ISSUE 178: EFFECT OF HURRICANE ANDREW ON TURKEY POINT

DESCRIPTION

On August 24, 1992, Hurricane Andrew hit south Florida and caused extensive onsite and offsite damage at Turkey Point. Following this hurricane which was classified Category 4, an NRC/industry team was organized to: (1) review the damage caused to the nuclear units; (2) review the licensee actions to prepare for, and recover from, the storm; and (3) compile lessons that might benefit other nuclear reactor facilities. Results of the team review were documented in NUREG-1474[1610] which was distributed to all power reactor licensees by INPO on June 10, 1993.

As a result of the EDO request for staff review of the NRC/industry report to determine the actions necessary for resolving the identified concerns, an action plan[1611] was established by NRR. Among the items in the plan were two that addressed whether there was the need to develop generic guidance for:

(1) licensees, to ensure that their offsite communication circuits can reliably survive or recover from the impact of a severe natural event such as a hurricane. (These circuits are required to provide reliable notification to offsite authorities of emergency conditions at the affected licensee's power reactor facility.)

(2) inspectors, to review licensees' preparation for, and response to, natural disasters, including industry pre-planned support.

This issue was identified in an NRR memorandum[1601] to RES in February 1996 and addressed the staff's efforts to increase its knowledge and understanding of hurricanes in order to increase its confidence in assessing levels of safety. Therefore, it was considered a Licensing Issue.[1731]

CONCLUSION

For Item 1, TI 2515/131, "Licensee Offsite Communication Capabilities," was issued[1136] in January 1996 to provide inspectors with guidance for collecting information on offsite notification circuits and, between February 1 and June 30, 1996, this TI was used at 17 plants. Data collected from these inspections and previous inspections were evaluated to determine if guidance to licensees in the form of generic communication was necessary to provide either survivability or rapid recoverability of the circuits from a severe natural event.[1770] For Item 2, the staff concluded that, from an emergency preparedness standpoint, sufficient guidance existed for reviewing licensee preparations in response to a hurricane or other external events.

The staff issued IN 93-53,[1612] Supplement 1, which expanded the scope of lessons learned to other external events and discussed existing regulatory guidance for various external events. The action to provide guidance for inspectors to address any vulnerabilities that may develop from the review of IPEEEs (GL 88-20, Supplement 4)[1222] was incorporated into Activity 1.3 (b) of the Probabilistic Risk Assessment Implementation Plan. Thus, this issue was resolved.

REFERENCES

1136. Memorandum to C. Miller from R. Borchardt, "Review of Temporary Instruction 2515/131, `Licensee Offsite Communication Capabilities,' for Deletion from the NRC Inspection Manual," December 3, 1996.

1222. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988, (Supplement 1) August 29, 1989, (Supplement 2) April 4, 1990, (Supplement 3) July 6, 1990, (Supplement 4) June 28, 1991, (Supplement 5) September 8, 1995.

1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996.

1610. NUREG-1474, "Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992," U.S. Nuclear Regulatory Commission and the Institute of Nuclear Power Operations, March 1993.

1611. Memorandum for J. Taylor from T. Murley, "Office of Nuclear Reactor Regulation (NRR) Plan for Generic Follow-on Actions - `Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992,'" July 22, 1993.

1612. NRC Information Notice 93-53, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," July 20, 1993, (Supplement 1) April 29, 1994.

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

1770. Memorandum for W. Russell from C. Miller, "Licensee Offsite Communication Capabilities; Results of Information Gathering Using Temporary Instruction," September 26, 1996.

ISSUE 179: CORE PERFORMANCE

DESCRIPTION

Core design is a fundamental component of plant safety because maintaining fuel integrity is the first principal safety barrier (i.e., fuel cladding, reactor coolant system boundary, or the containment) against serious radioactive releases. Likewise, safety analyses must be properly performed in order to verify, in conjunction with startup tests and normal plant parameter monitoring, that a core reload design is adequate and provide assurance that the reactor can be safely operated. QA activities are important to ensure that proper interfaces are established and that shortcuts are not taken that could degrade safety or quality.

Following a briefing of the EDO on March 29, 1994, NRR developed an action plan to address the review of fuel fabrication, core design, and reload analysis issues; this plan was a proactive approach to improve core performance in operating reactors. The briefing presented by the Reactor Systems Branch (SRXB), Division of Systems Safety and Analysis (DSSA), NRR, covered generic fuel and core performance issues and related evaluations of fuel failures; the Vendor Inspection Branch (VIB), Division of Reactor Inspection and Licensee Performance (DRIL), NRR, also participated in the briefing. Specific actions included in the action plan were: (1) evaluate fuel vendors' QA performance through performance-based inspections that evaluate the reload core design, safety analysis, licensing process, fuel assembly mechanical design, and fuel fabrication activities; (2) evaluate the QA performance of licensees that perform core reload analysis functions; (3) identify, document, and categorize core performance problems and root cause evaluations that will be further evaluated during the inspections, and provide input to licensee evaluations as well as support to regional enforcement actions, as appropriate; and (4) train and coordinate regional support staff participating in these activities; and (5) evaluate the results of these activities for use in formulating generic communications, revisions of regulatory guidance, regional inspection guidance, and other appropriate regulatory actions.

The action plan called for DSSA/NRR to identify one or more licensee inspections in each region to be performed, in coordination with the regional inspectors, to assess licensee performance in reload core analysis oversight and participation. The data acquired through licensee/vendor inspections was to be integrated with information supplied by the regions and other sources and evaluated for generic core performance indicators and industry conformance to existing regulatory requirements. The end product of the overall assessment was to include guidance for resident inspectors and regional staff and draft updates to Inspection Manual Chapters. The action plan identified ten vendor inspections to be performed by multi-disciplined inspection teams led by the Special Inspection Branch (PSIB), DISP/NRR, with contractor technical assistance. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

The action plan was intended to improve the NRC's ability to assess safety through inspections of fuel vendors, evaluation of licensees' reload analyses and core performance information, and regional training and interaction. Therefore, it was classified as a Licensing Issue.[1731]

CONCLUSION

The staff performed inspections of licensee reload analyses and nuclear fuel vendors ABB/CE, Framatome Cogema Fuels (formerly B&W Fuel Company), GE, Siemens Power Corporation, and Westinghouse. Licensee participation in reload analyses and data acquired during the vendor inspections were evaluated by the staff for generic impact and identification of emerging issues. The vendor and licensee inspections were coordinated with the Regions and training was provided to the regional staff. With the inclusion of the evaluation of licensee/vendor lead test programs for identification of core performance problems in the Program Plan for High Burn-Up Fuel[1748] (see Issue 170), all required actions were completed and the issue was closed out.[1747]

REFERENCES

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1636. Memorandum to A. Thadani from G. Holahan and R. Spessard, "Action Plan to Monitor, Review, and Improve Fuel and Core Components Operating Performance," October 7, 1994.

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

1747. Memorandum to G. Holahan from J. Wermiel, "Closeout of Core Performance Action Plan (TAC Nos. M91256, M91602)," February 16, 1999.

1748. Memorandum to Chairman Jackson, et al., from L. Callan, "Agency Program Plan for High-Burnup Fuel," July 6, 1998.

ISSUE 180: NOTICE OF ENFORCEMENT DISCRETION

DESCRIPTION

The NRC requires that a licensee operate its facility in compliance with the NRC's regulations and the specific facility's license. When a licensee fails to comply with the conditions of its license or the NRC's regulations, the staff will take enforcement action in accordance with its Enforcement Policy, NUREG-1600,[1661] (previously 10 CFR 2, Appendix C).

The NRC recognizes that it is not always possible to anticipate every contingency that might arise during the lifetime of a facility that might result in non-compliance with specific license conditions. In such instances, enforcement action may not be appropriate, even though, technically, a non-compliance situation may exist. For such circumstances, the Enforcement Policy provides for a specific type of enforcement discretion that is known as a Notice of Enforcement Discretion (NOED). The NOED policy indicates when the staff, under certain limited circumstances, may choose not to enforce compliance with a license condition when specific safety criteria are met. Staff guidance for implementing the NOED policy is contained in the NRC Inspection Manual Part 9900: Technical Guidance.

In May 1994, the NRC discovered some inconsistencies in the staff's implementation of the NOED policy and established a team to review the NOED policy, implementation process, and staff practices, and develop appropriate recommendations. At the same time, the Office of the Inspector General (OIG) conducted an independent assessment of NRC's compliance with the NOED policy and procedures. Both the NOED Review Team and the OIG audit team reviewed certain aspects of the NOED policy and related guidance contained in Inspection Manual Part 9900.

The reviews showed that, in general, the staff complied with its procedures and requirements for review and determinations relating to NOEDs and that staff actions reflected adequate consideration of radiological health and safety of the public and sound technical and safety bases. However, a number of areas were identified where improvements could be made to the NOED policy and its implementation. These areas involved changes to the NOED policy, staff's guidance and procedures for implementing the NOED policy, and other support/administrative aspects. Also, as part of its overall enforcement policy review, the NRC asked for, and received, public comments on the NOED policy as published in the Federal Register, 59 FR 49215, September 27, 1994. An action plan was initiated to address the recommendations and to implement those that were adopted. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

CONCLUSION

After reviewing the results of the evaluations and associated recommendations, the staff concluded that the existing NOED policy is technically sound and, therefore, need not be revised; the staff's conclusions were documented in SECY-95-078.[1660] The staff, however, determined that several aspects of the NOED guidance and procedures needed clarification to ensure proper

implementation of the policy. In addition to these recommendations, the staff subsequently identified other areas for improving the NOED guidance.

On November 2, 1995, the staff issued a revised NRC Inspection Manual Part 9900, Technical Guidance, which contains improved staff guidance for implementing the NOED policy. Also, on November 7, 1995, the NRC issued Administrative Letter, 95-05, "Revisions to Staff Guidance for Implementing NRC Policy on Notices of Enforcement Discretion," to inform the nuclear industry of this improved guidance.

This issue addressed the staff's efforts in clarifying existing requirements and guidance and, therefore, was considered a Licensing Issue[1731]; it was resolved with the issuance of the revised staff guidance.

REFERENCES

1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996.

1660. SECY-95-078, "Staff Actions to Address Recommendations Resulting From Recent Evaluations of the Notice of Enforcement Discretion (NOED) Policy and Process," March 29, 1995.

1661. NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," U.S. Nuclear Regulatory Commission, July 1995.

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

ISSUE 181: FIRE PROTECTION

DESCRIPTION

In February 1993, NRR completed a reassessment of the reactor fire protection review and inspection programs in response to programmatic concerns raised during the review of Thermo-Lag fire barriers and prepared a report.[1626] A fire protection task action plan (FP-TAP) was then prepared to implement the recommendations that resulted from this reassessment. The FP-TAP includes a wide range of technical and programmatic fire protection issues, including recommendations for action (Part I), recommendations for further study (Part II), confirmation issues (Part III), and lessons learned (Part IV). Staff actions to address the recommendations were submitted to the Commission in SECY-93-143[1627] and progress reports[1628,1629] were issued. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

Each operating reactor has an NRC-approved fire protection plan that, if properly implemented and maintained, satisfies 10 CFR 50.48 and GDC 3. The staff's focus was on developing the framework for future direction of the NRC fire protection program with emphasis on a fire protection functional inspection (FPFI) program, a plan for developing and implementing this program, and a plan for centralized management, by NRR, of the FPFI program and all other reactor fire protection work. The principal objective of these efforts was to ensure that the NRC has a strong, broad-based and coherent fire protection program which was commensurate with the issue's safety significance.

CONCLUSION

The staff developed the FPFI program inspection procedures and guidance and drafted recommendations for centrally managing all reactor fire protection reviews and inspections, using headquarters and regional staff qualified to perform such work. With contract assistance from BNL, the staff continued to develop a PRA model for the self-induced station blackout (SBO) study; BNL also drafted a report on risk-based approaches for evaluating fire mitigation features in nuclear power plants. Thus, this issue addressed the staff's efforts in improving its capability to make independent assessments of safety and, therefore, was considered a Licensing Issue.[1731]

REFERENCES

1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996.

1626. Memorandum for A. Thadani from G. Holahan, "Revision to Report on the Re-Assessment of the NRC Fire Protection Program," February 27, 1993.

1627. SECY-93-143, "NRC Staff Actions to Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program," May 21, 1993.

1628. SECY-95-034, "Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program," February 13, 1995.

1629. Memorandum to Chairman Jackson, et al., from J. Taylor, "Semiannual Report on the Status of the Thermo-Lag Action Plan and Fire Protection Task Action Plan," September 20, 1995.

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

ISSUE 182: GENERAL ELECTRIC EXTENDED POWER UPRATE

DESCRIPTION

Historical Background

A generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their existing licensed limit. In 1990, GE submitted licensing topical reports to initiate this program by proposing to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5%. Since 1990, the NRC reviewed and approved at least 9 power uprate requests in the generic BWR power uprate program. As a follow-on to this program, GE submitted a Licensing Topical Report, NEDC-32424, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) in February 1995 which outlined the methodology for implementation of an extended power uprate program of up to 120% of the original licensed thermal power. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

The issue did not affect public safety but could have an economic impact on the operation of plants with GE reactors. Therefore, it was classified as a Regulatory Impact issue.[1731]

CONCLUSION

An action plan was developed by NRR and described the strategy for completing both the generic and plant-specific reviews for extended power uprate submittals for BWRs. The staff believed that individual plant submittals for uprates would likely contain requests for an optimum power level specific to each plant and would be some value less than the full 120%. Specific actions included in the generic action plan were: (1) review ELTR1 and issue a staff position paper; (2) review ELTR2 and issue an SER; (3) review the lead plant application and issue an SER; and (4) develop a standard review procedure based on ELTR1, ELTR2, and the lead plant review.

The staff developed a position paper in February 1996 following its review of the ELTR1. The final SERs for both the ELTR2 and the lead plant (Monticello) were issued in September 1998. The standard review procedure for power uprates was later included in the Maine Yankee Lessons Learned Issues. After meeting all objectives of the action plan, the issue was closed out[1737] by NRR.

REFERENCES

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1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

1737. Memorandum to NRR Division Directors from D. Matthews, "Director's Quarterly Status Report," January 26, 1999.

ISSUE 183: CYCLE-SPECIFIC PARAMETER LIMITS IN TECHNICAL SPECIFICATIONS

DESCRIPTION

The objective of this task was to respond to the Regulatory Review Group (RRG) Item #55. The RRG recommendations were to provide quicker review of core reload codes and to revise existing TS to permit changes, in accordance with approved core topical reports, to take advantage of improved analyses without a license amendment by revising GL 88-16[1638] (Core Operating Limits Report [COLR] Guidance). The task was subsequently revised to address the first recommendation only by preparing a supplement to GL 83-11.[1637] This issue was identified in an NRR memorandum[1601] to RES in February 1996.

The RRG recommended actions had no impact on safety and were only intended to reduce schedule and resource requirements for NRC review of reactor core reloads and the reload analysis methodology. Thus, the issue addressed economic impacts on both licensees and the NRC and, therefore, was considered a Regulatory Impact issue.[1731]

CONCLUSION

In resolving the issue, the staff drafted a proposed supplement to GL 83-11,[1637] presenting criteria for licensees who wish to perform their own analyses using previously approved methods. By complying with these criteria, a licensee would eliminate the need to submit a topical report qualifying the use of a previously approved methodology. The supplement was published for public comment in the Federal Register (60 FR 54712) on October 25, 1995. However, the staff later concluded that reducing NRC oversight was not justified because of concerns for improper application of approved methods by licensees as well as increased complexities in core reload analysis due to mixed core designs. This conclusion was to be published in the Federal Register. Thus, the issue was resolved.

REFERENCES

1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996.

1637. NRC Letter to All Operating Reactor Licensees, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11)," February 4, 1983.

1638. NRC Letter to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

ISSUE 184: ENDANGERED SPECIES

DESCRIPTION

In 1973, Congress passed the Endangered Species Act for the protection of endangered or threatened species. In responding to a Commission memorandum of July 30, 1991, concerning efforts of the Commission, applicants, and licensees for the protection of endangered species in the vicinity of nuclear power facilities, it was identified that the NRC may not have completed all the necessary activities required by the Endangered Species Act for some of the facilities that have identified endangered species.

This issue was identified in an NRR memorandum[1601] to RES in February 1996 and called for the staff to develop a list of currently listed protected species in the vicinity of each nuclear power plant site, identify individual licensee programs and activities being conducted to further the conservation of protected species, and conduct informal or formal consultation with either the National Marine Fisheries Service or the Fish and Wildlife Service, as warranted, on any specific site. This issue addressed impacts on the environment of nuclear plants and, therefore, was considered an Environmental Issue.[1731]

CONCLUSION

An action plan was developed to determine the additional actions, if any, that needed to be taken at individual sites so that the NRC could meet its obligations under the Endangered Species Act. This plan called for the staff to: (1) evaluate plant-specific lists of endangered species and existing licensee commitments to further the conservation of the protected species; and (2) determine if informal or formal consultation with either the National Marine Fisheries Service or the Fish and Wildlife Service is warranted. A list of currently listed protected species in the vicinity of each nuclear power plant site was developed by the staff.

REFERENCES

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ISSUE 190: FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE

DESCRIPTION

The risk of failure from fatigue of various reactor coolant system components was studied under Issue 78 and later integrated into the NRR Fatigue Action Plan which was completed and documented in SECY-95-245.[547] The staff concluded that the risk from fatigue failure of the primary coolant pressure boundary components is very small; this conclusion was based on a plant life of 40 years. The impact of a license renewal period of 20 years on fatigue of metal components was to be considered in the resolution of Issue 166,[1564] but Issue 190 was established to address this subject separately.[928]

CONCLUSION

Based on the existence of an ongoing action plan[928] to address the safety concern and the NRR decision[1517] to pursue its resolution, the issue was considered nearly-resolved in August 1996. It was later given a high priority ranking in SECY-98-166.[1718]

In resolving the issue, the staff performed probabilistic analyses which showed low core damage frequencies resulting from fatigue failure of metal components. However, the nature of age-related degradation indicated the potential for an increase in the frequency of pipe leaks as plants continue to operate. Consistent with 10 CFR 54.21, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants: Contents of Application - Technical Information," licensees will have to address the effects of the reactor coolant system environment on component fatigue life, as aging management programs are formulated in support of license renewal. Thus, the issue was RESOLVED with no new or revised requirements.[1764]

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928. Memorandum for A. Thadani from T. Speis, "Generic Safety Issue (GSI)-166, 'Adequacy of Fatigue Life of Metal Components,'" August 26, 1996.

1517. Memorandum for J. Sniezek from T. Murley and E. Beckjord, "Resolution of Fatigue and Environmental Qualification Issues Related to License Renewal," April 1, 1993.

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.

1718. SECY-98-166, "Summary of Activities Related to Generic Safety Issues," July 6, 1998.

1764. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue 190, 'Fatigue Evaluation of Metal Components for 60-Year Plant Life,'" December 26, 1999.

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2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

4. NUREG-0572, "Review of Licensee Event Reports (1976-1978)," U.S. Nuclear Regulatory Commission, September 1979.

5. IE Circular No. 77-07, "Short Period During Reactor Startup," U.S. Nuclear Regulatory Commission, April 15, 1977. [9104240445]

6. IE Bulletin No. 79-12, "Short Period Scrams at BWR Facilities," U.S. Nuclear Regulatory Commission, May 31, 1979. [7906060168]

7. Memorandum for D. Ross from H. Richings, "RDA Statistical Analysis," June 17, 1975. [8105050833]

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9. Federal Register Notice 54 FR 16030, "Draft Regulatory Guide; Withdrawal," April 20, 1989.

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

12. Draft Regulatory Guide and Value/Impact Statement, Task SC 708-4, "Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety," U.S. Nuclear Regulatory Commission, February 1981. [9503290322]

13. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, September 1980.

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16. 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.

17. NUREG/CR-0255, "CONTEMPT-LT/028: A Computer Code for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, March 1979.

18. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," U.S. Atomic Energy Commission, May 1973. [7907100189]

19. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," U.S. Nuclear Regulatory Commission, November 1979, (Rev. 1) July 1981.

20. Memorandum for R. Fraley from R. Mattson, "ACRS PWR Question Regarding Effect of Pressurizer Heater Uncovery on Pressurizer Pressure Boundary Integrity," November 5, 1979. [8004100530]

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27. Memorandum for R. Baer from A. Thadani, "RRAB Preliminary Assessment of the Reactor Coolant Pump Seal Failure Problem," December 12, 1980. [8103050765]

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31. Memorandum for H. Denton from M. Ernst, "DST Evaluation of the Automatic Air Header Dump on Boiling Water Reactors," December 8, 1980. [8101230203]

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33. Memorandum for B. Sheron from M. Srinivasan, "Probabilities and Consequences of LOCA/Loss of Offsite Power (LOOP) Sequences," April 13, 1982. [8206300420]

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35. Memorandum for S. Hanauer from T. Murley, "Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power," February 25, 1981. [8110190723]

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APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING AND FUTURE REACTOR PLANTS

This appendix contains a listing of those residual GSIs that are applicable to operating and future reactor plants and includes: issues that have been resolved with requirements [I, NOTE 3(a)]; USI, HIGH- and MEDIUM-priority issues scheduled for resolution; nearly-resolved issues scheduled for resolution (NOTES 1 and 2); and issues that are scheduled for prioritization (NOTE 4). The priority designations for all issues are consistent with those listed in Table II of the Introduction. In accordance with 10 CFR 52.47(a)(1)(iv), any future application for design certification must contain proposed technical resolutions for the issues in this listing that are designated USI, HIGH, MEDIUM, NOTE 1, and NOTE 2. (In July 1998, the priority categories NOTES 1 and 2 were eliminated and all GSIs in these categories were given a HIGH priority ranking.[17,18]) Also included in this listing are those GSIs that were either prioritized or resolved with no impact on operating reactor plants but contain recommendations for future reactor plants (NOTE 6).

Legend

NOTES: 1	- Possible Resolution Identified for Evaluation (Discontinued 07-06-98)
2	- Resolution Available [Documented in NUREG, NRC Memorandum, SER or equivalent] (Discontinued 07-06-98)
3(a)	- Resolution Resulted in the Establishment of New Regulatory Requirements [Rule, Regulatory Guide, SRP Change, or equivalent]
4	- Issue to be Prioritized in the Future
6	- New Requirements for Future Plants Recommended
B&W	- Babcock & Wilcox Company
CE	- Combustion Engineering Company
GE	- General Electric Company
HIGH	- High Safety Priority
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
MEDIUM	- Medium Safety Priority
MPA	- Multiplant Action
NA	- Not Applicable
TBD	- To Be Determined
USI	- Unresolved Safety Issue
W	- Westinghouse Electric Corporation

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.2	Shift Technical Advisor	I	All	All	F-01	09/13/79	09/27/79
I.A.1.2	Shift Supervisor Administrative Duties	I	All	All		09/13/79	09/27/79
I.A.1.3	Shift Manning	I	All	All	F-02	07/31/80	06/26/80
I.A.1.4	Long-Term Upgrading	NOTE 3(a)	All	All		04/28/83	04/28/83
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-	-	-	-	-	-
I.A.2.1(1)	Qualifications - Experience	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(2)	Training	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	I	All	All	F-03	03/28/80	03/28/80
I.A.2.3	Administration of Training Programs	I	All	All		03/28/80	03/28/80
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-	-	-	-
I.A.2.6(1)	Revise Regulatory Guide 1.8	NOTE 3(a)	All	All		TBD	05/--/87
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	I	All	All		03/28/80	03/28/80
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-	-	-	-
I.A.4.1(2)	Interim Changes in Training Simulators	NOTE 3(a)	All	All		04/--/81	03/28/81
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-	-	-	-
I.A.4.2(1)	Research on Training Simulators	NOTE 3(a)	All	All		04/--/87	04/--/87
I.A.4.2(2)	Upgrade Training Simulator Standards	NOTE 3(a)	All	All		04/--/81	04/--/81
I.A.4.2(3)	Regulatory Guide on Training Simulators	NOTE 3(a)	All	All		04/--/81	04/--/81
I.A.4.2(4)	Review Simulators for Conformance to Criteria	NOTE 3(a)	All	All		03/25/87	03/25/87

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-	-	-	-
I.C.1(1)	Small Break LOCAs	I	All	All		09/13/79	09/13/79
I.C.1(2)	Inadequate Core Cooling	I	All	All	F-04	09/13/79	09/13/79
I.C.1(3)	Transients and Accidents	I	All	All	F-05	09/13/79	09/27/79
I.C.2	Shift and Relief Turnover Procedures	I	All	All		09/13/79	09/27/79
I.C.3	Shift Supervisor Responsibilities	I	All	All		09/13/79	09/27/79
I.C.4	Control Room Access	I	All	All		09/13/79	09/27/79
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	I	All	All	F-06	05/07/80	06/26/80
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	I	All	All	F-07	10/31/80	10/31/80
I.C.7	NSSS Vendor Review of Procedures	I	All	All		NA	06/26/80
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	I	All	All		NA	06/26/80
I.C.9	Long-Term Program Plan for Upgrading of Procedures	NOTE 3(a)	All	All		09/13/79	06/--/85
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	I	All	All	F-08	06/26/80	06/26/80
I.D.2	Plant Safety Parameter Display Console	I	All	All	F-09	06/26/80	06/26/80
I.D.5	Improved Control Room Instrumentation Research	-	-	-	-	-	-
I.D.5(2)	Plant Status and Post-Accident Monitoring	NOTE 3(a)	All	All		NA	12/--/80
I.F	QUALITY ASSURANCE						
I.F.2	Develop More Detailed QA Criteria	-	-	-	-	-	-
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(6)	Increase the Size of Licensees' QA Staff	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	NOTE 3(a)	All	All		NA	07/--/81
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	I	All	All		NA	06/26/80
I.G.2	Scope of Test Program	NOTE 3(a)	All	All		NA	07/--/81

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	I	All	All	F-10	09/13/79	09/27/79
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	I	All	All	F-11	09/13/79	09/27/79
II.B.3	Post-Accident Sampling	I	All	All	F-12	09/13/79	09/27/79
II.B.4	Training for Mitigating Core Damage	I	All	All	F-13	03/28/80	03/28/80
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	NOTE 3(a)	All	All		TBD	NA
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	NOTE 3(a)	All	All		TBD	01/25/85
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	I	All	All	F-14	09/13/79	09/27/79
II.D.3	Relief and Safety Valve Position Indication	I	All	All		07/21/79	09/27/79
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	I	NA	All	F15	03/10/80	03/10/80
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	I	NA	All	F-16, F-17	09/13/79	09/27/79
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	NOTE 3(a)	All	All		NA	07/--/81
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	I	NA	All		09/13/79	09/27/79
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	I	All	All	F-18	09/13/79	09/27/79
II.E.4.2	Isolation Dependability	I	All	All	F-19	09/13/79	09/27/79
II.E.4.4	Purging	-	-	-	-	-	-
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	NOTE 3(a)	All	All		11/28/78	NA
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	NOTE 3(a)	All	All		10/22/79	NA
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	NOTE 3(a)	All	All		09/27/79	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
			BWR	PWR			
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	NOTE 3(a)	NA	B&W			
II.E.5.2	B&W Reactor Transient Response Task Force	NOTE 3(a)	NA	B&W			
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	NOTE 3(a)	All	All		06/--/89	06/--/89
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	I	All	All	F-20, F-21 F-22, F-23 F-24, F-25	09/13/79	09/27/79
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	I	All	All	F-26	070/2/79	09/27/79
II.F.3	Instruments for Monitoring Accident Conditions	NOTE 3(a)	All	All		NA	12/--/80
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	All		09/13/79	09/27/79
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	NOTE 3(a)	All	All		07/31/91	07/31/91
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins	-	-	-	-	-	-
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	All		03/31/80	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All		03/31/80	03/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All		03/31/80	03/31/80
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	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	NOTE 3(a)	All	All			NA
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	NOTE 3(a)	All	All		01/01/81	01/01/81
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, <u>W</u>		03/31/80	NA
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	NOTE 3(a)	NA	CE, <u>W</u>		NA	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	NOTE 3(a)	NA	CE, <u>W</u>		NA	
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	NOTE 3(a)	NA	<u>W</u>			
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	NOTE 3(a)	NA	B&W		NA	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	NOTE 3(a)	NA	B&W		03/31/80	NA
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	NOTE 3(a)	NA	B&W		03/31/80	03/31/80

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	NOTE 3(a)	NA	B&W		03/31/80	03/31/80
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	NOTE 3(a)	All	NA		03/31/80	03/31/80
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA		03/31/80	03/31/80
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	NOTE 3(a)	NA	All		NA	
II.K.1(25)	Develop Operator Action Guidelines	NOTE 3(a)	NA	All		NA	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	NOTE 3(a)	NA	All		NA	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	NOTE 3(a)	NA	All		NA	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	NOTE 3(a)	NA	All		01/01/81	01/01/82
II.K.2	Commission Orders on B&W Plants	-	-	-	-	-	-
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	NOTE 3(a)	NA	B&W		NA	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	NOTE 3(a)	NA	B&W		NA	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	NOTE 3(a)	NA	B&W		NA	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	NOTE 3(a)	NA	B&W		NA	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	NOTE 3(a)	NA	B&W		NA	
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	NOTE 3(a)	NA	B&W		NA	
II.K.2(7)	Reevaluate Transient of September 24, 1977	NOTE 3(a)	NA	B&W		NA	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	I	NA	B&W	F-27	01/01/81	01/01/81
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	I	NA	B&W	F-28	01/01/81	01/01/81
II.K.2(11)	Operator Training and Drilling	I	NA	B&W	F-29	01/01/81	01/01/81
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	I	NA	B&W	F-30	01/01/81	01/01/81
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	I	NA	B&W	F-31	01/01/81	01/01/81
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	I	NA	B&W		06/01/80	06/01/80
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	I	NA	B&W	F-32	06/01/80	06/01/80

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	I	NA	B&W	F-33	NA	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	I	NA	B&W	F-34	01/01/81	NA
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	I	NA	B&W	F-35	01/01/81	NA
II.K.2(21)	LOFT L3-1 Predictions	NOTE 3(a)	NA	B&W		NA	
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-	-	-	-
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	I	NA	All	F-36	07/01/81	07/01/81
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	I	NA	All	F-37	01/01/81	01/01/81
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	I	All	All	F-38	04/01/80	04/01/80
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	I	NA	All	F-39, G-01	01/01/81	01/01/81
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	I	NA	B&W		01/01/81	01/01/81
II.K.3(9)	Proportional Integral Derivative Controller Modification	I	NA	W	F-40	07/01/80	07/01/80
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	I	NA	W	F-41		
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	I	All	All			
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	I	NA	W	F-42	07/01/80	07/01/80
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	I	GE	NA	F-43	10/01/80	10/01/80
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	I	GE	NA	F-44	01/01/81	NA
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	I	GE	NA	F-45	01/01/81	01/01/81
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	I	GE	NA	F-46	01/01/81	01/01/81
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	I	GE	NA	F-47	01/01/81	01/01/81
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	I	GE	NA	F-48	01/01/81	01/01/81
II.K.3(19)	Interlock on Recirculation Pump Loops	I	GE	NA	F-49	01/01/81	NA
II.K.3(20)	Loss of Service Water for Big Rock Point	I	GE	NA		01/01/81	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	F-50	01/01/81	01/01/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	F-51	01/01/81	01/01/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	01/01/82	01/01/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	01/01/82	01/01/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/01/80	10/01/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	01/01/82	01/01/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	04/01/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	01/01/83	01/01/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	01/01/83	01/01/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	01/01/81	01/01/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	01/01/81	01/01/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	07/01/80	07/01/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/01/80	NA
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-	-	-	-
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	I	All	All		10/10/79	08/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	-	-	-
III.A.1.2(1)	Technical Support Center	I	All	All	F-63	09/13/79	09/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	All	All	F-64	09/13/79	09/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	All	All	F-65	09/13/79	09/27/79
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-	-	-	-
III.A.2.1(1)	Publish Proposed Amendments to the Rules	NOTE 3(a)	All	All			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
III.A.2.2	Development of Guidance and Criteria	I	All	All	F-68		
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.3	Communications	-	-	-	-	-	-
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	NOTE 3(a)	All	All			
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	NOTE 3(a)	All	All			
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-	-	-	-
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	I	All	All		07/02/79	09/27/79
III.D.3	Worker Radiation Protection Improvement						
III.D.3.3	Inplant Radiation Monitoring	-	-	-	-	-	-
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	All	All	F-69	09/13/79	09/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(4)	Issue a Regulatory Guide	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.4	Control Room Habitability	I	All	All	F-70	05/07/80	06/26/80

TASK ACTION PLAN ITEMS

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
A-1	Water Hammer (former USI)	NOTE 3(a)	All	All		NA	03/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	NOTE 3(a)	NA	All	D-10	01/--/81	01/--/81
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	W		04/17/85	04/17/85
A-4	CE Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	CE		04/17/85	04/17/85
A-5	B&W Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	B&W		04/17/85	04/17/85
A-6	Mark I Short-Term Program (former USI)	NOTE 3(a)	GE	NA		12/--/77	NA
A-7	Mark I Long-Term Program (former USI)	NOTE 3(a)	GE	NA	D-01	08/--/82	08/--/82
A-8	Mark II Containment Pool Dyanmic Loads - Long Term Program (former USI)	NOTE 3(a)	GE	NA		08/--/81	08/--/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All		06/26/84	06/26/84

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
			BWR	PWR			
A-10	BWR Feedwater Nozzle Cracking (former USI)	NOTE 3(a)	All	NA	B-25	11/--/80	11/--/80
A-11	Reactor Vessel Materials Toughness (former USI)	NOTE 3(a)	All	All		10/--/82	NA
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	NOTE 3(a)	NA	All		NA	TBD
A-13	Snubber Operability Assurance	NOTE 3(a)	All	All	B-17, B-22	1980	1980
A-16	Steam Effects on BWR Core Spray Distribution	NOTE 3(a)	GE	NA	D-12	NA	
A-24	Qualification of Class 1E Safety Related Equipment (former USI)	NOTE 3(a)	All	All	B-60	08/--/81	08/--/81
A-25	Non-Safety Loads on Class 1E Power Sources	NOTE 3(a)	All	All		09/--/78	
A-26	Reactor Vessel Pressure Transient Protection (former USI)	NOTE 3(a)	NA	All	B-04	09/--/78	09/--/78
A-28	Increase in Spent Fuel Pool Storage Capacity	NOTE 3(a)	All	All		04/17/78	NA
A-31	RHR Shutdown Requirements (former USI)	NOTE 3(a)	All	All		05/--/78	10/01/78
A-35	Adequacy of Offsite Power Systems	NOTE 3(a)	All	All	B-23	06/02/77	1980
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	NOTE 3(a)	All	All	C-10, C-15	07/--/80	07/--/80
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	NOTE 3(a)	GE	NA		02/29/80	09/30/80
A-40	Seismic Design Criteria (former USI)	NOTE 3(a)	All	All		TBD	09/--/89
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	NOTE 3(a)	All	NA	B-05	02/--/81	02/--/81
A-43	Containment Emergency Sump Performance (former USI)	NOTE 3(a)	NA	All		NA	11/--/85
A-44	Station Blackout (former USI)	NOTE 3(a)	All	All		TBD	06/--/88
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	NOTE 3(a)	All	All		02/--/87	NA
A-47	Safety Implications of Control Systems (former USI)	NOTE 3(a)	All	All		09/20/89	09/20/89
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	NOTE 3(a)	All	W		12/--/81	12/--/81
A-49	Pressurized Thermal Shock (former USI)	NOTE 3(a)	NA	All	A-21	TBD	07/--/85
B-10	Behavior of BWR Mark III Containments	NOTE 3(a)	GE	NA		NA	09/--/84
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	NOTE 3(a)	All	All		03/--/78	
B-56	Diesel Reliability	NOTE 3(a)	All	All	D-19	06/--/93	06/--/93
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	NOTE 3(a)	All	All	B-45	04/20/81	
B-64	Decommissioning of Reactors	NOTE 3(a)	All	All		06/27/88	NA
B-66	Control Room Infiltration Measurements	NOTE 3(a)	All	All		NA	07/--/81
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	NOTE 3(a)	All	All		05/27/80	05/27/80

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
C-10	Effective Operation of Containment Sprays in a LOCA	NOTE 3(a)	All	All		NA	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	NOTE 3(a)	All	All		12/27/82	12/27/82
		NEW GENERIC ISSUES					
25.	Automatic Air Header Dump on BWR Scram System	NOTE 3(a)	All	NA		01/09/81	01/09/81
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	NOTE 3(a)	All	NA	B-65	08/31/81	08/31/81
41.	BWR Scram Discharge Volume Systems	NOTE 3(a)	All	NA	B-58	12/09/80	NA
43.	Reliability of Air Systems	NOTE 3(a)	All	All	B-107	08/08/88	08/08/88
45	Inoperability of Instrumentation Due to Extreme Cold Weather	NOTE 3(a)	All	All		NA	09/01/83
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	NOTE 3(a)	All	All	L-913	07/18/89	07/18/89
67.	Steam Generator Staff Actions	-	-	-	-	-	-
67.3.3	Improved Accident Monitoring	NOTE 3(a)	All	All	A-17	12/17/82	12/17/82
70.	PORV and Block Valve Reliability	NOTE 3(a)	NA	All		06/25/90	06/25/90
73.	Detached Thermal Sleeves	NOTE 3(a)	NA	W		NA	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	NOTE 3(a)	All	All	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85, B-86, B-87, B-88, B-89, B-90, B-91, B-92, B-93	07/08/83	TBD
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	NOTE 3(a)	All	NA	B-84	TBD	TBD
87.	Failure of HPCI Steam Line Without Isolation	NOTE 3(a)	All	All		06/28/89	06/28/89
89.	Stiff Pipe Clamps	NOTE 6	All	All	NA	NA	TBD
93.	Steam Binding of Auxiliary Feedwater Pumps	NOTE 3(a)	NA	All	B-98	10/--/85	10/--/85
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	NOTE 3(a)	NA	CE, W		06/25/90	06/25/90
99.	RCS/RHR Suction Line Valve Interlock on PWRs	NOTE 3(a)	NA	All	L-817	10/17/88	10/17/88
103.	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All		10/19/89	10/19/89
118.	Tendon Anchorage Failure	NOTE 3(a)	All	All	NA	NA	07/--/90
124.	Auxiliary Feedwater System Reliability	NOTE 3(a)	All	All		TBD	TBD
128.	Electrical Power Reliability	NOTE 3(a)	All	All		04/29/91	04/29/91
130.	Essential Service Water Pump Failures at Multiplant	NOTE 3(a)	NA	All		09/19/91	09/19/91

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
	Sites						
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-	-	-	-
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/--/95
156.	Systematic Evaluation Program	-	-	-	-	-	-
156.6.1	Pipe Break Effects on Systems and Components	HIGH	All	All		TBD	TBD
163.	Multiple Steam Generator Tube Leakage	HIGH	NA	All		TBD	TBD
168.	Environmental Qualification of Electrical Equipment	HIGH	All	All		TBD	TBD
170.	Fuel Damage Criteria for High Burnup Fuel	HIGH	All	All		TBD	TBD
172.	Multiple System Responses Program	HIGH	All	All		TBD	TBD
173.	Spent Fuel Storage Pool	-	-	-	-	-	-
173.A	Operating Facilities	HIGH	All	All		TBD	TBD
177.	Vehicle Intrusion at TMI	NOTE 3(a)	All	All		08/01/94	08/01/94
191.	Assessment of Debris Accumulation on PWR Sump Performance	HIGH	NA	All		TBD	TBD
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84

APPENDIX F

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS GSIs

This appendix documents those non-reactor GSIs identified, prioritized, and resolved by NMSS. As stated in SECY-98-001,[1724] the prioritization procedure for these issues is contained in NMSS Policy and Procedures Letter 1-57,[1725] "NMSS Generic Issues Program."

TABLE F.1

LISTING OF NMSS GSIs

This table contains the priority designations for all NMSS GSIs listed in Appendix F.

Legend

NOTES: 3(a) - Resolution Resulted in the Establishment of New Requirements
3(b) - Resolution Resulted in the Establishment of No New Requirements
4 - Issue to be Prioritized in the Future
HIGH - High Safety Priority
MEDIUM - Medium Safety Priority
LOW - Low Safety Priority

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0001	Door Interlock Failure Resulting from Faulty MicroSelectron-High Dose Rate Remote Afterloader	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0002	Significant Quantities of Fixed Contamination Remain in Krypton-85 Leak-Detection Devices After Venting	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0003	Corrosion of Sealed Sources Caused by Sensitization of Stainless Steel Source Capsules During Shipment	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0004	Overexposures Caused by Sources Stolen from Facility of Bankrupt Licensee	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0005	Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements With Victoreen Electrometers	Ramsey	NMSS/IMNS/IMOB	NOTE 3(a)		12/31/1998
NMSS-0006	Criticality in Low-Level Waste	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)	1	06/30/2000
NMSS-0007	Criticality Benchmarks Greater Than 5% Enrichment	Ramsey	NMSS/FCSS	LOW		12/31/1998
NMSS-0008	Year 2000 Computer Problem - Non-Reactor Licensees	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0009	Amersham Radiography Source Cable Failures	Ramsey	NMSS/IMNS	NOTE 3(b)		12/31/1998

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0010	Troxler Gauge Source Rod Weld Failures	Ramsey	NMSS/IMNS	MEDIUM		12/31/1998
NMSS-0011	Spent Fuel Dry Cask Weld Cracks	Ramsey	NMSS/SFPO	NOTE 3(b)		12/31/1998
NMSS-0012	Inadequate Transportation Packaging Puncture Tests	Ramsey	NMSS/SFPO	NOTE 3(b)	1	06/30/2000
NMSS-0013	Use of Different Dose Equivalent Models to Show Compliance	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0014	Surety Estimates for Groundwater Restoration at In-Situ Leach Fields	Ramsey	NMSS/DWM	MEDIUM		12/31/1998
NMSS-0015	Adequacy of 10 CFR 150 Criticality Requirements	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2000
NMSS-0016	Adequacy of 0.05 Weight Percent Limit in 10 CFR 40	Ramsey	NMSS/IMNS	MEDIUM		12/31/98

NMSS-0006: CRITICALITY IN LOW-LEVEL WASTE

DESCRIPTION

This issue was identified[1708] by NMSS and addresses the potential for special nuclear material containing unusual moderators to re-concentrate and form a critical mass in low-level waste disposal systems. The results of studies of two low-level waste sites indicated that criticality is not likely when the moderators silicon dioxide and water are present. However, the results of studies of containerized waste indicated that the presence of the moderators carbon and beryllium in excess of 5 times the mass of U^{235} served to reduce the areal density of fissile material required for criticality.

CONCLUSION

The issue was given a medium priority ranking and resolution was pursued.[1708] In October 1998, SECY-98-239 was issued with a recommendation to cease research on criticality concerns with unusual moderators, such as beryllium and graphite. The staff concluded that significant sources of unusual moderators were not present at low-level waste (LLW) facilities. Even if significant quantities of unusual moderators were present in LLW, the staff believed that it was unlikely that they would be commingled with special nuclear material to create a critical array. In December 1998, the Commission issued an SRM approving the staff's recommendation to cease further research. The staff was directed by the Commission to continue development of guidance on emplacement criticality controls at LLW facilities. Thus, the issue was resolved.[1771]

REFERENCES

1708. Memorandum to L. Shao from D. Cool, "Submittal of Generic Safety Issue," August 5, 1997.

1771. Memorandum to C. Rossi from D. Cool, "Status of NMSS Issues in the Generic Issue Management and Control System," June 25, 1999.

NMSS-0008: YEAR 2000 COMPUTER PROBLEM - NON-REACTOR LICENSEES

DESCRIPTION

The Year 2000 computer problem had the potential to pose a threat to public health, site safety and safeguards, and worker safety. Many computer systems could have potentially failed to recognize the change to a new century. The staff believed that these systems could have misread the year 2000 thereby causing them to fail or generate faulty data. To alert all licensees to the problem, IN 96-70[1710] was issued followed by IN 97-61,[1711] which specifically addressed medical licensees, veterinarians, and manufacturers/distributors of medical devices. This issue was identified[1709] by NMSS to ensure that actions were taken to ensure that the Year 2000 computer problem would be either eliminated or minimized for material licensees.

CONCLUSION

The issue was given a high priority ranking and resolution was pursued.[1709] Health and safety were not compromised at materials licensee facilities due to the transition into the year 2000, or by the Leap Day (February 29, 2000). A Lessons Learned report and the GL 98-03 closeout report were issued in March 2000. Thus, the issue was resolved.[546]

REFERENCES

546. Memorandum to C. Rossi from D. Cool, "NMSS Input for Second Quarter FY-2000 Update of the Generic Issues Management Control System," April 18, 2000.

1709. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," June 4, 1998.

1710. NRC Information Notice 96-70, "Year 2000 Effect on Computer System Software," December 24, 1996.

1711. NRC Information Notice 97-61, "U.S. Department of Health and Human Services Letter to Medical Device Manufacturers on the Year 2000 Problem," August 6, 1997.

NMSS-0012: INADEQUATE TRANSPORTATION PACKAGING PUNCTURE TESTS

DESCRIPTION

This issue was identified when two holders of Certificates of Compliance for shipping packages performed puncture tests using a bar that was not properly mounted, as specified in 10 CFR 71.73(c)(3). As a result, NRC Bulletin 97-02[1714] was issued to to holders of Certificates of Compliance for shipping packages under 10 CFR 71 with the request to review puncture test assessments for each certified package design. For designs based on physical tests, certificate holders were requested to determine whether the puncture tests were performed in accordance with 10 CFR 71.73(c)(3).

The responses to the Bulletin[1714] identified packages that had not been puncture-tested in full accordance with the regulations. In addition, there were some packages for which the certificate holders were unable to determine whether the tests were performed correctly. In these cases, the certificate holders submitted justification for continued use of their packages.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued.[1709] In response to Bulletin 97-02,[1714] some certificate holders retested their packages to demonstrate adequacy. It was determined that, even though previous tests had not been performed exactly as specified in the regulations, the differences had no significant effect on the test results. Based on these findings and the excellent safety record for these packages, the staff concluded that no further action was needed with respect to previously-approved packages. These packages were authorized for use under certain restrictions pursuant to 10 CFR 71.13. Thus, the issue was resolved.[1771]

REFERENCES

1709. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," June 4, 1998.

1714. NRC Bulletin 97-02, "Puncture Testing of Shipping Packages Under 10 CFR Part 71," September 23, 1997.

1771. Memorandum to C. Rossi from D. Cool, "Status of NMSS Issues in the Generic Issue Management and Control System," June 25, 1999.

NMSS-0013: USE OF DIFFERENT DOSE EQUIVALENT MODELS TO SHOW COMPLIANCE

DESCRIPTION

This issue was identified[1723] by NMSS to allow licensees to establish ALIs and DACs for use at their facilities based on new models to show compliance. NMSS believed that 10 CFR 20 needed to be revised so that Appendix B and all references to variables associated with converting dose to dose equivalent, such as weighting factors and quality factors, are removed thus allowing dose equivalent limits to remain. The revision to 10 CFR 20 was expected to eliminate all aspects of regulations that cause elusive use of particular mathematical models, i.e., ICRP 26 and 30. Simultaneously, a Regulatory Guide would be developed to contain NRC-acceptable models and parameters, including the ALIs and DACs and concentrations removed from Appendix B of 10 CFR 20, that could be used by the majority of licensees to show compliance.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued. The staff considered initiating a major rulemaking to remove all dose modeling parameters from 10 CFR 20 and issue them as guidance. However, only a few licensees requested permission to use newer models for dose calculations, and initiating such a significant rulemaking would have been a resource-intensive effort. In addition, there were great uncertainties in the doses calculated by the different models. The level of uncertainty in newer models was no better than older models for many isotopes. The primary benefit of the rulemaking would have been to minimize requests for exemptions from 10 CFR 20. It was believed that rulemaking would have established a means of authorizing the use of new dose models for those relatively few licensees that requested the use of new models. Instead of pursuing rulemaking, the staff decided to address licensee requests on a case-by-case basis.[1772]

REFERENCE

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

1772. Memorandum to C. Rossi from D. Cool, "Closure of NMSS Generic Issue," May 18, 1999.

NMSS-0015: ADEQUACY OF 10 CFR 150 CRITICALITY REQUIREMENTS

DESCRIPTION

As of July 1998, SNM-bearing low-level waste was being disposed of at the following three facilities: (1) Envirocare in Clive, Utah; (2) U.S. Ecology in Hanford, Washington; and (3) Chem-Nuclear in Barnwell, South Carolina. Until 1997, the Hanford and Barnwell facilities were licensed by NRC under 10 CFR 70 to possess and dispose of greater than critical mass quantities of SNM. These facilities requested that the SNM possession limits be reduced to the 10 CFR 150.11 limits and that the NRC licenses be transferred to the respective Agreement States. The licenses were amended as requested and transferred to the respective Agreement States. Thus, all three facilities are currently regulated by Agreement States.

Emplacement criticality safety was addressed in 10 CFR 61.16(b)2 which requires licensees to describe proposed procedures for avoiding accidental criticality for both storage and emplacement. However, at the time this issue was identified, this portion of 10 CFR 61 was not a compatibility requirement for Agreement States. In conjunction with considering changing the compatibility requirement of 10 CFR 61.16(b), NRC planned to develop guidance on emplacement criticality which could be used at existing and future low-level disposal sites. This issue was identified[1723] by NMSS to develop guidance on emplacement criticality that could be used at existing and future low-level waste disposal sites by licensees, Agreement States, and the NRC.

CONCLUSION

The issue was given a medium priority ranking and resolution was pursued. SECY-99-272 was issued with recommendations to revise the Agreement State compatibility requirements in 10 CFR 61 and issue new emplacement criticality guidance. In an SRM dated January 5, 2000, the Commission disapproved the staff's proposal and directed the staff to cease work on the issue. The Commission concluded that the theoretical scenarios developed by the staff were unrealistic and failed to justify further efforts. Thus, the issue was resolved[546] with the Commission's decision.

REFERENCES

546. Memorandum to C. Rossi from D. Cool, "NMSS Input for Second Quarter FY-2000 Update of the Generic Issues Management Control System," April 18, 2000.

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

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11. ABSTRACT *(200 words or less)*

The report presents the safety priority ranking for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, and DROP, and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolution of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

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