECY-85-160,[815] provided for a comprehensive review of selected operating reactors to address all pertinent safety issues and to provide an integrated cost-effective implementation plan for making needed changes. Under ISAP, each plant would be subject to an integrated assessment of safety topics, a probabilistic safety assessment, and an evaluation of operating experience.

NRC guidance, as described in the Severe Accident Policy Statement (see Item II.B.8), states that OLs will be expected to perform plant-specific PRAs in order to discover instances of particular vulnerability to a core-melt or poor containment performance, given a core-melt. Thus, this item was RESOLVED and no new requirements were established.[816]

REFERENCES

0001.	SECY-81-513, "Plan for Early Resolution of Safety Issues," U.S. Nuclear Regulatory Commission, August 25, 1981. [8109140067]
0048.	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.
0064.	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.
0275.	NUREG/CR#1614, "Approaches to Acceptable Risk: A Critical Guide," U.S. Nuclear Regulatory Commission, September 1980.
0276.	NUREG/CR#1539, "A Methodology and a Preliminary Data Base for Examining the Health Risks of Electricity Generation from Uranium and Coal Fuels," U.S. Nuclear Regulatory Commission, August 1980.
0277.	NUREG/CR#1930, "Index of Risk Exposure and Risk Acceptance Criteria," U.S. Nuclear Regulatory Commission, February 1981.
0278.	NUREG/CR#1916, "A Risk Comparison," U.S. Nuclear Regulatory Commission, February 1981.
0279.	NUREG/CR#2040, "A Study of the Implications of Applying Quantitative Risk Criteria in the Licensing of Nuclear Power Plants in the U.S.," U.S. Nuclear Regulatory Commission, March 1981.
0814.	SECY-84-133, "Integrated Safety Assessment Program (ISAP)," U.S. Nuclear Regulatory Commission, March 23, 1984. [8404100072]
0815.	SECY-85-160, "Integrated Safety Assessment Program—Implementation Plan," U.S. Nuclear Regulatory Commission, May 6, 1985. [8505230571]
0816.	Memorandum for W. Dircks from H. Denton, "Close-out of Generic Issues II.C.2, 'Continuation of IREP' and IV.E.5, 'Assess Currently Operating Reactors,'" September 25, 1985. [9909290069]
0954.	Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Items," November 13, 1986.
0970.	NUREG/CR-3568, "A Handbook for Value-Impact Assessment," U.S. Nuclear Regulatory Commission, December 1983.

1858.	Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.
1967.	SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [*ML111590814*]

Task IV.F: Financial Disincentives to Safety ( )

The objective of this task is to enhance public safety through the reduction of disincentives to safety resulting from financial pressures on the utility at the construction, operation, and decommissioning stages.

ITEM IV.F 1: INCREASED OIE SCRUTINY OF THE POWER-ASCENSION TEST PROGRAM

DESCRIPTION

Historical Background

As part of the post-TMI actions,[48] the staff was to explore the possible disincentives to safety which could result from financial pressures on the utility during construction and transition to the operating stages.

Safety Significance

It is possible that, in order to avoid delay in commercial operation, some short-cuts may be made which could impact safety.

Possible Solutions

As part of the TMI Action Plan,[48] the NRC committed to increase inspections of the startup test program and power ascension test programs at plants that have been completed and are awaiting operating licenses. This included having NRC personnel witness all tests on all shifts.

Accordingly, OIE reported[239] that procedures have been issued to increase inspection coverage during power ascension testing. Reactor Inspection Program 2514/01, Revision 2, calls for NRC to witness portions of tests on all shifts and these inspection requirements have been incorporated into the OIE Manual.[247] All work required by this item has been Completed.[239,379,406]

CONCLUSION

This item has been RESOLVED with changes in the NRC procedures that address the scrutiny of power-ascension test programs.

ITEM IV.F.2: EVALUATE THE IMPACTS OF FINANCIAL DISINCENTIVES TO THE SAFETY OF NUCLEAR POWER PLANTS

DESCRIPTION

Historical Background

The purpose of this TMI Action Plan[48] item is to study the recommendations of the NRC/TMI Special Inquiry Group and focus on questions such as:

(1)	Does the financial status of a utility impact safety or indicate when impacts of a safety nature may occur?
(2)	Would continuing evaluation of a licensee's financial condition be a useful method to alert IE to times when the licensee might be tempted to cut corners or are there more pragmatic actions that accomplish this objective?
(3)	Will improved communications with economic regulatory agencies, such as NARUC, PUCs, IRS, and FERC sufficiently increase their understand-ing of a sensitivity to safety matters and financial disincentives?
(4)	Do the requirements of the various financial regulatory agencies result in reducing nuclear safety and, if so,

	how could improvements in financial regulation best be achieved?

Safety Significance

In SECY-81-168B,[232] the staff summarized the results of their discussions with various economic regulatory agencies and outlined other possible areas of investigation. Generally, the staff acknowledges that some financial disincentives exist but considers their impact on safety to be small, particularly when considered relative to other regulatory provisions which assure safety, such as plant Technical Specifications that must be complied with to maintain an operating license. Further, any financial benefits that might be associated with taking safety risks are considered small compared to the potential financial impact of plant investment cost and accident cleanup costs associated with safety risks.

Other financial issues which relate to safety have been resolved by separate rulemaking. Specifically, a rule has been published, 10 CFR 50.54(W),[197] which requires licensees to maintain, as a minimum, specified amounts of commercially available onsite property damage insurance.

CONCLUSION

This item has been RESOLVED.

REFERENCES

0048.	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.
0197.	*Federal Register* Notice 45 FR 37011, "Decommissioning of Nuclear Facilities Regulation (10 CFR Parts 30, 40, 50, and 70)," May 30, 1980.
0232.	SECY-81-168B, "Response to Commission Request for Information on Financial Considerations in Licensing Proceedings," U.S. Nuclear Regulatory Commission, July 13, 1981. [8107310227]
0239.	Memorandum for W. Dircks from V. Stello, "TMI Action Plan—Status Report," December 19, 1980. [8205260193]
0247.	NUREG/CR#5669, "Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators," U.S. Nuclear Regulatory Commission, July 1991.
0379.	Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]
0406.	Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan—Status Report," March 4, 1982. [8204290601]

Task IV.G: Improve Safety Rulemaking Procedures ( )

The objectives of this task are to improve NRC rulemaking procedures to provide a greater opportunity for public participation, to assure a periodic and systematic reevaluation of NRC rules, and to include appropriate provision for backfitting in all new regulations. (Item V.12 discusses related action assessing the delegation of rulemaking authority to members of the staff.)

ITEM IV.G.1: DEVELOP A PUBLIC AGENDA FOR RULEMAKING

DESCRIPTION

At the time NUREG-0660[48] was prepared, the NRC was issuing quarterly status reports on petitions for rulemaking and proposed rules and a status summary report listing those regulations under development by RES. Also published were advance notices of proposed rulemaking on major NRC actions.

The TMI Action Plan[48] called for the publication of a semiannual agenda for significant rulemaking actions as required by Executive Order 12044. The criteria for determining significant regulations were to be developed by RES and the publication of the agenda was to be accomplished by ADM, after consultation with other program offices.

The first semiannual regulatory agenda was published in the Federal Register[261] in October 1981 and subsequent publications were made in April and October 1982.[293,294] Future publications of the agenda will be made in the Federal Register in April and October every year. This agenda describes the need and legal basis for each regulation and indicates the status of each regulation on the agenda (or previous editions of the agenda) until the issuance of final rules. This item is not directly related to safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM IV.G.2: PERIODIC AND SYSTEMATIC REEVALUATION OF EXISTING RULES

DESCRIPTION

NRC will comply with the intent of Executive Order 12044, which requires a periodic and systematic reevaluation of existing rules and that regulations be written in plain English. NRC will first review its rules for content, quality,

and clarity and, at a later date, will review the regulations as a body for proper structure. The initial review will concentrate on areas where rules are broadly affected by the accident at Three Mile Island. This initial review should be completed within five years and should be repeated every five years thereafter.

Since the thrust of this issue is directed at assuring that regulations are clearly stated and easily understood, there is no direct public or occupational risk reduction which could be attributed to this item.

Resolution of this issue should result in less confusion and conflict in the preparation and review of license applications which, in turn, would result in cost and time savings to both the industry and the NRC staff.

CONCLUSION

The NRC rules most directly associated with TMI accident considerations include operator training, emergency planning, environmental monitoring, radiation protection, and consistent treatment of fission product release from fuel cladding failure. The rules in these areas have been reviewed and actions taken as needed. Additionally, systematic reviews of other regulations have been carried out or are underway under several programs throughout the NRC.

A RES program was initiated in 1984 to review selected existing regulatory requirements in terms of risk effectiveness. A contractor report (NUREG/CR-4330,[971] Volumes 1 and 2) was published in 1986: Volume 1 summarized the results of a survey to identify regulatory requirements that may have marginal importance to safety; Volume 2 provided the results of a detailed evaluation in terms of risk, dose, and cost for assumed changes in requirements for three regulatory areas. The NRC staff is currently evaluating these reports and rule changes will be made where appropriate. Four additional regulatory areas are currently under evaluation by the contractor.

A program to systematically review Technical Specifications was completed by a NRC task force in 1985. Recommendations were made to improve the efficiency and effectiveness of Technical Specifications. These recommendations are being implemented by the Technical Specifications Coordination Branch, Division of Human Factors Technology, Office of Nuclear Reactor Regulation.

Another program was initiated in 1984 by the EDO following the UCLA relicensing hearings. The purpose of this program is to identify inconsistencies among safeguards (security) regulations, regulatory guides, inspection procedures, licensing criteria, and other guidance. These will be modified as needed.

Based on the ongoing NRC programs that have been established to systematically evaluate existing rules, RES determined that this Licensing Issue has been resolved.[954]

ITEM IV.G.3: IMPROVE RULEMAKING PROCEDURES

DESCRIPTION

NRC will reevaluate the rulemaking process to ensure that it is properly focused on resolving important safety issues and that the procedures are clear, understandable, efficient, and well-publicized. NRC will then consider a proposal to codify in NRC regulations and practice a procedure under which all new rules would include consideration of backfitting to existing plants.

The establishment of the Committee for the Review of Generic Requirements (CRGR) and the limited delegation of Commission rulemaking authority to the Office of the Executive Director for Operations[370] have implemented changes in the rulemaking procedure which are in direct response to this issue. As a part of the revised process, value/impact analyses[371] will be required for all proposed or final rules which would (a) likely have an effect on the economy of greater than $100M in direct and indirect costs, (b) likely result in a significant adverse effect on the public health, safety or environment, or (c) result in a substantial increase in cost or prices. Value/impact analyses will include backfitting considerations. In addition, the Regulatory Reform Task Force has recently recommended a change to the backfit rule (10 CFR 50.109) which would redefine the term "backfitting" as applied to those plants which have received a construction permit and would require an analysis to establish that backfitting a new or revised requirement would result in a substantial increase in the protection of public health and safety.[372] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM IV.G.4: STUDY ALTERNATIVES FOR IMPROVED RULEMAKING PROCESS

DESCRIPTION

NRC will study alternatives to the present rulemaking procedures for the purpose of improving the Commission's rulemaking efforts.

Several means to enhance the Commission's rulemaking efforts have been addressed, in part, in NUREG-0499,[373] including Supplement 1, and in the GC/OPE Memorandum to the Commission on "Review of Delegations of Authority Within NRC."[374] In addition, the Commission has delegated substantial rulemaking authority to the EDO[370] (See Item IV.G.3). A number of improvements to the rulemaking process have already been made as indicated by the completion of Item IV.G.3. However, there will always be a need to investigate and evaluate possible changes to the process as an ongoing activity. This ongoing need has been recognized and has been dealt with by chartering the Regulatory Analysis Branch of RES with the responsibility for the investigation and evaluation of proposed improvements to the rulemaking process.[375] This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

Inasmuch as initial improvements have been studied and implemented, and an institutional change has been made to provide for the continuing investigation and evaluation of improved rulemaking procedures, this Licensing Issue has been resolved.

REFERENCES

0048.	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.
0261.	*Federal Register* Notice 46 FR 53594, "NRC Regulatory Agenda," October 29, 1981.
0293.	*Federal Register* Notice 47 FR 18508, "NRC Regulatory Agenda," April 29, 1982.
0294.	*Federal Register* Notice 47 FR 48960, "NRC Regulatory Agenda," October 28, 1982.
0370.	SECY-81-676, "Delegation of Rulemaking Authority to the EDO," U.S. Nuclear Regulatory Commission, December 3, 1981. [8201110403]
0371.	SECY-82-187, "Revised Guidelines for Value-Impact Analyses," U.S. Nuclear Regulatory Commission, May 7, 1982. [8205130275]
0372.	SECY-82-447, "Draft Report of the Regulatory Reform Task Force," U.S. Nuclear Regulatory Commission, November 3, 1982. [8211160547]
0373.	NUREG#0499, "Preliminary Statement on General Policy for Rulemaking to Improve Nuclear Power Plant Licensing," U.S. Nuclear Regulatory Commission, December 1978.
0374.	Memorandum for J. Hendrie from L. Bickwit, "Review of Commission Delegation of Authority," October 4, 1979. [8001150518]
0375.	Memorandum for R. Minogue from R. Bernero, "Charter of the Regulatory Analysis Branch," October 9, 1981. [8110280720]
0954.	Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Items," November 13, 1986.
0971.	NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1986, (Vol. 2) June 1986.

Task IV.H: NRC Participation in the Radiation Policy Council ( )

The objective of this task is to respond to the President's request for NRC participation in the Radiation Policy Council.

ITEM IV.H.1: NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL

DESCRIPTION

The Radiation Policy Council, a policy coordinating body chaired by the EPA Administrator, has, at President Carter's December 1979 request, had an NRC representative on it. However, the Council has been inactive and has had no funding in FY 1982. In August 1982 it was abolished.[283]

Functions similar to those of the Council have recently been assumed by the newly created Committee for Interagency Radiation Policy Coordination, which reports to the Director, Office of Science and Technology Policy.[284] An NRC representative and alternate have been designated.[285,286] At this writing, the Committee has not yet met. This item is not directly related to public safety and, therefore, is considered a licensing issue.

CONCLUSION

NRC representation on the Committee for Interagency Radiation Policy Coordination is a required part of the programmatic management and interagency coordination for NRC's mission in the radiation safety area. By appointment of an NRC representative, this continuing function has been instituted. Thus, this Licensing Issue has been resolved.

REFERENCES

0283.	*Federal Register* Notice 47 FR 36099, "Executive Order 12379 of August 17, 1982, Termination of Boards, Committees, and Commissions," August 19, 1982.
0284.	Letter to N. Palladino (U.S. Nuclear Regulatory Commission) from G. Keyworth (Office of State and Tribal Programs), July 21, 1982. [9705190213]
0285.	Letter to G. Keyworth (Office of State and Tribal Programs) from N. Palladino (U.S. Nuclear Regulatory Commission), July 23, 1982. [9705190203]
0286.	Letter to T. Pestorius (Office of State and Tribal Programs) from R. Minogue (U.S. Nuclear Regulatory Commission), August 27, 1982. [9104170201]

Task V.A: Development of Safety Policy ( )

The objective of this task was the further delineation of substantive safety policy by the NRC.

ITEM V.A.1: DEVELOP NRC POLICY STATEMENT ON SAFETY

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to develop more explicit articulation of policy with respect to the fundamental issues of public health and safety. This development was to include some general approach to risk acceptability and safety/cost trade-offs and, to the extent that these reasonably lent themselves to articulation, quantitative safety goals, safety improvement goals, and standards for review of past actions in light of new rules and improved practices. This item was originally identified as Item 1 in Chapter V but was made Item V.A.1 in Rev. 1 to NUREG-0660.[48]

The Commission issued a "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" for public comment in February 1982.866 In March 1983, the policy statement867 on the "Safety Goal Development Program" was issued and a 2-year evaluation period began. Follow-up work completed since that time resulted in the publication of NUREG-0880,[69] Rev. 1, in May 1983. This report addressed the development and rationale of the Commission's policy statement, the public comments on the earlier draft[866] of the policy statement, and the NRC plan for evaluating the policy statement. The purpose of the evaluation was to develop recommendations on the future use of safety goals in regulation and licensing. In August 1986, the Commission issued the policy statement.[939]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue was resolved with the publication of the policy statement[939] in August 1986.

REFERENCES

0048.	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.
0069.	NUREG#0880, "Safety Goals for Nuclear Power Plants: A Discussion Paper," U.S. Nuclear Regulatory Commission, February 1982, (Rev. 1) May 1983.
0866.	*Federal Register* Notice 47 FR 7023, "Proposed Policy Statement on Safety Goals for Nuclear Power Plants," February 17, 1982.
0939.	*Federal Register* Notice 51 FR 28044, "Safety Goals for the Operations of Nuclear Power Plants," August 4, 1986.

Task V.B: Possible Elimination of Non-Safety Responsibilities ( )

The objective of this task was the elimination of nonsafety responsibilities from NRC jurisdiction, if appropriate.

ITEM V.B.1: STUDY AND RECOMMEND, AS APPROPRIATE, ELIMINATION OF NON-SAFETY RESPONSIBILITIES

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to review nonsafety and nonsafeguard regulatory review responsibilities including antitrust, NEPA, and export licensing. The Commission was to examine whether removal of these responsibil-ities would leave gaps in Federal regulation and whether they should be transferred to other agencies. This item was originally identified as Item 2 in Chapter V but was made Item V.B.1 in Rev. 1 to NUREG-0660.[48]

The Commission, in 1980, agreed that transfer of export licensing functions to the Executive Branch would constitute a prudent course.[48] However, the current Commissioners have never been asked to express views on this issue. No legislation transferring the export licensing function has been enacted. In 1980, the Commission decided not to seek transfer of other nonsafety responsibilities.[48] Again, the current Commissioners have never been asked to express views on this issue.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue was resolved and the Commission's decisions were published in NUREG-0660,[48] Rev. 1.

REFERENCES

0048.	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.

Task V.C: Advisory Committees ( )

The objective of this task was to strengthen the role of advisory committees in Commission activities.

ITEM V.C.1: STRENGTHEN THE ROLE OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to strengthen the role of the ACRS by seeking legislation which would eliminate its compulsory jurisdiction and by considering ACRS views on the President's Commission recommendations regarding its role. This item was originally identified as Item 3 in Chapter V but was made Item V.C.1 in Rev. 1 to NUREG-0660.[48]

In a letter[868] to the Commission in January 1980, the ACRS agreed with the recommendations that the role of ACRS be strengthened. In April 1981, 10 CFR Part 2 was revised to provide for ACRS participation in rulemakings involving safety issues.869

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue was resolved with the revision to 10 CFR Part 2.

ITEM V.C.2: STUDY NEED FOR ADDITIONAL ADVISORY COMMITTEES

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to determine whether NRC should establish additional advisory committees such as a citizens' advisory committee or a general advisory committee similar to that of the Atomic Energy Commission. This item was originally identified as Item 4 in Chapter V but was made Item V.C.2 in Rev. 1 to NUREG-0660.[48]

In July 1980, the Commission decided that no further advisory committees were needed.[871] At the Commission's request, the Offices of Public Affairs, Policy Evaluation, and General Counsel developed methods to ensure that a broader spectrum of representatives of the public and other outside organizations appear before the Commission on a periodic basis.[870] This plan was presented to the Commission in October 1980 and was subsequently approved.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.C.3: STUDY THE NEED TO ESTABLISH AN INDEPENDENT NUCLEAR SAFETY BOARD

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to study the need to establish a Nuclear Safety Board that would independently investigate nuclear accidents and important incidents and would monitor and evaluate the quality of NRC's regulatory process. This item was originally identified as Item 8 in Chapter V but was made Item V.C.3 in Rev. 1 to NUREG-0660.[48]

Commission consideration of an independent board to investigate nuclear accidents preceded the TMI-2 accident. In August 1978, the NRC Chairman responded[921] to a series of questions on a nuclear accident board posed by Congress.[922] After the TMI-2 accident, the President established a Nuclear Safety Oversight Committee by Executive Order 12202 in March 1980.[923] This committee was terminated on September 30, 1980. In July 1980, the Commission decided that an independent Nuclear Safety Board was not needed.[871] Subsequently, in 1984 Congress directed the NRC to conduct a study of the need for and feasibility of an independent organization responsible for conducting investigations of significant safety events at NRC-licensed facilities. In response, the NRC contracted with BNL; BNL submitted its final report to the NRC in February 1985.

BNL recommended the establishment of a statutory office of nuclear safety, headed by a Director reporting to the Commission. However, the study stated current practices for investigations of operating events have been conducted in a "proficient and technically competent" manner. While BNL suggested a number of improvements for event investigations, it was noted that, for the most part, these improvements could be implemented within the present organizational structure. Many of the improvements recommended by BNL have been adopted as part of the NRC Incident Investigation Program. Based on the Commission's review of the BNL report and other studies of the issue, the Commission believed that there were no major deficiencies in the NRC accident investigation program that would warrant formation of an independent Nuclear Safety Board. (Testimony by Chairman Zech to the Subcommittee on Energy and the Environment, Committee on Interior and Insular Affairs, U.S. House of Representatives, concerning Licensing Reform and Other Matters, July 22, 1986.)

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.
0868.	Letter to J. Ahearne from M. Plesset, "Recommendations of President"s Commission on ACRS Role," January 15, 1980. [8002150071]
0870.	Memorandum for Commissioner Ahearne et al. from L. Bickwit et al., "TMI Action Plan, Chapter V, Formal Procedures for Ensuring Periodic Public Interaction," October 2, 1980.
0871.	Memorandum for W. Dircks et al. from J. Hoyle, "Staff Requirements—Discussion of Action Plan, Chapter V (See SECY-80-230B), 2:00 p.m. Monday, July 7, 1980, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," July 9, 1980. [8012030928]
0921.	Letter to the Honorable Morris K. Udall from Joseph M. Hendrie, August 7, 1978. [7901030172, 8001230259]
0922.	Letter to Joseph Hendrie from Morris K. Udall, January 27, 1978. [8007210279, 8007180431]
0923.	Memorandum for J. Taylor from D. Morrison, "Resolution of Generic Safety Issue 15, 'Radiation Effects on Reactor Vessel Supports,'" May 29, 1996. [9606190081]

Task V.D: Licensing Process (Rev. 1) ( )

The objective of this task was to enhance public participation in, and make needed reforms to, the nuclear licensing process.

ITEM V.D.1: IMPROVE PUBLIC AND INTERVENOR PARTICIPATION IN THE HEARING PROCESS

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to assess alternative methods to enhance public and intervenor participation in the hearing process by undertaking a pilot program for intervenor funding in accordance with the FY-81 budget request and by studying the concept of an Office of Hearing Counsel, as described by the President's Commission recommendation, and other concepts of Public Counsel (such as the Office of Public Counsel recommended by the NRC Special Inquiry Group or concepts used by some Public Service Commissions). If such concepts proved to be desirable, the Commission was to propose the needed legislation. This item was originally identified as Item 5 in Chapter V but was made Item V.D.1 in Rev. 1 to NUREG-0660.[48]

The NRC sought authorization to establish a pilot program[872] to fund intervenors in its budget request for FY-81. Congress not only failed to enact such legislation, but included a provision in NRC's 1981 Appropriations Act (Public Law 96-367)[924] which precluded the use of funds to pay the expenses of, or otherwise compensate, parties intervening in NRC proceedings. After enactment of this legislation and issuance of an opinion by the Comptroller General on December 3, 1980, the NRC terminated[873, 874] a one-year pilot program it had established to provide intervenors certain forms of procedural assistance, such as free hearing transcripts and service of documents. Congress also barred the NRC from funding intervenors in FY-82 and FY-83. Prior to Congressional action, OGC had begun a review of the desirability of creating an Office of Public Counsel. After Congress prohibited intervenor funding, OGC terminated its review.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence is assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.D.2: STUDY CONSTRUCTION-DURING-ADJUDICATION RULES

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to complete rulemaking on whether construction should be permitted while challenges to a construction permit authorized by a licensing board are under administrative adjudication. This item was originally identified as Item 6 in Chapter V but was made Item V.D.2 in Rev. 1 to NUREG-0660.[48]

Following the TMI-2 accident, the Commission suspended in part its so-called immediate effectiveness rule. This rule had authorized the issuance of reactor construction permits or operating licenses immediately upon receipt of a favorable licensing Board decision, notwithstanding the filing of administrative appeals. In its place, the Commission instituted a mandatory review procedure for such decisions. In 1981, the rule was partially reinstated with respect to decisions authorizing the issuance of a reactor operating license. The rule, as applied to decisions authorizing reactor construction, has been the subject of a separate rulemaking.

The Commission published a notice of proposed rulemaking and requested comments on several options for amending the immediate effectiveness rule for construction permit decisions.[875] On October 25, 1982, the Commission published a proposed rule that would make the effectiveness review procedures for construction permits conform to those for operating licenses.[876] The Commission noted that it was still considering the various options presented and that revisions might be proposed later as part of broader reforms to the Commission's hearing process. As a result of further consideration, the Commission now has pending before it a new rulemaking proposal relative to immediate effectiveness reviews for both construction permits and operating licenses.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

Staff stated in the Supplement to this report published in 1986 that a solution to this Licensing Issue was available, but the item had not been resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM V.D.3: REEXAMINE COMMISSION ROLE IN ADJUDICATION

DESCRIPTION

This NUREG-0660[48] Rev. 1 item called for the Commission to review its role in adjudications to examine the extent of Commission involvement in licensing proceedings and to eliminate any undesirable and unnecessary insulation of the Commission from decision-making activities of the staff. This item was originally identified as Item 17 in Chapter V but was made Item V.D.3 in Rev. 1 to NUREG-0660.[48]

The Commission's role in adjudication is addressed under three topics: the immediate effectiveness review, the appellate process structure, and communications between the Commission and the staff.

Immediate Effectiveness Reviews: Following the TMI-2 accident, the Commission promulgated amendments to its regulations (10 CFR 2.764) which increased the Commission's role in adjudications. Under the revised regulations, decisions by Atomic Safety and Licensing Boards (ASLB) which authorize a utility to operate a facility at full power do not become effective upon issuance. Instead, the Commission conducts an "immediate effectiveness review" to determine whether the ASLB decision should be effective during the pendency of administrative appeals. The Commission seeks to complete these reviews within 30 days of the ASLB decision, or on an otherwise timely basis when the applicant has not completed construction or is not otherwise ready to operate at full power.

In 1981, the Commission established a Regulatory Reform Task Force to examine the NRC's licensing process. This Task Force recommended a different approach; it advocated the "immediate effectiveness" rule that was in place prior to the TMI-2 accident, i.e., construction permits and operating licenses should be issued on the basis of favorable ASLB decisions with an immediate effectiveness review by the Commission. In October 1982, the Commission issued for public comment a Notice of Proposed Rulemaking[876] which, if adopted, would extend the immediate effectiveness review procedures to ASLB decisions which authorize the issuance of construction permits or limited work authorizations.

As is indicated in the discussion under Item V.D.2 above, the Commission now has pending before it a new rulemaking proposal relative to immediate effectiveness reviews for operating licenses.

Structure of the Appellate Process: The Commission has a three-tier adjudicatory system. Matters are first heard by an ASLB, followed in most cases by a mandatory review by an ASLAB and then by a discretionary Commission review. In December 1979, OGC prepared a study of the Commission's appellate system. One option examined, but not recommended, was to increase the Commission's adjudicatory role by eliminating the ASLAB. After receiving public comments on the study, the Commission decided not to abolish ASLAB review. The Regulatory Reform Task Force recommended to the Commission that it remove the ASLAB as an intermediate appeal body, but assign it responsibility of drafting Commission adjudicatory orders. The Commission did not adopt this recommendation.[984]

Communications Between the Commission and the Staff: The Commission's Regulatory Reform Task Force recommended that the Commission modify its separation of functions (10 CFR 2.719) and **ex parte** rules (10 CFR 2.780) to permit greater communication between the Commission and the staff on matters under adjudication.

On March 26, 1986, the Commission published a proposed rule to revise the Commission's separation of functions and **ex parte** rules.[877] Present rules preclude communications between the Commission and any NRC staff member concerning a substantive matter at issue in a formal adjudicatory proceeding. Under this proposed rule, only those members of the NRC staff who are involved in an "investigative or litigative" function relative to a particular proceeding would be barred from communicating with the Commission on disputed issues in the proceeding, thereby allowing for much wider Commission access to staff expertise.

On November 2, 1983, the Commission published in the Federal Register an Advanced Notice of Proposed Rulemaking on the role of the NRC staff in the licensing process.[985]After evaluating the public comments, the Commission determined that no change should be made in the staff's role and accordingly withdrew its Advance Notice of Proposed Rulemaking.[986]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

Staff stated in the Supplement to this report published in 1986 that a portion of this item had not been resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issuess are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM V.D.4: STUDY THE REFORM OF THE LICENSING PROCESS

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to study alternatives to reform the licensing process. One suggested reform would abolish the present two-step process for initial licensing and would substitute a one-step process with increased public involvement prior to the hearing. It would also involve continued NRC jurisdiction after issuance of the single permit to verify that plant construction conforms with plans and permit specifications. The Commission was to study the standardization of nuclear power plants and consider suspending review and proceedings for applications for CPs and LWAs until the reform issues were resolved. This item was originally identified as Item 9 in Chapter V but was made Item V.D.4 in Rev. 1 to NUREG-0660.[48]

In its first formal response to the President's Commission on the TMI-2 accident, the Commission noted that a revision of licensing procedures to emphasize early and effective resolution of safety issues would require legislation (NUREG-0632).[878] Prior to forwarding proposed legislation to the Congress, the Commission took steps to improve the balance and efficiency of the power reactor licensing process. In May 1981, a statement of policy on the conduct of licensing proceedings was issued describing procedural devices which could expedite the hearings and providing Commission guidance on the use of such tools.[879] In addition, the Commission's rules of practice (10 CFR 2) were amended to expedite certain aspects of adjudicatory proceedings. Two rules were promulgated in 1982: (1) elimination of the need for power and alternative energy source issues from reactor operating license proceedings; and (2) elimination of the requirements for the review of financial qualifications of state-regulated public utilities applying for permits or licenses. Both of these rules were expected to further expedite licensing hearings. In view of the limitations of rulemaking as a means of reforming the nuclear power plant licensing process, the Commission proceeded to develop proposals for statutory changes that would accomplish the desired reforms.

In November 1981, the Commission established the Regulatory Reform Task Force[880] to review the NRC's licensing process. As a result of the efforts of this group and senior NRC officials, the Commission in June 1982 issued for public comment a draft of proposed legislation, "Nuclear Standardization Act of 1982," which included provisions for one-step licensing, issuance of a combined construction permit and operating license, and licensing of standardized plant. After review and consideration of the public comments and comments provided by an Ad Hoc Committee for the Review of Nuclear Reactor Licensing Reform Proposals, the Commission developed a draft bill, "Nuclear Power Reactor Licensing Reform Act of 1983," and on February 21, 1983 forwarded it to the Congress.[881] The 98th Congress did not act on the Commission's 1983 legislative proposal. The Commission submitted a revised proposal to the 99th Congress in 1985, but again Congress did not act.

The Regulatory Reform Task Force proposed that a number of reforms be accomplished via rulemaking: (1) amendment of 10 CFR 50 to modify the backfitting provision and associated sections applicable to reactors; (2) amendment of 10 CFR 2 and 10 CFR 50 to improve the quality of the hearing process; (3) amendment of 10 CFR 2 regarding separation of functions and **ex parte** communications in on-the-record adjudications; and (4) amendment of 10 CFR 2 to limit NRC staff participation as a party in contested initial license proceedings to issues on which the staff disagrees with the license applicant.

The Commission on September 20, 1983 issued a policy statement[882] on revising the backfitting process. It also issued an Advanced Notice of Proposed Rulemaking[883] on the backfitting process and presented a number of questions for public response. The final rule[884] on the backfitting process was published in September 1985.

The Commission on November 23, 1983 issued an Advance Notice of Proposed Rulemaking on amending its rules of practice (10 CFR 2) to change the staff's role in adjudicatory licensing hearings summarized this issue and presented a number of options for rulemaking and solicited public response to a set of questions.877 The Commission withdrew this notice after determining that no change in the staff role was warranted.[986]

The Commission on April 12, 1984 published[987] a Federal Register notice soliciting public comments on the changes to the hearing process proposed by the Regulatory Reform Task Force. After reviewing the public comments, the Commission determined that four of the proposals merited further consideration. These were published as a proposed rule.[988] The comment on October 17, 1986 and final action on the proposals is expected in early 1987.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

Staff stated in the Supplement to this report published in 1986 that this item was only partially resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

REFERENCES

0048.	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.
0872.	*Federal Register* Notice 45 FR 49535, "10 CFR Part 2, Procedural Assistance in Adjudicatory Licensing Proceedings," July 25, 1980.
0873.	*Federal Register* Notice 46 FR 13681, "10 CFR Part 2, Domestic Licensing Proceedings; Procedural Assistance Program," February 24, 1981.
0874.	Memorandum for L. Bickwit from S. Chilk, "SECY-81-391—Provision of Free Transcripts to All Full Participants in Adjudicatory Proceedings: May 11, 1981 Comptroller General Decision," February 25, 1982.
0875.	*Federal Register* Notice 45 FR 34279, "10 CFR Parts 2, 50, Possible Amendments to `Immediate Effectiveness Rule,'" May 22, 1980.
0876.	*Federal Register* Notice 47 FR 47260, "10 CFR Part 2, Commission Review Procedures for Power Reactor Construction Permits; Immediate Effectiveness Rule," October 25, 1982.

0877.	*Federal Register* Notice 51 FR 10393, "10 CFR Parts 0 and 2, Revision of Ex Parte and Separation of Functions Rules Applicable to Formal Adjudicatory Proceedings," March 26, 1986.
0878.	NUREG-0632, "NRC Views and Analysis of the Recommendations of the President"s Commission on the Accident at Three Mile Island," U.S. Nuclear Regulatory Commission, November 1979.
0879.	*Federal Register* Notice 46 FR 28533, "Statement of Policy on Conduct of Licensing Proceedings," May 27, 1981.
0880.	Memorandum for All Employees from N. Palladino, "Regulatory Reform Task Force," November 17, 1981.
0881.	Letter to the Honorable Thomas P. O"Neill, Jr. from N. Palladino, February 21, 1983.
0882.	*Federal Register* Notice 48 FR 44173, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.
0883.	*Federal Register* Notice 48 FR 44217, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.
0884.	*Federal Register* Notice 50 FR 38097, "10 CFR Parts 2 and 50, Revision of Backfitting Process for Power Reactors," September 20, 1985.
0924.	SECY-96-107, "Uniform Tracking of Agency Generic Technical Issues," U.S. Nuclear Regulatory Commission, May 14, 1996. [9605230140]
0984.	Memorandum for J. Tourtelotte et al. from S. Chilk, "Addendum to SRM M841218—Briefing and Discussion on the Hearing Process, 2:00 p.m., Tuesday, December 18, 1984, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," January 31, 1985. [8502060511]
0985.	*Federal Register* Notice 48 FR 50550, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings," November 2, 1983.
0986.	*Federal Register* Notice 51 FR 36811, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings," October 16, 1986.
0987.	*Federal Register* Notice 49 FR 14698, "10 CFR Parts 2 and 50, Request for Public Comment on Regulatory Reform Proposal Concerning the Rules of Practice, Rules for Licensing of Production and Utilization Facilities," April 12, 1984.
0988.	*Federal Register* Notice 51 FR 24365, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings—Procedural Changes in the Hearing Process," July 3, 1986.
1858.	Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.
1967.	SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [*ML111590814*]

Task V.E: Legislative Needs (Rev. 1) ( )

The objective of this task was to evaluate legislative needs evidenced by and from TMI.

ITEM V.E.1: STUDY THE NEED FOR TMI-RELATED LEGISLATION

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to study the need for legislation with respect to the following:

(1)	Clarification of NRC authority to issue a license amendment prior to a hearing when necessary to ensure the health and safety of the public;
(2)	Determination of whether NRC should seek an amendment to the Sunshine Act to reduce the Act's requirements for Commission meetings during an emergency;
(3)	Determination of NRC's current legal authority to take over and conduct cleanup actions at a nuclear facility and with respect to the Federal Government's (a) liability for damages occurring during a cleanup conducted by NRC, and (b) entitlement to reimbursement for cleanup costs;
(4)	The continuing desirability of the Price-Anderson Act in two areas: (a) extraordinary nuclear occurrence, and (b) limitation of liability;
(5)	Desirability of creating a new category of license to be issued in place of an operating license for a facility during an extended recovery period following a major accident;
(6)	The need for new or modified NRC authority to address the establishment of a chartered national operating company or consortium.

This item was originally identified as Item 7 in Chapter V but was made Item V.E.1 in Rev. 1 to NUREG-0660.[48]

The following is a discussion of NRC actions on the six subtasks of this item:

(1)	Section 12 of Public Law 97-415[928] was amended in 1983 and clarified Commission authority under Section 189a of the Atomic Energy Act of 1954. This amendment, commonly referred to as the "Sholly Amendment," clarified NRC authority to issue a license amendment prior to a hearing when necessary to ensure the health and safety of the public. Thus, this subtask was resolved.
(2)	The NRC Reorganization Plan No. 1 of 1980[929] directed the Chairman to act for the Commission in emergencies. This legislation nullified the need for any amendment to the Sunshine Act which originally required Commission meetings during emergencies. Thus, this subtask was resolved.

(3)	In November 1980, NRC issued NUREG-0689[931] which adddressed NRC's legal authority over cleanup activities. After receiving this document, the Commission has not sought any legislation to augment or clarify its authority. Thus, this subtask has been resolved.
(4)	The Congress is expected to pass, in the next legislative session, a revision to the Price-Anderson Act which will alter the limitation on liability provisions. It is likely that the extraordinary nuclear occurrence (ENO) provisions will remain in slightly modified form. The Commission published a proposed amendment to 10 CFR 140 revising its criteria for an ENO in April of 1985.[989] A final rule is expected in 1987. This subtask will be resolved when a final rule is approved by the Commission.
(5)	Although it might be convenient to have a special category of license for a facility engaged in extended recovery operations, it has been the Commission's experience with the TMI-2 cleanup phase that NRC's authority to issue orders and license amendments provides adequate flexibility for conducting recovery operations within the framework of the preexisting facility license. Accordingly, the staff determined that there was no need to develop a new license category. Thus, this subtask has been resolved.
(6)	This subtask called for the formation of an industry-wide consortium to operate the nuclear plants of utilities that are unable to meet the increased regulatory demands resulting from the TMI-2 accident. The Commission has not sought legislation in this area. Thus, this subtask has been resolved.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

Staff stated in the Supplement to this report published in 1986 that this item would be resolved when Subtask (4) was completed. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

REFERENCES

0048.	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.
1858.	Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967.	SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [*ML111590814*]

Task V.F: Organization and Management ( )

The objective of this task was to improve Commission organization and management.

ITEM V.F.1: STUDY NRC top MANAGEMENT STRUCTURE AND PROCESS

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to hire an independent management consulting firm to examine the internal management approaches and procedures used by the Commissioners to execute their responsibilities and to examine possible improvements in the Commission's efficiency and effectiveness.

The FY-80 NRC Authorization Act required the Commission to contract for the completion by July 1981 of an independent review of the Commission's management structure, processes, procedures, and operations at all levels of agency management. This item was originally identified as Item 10 in Chapter V but was made Item V.F.1 in Rev. 1 to NUREG-0660.[48]

In May 1980, the Commission developed and issued a Policy and Planning Guidance (PPG)[210] document to provide direction to the staff on general policies and objectives of the agency. The PPG provides a common basis for establishing agencywide priorities and is instrumental in shaping NRC programs and plans. The PPG is updated and revised annually.

In support of the PPG, the EDO Program Guidance was developed to help determine appropriate resource needs through the budget process. Together, the PPG and the EDO Program Guidance form the basis for agency planning and program development. The EDO Program Guidance is updated and revised annually.

In an NRC letter[934] to the Congress in March 1981, it was indicated that Congressional concerns involving the Commission's management structure, processes, and operations were adequately addressed by actions already underway. Plans to have a consulting firm examine internal management were therefore dropped.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.F.2: REEXAMINE ORGANIZATION AND FUNCTIONS OF THE NRC OFFICES

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to examine the organization and functions of the NRC offices to identify possible improvements in the overall efficiency and effectiveness of NRC including (1) an evaluation of the consolidation of all NRC resources and activities for monitoring operating reactors in a single office; (2) the reorganization of NRR to elevate human factors in criteria development and systems evaluation to a level of prominence equivalent to that of the safety equipment; (3) the reorganization of OIE to increase inspection and enforcement effectiveness; (4) the establishment of an integrated program for modifying regulatory requirements based on systemic identification and assessment of safety issues; and (5) the use of technical consultants to increase staff capability in discrete technical areas. This item was originally identified as Item 11 in Chapter V but was made Item V.F.2 in Rev. 1 to NUREG-0660.[48]

Over the past several years, a number of organizational changes have taken place at NRC to improve the effectiveness and efficiency of the agency. Some of the major changes are listed below.

Monitoring Operating Reactors: AEOD was created to identify and communicate lessons of operating experience to all appropriate parties. ORAB was also created in NRR to perform systematic assessments of operating reactor experience. Following the NRR reorganization in November 1985, ORAB was made the Operating Reactors Assessment Staff (ORAS).

Human Factors Considerations: The Division of Human Factors Safety (DHFS) was created within NRR to provide increased emphasis on the benefits and problems represented by the human element in nuclear operations. Following the NRR reorganization in November 1985, DHFS was made the Division of Human Factors Technology (DHFT).

Inspection and Enforcement:

(1) The emergency preparedness function was transferred to OIE and the Division of Emergency Preparedness and Engineering Response (DEPER) was created. An OIE component to manage NRC's incident response operations and planning efforts was also added.

(2) Improvements and intensification of inspections at operating reactor sites and at plants under construction were made that included (a) more inspection effort at operating power reactors directed toward verification of licensees' implementation and completion of actions specified in the TMI Action Plan; (b) special attention in the construction inspection program focused on quality assurance, on-site design, and review of as-built structures and systems; (c) raised limit on NRC fine for a single violation from $5,000 to as much as $100,000 per day with no ceiling on the total fine for any 30-day period; (d) NRC Policy changes that strengthened enforcement measures to prohibit operations by licensees who fail to meet adequate levels of protection and made non-compliance more expensive than compliance; (e) various improvements to inspector training; (f) studies by SNL, the results of which were used to further increase the effectiveness and safety efficiency of the operating reactor inspection program (NUREG/CR-1368[935] and NUREG/CR-1369[936]); (g) revisions to inspection programs making safety verification the highest priority; (h) at least one inspector assigned to every site with an operating power reactor and every site where construction activities are in progress; and (i) identification of licensee management control problems through NRC team appraisals. These changes are documented in NRC Manual Chapter 0127.[335]

Safety Issues: (1) A systematic review has been performed of all candidate issues from the TMI investigations and continues to be performed for issues identified from operating reactor experience; (2) Generic issues are integrated into an agencywide program according to priority based on potential safety significance and cost of implementation; (3) NRC uses a 10-criteria evaluation process to determine which of the safety issues could be designated as USIs. Task Action Plans are developed to resolve those issues identified for further pursuit.

Technical Consultants: (1) NRR technical assistance in Operating Reactors and Casework more than tripled from FY-79 to FY-81 mostly because of the TMI-2 accident impact. This increase reflected additional technical assistance from such technical experts as the Franklin Research Institute to complete the growing number of reactor licensing amendments/actions. Efforts were also augmented for casework activities through technical support from contractors such as SAI and from the DOE National Labs; (2) Additional members were added to the panel of ASLBs which, in many cases, were reconstituted to minimize schedule conflicts.

Other Reorganizations: (1) During 1981, the Offices of Nuclear Regulatory Research (RES) and Standards Development (SD) were consolidated into a single RES Office. This change made the research function more responsive to the regulatory needs of the agency, by more direct application of research to regulations and rules, and made more effective use of staff resources. (2) A new position (Deputy Executive Director for Regional Operations and Generic Requirements) was created in 1981 in the EDO office to bring tighter control and coordination to new regulatory requirements and to help manage the enlarged role of NRC regional offices. (3) Also in 1981, regional operations were expanded to improve the quality of regulation by transferring a number of headquarters functions to the regions. Regional Administrators report directly to the EDO and act as key agents in NRC's interactions with licensees. This change is documented in NRC Manual Chapter 0128.[335]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.F.3: REVISE DELEGATIONS OF AUTHORITY TO STAFF

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to improve NRC's organizational and management capabilities for effective pursuit of safety goals by clarifying and, as necessary, revising delegations of authority to the staff. This item was originally identified as Item 12 in Chapter V but was made Item V.F.3 in Rev. 1 to NUREG-0660.[48]

The Commission delegated substantial rulemaking authority to RES. To reflect the requirements of the President's Reorganization Plan No. 1 of 1980,[929] NRC reviewed and changed manual chapters[335] that dealt with the delegation of authority to staff office directors. These changes were documented in an EDO memorandum to the Chairman in December 1980.[937] NRC also reviewed delegations of authority to Commission-level offices and concluded no changes were required. This conclusion was documented in SECY-80-497.[938] Additional rulemaking authority was delegated to the staff in 1985.[992]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.F.4: CLARIFY AND STRENGTHEN THE RESPECTIVE ROLES OF CHAIRMAN, COMMISSION, AND EXECUTIVE DIRECTOR FOR OPERATIONS

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to clarify and strengthen the respective roles and authorities of the Chairman as chief executive officer, the Commission as head of the agency, and the EDO as chief staff officer. This item was originally identified as Item 13 in Chapter V but was made Item V.F.4 in Rev. 1 to NUREG-0660.[48]

The President's Reorganization Plan of 1980[929] served to strengthen the authority of the NRC Chairman relative to the Commission. For example, the Chairman is the official spokesman and principal executive officer of the Commission and directs and delegates various functions to the EDO who reports to the Chairman on all matters. Two Commission-level offices (Public Affairs and Congressional Affairs) also report directly to the Chairman. The Commission retains responsibility for policy formulation, rulemaking, orders, and adjudication. The Chairman initiates personnel actions, subject to Commission approval, for heads of offices reporting directly to the Commission, for the EDO, and for the heads of the major program offices. The Chairman directs and delegates to the EDO responsibility for all administrative functions, distribution of business, preparation of reorganization proposals and budget estimates, allocation of funds, and personnel matters other than those affecting the major program offices and certain other offices reporting to the Commission.

The EDO position was also strengthened relative to the program staff. For example, all program offices and regions report to the EDO. The EDO keeps the Commission fully and currently informed through the Chairman. The EDO is to be consulted regarding actions affecting the program offices and regional offices. These procedures are documented in NRC Manual Chapter 0103.[335]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.F.5: AUTHORITY TO DELEGATE EMERGENCY RESPONSE FUNCTIONS TO A SINGLE COMMISSIONER

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to seek authority to delegate specific management responsibilities to an individual Commissioner in the event of defined emergencies. (See also Item III.A.3.1 in which NRC is to develop its role in responding to nuclear emergencies.) This item was originally identified as Item 14 in Chapter V but was made Item V.F.5 in Rev. 1 to NUREG-0660.[48]

The Commission's emergency response functions were transferred to its Chairman, as documented in the Reorganization Plan No. 1 of 1980.[929] The responsibilities of the Commission and staff are set forth in detail in NRC Manual Chapter 0502.[335]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.
0210.	NUREG#0885, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission, (Issue 1) January 1982, (Issue 2) January 1983, (Issue 3) January 1984, (Issue 4) February 1985, (Issue 5) February 1986, (Issue 6) September 1987.
0335.	Memorandum for J. Taylor from D. Morrison, 'Resolution of Generic Safety Issue 83, "Control Room Habitability,"' June 17, 1996. [9607250277]
0929.	Regulatory Guide 1.139, "Guidance for Residual Heat Removal," U.S. Nuclear Regulatory Commission, May 1978.
0934.	Letter to the Honorable Alan Simpson from Joseph Hendrie, March 24, 1981. [8104030556]
0935.	NUREG/CR-1368, "Development of a Checklist for Evaluating Maintenance, Test and Calibration Procedures Used in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1980.
0936.	NUREG/CR-1369, "Procedures Evaluation Checklist for Maintenance, Test and Calibration Procedures," U.S. Nuclear Regulatory Commission, May 1980.
0937.	Memorandum for Chairman Ahearne from W. Dircks, "Manual Chapters Delegation of Authority to Staff Office Directors," December 23, 1980.
0938.	SECY-80-497, "Review of Delegations of Authority and Other Documentation," U.S. Nuclear Regulatory Commission, November 10, 1980. [8011190612]
0992.	*Federal Register* Notice 50 FR 42145, "10 CFR Part 1, Statement of Organization and General Information," October 18, 1985.

Task V.G: Consolidation of NRC Locations ( )

The objective of this task was to achieve a single location for the NRC's headquarters office.

ITEM V.G.1: ACHIEVE SINGLE LOCATION, LONG-TERM

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to break the impasse hindering the location of NRC and its major headquarters staff components in a single location (a single building or an adjacent group of buildings). The accomplishment of this objective was essential, among other purposes, to minimize adverse disruption of NRC headquarters upon installation of the NRC terminal of the nuclear data link and of headquarters computer and simulator equipment. (See Item III.A.3.4.) This item was originally identified as Item 15 in Chapter V but was made Item V.G.1 in Rev. 1 to NUREG-0660.[48]

Since its inception as an independent regulatory agency, the NRC has sought to consolidate its headquarters' staff in the Washington, D.C. area. Further emphasis was placed on this effort as a result of the TMI-2 accident and the continuing recommendations of various House and Senate Committees, special commissions, OMB, and the GAO. As a result of past discussions between senior NRC and GSA officials, both agencies acknowledged the benefits of NRC headquarters consolidation and agreed to develop alternative options that would accommodate both short- and long-term housing solutions. On November 6, 1986, GSA concluded negotiations with White Flint North Limited Partnership to purchase an 18-story building (One White Flint North, 11555 Rockville Pike, Rockville, Maryland) for the consolidation of NRC.

The prospectus for completion of the NRC consolidation at White Flint has been favorably reported on by OMB and by the House Committee on Public Works. Under the terms of the Public Buildings Act of 1959, only Committee action is required for Congressional approval of the project. With Congressional approval, the GSA proposes to lease approximately 300,000 additional square feet of space in Rockville to complete consolidation. Congressional approval of the prospectus would authorize GSA to negotiate extension of existing leases of NRC-occupied buildings until relocation is completed.

The purchase contract for One White Flint North includes an option to lease a second building of similar size to be constructed on an adjacent portion of the One White Flint North site. On November 14, 1986, the government exercised this option and the developer will, under the terms of the contract, construct the second building within 30 months of the execution of the option. An option to purchase the second building may be exercised by the government in the 5th year of the 20-year lease. At present the NRC continues to be dispersed in 10 buildings located in Washington, D.C. and Maryland.

CONCLUSION

This Licensing Issue has been resolved.

ITEM V.G.2: ACHIEVE SINGLE LOCATION, INTERIM

DESCRIPTION

This NUREG-0660,[48] Rev. 1 item called for the Commission to promptly reduce the distance between NRC headquarters offices by the consolidation of NRC offices in the Matomic Building (1717 H Street, N.W., Washington, D.C.) and in some of its present Bethesda locations. This move was intended to house the NRC program offices in one building. The agencies leaving 1717 H Street were to occupy either space vacated as a result of the NRC movement from suburban areas or other undetermined space. This item was originally identified as Item 16 in Chapter V but was made Item V.G.2 in Rev. 1 to NUREG-0660.[48]

After extensive planning for the interim consolidation of major organizational elements of NRC in the Matomic Building, GSA withdrew the building from consideration for assignment in March 1982 based upon economic considerations. In September 1983, GSA further informed NRC that they were considering the relocation of all federal tenants as a result of the age and general condition of the Matomic Building.

Interim consolidation planning, the potential forced relocation from the Matomic Building, and preparation of the concurrent Agency Space Reduction Program which responds to Federal Property Management Temporary

Regulations D-68 and D-70 (Work Space Management Reforms) were considered in the interim housing of NRC headquarters' staff in a minimum number of locations in Bethesda, Maryland, and Washington, D.C. Achievement of a single, interim location was never accomplished and has been overtaken by NRC efforts toward achieving a single, long-term location.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

0048.	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.