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SUPPLEMENT 16 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"
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INTRODUCTION

I. BACKGROUND

History

On October 8, 1976, the Commission directed the staff to develop "a program plan for resolution of generic issues and completion of technical projects." The Commission further requested that "this plan should include: task schedules ... task priority and manpower requirements (with proportions of staff contract efforts explicitly identified)." On December 12, 1977, the Energy Reorganization Act of 1974 was amended by Congress through Public Law 95-209 to include, among other things, a new Section 210 as follows:

> UNRESOLVED SAFETY ISSUES PLAN
>
> Sec. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

In order to meet both Commission and Congressional directives, the staff developed a generic issues program that provided for the identification of generic issues, the assignment of priorities, the development of detailed action plans, projections of dollar and manpower costs, continuous high level management oversight of progress, and public dissemination of information related to the issues as they progressed. This program was published in NUREG-0410[387] in January 1978 and, shortly thereafter, the Commission issued a Policy Statement[1190] on the NRC "Program for Resolution of Generic Issues Related to Nuclear Power Plants."

The NRC generic issues program published in NUREG-0410[387] was considerably broader than the "Unresolved Safety Issues Plan" required by Section 210. It included plans for the resolution of generic environmental issues, for the development of improvements in the reactor licensing process, and for consideration of less conservative design criteria or operating limitations in areas where existing requirements might be unnecessarily restrictive or costly.

The first attempts by the staff to implement the generic issues program stated in NUREG-0410[387] were based largely on engineering judgments. This qualitative effort to rank unresolved generic issues continued through two phases:

(1) In 1977, all issues were classified into four categories according to importance, from "significant" to "little or no importance."

(2) In the early part of 1978, the issues were reclassified into Groups 1 through 8 by type rather than by order of importance.

Later in 1978, the staff began to take a quantitative approach by using risk assessment to place the issues into four categories ranging from I (potential high risk items) to IV (items not directly related to risk). With increased confidence in this risk assessment approach, the staff introduced a more comprehensive quantitative system in early 1979. Points were assigned to each issue based on an assessment of safety significance, environmental significance, licensing effectiveness, deadline pressure, and retrofit versus forward-fit. Although the point system was still quite subjective, it was nevertheless a major improvement over the previous methods used.

In the aftermath of the Three Mile Island Unit 2 (TMI-2) accident, many new generic issues were raised and the staff came to the conclusion that the point system was too subjective to be used for ranking the issues. One of the TMI Action Plan[48] items, IV.E.2, called for the staff to develop a plan for the early resolution of safety issues. It was in resolving this issue that the staff developed a quantitative "prioritization" methodology whereby a numerical priority score could be assigned to each generic safety issue. With this approach, priorities were to be based on an evaluation of the estimated risk reduction associated with the potential change in requirements that could result from resolution of an issue and the estimated costs to the NRC and the industry in implementing such a change. This methodology was submitted to the Commission for information in SECY-81-513.[2] In April 1983, this approach was refined and resubmitted to the Commission for approval in SECY-83-221.[1188] After Commission review, approval to use the methodology was given in November 1983.[1189]

In April 1993, after approximately ten years of experience with the methodology, adjustments were made in the numerical thresholds, while retaining the basic features of the method. These adjustments involved raising risk thresholds and simplifying the way in which costs entered the priority rankings. What motivated the raising of risk thresholds was the observation[1479] that, of the issues resolved, only 3 of the 27 MEDIUM-priority and about half of the HIGH-priority issues resulted in decisions to take regulatory action, i.e., in retrospect, it appeared that resources had been devoted to resolving a large number of issues with no resulting safety improvement. This outcome must be interpreted with the qualification that generic issue resolution efforts that have not led to regulatory action have, nevertheless, in many instances, produced safety benefits through licensee actions taken voluntarily, in consideration of the issues raised, or in response to interim guidance. However, the extent of these benefits, when they occurred, was generally in proportion to the priority rank and MEDIUM-priority issues usually resulted in marginal improvements. The proposed revisions were submitted to the Commission in SECY-93-108[1479]; in July 1993, Commission approval was obtained.[1505]

The threshold adjustments were intended to cause the prioritization process to model the resolution process without the earlier, apparently excessive margin for initial uncertainties, to reduce resolution efforts that do not produce safety improvements, while still ensuring attention to issues that require it. The raising of the numerical safety thresholds is accompanied by strengthened attention to uncertainties and special considerations, to help recognize instances when a priority rank higher than the indication from the new numerical formula is warranted, the objective being to improve the efficiency of the prioritizations without impairing their prudence.

The priority ranking chart and risk thresholds used in prioritization analyses completed before August 1, 1993 are shown in Appendix C.

The simplification of the way in which costs enter reflects the confirmation from experience that risk significance is indeed the primary factor in priority ranking, with a more bounded role for safety-cost trade-offs.

Operating Plan

The initial work in prioritizing issues was essentially done by various Staff Working Groups. Following a reorganization of the Office of Nuclear Reactor Regulation (NRR) in April 1980, the lead responsibility for prioritization was assigned to the Safety Program Evaluation Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation (SPEB/DST/NRR).

The 1983 NRC Policy and Planning Guidance (NUREG-0885, Issue 2),[210] in addressing the area of Coordinating Regulatory Requirements (Planning Guidance, Item 5, p.6) called for "...a priority list of generic safety issues including TMI-related issues based on the potential safety significance and cost of implementation of each issue..." to be submitted to the Commission for approval. Using the prioritization methodology outlined below, this list was developed by SPEB in response to the Planning Guidance and forwarded to the Commission in SECY-83-221.[1186]

After another NRR reorganization in November 1985, this task was assigned to the Safety Program Evaluation Branch, Division of Safety Review and Oversight (SPEB/DSRO/NRR). Following an NRC reorganization in April 1987, the responsibility for preparing and maintaining the list of generic safety issues and their priority was assigned to the Advanced Reactors and Generic Issues Branch, Division of Regulatory Applications, Office of Nuclear Regulatory Research (ARGIB/DRA/RES). Currently, this responsibility is with the Division of Safety Issue Resolution, Office of Nuclear Regulatory Research (DSIR/RES).

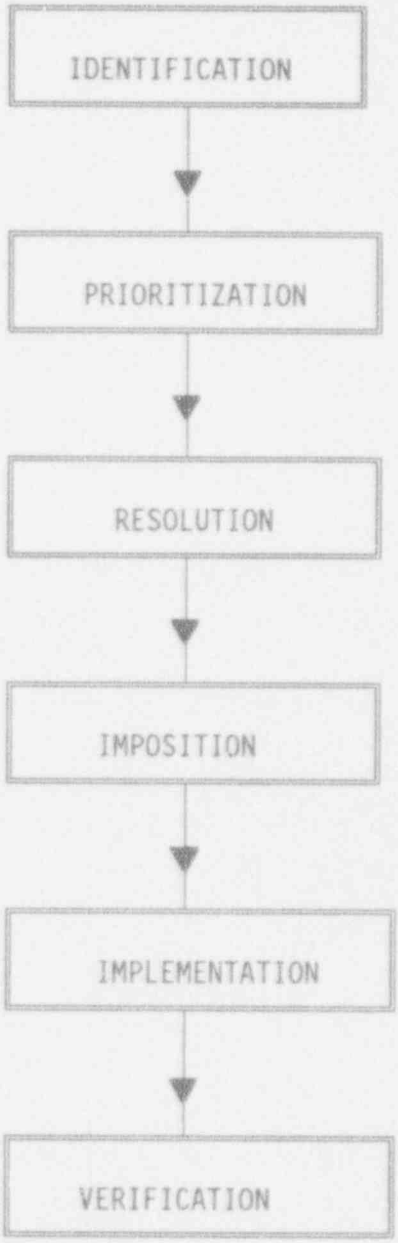
The prioritization of generic issues is an ongoing staff function that has been reflected annually in the NRC Policy and Planning Guidance.[210] This document was superseded in 1987 by the NRC Five-Year Plan.

II. GENERIC ISSUES PROGRAM

After issuance of the Policy Statement[1190] in 1978, the NRC program to resolve generic issues underwent many reviews and changes. As a result, the Commission concluded in April 1989 that the 1978 Policy Statement no

Exhibit A

GENERIC ISSUES PROGRAM



longer reflected the NRC's generic issues program and withdrew it from the public record.[1191] The current generic issues program consists of six separate and distinct steps: identification, prioritization, resolution, imposition, implementation, and verification (See Exhibit A). An explanation of each of these six steps is given below.

Identification

Generic concerns may be identified by individuals or organizations within the NRC staff or by the Advisory Committee on Reactor Safeguards (ACRS), the nuclear power industry, or the public. RES Office Letter No. 1 (OL #1)[1192] provides a procedure and suggested content for individuals or organizational units within the NRC to request consideration of a concern as a new generic issue. This procedure may also be used by parties outside the NRC to express their concerns to the staff for consideration as potential generic issues. Sources of potential generic issues are many and varied and include, but are not limited to, the following: evaluation of safety-related research, risk assessment analyses, and public and industry concerns.

Prioritization

This report focuses on the prioritization step of the generic issues program which is explained in detail in Paragraph III below.

Resolution

After an issue has been prioritized and approved for resolution, the first task is the development of a plan to delineate the work to be done, assignment of major responsibilities, identification of project resource needs, and scheduling of milestone dates. These activities vary in scope and depth in accordance with issue priority and the depth of information on a given issue. The second task involves development of a technical solution. Typically, the information used to resolve an issue comes from experience data, experiments, tests, analyses, and probabilistic risk assessments (PRAs). The results of such work or the technical findings may be published in contractor and staff NUREG reports which are made Exhibit A available through the NRC Public Document Room (PDR), Washington, D.C., or the National Technical Information Service, Department of Commerce, Springfield, Virginia.

In the final stage of resolution, the technical findings are used as a basis to develop a proposed resolution for the issue involving a change to NRC requirements or guidance. Several alternatives may be considered. A regulatory analysis, including a detailed cost/benefit analysis of each practical alternative, and consideration of the best methods of imposition, implementation, and verification are used in selecting a proposed resolution. If a backfit is proposed, first, a determination is made as to whether the backfit is required to provide adequate protection to the health and safety of the public or simply provides for enhancement of public health and safety. If it is determined that the backfit is necessary to provide an adequate level of protection, the backfit will be imposed regardless of the costs to achieve it. If it is determined that the backfit provides for enhancement of public health and safety, a generic analysis is required that treats the nine factors specified in 10 CFR 50.109(c). Once the cognizant NRC Office Directors have agreed to a proposed resolution, it is then forwarded to the Committee for the Review of Generic Requirements (CRGR), the ACRS, the Executive Director for Operations (EDO), and the Commission for review and approval as appropriate. Changes to regulations, Policies, the Standard Review Plan (SRP), and Regulatory Guides are published in the Federal Register for public comment. Comments received are then incorporated, as appropriate, with the final product published in the Federal Register. Resolution of a generic issue can take from several months to a few years depending on the length of time required by the deliberations involved at each of the above steps.

RES Office Letter No. 3[1194] describes the procedure to be followed in the resolution of a generic issue, denotes the required elements of the resolution plan and resolution package, and identifies review procedures and organizational responsibilities for the approval of the resolution of a generic issue. Guidance for the preparation, review, and required content of the regulatory analysis portion of the resolution package is provided in RES Office Letter No. 2.[1193] Milestone information and reporting requirements as well as organizational responsibilities for the tracking of generic issue resolution are provided in OL #1.[1192] All issues scheduled for resolution are tracked through the resolution process by the Generic Issue Management Control System (GIMCS) which is updated quarterly and placed in the PDR.

Imposition

Imposition is the step in the generic issues program where each affected licensee and/or applicant is required or guided to prepare a schedule for implementing the generic issue resolution consistent with a Rule, Policy, Regulatory Guide, generic letter, bulletin, and/or licensing guidance developed during the resolution stage. Normally, NRC requirements, policies, and/or guidance will not provide for NRC consideration of a licensee's modifications prior to their implementation at an affected plant. This facilitates completion of plant modifications to enhance safety within two refueling outages, not to exceed three years after issuance of NRC requirements, policies, and/or guidance. However, in a few exceptional cases, licensees may be expected to submit (normally for NRC approval) their plans (including schedules) for plant modifications prior to their implementation. In all cases, licensees will be expected to certify in writing to the NRC that plant modifications have been completed.

For the exceptional cases, the staff reviews each applicant's and/or licensee's submittal with regard to proposed modifications to site, equipment, structures, procedures, technical specifications, operating instructions, etc. and schedules proposed for the accomplishment of the modifications. For backfits, imposition is complete when each affected licensee has committed to compliance actions and schedules for implementing these actions. For forward-fits, the imposition of a generic issue resolution is complete when the new requirement or guidance becomes effective as an integral part of NRC regulations, policies, and/or guidance.

During this stage, a resolved GSI is identified as a Multiplant Action (MPA) for licensee action. The imposition status of all MPAs is tracked in the Safety Issue Management System (SIMS).

Implementation

Implementation is the step in the generic issues program where the affected licensees perform the actions on existing plants to satisfy the commitments made during the imposition stage. These may include modifications/additions to equipment, structures, procedures, technical specifications, operating instructions, etc. No later than 30 days after each affected licensee has completed all of the actions required for a particular generic issue resolution, and the modified/additional system is fully operational, the licensee is required to certify in writing to the NRC that plant modifications have been completed in accordance with NRC requirements, policies, and/or guidance. When all affected licensees have officially notified the NRC of completion of all required/committed actions, the implementation stage is complete, unless it is determined by the staff from subsequent verification inspection that additional licensee actions are needed for compliance.

Verification

The verification step consists of three parts. First, the portions of a licensee's actions, if any, that warrant NRC inspection must be determined. This decision is made during the resolution stage based on the judgment of the safety significance of the issue relative to other matters in the inspection program, licensee performance, and the resources needed to accomplish a meaningful inspection. Next, as necessary, inspection instructions are prepared to ensure that the NRC inspection is performed in a consistent and appropriate manner at all affected plants; the inspection, by its very nature, is an audit. Therefore, carefully thought-out instructions must be provided to the NRC inspectors so that the maximum safety benefit is achieved for the limited resources devoted to this effort. The third part of the verification process is the actual verification and documentation of the results in an inspection report. Physical inspections are performed on an audit basis in a manner consistent with general inspection procedures which involve a sampling of changes made by licensees or applicants, as opposed to a 100% inspection of all actions. Verification of licensee implementation of generic issue resolution is reported by the staff in SIMS.

III. PRIORITIZATION

Purpose and Scope

The primary purpose of prioritization is to assist in the timely and efficient allocation of resources to those safety issues that have a high potential for reducing risk and in decisions to remove from further consideration issues that have little safety significance and hold little promise of worthwhile safety enhancement. However, issues of such gravity that consideration of immediate action is called for are excluded from prioritization because of the compressed time scale in which decisions for such issues must be made. Generally, immediate action takes the form of a

Bulletin or Order. Both operating and future plants are considered in the priority ranking process.

Prioritization focuses on generic <u>safety</u> issues (GSIs) i.e., safety concerns that may affect the design, construction, or operation of all, several, or a class of nuclear power plants and may have the potential for safety improvements and promulgation of new or revised requirements or guidance. However, the method can be used to identify changes in current requirements that could significantly reduce the impact (usually cost) on licensees without any substantial change in public risk. Issues of this type are classified as <u>Regulatory Impact</u> issues (RI) to clearly differentiate them as not improving the safety of nuclear power plants but, nevertheless, possibly worthwhile.

In order to identify GSIs, all issues originated in accordance with OL #1[1192] are reviewed to determine their safety significance. Issues that primarily concern environmental protection or the licensing process and do not involve significant safety improvement elements are classified accordingly and noted for separate consideration outside the GSI priority ranking scheme. These issues are classified as either environmental issues or licensing issues. <u>Environmental</u> issues (EI) involve impacts on the human environment and the values sought to be protected by the National Environmental Policy Act (NEPA). <u>Licensing</u> issues (LI) are not directly related to protecting public health and safety or the environment, but relate to: (1) increasing the staff's knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety; (2) improving or maintaining the NRC capability to make independent assessments of safety; (3) establishing, revising, and carrying out programs to identify and resolve GSIs; (4) documenting, clarifying, or correcting current requirements and guidance; and (5) improving the effectiveness or efficiency of the review of applications.

The list of issues subjected to prioritization contains the following groups:

(1) TMI Action Plan items identified for development in NUREG-0660[48]; these issues are covered in Section 1. The priority recommendations in this report exclude those issues that were designated for implementation in NUREG-0737.[98]

(2) Task Action Plan items identified in NUREG-0371[2] and NUREG-0471,[3] plus the subsequently added issues A-42 through A-49 that were designated as Unresolved Safety Issues (USIs); these issues are covered in Section 2. However, issues designated as USIs were excluded from prioritization because of the high-priority attention they were given based on priority decisions previously made. In the future, USIs will come from issues that have been prioritized.

(3) New Generic issues identified by the staff, ACRS, or others; these issues are covered in Section 3. All new issues identified will be prioritized and included in Section 3 and published in future supplements to this report.

(4) Human Factors Program Plan (HFPP) items identified for development in NUREG-0985[603]; these items are covered in Section 4.

(5) Chernobyl Issues identified in NUREG-1251[1195]; these issues are covered in Section 5.

A comprehensive listing of all issues in the above five groups is given in Table II which includes the following information for each issue: (1) the NRC person responsible for the prioritization evaluation; (2) the lead NRC office, division, and branch responsible for reviewing the prioritization analysis and/or resolving the issue; (3) the priority ranking or status; (4) the latest version of the evaluation; (5) the issuance date of the latest version of the evaluation; and (6) the MPA number for those issues that have been resolved and require licensee actions. A summary of the number of issues in each category is shown in Table III. A cross-reference listing of reports prepared by the Office for Analysis and Evaluation of Operational Data (AEOD) and their corresponding generic issues is provided in Table IV.

How the Work Is Done

The work is done, in accordance with the criteria described below, by the responsible NRC Branch in consultation with others in the NRC with knowledge of the issues or expertise in the technical disciplines involved. In a number of instances, technical or cost information is obtained from industry and other outside sources. The Battelle Pacific Northwest Laboratories (PNL), under a technical assistance contract, developed detailed methods to quantify safety benefits and costs and provided safety-benefit analyses and cost information for many of the issues. The responsible NRC Branch, with internal consultations as necessary, reviews and applies the PNL-supplied technical factors, in conjunction with additional factors, in developing the priority rankings and recommendations.

Systematic peer review of each prioritization evaluation within the NRC contributes to the assurance that the analysis is complete and accurate and that the judgments are soundly based. This review is done in two stages. First, each analysis is reviewed by the NRC organizational unit or units whose area of responsibility or specialized knowledge is substantially involved. Second, any comments made are then resolved, where practical, and factored into the analysis, as appropriate. Upon completion of peer review, the analysis is then finalized and prepared for approval by the responsible Office Director. Once approved, it is placed in the PDR and published in a future supplement to this report, after which, additional comments from the ACRS, the industry, and the public are considered in any further reassessment of the issue's priority.

Priority Categories: Their Meaning and Proposed Use

Four priority rankings are used: HIGH, MEDIUM, LOW, and DROP. They are intended for use in guiding allocation of NRC resources and scheduling of efforts to resolve the various issues, in conjunction with other pertinent factors such as: (1) the nature, extent, and availability of manpower and material resources estimated to be required; (2) length of time needed to resolve; (3) conflicts in resource allocation and scheduling among items of comparable priority; (4) status of affected reactors; and (5) budget constraints.

A HIGH priority ranking means that strong efforts to achieve the earliest practical resolution are appropriate. This is because: (a) an important safety concern may be involved (though generally the concern is not severe enough to require prompt plant shutdown); or (b) the uncertainty of the safety assessment is unusually large and an upper-bound risk assessment would indicate an important safety concern. All unresolved HIGH priority issues are periodically reviewed in accordance with the criteria stated in NUREG-0705[44] for possible designation as USIs. A USI is defined as a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected.[166] In accordance with Section 210 of the Energy Reorganization Act of 1974, progress on the resolution of USIs is reported to Congress in each NRC Annual Report.

A MEDIUM priority ranking means that no safety concern demanding high-priority attention is involved, but there is believed to be potential for safety improvements or reductions in uncertainty of analysis that may be substantial and worthwhile. Efforts at resolution should be planned, perhaps over the ensuing years, but on a basis of not interfering with pursuit of HIGH-priority generic issues or other high-priority work.

A LOW priority ranking means that no safety concerns demanding at least MEDIUM-priority attention are involved and there is little or no prospect of safety improvements that are both substantial and worthwhile. When the prioritization process results in a LOW priority ranking for an issue, approval of this ranking by the responsible Office Director signifies that the issue has been eliminated from further pursuit. However, in accordance with SRM 871021A,[1493] the staff conducts a periodic review of existing LOW-priority GSIs to determine whether there is any new information that would necessitate reassessment of the original prioritization evaluations.

The DROP category covers proposed issues that are without merit or whose significance is clearly negligible. Issues are also DROPPED from further consideration if it is determined that their safety concerns have been addressed in previously prioritized or resolved issues. When the prioritization process results in a DROP priority ranking for an issue, approval of this ranking by the responsible Office Director signifies that the issue has been eliminated from further pursuit.

An issue is considered resolved, indicated by NOTE 3 in Table II, when its resolution has resulted in either: (a) the establishment of regulatory requirements or guidance (by Rule, SRP[11] change, or equivalent); or (b) a documented authoritative decision that no change in requirements is warranted. Priority rankings are not assigned to issues that have been resolved. However, in those cases where issues were resolved after having been identified for further pursuit by the prioritization process, the related calculations have been retained in the text of this document for future use.

Priority rankings are not assigned to issues that are <u>nearly-resolved</u> (denoted by NOTES 1 and 2 in Table II) because approval of changes to requirements, based on the resolution of an issue, requires that a detailed value/impact evaluation of the safety benefit, implementation costs, and other relevant factors be made. Prioritization would duplicate this value/impact analysis, but in a less comprehensive manner. Therefore, the effort that would be needed to prioritized an issue is devoted to completing the final evaluation of the issue, rather than making a tentative judgment as to the importance and value of the issue. Possible resolution of an issue is considered to be identified, indicated by NOTE 1 in Table II, when a possible technical resolution is under evaluation and the evaluation is nearing completion. Further work may be required as part of the review and approval process before a change in requirements or guidance is issued. Resolution of an issue is considered available, indicated by NOTE 2 in Table II, when proposed or recommended changes to requirements or guidance are documented in a NUREG report, NRC memorandum, Safety Evaluation Report (SER), or equivalent.

Priority rankings are also not assigned to those issues whose safety concerns are determined to be covered (at the time of prioritization) in other issues of broader scope that are being prioritized or are being resolved. Issues in this category are integrated into the issues of broader scope. A detailed listing of all such issues is given in Table V.

<u>Criteria For Assigning Priorities</u>

1. <u>Basic Approach</u>

The method of assigning priority rank involves two primary elements:

(i) the estimated safety importance of the issue; and (ii) the estimated cost of developing and implementing a resolution. Special considerations may influence the proper use of the estimates. These elements are applied as follows:

(a) The issue is identified and defined. Since issues are often complex and interrelated with other issues, careful definition of an issue's scope and bounds is essential in arriving at a sound and applicable assessment.

(b) A quantitative estimate is made of the safety importance of the issue, measured in terms of the risk (the product of accident probabilities and radiological consequences) attributable to the issue and the decrease in that risk that may be attainable by resolving the issue.

(c) A quantitative estimate is made of the cost of resolution.

(d) A numerical impact/value ratio is calculated by dividing the estimated cost entailed by the estimated potential risk reduction. The ratio measures the safety value received in return for the cost impact incurred.

(e) A priority rank (HIGH, MEDIUM, LOW, or DROP) is obtained by application of criteria in which both the safety significance of the issue and the impact/value ratio are taken into account. The ratio is not always directly applied to determine the priority rankings. In some cases, the safety significance of the issue is so great that it demands a HIGH priority, or so minor that only a LOW priority (or a decision to DROP) is warranted irrespective of the impact/value assessment.

(f) The priority ranking is reviewed and modified, if appropriate, in light of any special factors (discussed below) that: (i) might bring into question the applicability of the necessarily simplified calculation technique; and (ii) call for special consideration of NRC management decisions or large uncertainties in the quantitative estimates.

In summary, while the method has a quantitative emphasis, the calculated numerical values are used as an aid to judgment and not as determinative of the ranking results. The nature of the specific issue, the quality of the data base, and the scope of the necessarily limited analysis determine in each case the dependability of the numerical indications as a judgment aid.

2. Safety Significance

The safety significance of an issue is represented by the reduction in risk that resolution could effect. Risk is ordinarily expressed here in terms of the product of the frequency of an accident occurrence and the public dose (in person-rem) that would result in the event of the accident. If more than one accident scenario is important within the necessarily rough risk estimates, the risks are summed.

The potential risk reduction calculated in this way is used in calculating the impact/value ratio as part of the simplified impact/value analysis, discussed in Paragraph III.3 below. It is also used directly as a measure of safety significance, as discussed in Paragraph III.4 below, in arriving at a priority rank that is influenced by the safety significance of an issue as well as by the estimated value/impact relation of a projected solution, or is determined on the basis of safety significance alone.

The person-rem-based risk reduction estimate may not be the only appropriate measure of an issue's safety significance in all cases. For example, when a possible core damage is involved but release outside containment would be minor or highly improbable, contribution to the core-damage probability may well be more indicative of safety significance. Provision is made, as described in Paragraph III.4 below, for use of alternative measures of safety significance in determining a priority ranking when such alternative measures are useful.

3. Impact/Value Relation

a. The Impact/Value Ratio Formula

To the extent reasonably possible, quantitative estimates are made of the possible solutions to a GSI by calculating an Impact/Value Ratio that reflects the relation between the risk reduction value expected to be achieved and the associated cost impact. The formula for the impact/value ratio (R) is:

$$R = \frac{\text{Cost}}{\text{Safety Benefit}}$$

where the safety benefit is the estimated risk reduction (event frequency x public dose averted) that may be achieved, and the cost is that thought necessary to develop and implement a resolution in the number of plants involved. The scoring computation for any issue is then:

$$R = \frac{C}{NFTD}$$

where, N = number of reactors involved
T = average remaining life of the affected plants (years)
F = the accident frequency reduction (event/reactor-year)
D = public dose from the radioactive material released from containment (person-rem)
C = total cost of developing and implementing the resolution of the issue for all plants affected (dollars).

The total cost (C) includes both the cost of developing the generic solution, typically NRC cost, and the cost of implementing the possible solution at all affected plants, typically industry cost, including design, equipment, installation, test, operation, and maintenance. The priority ratio (R) has the units of dollars per person-rem.

Simplified calculations usually suffice, since only an approximate impact/value ratio is required. Reference should be made to the current version of the Value-Impact Handbook[970] where necessary to supplement the general guidelines provided below.

b. Rationale for the Formula

The qualitative diversity of factors entering impact/value analyses in support of GSI prioritization, together with inevitable quantitative uncertainties, make any of various possible impact/value score formulas necessarily imperfect. Accordingly, provisions are made to compensate for those imperfections to the extent practical (as discussed in Paragraph III.5 below).

The formula selected measures a total-cost/total-safety-benefit relation. As discussed herein, it is applied within limits set by other possible considerations where a safety issue is either too important to depend on safety-cost tradeoffs or too trivial to merit attention at all. Two principal arguments favor a formula of this type:

(1) The denominator is designed as a direct measure of the safety values that it is NRC's primary mission to protect. The numerator is designed to measure the overall cost impact, including industry as well as NRC costs, and should thus reflect the entire public interest in economy. The resulting impact/value ratio should, subject to the stated caveats, reasonably approximate measuring the overall public interest in safety value received for total resources expended.

(2) The allocation of national resources, which in most cases are primarily industry resources, is optimized.

c. Risk Estimates

The risk estimates developed for GSIs are useful as rough approximations for comparative purposes, but are not necessarily applicable to the assessment of absolute levels of risk attributable to particular issues. Similarly, the impact/value ratios provide, for the limited purpose of prioritization, tentative assessments of relative potential for cost-effective resolution. They are not intended to be applied as impact/value determinations for any regulatory proposal that may ultimately result from efforts to resolve an issue. In addition, the assumed resolutions are not intended to prejudge the final resolutions, but are only assumptions that are necessary to perform quantitative analyses.

The basis of frequency estimates generally involves the following:

(1) Identification of the specific events which are the basis for the concern, for which the consequences are to be established, and which are to be eliminated or ameliorated by a proposed technical solution

(2) Use of event sequence diagrams, fault trees, or decision trees, if possible

(3) Identified references and calculations, or stated assumptions for the numbers used

(4) Consideration of the probability of common mode as well as random independent failures.

Exhibit B

Release Category	Release (Curies)	Estimated Public Dose** (Person-rem)
PWR-1	1.2×10^9	5,400,000
PWR-2	9.3×10^8	4,800,000
PWR-3	5.2×10^8	5,400,000
PWR-4	2.8×10^8	2,700,000
PWR-5	1.3×10^8	1,000,000
PWR-6	1.0×10^8	150,000
PWR-7	2.1×10^6	2,300
PWR-8*	7.7×10^5	75,000
PWR-9*	1.1×10^3	120
BWR-1	1.1×10^9	5,400,000
BWR-2	1.1×10^9	7,100,000
BWR-3	5.0×10^8	5,100,000
BWR-4	2.1×10^8	610,000
BWR-5*	1.7×10^5	20

* Non-core-melt (Other release categories involve core-melt).

** The Release value (Curies) and Estimated Public Dose (Person-rem) will be updated in the future to be consistent with the ongoing evaluation to revise the Source Term following a postulated severe accident.

Where possible numerical estimates are made based on operating experience, usually Licensee Event Reports (LERs). Other sources include prior PRAs and other risk and reliability studies. Some numbers are based on engineering judgment; in such cases, the basis for that judgment is stated.

For the identified end event(s), the expected radiological consequences are expressed in person-rem generally based on the radioactive release categories described in WASH 1400[16] (Appendix VI, pp. 2-1 to 2-5), reproduced as Appendix A to this report. Exhibit B gives estimated curies released and approximate population doses for each release category. The computer program CRAC2, applied to a typical midwest site (Braidwood) meteorology, was used for the dose calculations. However, the calculated doses were adjusted to reflect the mean of the population density within a 50-mile radius of U.S. nuclear power plants.[64] Assumptions and parameters used for the calculations at this stage (Step (b) described under "Basic Approach")were as follows:

- Consequences are represented by the whole body population dose (person-rem) received within 50 miles of the site.

- An exclusion area of 1/2 mile was assumed with a uniform population density of 340 persons per square mile beyond 1/2 mile. This is the mean 50-mile radius population density projected for the year 2000 (NUREG-0348, p.T52).[70]

- Evacuation of people was not considered because of the possible large variations in evacuation capability for each plant site.

- All exposure pathways were included in the basis of the tabulated numbers except ingestion pathways, i.e., interdiction of contaminated foods was assumed. (Farmland usage parameters for the State of Illinois were used for separate ingestion pathway calculations where made.)

- Meteorological data was taken from the U.S. National Weather Service station at Moline, Illinois.

The person-rem factors for each release category are given in Exhibit B. Although generally used, consequence estimates were not solely based on these factors. Other factors were used in some cases when more appropriate.

An estimated occupational dose of 20,000 person-rem from postaccident cleanup, repair, and refurbishment is also considered.

Where significant occupational radiological exposure (ORE) is incurred or averted in implementing current requirements or the proposed resolution of a GSI, such exposure is taken into account but stated separately. Where more direct issue-specific ORE information is lacking, dose estimates are obtained by assuming an average dose rate of 2.5 millirem/hour (based on the PNL analysis[64] cited above) and multiplying by the estimated number of man-hours involved.

A second factor is that the risk associated with an issue is more likely to be overestimated than underestimated. Where risk estimates are widely uncertain, a reasonably conservative value of risk reduction is generally selected to help assure adequate priority to issues that may warrant attention.

The sum of the estimated risks of all the separate issues will likely exceed the present estimate of the total risk of nuclear power plants because of two factors. First, individual accident sequences can be affected by more than one issue. The resolution of one issue would reduce the probability or consequences of a certain set of accident sequences. Some or even all of these sequences could be the same as some or even all of the sequences affected by another issue. However, issues are assessed independently and this interaction of their risk significance is not ordinarily considered. This interaction is strongest for issues related to human factors, since human error affects almost all sequences. The sum of the reductions in core-melt frequency estimated for all of the human factors-related issues may be as much as twice as great as the total human factors contribution to total risk. However, most of the issues not related to human factors are much less strongly interrelated.

d. Cost Estimates

Because cost estimates are used here only in relation to risk estimates which are generally subject to more or less wide uncertainties, only approximate costs are needed.

No separate estimates are generally made for offsite property damage; reasonably conservative use of the public dose estimates is an adequate surrogate in this application. Furthermore, there is no readily-available data on offsite damage that is realistic and detailed enough to make estimates meaningful, reasonably accurate, and generically applicable. If unusual or special offsite effects are not adequately represented by the public dose in some issues, this fact will be considered separately and explicitly in evaluating such issues.

The expected technical solution on which the cost estimate is based is identified. Estimated costs are established by collecting available data regarding engineering, procurement, installation, testing, and periodic inspection and maintenance. Where data are non-existent, estimates are based on judgments by the experts involved. Assumptions and estimated uncertainties are identified. Costs are estimated in 1982 dollars.

NRC costs include the following: (1) issue identification, analysis, resolution, and report issuance; (2) research to establish proposed specific changes to licensing requirements (or to determine that no change is required); (3) technical assistance contracts (including associated NRC effort); (4) discussions and correspondence with industry owners' groups; (5) plant reviews; and (6) preparation and review of SERs and requirement documents. The estimated cost of NRC professional time is based on $100,000 per person-year.

The costs to industry generally consist of some combination of the following: (1) licensing; (2) design; (3) equipment procurement; (4) installation; (5) testing, inspection, monitoring, and periodic maintenance; and (6) plant downtime to effect a change, taken as the cost of replacement power at $300,000/day. Industry manpower costs are ordinarily taken as $100,000 per person-year.

Averted plant damage costs may affect the priority of a GSI. Estimates for such averted costs are multiplied by the accident frequency and used as negative costs, i.e., subtracted from the (positive) costs of implementing the resolution of the issue.[1473] The averted costs may include those of averted equipment failures, limited-time plant outage, or limited plant-contamination cleanup. In the extreme, they can also include averted permanent loss of the plant, estimated at approximately $2 billion present worth. This estimate for a "generic" plant includes the costs of both plant-wide cleanup and permanent loss of use of the plant, discounted to present worth based on a 7% real discount rate. This figure is multiplied in each case by the reduction in frequency of such events that would be brought about by resolution of the GSI. The plant loss estimate includes allowance for typical plant age at the time of the accident as well as replacement power costs together with apportioned cost of a replacement plant. The plant-wide cleanup estimate reflects cleanup to the point at which the plant is ready for decommissioning or refurbishing for restart.[393] Refurbishing costs, when restart is more economical than decommissioning, would depend on the nature of the accident and could range from a fraction of the total plant loss figure to a cost approaching that figure.

Some fixed costs are one-time, initial costs; others may occur at future times. Future costs are discounted to present worth at a 7% rate. Where costs are continuous or periodically recurring throughout a plant's remaining life, the periodic cost is taken into account using an approximation of the present worth of the continuing (or repetitive) costs for plants with remaining operating lives of 20 years or longer.

e. Uncertainty Bounds

Major sources of uncertainty in the priority score are identified and judgments as to their quantitative significance are indicated as information warrants. Where data warrant, the method described in NUREG/CR-

2800,[64] Section 5, for the general case of combining uncertainties for random variables with unknown distributions (as well as some special cases) are used. [See also Paragraph III.5(a)]. Most often, however, a rigorous uncertainty analysis is not warranted. In most cases, the uncertainty in the point estimates of risks and costs is known to be large. However, sufficient information is not usually available to make a meaningful quantitative analysis of the uncertainty bounds of these point estimates. Decisions are tempered by the knowledge that the uncertainty is generally large. This knowledge was also used in developing the chart of tentative priority rankings (Figure 1). The wide spread between a level of risk, for example, at which an issue would be ranked as having a high priority and the level at which an issue would be ranked as low priority (a factor of 100) is partially based on the recognition that the uncertainties are large. In cases where uncertainty has a special character or importance, this is discussed and considered in the conclusion of the analysis of the GSI.

4. Priority Ranking

(a) Priority Ranking Chart

A chart showing how the tentative priority rankings are derived from the safety significance of an issue and its impact/value ratio is presented in Figure 1. The thresholds on the chart are discussed in Paragraphs III.4(b) and III.4(c) below. A revision to the $1,000/person-rem figure is currently being considered as part of a program to revise the NRC's guidelines for the performance of regulatory analyses. This figure will be updated, as necessary, when the revised guidelines are approved.

(b) Preliminary Screening for Safety Significance

The determination of a priority rank starts with a triage based on safety significance, i.e., the incremental risk associated with the issue. For a reduction in core damage frequency (ΔCDF) greater than 10^{-4} per reactor-year (RY), a HIGH priority is assigned on the basis of safety importance alone, regardless of other considerations, such as an initially estimated high cost, which might result in a low priority score.

At the other extreme, an issue's safety significance could be too minor to warrant diversion of attention from more important safety issues even if it has a low impact/value ratio because an inexpensive solution is believed to be available. Below a minimal safety significance threshold, the priority would always be DROP; where the potential risk reduction is trivial, there can be no basis for regulatory action on safety grounds.

In between, there may be issues of less extreme importance or unimportance, for which a HIGH, MEDIUM, LOW, or DROP priority may be appropriate, based on consideration of the impact/value relation as well as safety significance. As indicated in Figure 1, a HIGH priority may be assigned to an issue exclusively on the basis of a high safety significance; the threshold shown on the chart is $\Delta CDF=10^{-4}/RY$. For an issue with a safety significance lower than the threshold for an always-HIGH priority but at least 10% of that threshold ($\Delta CDF=10^{-5}/RY$), the chart indicates a HIGH or MEDIUM priority based on cost trade-offs. At the low-risk end of the abscissa, the priority rank indicated is always DROP for $\Delta CDF<10^{-7}/RY$. Cost trade-offs enter in the 10^{-7} to $10^{-4}/RY$ ΔCDF range, as discussed in Section 4(c) below.

The abscissa in Figure 1 provides a measure of an issue's estimated safety significance in terms of the change (Δ) in CDF attributable to resolution of the issue. This is often the most useful safety significance measure in GSI prioritization, though for some issues other measures may be required or appropriate. For example, a measure based on radiological consequences (probability-averaged over the remaining reactor life) is used when the issue under consideration involves containment bypass or relates to containment performance or other features or actions to mitigate the radiological consequences of a core damage. Also, the thresholds may need to accommodate the possible influence of the number of reactors affected on the appropriate priority ranking. Therefore, Figure 1 is repeated in Figure 2, with auxiliary abscissae providing additional measures of safety significance. These are used when the principal abscissa is inapplicable, or when an auxiliary abscissa leads to a higher priority indication.

Thus, the abscissae for total effect on all plants are considered when more than 30 plants are affected.

(c) Impact/Value Ratio Thresholds

When the safety significance is in the intermediate range discussed above, i.e., ΔCDF between 10^{-7} and $10^{-4}/RY$, or between 0.1% and 100% of the threshold for an always-HIGH priority, the impact/value ratio (R) is taken into account in the ranking indicated by the chart (Figure 1). This is done as follows:

(1) In the range of 10% to 100% of the threshold for an always-HIGH priority, the indicated priority is HIGH if R is below $1,000/person-rem; otherwise, the indicated priority is MEDIUM.

FIGURE 1

PRIORITY RANKING

Impact/Value ($/Person-Rem)

> 1,000	DROP	DROP	LOW	MEDIUM	HIGH
1,000					
< 1,000	DROP	LOW	MEDIUM	HIGH	HIGH

10^{-7} 10^{-6} 10^{-5} 10^{-4} ΔCDF/RY

(2) In the range of 1% to 10% of the threshold for an always-HIGH priority, the indicated priority is MEDIUM if R is below $1,000/person-rem; otherwise, the indicated priority is LOW.

(3) In the range of 0.1% to 1% of the always-HIGH threshold, the indicated priority is LOW or DROP, depending on whether R is below or above $1,000/person-rem.

5. Other Considerations

The formula-based rankings represent the primary concern of the NRC: public safety. The secondary concern is the impact on licensees, evaluated in terms of cost. However, the tentative priority rankings are subject to the limitations of an often incomplete and imprecise data base and to possible distortions due to the nature of the necessarily highly simplified quantitative formula underlying them. Special situations with respect to some issues may cause added difficulty in priority assignment. While the formula-based tentative rankings generally indicate that the safety significance is sufficient to justify NRC action, other considerations not adequately reflected, or not reflected at all, in the numerical formula are often needed to corroborate or adjust the results. Decision-making is helped by explicit identification of such other considerations and explanation of how they bear on the resulting final priority ranking, whether the effect is one of corroborating or of changing the estimates.

Listed below are some factors that may be important in arriving at a sound priority ranking and may lead to adjustment of a tentative, formula-derived ranking. Possible effects of occupational doses and uncertainty bounds [(a)(1), (a)(2), and (b)(1) below] require particularly careful consideration for all issues. The factors listed are not considered all-inclusive. Others thought significant are discussed and, when practical, quantified appropriately in the overall risk significance measure and impact/value ratio along with their associated uncertainties. Sometimes, there are special considerations that are quite specific to an issue or some aspect of it. However, it should be noted that, in determining an issue's priority, those factors that relate to safety are given the most consideration. The following is a partial list of other factors to be considered:

(a) Special risk and cost aspects not included in or potentially masked by the numerical formulas:

(1) The net change in occupational doses entailed by implementing the current versus the proposed requirements.

(2) Any significant non-radiation-related occupational risk affected by the proposed resolutions.

FIGURE 2

Priority Ranking With Auxiliary Abscissae

Impact/Value ($/Person-Rem)

> 1,000	DROP	DROP	LOW	MEDIUM	HIGH
1,000					
< 1,000	DROP	LOW	MEDIUM	HIGH	HIGH

10^{-7}	10^{-6}	10^{-5}	10^{-4}	ΔCDF/Year
3×10^{-6}	3×10^{-5}	3×10^{-4}	3×10^{-3}	ΔCDF/Year (Total, All Affected Reactors)
10^{1}	10^{2}	10^{3}	10^{4}	ΔPerson-Rem/Reactor (Offsite)
3×10^{2}	3×10^{3}	3×10^{4}	3×10^{5}	ΔPerson-Rem (Total Offsite, All Affected Reactors)

(3) Loss or severe degradation of a layer in the defense-in-depth concept (e.g., one mode of core cooling or containment cooling)

(4) Issues for which solutions of widely differing costs may be applicable to different classes of plants or various plants are otherwise affected in vastly different ways.

(b) Factors related to uncertainties stemming from an incomplete or imprecise data base for the priority formula:

(1) Uncertainty bounds, imbalance in uncertainty factors, certainty of cost to fix versus uncertainty that safety is really improved and the true extent of such improvement.

(2) Situations where uncertainty is extraordinarily large (in accident probability, consequences, or cost, or any or all of these). If there are large uncertainties in either the numerator or the denominator, the mean of the impact/value ratio (mean ratio) should be used with caution in assigning a priority ranking. The ratio of the means is a good approximation to the mean ratio provided only that the uncertainty in the denominator is small. However, if the uncertainty in the denominator is large, then the ratio of the means is a poor estimate of the mean ratio.

(3) Problems which are ill-defined and problems for which solutions are not evident so that at least the resources necessary to understand the problem are assigned.

(4) The potential for a proposed change to affect more than one accident or transient sequence, thus affecting risk to a greater or lesser degree than assessed in the description of the issue; notably, the potential for a new safety decrement, or increase in risk, due to unidentified effects of a proposed change, or added complexity, or for other reasons.

(5) Circumstances imparting unusual significance to accident consequences (such as ingestion pathway effects) or mitigating measures (such as evacuation) that are not directly included in the public dose calculations.

(6) Potential for human intervention, using available equipment.

(7) Acute knowledgeable professional controversy concerning the importance of an issue or modes of dealing with it.

(c) Change with passage of time:

  (1) The effect of license renewal should be considered in every prioritization. The effect, if any, on the priority rank of an additional 20 years of operation should be separately stated.

  (2) Potential substantial deterioration of the impact/value ratio while awaiting regulatory resolution (e.g., a potential design fix that is inexpensive to apply before construction, much more expensive after the plant is largely built, and extremely expensive and problematical to apply to an operating plant).

  (3) The amount of resources already spent on an issue, and how close to completion it may be; the value of continuity in efforts to resolve an issue.

  (4) The span of time predicted to resolve an issue and implement the resolution.

  (5) The clarity of an "issue" and the objectivity with which it is currently defined. (Perhaps additional research effort is necessary to identify and define a specific risk reduction of interest.)

  (6) Change of perceptions (of safety importance or impact/value relation or some special issue-peculiar factor) in the course of time.

Generally, in situations of large doubt or conflicting indications, the highest priority rank reasonably consistent with the nature of an issue is assigned. Thus, where no solution is evident, assignment of a priority consistent with the safety significance of the issue may lead to a search for resolution or mitigation at an acceptable cost. Generally, should uncertainties narrow or perceptions change in the course of time, the priority rankings can be reexamined in the light of new developments and retained or changed. When different classes of plants are expected to be very differently affected by a potential resolution, the priority assignment is governed by the class of plants for which resolution is most worthwhile and urgent. (Resolution in such cases can involve a new requirement for some class of plants and no action for others.) Where resolution differs for different classes of plants, differing priorities may be assigned.

6. Concluding Remarks

The criteria and estimating process on which the priority rankings are based are neither rigorous nor precise. Considerable application of professional judgment, sometimes guided by good information but often tenuously based, occurs at a number of stages in the process when numerical values are selected for use in the formula calculations and when other considerations are taken into account in corroborating or changing a priority ranking. What is important in the process is that it is systematic, that it is guided by analyses that are as quantitative as the situation reasonably permits, and that the bases and rationale are explicitly stated, providing a "visible" information base for decision. The impact of imprecision is blunted by the fact that only approximate rankings (in only four broad priority categories) are necessary and sought.

IV. RESULTS OF PRIORITIZATION

The results of the prioritization and resolution of all issues contained in this report are summarized and tabulated by group in Table III. In addition, a listing of those issues that affect operating and future plants is given in Appendix B. This appendix reflects the results of prioritization and resolution and only includes: (1) issues that have been resolved with new requirements [NOTE 3(a)]; (2) USI, HIGH and MEDIUM priority issues that are being resolved; (3) nearly-resolved issues (NOTES 1 and 2); (4) issues that are scheduled for prioritization and whose impact is not yet known (NOTE 4); and (5) issues that were resolved without requirements for operating plants but with staff requirements for future plants under development.

REFERENCES

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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. Nuclear Regulatory Commission, October 1975.

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70. NUREG-0348, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," U.S. Nuclear Regulatory Commission, November 1979.

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1191. Federal Register Notice 54 FR 24432, "Program for Resolution of Generic Issues Related to Nuclear Power Plants; Policy Statement," June 7, 1989.

1192. Memorandum for R. Fraley, et al., from E. Beckjord, "RES Office Letter No. 1, 'Procedure for Identification, Prioritization, and Tracking of the Resolution of Generic Issues,'" December 3, 1987, (Rev. 1) March 22, 1989.

1193. Memorandum for R. Fraley, et al., from E. Beckjord, "RES Office Letter No.2, 'Procedures for Obtaining Regulatory Impact Analysis Review and Support,'" November 18, 1988.

1194. Memorandum for R. Fraley et al., from E. Beckjord, "RES Office Letter No. 3 Procedure and Guidance for the Resolution of Generic Issues," May 10, 1988, (Rev. 1) December 21, 1988, (Rev. 2) March 27, 1989.

1195. NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1989, (Vol. 2) April 1989.

1473. Memorandum for J. Taylor from S. Chilk, "SECY-89-102 - Implementation of the Safety Goals," June 15, 1990.

1479. SECY-93-108 "Revised Guidelines for Prioritization of Generic Safety Issues, "April 28, 1993.

1493. Memorandum for V. Stello from S. Chilk, "Staff Requirements - Briefing on Status of Unresolved. Safety/Generic Issues, 10:00 a.m. Wednesday, October 21, 1987, Commissioner Conference Room D.C. Office (Open to Public Attendance," November 6, 1987.

1505. Memorandum for J. Taylor from S. Chilk, "SECY-93-108 - Revised Guidelines for Prioritization of Generic Safety Issues," July 23, 1993.

TABLE II

LISTING OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS ISSUES

This table contains the priority designations for all issues listed in this report. For those issues found to be covered in other issues described in this document, the appropriate notations have been made in the Safety Priority Ranking column, e.g., I.A.2.2 in the Safety Priority Ranking column means that Item I.A.2.6(3) is covered in Item I.A.2.2. For those issues found to be covered in programs not described in this document, the notation (S) was made in the Safety Priority Ranking column. For resolved issues that have resulted in new requirements for operating plants, the appropriate multiplant licensing action number is listed. The licensing action numbering system bears no relationship to the numbering systems used for identifying the prioritized issues. An explanation of the classification and status of the issues is provided in the legend below.

Legend

NOTES: 1 - Possible Resolution Identified for Evaluation
2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent)
or (b) No New Requirements
4 - Issue to be Prioritized in the Future
5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion

HIGH - High Safety Priority
MEDIUM - Medium Safety Priority
LOW - Low Safety Priority
DROP - Issue Dropped as a Generic Issue
EI - Environmental Issue
I - Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI - Licensing Issue
MPA - Multiplant Action
NA - Not Applicable
RI - Regulatory Impact Issue
S - Issue Covered in an NRC Program Outside the Scope of This Document
USI - Unresolved Safety Issue

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I	2	12/31/86	F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I	2	12/31/86	
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I	2	12/31/86	F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFO/HFBR	NOTE 3(a)	2	12/31/86	
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	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I	5	12/31/87	
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI (NOTE 3)	5	12/31/87	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	5	12/31/87	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	5	12/31/87	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	5	12/51/87	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	5	12/31/86	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	5	12/31/86	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DRAO/HFSB	NOTE 3(b)	5	12/31/86	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/86	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	5	12/31/86	NA
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-			
I.A.4.1(1)	Short-Term Study of Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(b)	5	06/30/88	NA
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	5	06/30/88	
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	5	06/30/88	
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3(a)	5	06/30/88	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3(a)	5	06/30/88	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DLPQ/LOLB	NOTE 3(a)	5	06/30/88	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	5	06/30/88	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI (NOTE 3)	5	06/30/88	NA
I.B.	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	3	12/31/86	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	3	12/31/86	NA
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants	-	-	-			
I.B.1.2(1)	Prepare Draft Criteria	-	NRR/DHFS/LQB	NOTE 3(b)	3	12/31/86	NA
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	NOTE 3(b)	3	12/31/86	NA
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	NOTE 3(b)	3	12/13/86	NA
I.B.1.3	Loss of Safety Function	-	-	-			
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I	3	12/31/86	
I.C.1(2)	Inadequate Core Cooling	-	NRR	I	3	12/31/86	F-04
I.C.1(3)	Transients and Accidents	-	NRR	I	3	12/31/86	F-05
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	3	12/31/86	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I	3	12/31/86	
I.C.3	Shift Supervisor Responsibilities	-	NRR	I	3	12/31/86	
I.C.4	Control Room Access	-	NRR	I	3	12/31/86	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I	3	12/31/86	F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I	3	12/31/86	F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	NOTE 3(b)	3	12/31/86	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I	5	12/31/89	F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I	5	12/31/89	F-09
I.D.3	Safety System Status Monitoring	Thatcher	RES/DE/MEB	MEDIUM	5	12/31/89	
I.D.4	Control Room Design Standard	Thatcher	RES/DRPS/RHFB	NOTE 3(b)	5	12/31/89	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	5	12/31/89	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	5	12/31/89	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DE/MEB	NOTE 1	5	12/31/89	
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	5	12/31/89	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	RES/DRPS/RHFB	LI (NOTE 5)	5	12/31/89	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	5	12/31/89	NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	1	6/30/84	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	1	6/30/84	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	1	6/30/84	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	1	6/30/84	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	Pittman	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/89	NA
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/09	NA
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements		NRR/DHFS/PSRB	I	2	06/30/89	
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PSRB	NOTE 3(a)	2	06/30/89	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	V'Molen	NRR/DE/SAB	NOTE 3(b)	1	12/31/84	NA
II.A.2	Site Evaluation of Existing Facilities	V'Molen	NRR/DE/SAB	V.A.1	1	12/31/84	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I	3	12/31/91	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I	3	12/31/91	F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I	3	12/31/91	F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	3	12/31/91	F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.B.5(1)	Behavior of Severely Damaged Fuel	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.5(2)	Behavior of Core-Melt	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	NOTE 3(a)	3	12/31/91	
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8	3	12/31/91	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/DRAO/RAMR	NOTE 3(a)	3	12/31/91	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRAO/RRB	NOTE 3(b)	2	12/31/88	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	NOTE 3(b)	2	12/31/88	NA
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17	2	12/31/88	NA
II.C.4	Reliability Engineering	Pittman	RES/DRPS/RHFB	NOTE 3(b)	2	12/31/88	NA
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I	1	06/30/89	F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW	1	06/30/89	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I	1	06/30/89	
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	1	12/31/86	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	1	12/31/86	F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)	1	12/31/86	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)	1	12/31/85	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	NOTE 3(b)	1	12/31/85	NA
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW	1	12/31/85	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR/DL	I	1	06/30/91	
II.E.3.2	Systems Reliability	V'Molen	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)	1	06/30/91	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I		06/30/88	F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I		06/30/88	F-19
II.E.4.3	Integrity Check	Milstead	RES/DRPS/RPSI	NOTE 3(b)		06/30/88	NA
II.E.4.4	Purging	-	-	-			

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(a)		06/30/88	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)		06/30/88	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)		06/30/88	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)		06/30/88	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)		06/30/88	NA
<u>II.E.5</u>	<u>Design Sensitivity of B&W Reactors</u>						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3(a)	1	12/31/84	
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 3(a)	1	12/31/84	
<u>II.E.6</u>	<u>In Situ Testing of Valves</u>						
II.E.6.1	Test Adequacy Study	Thatcher	RES/DE/EIB	NOTE 3(a)	1	06/30/89	
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	2	06/30/89	F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I	2	06/30/89	F-26
II.F.3	Instruments for Monitoring Accident Conditions	V'Molen	RES/DFO/ICBR	NOTE 3(a)	2	06/30/89	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP	2	06/30/89	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DE	LI (NOTE 3)	2	06/30/89	NA
<u>II.G</u>	<u>ELECTRICAL POWER</u>						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I			
<u>II.H</u>	<u>TMI-2 CLEANUP AND EXAMINATION</u>						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3(b)		11/30/83	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DRAA/AEB	HIGH		11/30/83	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	II.H.2		11/30/83	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SEBR	LI (NOTE 3)		11/30/83	NA
<u>II.J</u>	<u>GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES</u>						
<u>II.J.1</u>	<u>Vendor Inspection Program</u>						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.3	Assign Resident inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.3.2	Issue Regulatory Guide	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	AEOD/DSP/ROAB	NOTE 3(a)	1	12/31/91	NA
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	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training Instructions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(5)	Safety-Related Valve Position Description	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	Emrit	NRR	NOTE 3(a)		12/31/84	-
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	Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	Phases of, the TMI-2 Accident						
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	Emrit	NRR	NOTE 3(a)		12/31/84	-
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	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Emrit	NRR	NOTE 3(a)		12/31/84	-
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	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2	Commission Orders on B&W Plants	-	-	-			
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	Emrit	NRR	II.E.1.1, II.E.1.2		12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	Emrit	NRR	I		12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	Emrit	NRR	I		12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	Emrit	NRR	I		12/31/84	F-33
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	Emrit	NRR	I		12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	Emrit	NRR	I		12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-			
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	Emrit	NRR	I		12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	Emrit	NRR	II.C.1, II.C.2, II.C.3		12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	Emrit	NRR	I		12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3		12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	Emrit	NRR	I		12/31/84	-
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	Emrit	NRR	I		12/31/84	F-40

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	Emrit	NRR	I		12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	Emrit	NRR	I		12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	Emrit	NRR	I		12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	Emrit	NRR	I		12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-45
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	Emrit	NRR	I		12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	Emrit	NRR	I		12/31/84	F-47
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Emrit	NRR	I		12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	Emrit	NRR	I		12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	Emrit	NRR	I		12/31/84	-
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	Emrit	NRR	I		12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	Emrit	NRR	I		12/31/84	F-51
II.K.3(23)	Central Water Level Recording	Emrit	NRR	I.D.2, III.A.1.2(1), III.A.3.4		12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	Emrit	NRR	I		12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	Emrit	NRR/DSI	II.E.2.1		12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	Emrit	NRR	I		12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	Emrit	NRR	I		12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	Emrit	NRR	I		12/31/84	F-56
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Emrit	NRR	I		12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Emrit	NRR	I		12/31/84	F-58
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	Emrit	NRR	II.C.1		12/31/84	NA
II.K.3(34)	Relap-4 Model Development	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	Emrit	NRR	I.C.1(3)		12/31/84	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	Emrit	NRR	II.K.2(16)		12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	Emrit	NRR	II.K.2(15)		12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Emrit	NRR	I		12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	Emrit	NRR	I		12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	Emrit	NRR	I		12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	Emrit	NRR	I.C.1(3), II.E.2.2		12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	Emrit	NRR	II.C.1, II.C.2		12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9		12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	Emrit	NRR/DHFS/PSRB	I.C.7, I.C.9		12/31/84	NA
II.K.3(51)	Symptom-Based Emergency Procedures	Emrit	NRR/DHFS/PSRB	I.C.9		12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	Emrit	NRR	I.B.1.1, I.C.2, I.C.5		12/31/84	NA
II.K.3(53)	Two Operators in Control Room	Emrit	NRR	I.A.1.3		12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	Emrit	NRR	I.A.4.1(2)		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	Emrit	NRR	I.C.1(3), I.D.2, I.D.3		12/31/84	NA
II.K.3(56)	Simulator Training Requirements	Emrit	NRR/DHFS/OLB	I.A.2.6(3), I.A.3.1		12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	Emrit	NRR	I		12/31/84	F-62
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short-Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-	2	06/30/91	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I	2	06/30/91	
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	2	06/30/91	
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB	I	2	06/30/91	F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-	2	06/30/91	
III.A.1.3(1)	Workers	Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
<u>III.A.2</u>	<u>Improving Licensee Emergency Preparedness-Long Term</u>						
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	-	RES	I			
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	I			
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	I			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I			F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I			F-68
<u>III.A.3</u>	<u>Improving NRC Emergency Preparedness</u>						
III.A.3.1	NRC Role in Responding to Nuclear Emergencies	-	-	-			
III.A.3.1(1)	Define NRC Role in Emergency Situations	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6	Interaction of NRC and Other Agencies	-	-	-			
III.A.3.6(1)	International	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
<u>III.B</u>	<u>EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS</u>						
III.B.1	Transfer of Responsibilities to FEMA	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities	-	-	-			
III.B.2(1)	The Licensing Process	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public	-	-	-			
III.C.1(1)	Review Publicly Available Documents	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing With the News Media	-	-	-			
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	Pittman	PA	LI (NOTE 3)		11/30/83	NA
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	-	NRR	I	1	12/31/88	-
III.D.1.1(2)	Review Information on Provisions for Leak Detection	Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)	1	12/31/88	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA
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	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	-	-	-			
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	Emrit	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA
III.D.2.3	Liquid Pathway Radiological Control	-	-	-			
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	2	12/31/85	NA
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	2	12/31/85	NA
III.D.3	Worker Radiation Protection Improvement						
III.D.3.1	Radiation Protection Plans	V'Molen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/87	NA
III.D.3.2	Health Physics Improvements	-	-	-			
III.D.3.2(1)	Amend 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.3	In-plant Radiation Monitoring	-	-	-			
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling	-	NRR/DL	I	2		F-69
Instrumentation							
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.4	Control Room Habitability	-	NRR/DL	I			F-70
III.D.3.5	Radiation Worker Exposure	-	-	-			
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	[illegible] Lessons Learned from TMI to Other NRC Programs	Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA
IV.E	SAFETY DECISION-MAKING						
IV.E.1	Expand Research on Quantification of Safety Decision-Making	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.2	Plan for Early Resolution of Safety Issues	Emrit	NRR/DST/SPEB	LI (NOTE 3)	2	12/31/86	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	Colmar	RES/DRA/RABR	LI (NOTE 5)	2	12/31/86	NA
IV.E.4	Resolve Generic Issues by Rulemaking	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.5	Assess Currently Operating Reactors	Matthews	NRR/DL/SEPB	NOTE 3(b)	2	12/31/86	NA
IV.F	FINANCIAL DISINCENTIVES TO SAFETY						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	Thatcher	OIE/DQASIP	NOTE 3(b)	1	12/31/86	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	Matthews	SP	NOTE 3(b)	1	12/31/86	NA
IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)	1	12/31/86	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.H	NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL						
IV.H.1	NRC Participation in the Radiation Policy Council	Sege	RES/DHSWM/HEBR	LI (NOTE 3)		11/30/63	NA
V.A	DEVELOPMENT OF SAFETY POLICY						
V.A.1	Develop NRC Policy Statement on Safety	Emrit	GC	LI (NOTE 3)		12/31/86	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
V.B	POSSIBLE ELIMINATION OF NONSAFETY RESPONSIBILITIES						
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C	ADVISORY COMMITTEES						
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D	LICENSING PROCESS						
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D.2	Study Construction-During-Adjudication Rules	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.3	Reexamine Commission Role in Adjudication	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.4	Study the Reform of the Licensing Process	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G.2	Achieve Single Location, Interim	Emrit	GC	LI (NOTE 3)		12/31/86	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-10
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-4	CE Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-5	B&W Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No
A-6	Mark I Short-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-7	Mark I Long-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-01
A-8	Mark II Containment Pool Dyanmic Loads Long-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	
A-14	Flaw Detection	Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former (USI)	Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	Milstead	RES/DSR/HFB	LI (NOTE 5)	1	06/30/91	NA
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	NA
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	V'Molen	NRR/DSI/CSB	LOW		11/30/83	NA
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	V'Molen	NRR/DSI/CSB	DROP		11/30/83	NA
A-23	Containment Leak Testing	Matthews	NRR/DSI/CSB	RI (NOTE 5)		11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	Thatcher	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-26	Reactor Vessel Pressure Transient Protection (former (USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI (NOTE 5)		11/30/83	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	Colmar	NRR/DE/SGEB	NOTE 3(a)		11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	Colmar	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/89	NA
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-32	Missile Effects	Pittman	NRR/DE/MTEB	A-37, A-38, B-68		11/30/83	NA
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	EI(NOTE 3)		11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3		11/30/83	NA
A-35	Adequacy of Offsite Power Systems	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	Emrit	NRR/DSI/GIB	NOTE 3(a)	1	06/30/85	C-10, C-15
A-37	Turbine Missiles	Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	Sege	NRR/DSI/ASB	LOW		11/30/83	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	6/30/85	
A-40	Seismic Design Criteria (former USI)	Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	NA
A-41	Long-Term Seismic Program	Colmar	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	Emrit	NRR/DSRO/EIB	NOTE 3(a)	1	12/31/87	
A-47	Safety Implications of Control Systems (former USI)	Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	Emrit	NRR/DSRO/RSIB	NOTE 3(a)	1	12/31/87	A-21
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	EI (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (NOTE 3)		11/30/83	NA
B-4	ECCS Reliability	Emrit	NRR/DSI/RSB	II.E.3.2		11/30/83	NA
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DSRO/EIB	119.1		12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (NOTE 3)		11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	Riggs	NRR/DSI/RSB	DROP		11/30/83	NA
B-9	Electrical Cable Penetrations of Containment	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	V'Molen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	Emrit	NRR/DST/GIB	A-48		11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (NOTE 3)		11/30/83	NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Emrit	NRR/DE/MEB	A-18		11/30/83	NA
B-17	Criteria for Safety-Related Operator Actions	Milstead	RES/DRPS/RHFB	MEDIUM	2	12/31/86	
B-18	Vortex Suppression Requirements for Containment Sumps	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
B-19	Thermal-Hydraulic Stability	Colmar	NRR/DSI/CPB	NOTE 3(b)		6/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI (NOTE 5)		11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-22	LWR Fuel	Emrit	RES/DSIR/RPSIB	DROP		06/30/91	NA
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Equipment	Emrit	NRR	A-46		11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	LI (NOTE 3)	1	06/30/91	NA
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	LI (NOTE 3)	1	06/30/89	NA
B-32	Ice Effects on Safety-Related Water Supplies	Pittman	NRR/DE/EHEB	153	1	06/30/91	NA
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)		11/30/83	NA
B-34	Occupational Radiation Exposure Reduction	Emrit	NRR/DSI/RAB	III.D.3.1		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	-	NRR/DSI/METB	LI (NOTE 5)		11/30/83	
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	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	EI (NOTE 5)		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	EI (NOTE 5)		11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-47	Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	Colmar	NRR/DE/MTEB	DROP		11/30/83	NA
B-48	BWR Control Rod Drive Mechanical Failures	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI (NOTE 5)		11/30/83	
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	RI (NOTE 3)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	Emrit	NRR/DE/MEB	A-40		11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	Emrit	NRR/DST/GIB	A-2		11/30/83	NA
B-53	Load Break Switch	Sege	NRR/DSI/PSB	RI (NOTE 3)		11/30/83	
B-54	Ice Condenser Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	V'Molen	RES/DE/EIB	MEDIUM		11/30/83	
B-56	Diesel Reliability	Milstead	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/93	D-19
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	Colmar	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	RI (NOTE 3)	1	6/30/85	E-04,E-05
B-60	Loose Parts Monitoring Systems	Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	Pittman	RES/DRAA/PRAB	MEDIUM		11/30/83	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
B-64	Decommissioning of Reactors	Colmar	RES/DE/MEB	NOTE 2		11/30/83	
B-65	Iodine Spiking	Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	Matthews	NRR/DSI/AEB	NOTE 3(a)		11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	Colmar	NRR/DSI/METB	III.D.2.1		11/30/83	NA
B-68	Pump Overspeed During LOCA	Riani	NRR/DSI/ASB	DROP		11/30/83	NA
B-69	ECCS Leakage Ex-Containment	Riani	NRR/DSI/METB	III.D.1.1(1)		11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	Coolant Pumps						
B-71	Incident Response	Riani	NRR	III.A.3.1		11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI (NOTE 5)		11/30/83	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	Thatcher	NRR/DE/MEB	C-12		11/30/83	NA
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Milstead	NRR/DE/EQB	NOTE 3(a)		11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	Emrit	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
C-3	Insulation Usage Within Containment	Emrit	NRR/DST/GIB	A-43	1	06/30/91	NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-6	LOCA Heat Sources	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	NA
C-8	Main Steam Line Leakage Control Systems	Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	LI (NOTE 3)		06/30/88	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	LOW		11/30/83	NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	RES/DRA/ARGIB	DROP		12/31/88	NA
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
	NEW GENERIC ISSUES						
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Diab	RES/DSIR/EIB	DROP	1	12/31/92	NA
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	V'Molen	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
7.	Failures Due to Flow-Induced Vibrations	V'Molen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	Emrit	NRR/DE/MTEB	NOTE 3(b)	1	12/31/85	NA
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	HIGH	2	12/31/89	
16.	BWR Main Steam Isolation Valve Leakage Control Systems	Milstead	NRR/DSI/ASB	C-8		11/30/83	NA
17.	Loss of Offsite Power Subsequent to a LOCA	Colmar	NRR/DSI/PSB, ICSB	DROP		11/30/83	NA
18.	Steam Line Break with Consequential Small LOCA	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	Sege	NRR/DST/GIB	A-47		11/30/83	NA
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	06/30/84	NA
21.	Vibration Qualification of Equipment	Riggs	NRR/DE/EIB	DROP	2	06/30/91	NA
22.	Inadvertent Boron Dilution Events	V'Molen	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
23.	Reactor Coolant Pump Seal Failures	Riggs	RES/DE/EIB	HIGH		11/30/83	
24.	Automatic ECCS Switchover to Recirculation	Milstead	NRR/DSIR/RPSIB	MEDIUM	1	12/31/91	
25.	Automatic Air Header Dump on BWR Scram System	Milstead	NRR/DSI/RSB	NOTE 3(a)		11/30/83	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	Emrit	NRR/DSI/ASB	17		11/30/83	NA
27.	Manual vs. Automated Actions	Pittman	NRR/DSI/RSB	B-17		11/30/83	NA
28.	Pressurized Thermal Shock	Emrit	NRR/DST/GIB	A-49		11/30/83	NA
29.	Bolting Degradation or Failure in Nuclear Power Plants	V'Molen	RES/DSIR/EIB	NOTE 3(b)	1	12/31/91	NA
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA
31.	Natural Circulation Cooldown	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	Emrit	NRR/DSI/ASB	51		11/30/83	NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	Pittman	NRR/DSI/ICSB	A-47		11/30/83	NA
34.	RCS Leak	Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	V'Molen	NRR/DSI/CPB, RSB	LOW	1	06/30/85	NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	3	06/30/91	NA
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Colmar	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1(2)	1	06/30/85	NA
38.	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	Emrit	RES/DSIR/RPSIB	DROP	1	12/31/91	NA
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25		11/30/83	NA
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Colmar	NRR/DSI/ASB	NOTE 3(a)	1	06/30/84	B-65
41.	BWR Scram Discharge Volume Systems	V'Molen	NRR/DSI/RSB	NOTE 3(a)		11/30/83	B-58

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	I.C.1	1	06/30/85	NA
43.	Reliability of Air Systems	Milstead	RES/DSIR/RPSI	NOTE 3(a)	2	12/31/88	
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43	1	12/31/88	NA
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3(a)	2	06/30/91	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76		11/30/83	NA
47.	Loss of Offsite Power	Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)		11/30/83	
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
49.	Interlocks and LCOs for Redundant Class 1E Tie-Breakers	Sege	NRR/DSI/PSB	128	3	06/30/91	NA
50.	Reactor Vessel Level Instrumentation in BWRs	Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	
52.	SSW Flow Blockage by Blue Mussels	Emrit	NRR/DSI/ASB	51		11/30/83	NA
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	V'Molen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	Colmar	NRR/DE/MEB	II.E.6.1	1	06/30/85	NA
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Emrit	NRR/DSI/PSB	DROP	2	06/30/91	NA
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	Colmar	NRR/DHFS/HFEB	A-47, I.D.1		11/30/83	NA
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	Milstead	RES/DRA/ARGIB	MEDIUM	1	06/30/88	
58.	Inadvertent Containment Flooding	Sege	NRR/DSI/ASB, CSB	DROP		11/30/83	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	Emrit	NRR/DST/TSIP	RI (NOTE 5)	1	06/30/85	NA
60.	Lamellar Tearing of Reactor Systems Structural Supports	Colmar	NRR/DST/GIB	A-12		11/30/83	NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/86	NA
62.	Reactor Systems Bolting Applications	Riggs	RES/DSIR/EIB	29	1	12/31/88	NA
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	Pittman	RES/DRA/ARGIB	DROP	1	06/30/90	NA
64.	Identification of Protection System Instrument Sensing Lines	Thatcher	NRR/DSI/ICSB	NOTE 3(b)		11/30/83	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	V'Molen	NRR/DSI/ASB	23	1	12/31/86	NA
66.	Steam Generator Requirements	Riggs	NRR/DEST/EMTB	NOTE 3(b)	2	12/31/88	NA
67.	Steam Generator Staff Actions	-	-	-		-	
67.2.1	Integrity of Steam Generator Tube Sleeves	Riggs	NRR/DE/MEB	135	3	06/30/91	NA
67.3.1	Steam Generator Overfill	Riggs	NRR/DST/GIB NRR/DSI/RSB	A-47, I.C.1	3	06/30/91	NA
67.3.2	Pressurized Thermal Shock	Riggs	NRR/DST/GIB	A-49	3	06/30/91	NA
67.3.3	Improved Accident Monitoring	Riggs	NRR/DSI/ICSB	NOTE 3(a)	3	06/30/91	A-17

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
67.3.4	Reactor Vessel Inventory Measurement	Riggs	NRR/DSI/CPB	II.F.2	3	06/30/91	NA
67.4.1	RCP Trip	Riggs	NRR/DSI/RSB	II.K.3(5)	3	06/30/91	G-01
67.4.2	Control Room Design Review	Riggs	NRR/DHFS/HFEB	I.D.1	3	06/30/91	F-08
67.4.3	Emergency Operating Procedures	Riggs	NRC/DHFS/PSRB	I.C.1	3	06/30/91	F-05
67.5.1	Reassessment of Radiological Consequences	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	3	06/30/91	NA
67.5.2	Reevaluation of SGTR Design Basis	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	3	06/30/91	NA
67.5.3	Secondary System Isolation	Riggs	NRR/DSI/RSB	DROP	3	06/30/91	NA
67.6.0	Organizational Responses	Riggs	OIE/DEPER/IRDB	III.A.3	3	06/30/91	NA
67.7.0	Improved Eddy Current Tests	Riggs	RES/DE/EIB	135	3	06/30/91	NA
67.8.0	Denting Criteria	Riggs	NRR/DE/MTEB	135	3	06/30/91	NA
67.9.0	Reactor Coolant System Pressure Control	Riggs	NRR/DSI/GIB NRR/DSI/RSB	A-45, I.C.1 (2,3)	3	06/30/91	NA
67.10.0	Supplemental Tube Inspections	Riggs	NRR/DL/ORAB	LI (NOTE 5)	3	06/30/91	NA
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	Pittman	NRR/DSI/ASB	124	3	06/30/91	NA
69.	Make-up Nozzle Cracking in B&W Plants	Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70.	PORV and Block Valve Reliability	Riggs	RES/DE/EIB	NOTE 3(a)	3	06/30/91	
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Pittman	RES/DRA/ARGIB	LOW	1	06/30/90	NA
72.	Control Rod Drive Guide Tube Support Pin Failures	Riggs	RES	DROP	1	06/30/91	NA
73.	Detached Thermal Sleeves	Emrit	RES/DSIR/EIB	NOTE 3(a)	2	12/31/92	NA
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Emrit	RES/DRA/ARGIB	NOTE 3(a)	1	06/30/90	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
76.	Instrumentation and Control Power Interactions	Zimmerman	RES/DSIR/EIB	DROP	2	06/30/93	NA
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Rourk	RES/DSIR/EIB	MEDIUM	1	12/31/92	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	RES/DSIR/EIB	NOTE 3(b)	2	12/31/92	NA
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	V'Molen	NRR/DSI/RSB, ASB, CPB	LOW	1	06/30/91	NA
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	Rourk	RES/DSIR/EIB	LOW	3	12/31/92	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/89	NA
83.	Control Room Habitability	Emrit	RES/DRAA/SAIB	NOTE 1	1	12/31/86	
84.	CE PORVs	Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85.	Reliability of Vacuum Breakers Connected to Steam	Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	Discharge Lines Inside BWR Containments						
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	Pittman	RES/DSIR/EIB	NOTE 3(a)	1	12/31/91	
88.	Earthquakes and Emergency Planning	Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	Chang	RES/DSIR/EIB	LOW	1	06/30/93	NA
90.	Technical Specifications for Anticipatory Trips	V'Molen	NRR/DSI/RSB, ICSB	LOW		12/31/84	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	V'Molen	NRR/DSI/RSB, CPB	LOW		12/31/84	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	Pittman	RES/DSIR/RPSI	NOTE 3(a)		06/30/90	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	RES/DRA/ARGIB	NOTE 3(b)		06/30/90	NA
96.	RHR Suction Valve Testing	Milstead	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	V'Molen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	
100.	Once-Through Steam Generator Level	Jackson	RES/DSIR/EIB	DROP	1	12/31/91	NA
101.	BWR Water Level Redundancy	V'Molen	RES/DE/EIB	NOTE 3(b)	1	06/30/89	NA
102.	Human Error in Events Involving Wrong Unit or Wrong Train	Emrit	NRR/DLPQ/LPEB	NOTE 3(b)	2	12/31/88	NA
103.	Design for Probable Maximum Precipitation	Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	NA
104.	Reduction of Boron Dilution Requirements	Pittman	RES/DRA/ARGIB	DROP		12/31/88	NA
105.	Interfacing Systems LOCA at LWRs	Milstead	RES/DE/EIB	NOTE 3(b)	2	06/30/93	NA
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Milstead	RES/DRPS	MEDIUM		12/31/87	
107.	Main Transformer Failures	Milstead	RES/DRA/ARGIB	LOW	1	06/30/91	NA
108.	BWR Suppression Pool Temperature Limits	Colmar	NRR/DSI/CSB	RI (NOTE 3)		06/30/85	NA
109.	Reactor Vessel Closure Failure	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
110.	Equipment Protective Devices on Engineered Safety Features	Diab	RES/DSIR/EIB	DROP		12/31/92	NA
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	Riggs	NRR/DE/MTEB	LI (NOTE 5)	1	06/30/91	NA
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	Pittman	NRR/DSI/ICSB	RI (NOTE 3)		12/31/85	NA
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Riggs	RES/DSIR/EIB	NOTE 3(b)	1	12/31/92	NA
114.	Seismic-Induced Relay Chatter	Riggs	NRR/DSRO/SPEB	A-46	1	06/30/91	NA
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	Milstead	RES/DRPS/RPSI	NOTE 3(b)		06/30/89	NA
116.	Accident Management	Pittman	RES/DRA/ARGIB	S		06/30/91	NA
117.	Allowable Time for Diverse Simultaneous Equipment Outages	Pittman	RES/DRA/ARGIB	DROP		06/30/90	NA
118.	Tendon Anchorage Failure	Shaukat	RES/DSIR/EIB	NOTE 3(a)		12/31/92	NA

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Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
119.	Piping Review Committee Recommendations	-	-	-			
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	Riggs	NRR/DE	RI (NOTE 3)	2	06/30/93	NA
119.2	Piping Damping Values	Riggs	NRR/DE	RI (DROP)	2	06/30/93	NA
119.3	Decoupling the OBE from the SSE	Riggs	NRR/DE	RI (S)	2	06/30/93	NA
119.4	BWR Piping Materials	Riggs	NRR/DE	RI (NOTE 5)	2	06/30/93	NA
119.5	Leak Detection Requirements	Riggs	NRR/DE	RI (NOTE 5)	2	06/30/93	NA
120.	On-Line Testability of Protection Systems	Milstead	RES/DRA/ARGIB	NOTE 3(b)	1	06/30/93	NA
121.	Hydrogen Control for Large, Dry PWR Containments	Emrit	RES/DSIR/SAIB	NOTE 3(b)	1	12/31/92	NA
122.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Short-Term Actions						
122.1	Potential Inability to Remove Reactor Decay Heat	-	-	-			
122.1.a	Failure of Isolation Valves in Closed Position	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.1.b	Recovery of Auxiliary Feedwater	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.1.c.	Interruption of Auxiliary Feedwater Flow	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.2	Initiating Feed-and-Bleed	V'Molen	NRR/DEST/SRXB	NOTE 3(b)	3	06/30/91	NA
122.3	Physical Security System Constraints	V'Molen	NRR/DSRO/SPEB	LOW	3	06/30/91	NA
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	Milstead	RES/DSIR/SAIB	DROP		12/31/91	NA
124.	Auxiliary Feedwater System Reliability	Emrit	NRR/DEST/SRXB	NOTE 3(a)	3	06/30/91	
125.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Long-Term Actions	-	-	-			
125.I.1	Availability of the Shift Technical Advisor	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.2	PORV Reliability	-	-	-	6	12/31/89	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	V'Molen	NRR/DSRO/SPEB	70	6	12/31/89	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	V'Molen	NRR/DSRO/SPEB	70	6	12/31/89	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	V'Molen	NRR/DSRO/SPEB	A-45	6	12/31/89	NA
125.I.3	SPDS Availability	Milstead	RES/DRA/ARGIB	NOTE 3(b)	6	12/31/89	NA
125.I.4	Plant-Specific Simulator	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.7	Operator Training Adequacy	-	-	-			
125.I.7.a	Recover Failed Equipment	Pittman	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.7.b	Realistic Hands-On Training	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.1	Need for Additional Actions on AFW Systems	-	-	-			
125.II.1.a	Two-Train AFW Unavailability	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	124	6	12/31/89	NA
125.II.1.c	NUREG-0737 Reliability Improvements	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.4	Thermal Stress of OTSG Components	Riggs	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	
125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss of All Feedwater	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	V'Molen	RES/DRPS/RPSI	NOTE 3(b)	6	12/31/89	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.9	Enhanced Feed-and-Bleed Capability	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.10	Hierarchy of Impromptu Operator Actions	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.12	Adequacy of Training Regarding PORV Operation	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.13	Operator Job Aids	Pittman	NRR/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	V'Molen	NRR/DSRO/SPEB	LOW	6	12/31/89	NA
126.	Reliability of PWR Main Steam Safety Valves	Riggs	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
127.	Maintenance and Testing of Manual Valves in Safety-Related Systems	Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/91	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
130.	Essential Service Water Pump Failures at Multiplant Sites	Riggs	RES/DSIR/RPSIB	NOTE 3(a)	1	12/31/91	
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse-Designed Plants	Riggs	RES/DRA/ARGIB	S	1	06/30/91	NA
132.	RHR System Inside Containment	Su	RES/DSIR/SAIB	DROP		12/31/92	NA
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	Emrit	RES/DSIR/EIB	NOTE 3(b)	2	12/31/91	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
138.	Deinerting of BWR Mark I and II Containminants During Power Operations Upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	Milstead	RES/DSIR/SAIB	LOW		12/31/91	NA
139.	Thinning of Carbon Steel Piping in LWRs	Riggs	RES/DRA/ARGIB	RI (NOTE 3)		12/31/88	NA
140.	Fission Product Removal Systems	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	Milstead	RES/DSIR/EIB	NOTE 3(b)	2	06/30/93	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
143.	Availability of Chilled Water Systems and Room Cooling	Milstead	RES/DRA/ARGIB	HIGH		06/30/91	
144.	Scram Without a Turbine/Generator Trip	Hrabal	RES/DSIR/EIB	LOW		12/31/92	
145.	Actions to Reduce Common Cause Failures	Rasmuson	RES/DSIR/SAIB	NOTE 1		12/31/92	
146.	Support Flexibility of Equipment and Components	Chang	RES/DSIR/EIB	NOTE 4		(later)	
147.	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	Milstead	RES/DSIR/SAIB	LI (NOTE 5)		12/31/92	NA
148.	Smoke Control and Manual Fire-Fighting Effectiveness	[illegible]jekas	RES/DSIR/RPSIB	LI (NOTE 5)		12/31/92	NA
149.	Adequacy of Fire Barriers	Emrit	RES/DSIR/EIB	LOW		12/31/92	NA
150.	Overpressurization of Containment Penetrations	Milstead	RES/DSIR/SAIB	DROP		12/31/91	NA
151.	Reliability of Anticipated Transient Without SCRAM Recirculation Pump Trip in BWRs	Milstead	RES/DSIR/SAIB	NOTE 3(b)	2	12/31/92	NA
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	Emrit	RES/DSIR/EIB	LOW		06/30/93	NA
153.	Loss of Essential Service Water in LWRs	Riggs	RES/DRA/ARGIB	NOTE 3(b)		06/30/93	NA
154.	Adequacy of Emergency and Essential Lighting	Woods	RES/DSIR/SAIB	LOW		12/31/92	NA
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-		-	-
155.1	More Realistic Source Term Assumptions	Emrit	RES/DSIR/EIB	NOTE 2	1	06/30/93	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	Emrit	RES/DSIR/EIB	RI (NOTE 5)	1	06/30/93	NA
155.3	Improve Design Requirements for Nuclear Facilities	Emrit	RES/DSIR/EIB	DROP	1	06/30/93	NA
155.4	Improve Criticality Calculations	Emrit	RES/DSIR/EIB	DROP	1	06/30/93	NA
155.5	More Realistic Severe Reactor Accident Scenario	Emrit	RES/DSIR/EIB	DROP	1	06/30/93	NA
155.6	Improve Decontamination Regulations	Emrit	RES/DSIR/EIB	DROP	1	06/30/93	NA
155.7	Improve Decommissioning Regulations	Emrit	RES/DSIR/EIB	DROP	1	06/30/93	NA
156.	Systematic Evaluation Program	-	-	-		-	-
156.1.1	Settlement of Foundations and Buried Equipment	Chang	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.1.2	Dam Integrity and Site Flooding	Chen	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	Chen	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.1.4	Industrial Hazards	Ferrell	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.1.5	Tornado Missiles	Chen	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.1.6	Turbine Missiles	Emrit	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.2.1	Severe Weather Effects on Structures	Chen	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.2.2	Design Codes, Criteria, and Load Combinations	Kirkwood	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.2.3	Containment Design and Inspection	Shaukat	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.2.4	Seismic Design of Structures, Systems, and Components	Chen	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.1.1	Shutdown Systems	Woods	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.1.2	Electrical Instrumentation and Controls	Woods	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.2	Service and Cooling Water Systems	Su	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.3	Ventilation Systems	Burdick	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.4	Isolation of High and Low Pressure Systems	Burdick	RES/DSIR/SAIB	DROP	2	06/30/93	NA
156.3.5	Automatic ECCS Switchover	Milstead	RES/DSIR/SAIB	24	2	06/30/93	NA
156.3.6.1	Emergency AC Power	Emrit	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.3.6.2	Emergency DC Power	Rourk	RES/DSIR/EIB	LOW	2	06/30/93	NA
156.3.8	Shared Systems	Emrit	RES/DSIR/EIB	DROP	2	06/30/93	NA
156.4.1	RPS and ESFS Isolation	Emrit	RES/DSIR/EIB	142	2	06/30/93	NA
156.4.2	Testing of the RPS and ESFS	Chang	RES/DSIR/SAIB	120	2	06/30/93	NA
156.6.1	Pipe Break Effects on Systems and Components	Page	RES/DSIR/EIB	NOTE 4		(later)	
157.	Containment Performance	Shaperow	RES/DSIR/SAIB	NOTE 3(b)		12/31/92	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
158.	Performance of Power-Operated Valves Under Design Basis Conditions	Cheh	RES/DSIR/SAIB	NOTE 4		(later)	
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	Cheh	RES/DSIR/SAIB	NOTE 4		(later)	
160.	Spurious Actions of Instrumentation Upon Restoration of Power	Chang	RES/DSIR/EIB	NOTE 4		(later)	
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	Rourk	RES/DSIR/EIB	DROP		06/30/93	NA
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit is Shut Down	Cheh	RES/DSIR/SAIB	NOTE 4		(later)	
163.	Multiple Steam Generator Tube Leakage	Burdick	RES/DSIR/SAIB	NOTE 4		(later)	
164.	Neutron Fluence in Reactor Vessel	Emrit	RES/DSIR/EIB	DROP		06/30/93	NA
165.	Safety and Safety/Relief Valve Reliability	Hrabal	RES/DSIR/EIB	NOTE 4		(later)	
166.	Adequacy of Fatigue Life of Metal Components	Emrit	RES/DSIR/EIB	NOTE 1		06/30/93	
167.	Combustible Gas Storage Facilities	TBD	RES/DSIR	NOTE 4		(later)	
168.	Environmental Qualification of Electrical Equipment	Emrit	RES/DSIR/EIB	NOTE 1		06/30/93	
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF2	TRAINING						
HF2.1	Evaluate Industry Training	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	3	06/30/91	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	3	06/30/91	NA
HF4.3	Criteria for Safety-Related Operator Actions	Pittman	NRR/DHFT/HFIB	B-17	3	06/30/91	NA
HF4.4	Guidelines for Upgrading Other Procedures	Pittman	RES/DRPS/RHFB	HIGH	3	06/30/91	NA
HF4.5	Application of Automation and Artificial Intelligence	Pittman	NRR/DHFT/HFIB	HF5.2	3	06/30/91	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	Pittman	RES/DRPS/RHFB	NOTE 3(b)	2	06/30/93	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	Pittman	RES/DRPS/RHFB	NOTE 3(b)	2	06/30/93	NA
HF5.3	Evaluation of Operational Aid Systems	Pittman	NRR/DHFT/HFIB	HF5.2	2	06/30/93	NA
HF5.4	Computers and Computer Displays	Pittman	NRR/DHFT/HFIB	HF5.2	2	06/30/93	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
		CHERNOBYL ISSUES					
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls to Ensure That Procedures Are Followed and That Procedures Are Adequate	-	-				
CH1.1A	Symptom-Based EOPs	Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)		06/30/89	NA
CH1.1B	Procedure Violations	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.2	Approval of Tests and Other Unusual Operations	-	-				
CH1.2A	Test, Change, and Experiment Review Guidelines	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.2B	NRC Testing Requirements	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.3	Bypassing Safety Systems	-	-				
CH1.3A	Revise Regulatory Guide 1.47	Emrit	RES/DE/EMEB	LI (NOTE 5)		06/30/89	NA
CH1.4	Availability of Engineered Safety Features	-	-				
CH1.4A	Engineered Safety Feature Availability	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	Emrit	[illegible]SR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Lead Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				
CH3.2A	Filtered Venting	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				
CH4.3A	Ingestion Pathway Protective Measures	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation	-	-				
CH4.4A	Decontamination	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4B	Relocation	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH5	SEVERE ACCIDENT PHENOMENA						
CH5.1	Source Term	-	-				
CH5.1A	Mechanical Dispersal in Fission Product Release	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.1B	Stripping in Fission Product Release	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.2	Steam Explosions	-	-				
CH5.2A	Steam Explosions	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.3	Combustible Gas	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6	GRAPHITE-MODERATED REACTORS						
CH6.1	Graphite-Moderated Reactors	-	-				
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.1B	Structural Graphite Experiments	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.2	Assessment	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA

TABLE III

SUMMARY OF THE PRIORITIZATION OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, HUMAN FACTORS ISSUES, AND CHERNOBYL ISSUES

Legend

NOTES:

1 - Possible Resolution Identified for Evaluation
2 - Resolution Available
3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
4 - Issues to be Prioritized in the Future
5 - Issues that are not GSIs but Should be Assigned Resources for Completion

DROP - GSI Dropped from Further Pursuit
EI - Environmental Issue
GSI - Generic Safety Issue
HIGH - High Safety Priority
I - TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI - Licensing Issue
LOW - Low Safety Priority
MEDIUM - Medium Safety Priority
RI - Regulatory Impact Issue
USI - Unresolved Safety Issue

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	COVERED IN OTHER ISSUES	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
			NOTE 1	NOTE 2	NOTE 3								
TMI ACTION PLAN ITEMS (369)													
GSI	87	46	1	0	129	0	1	1	12	9	-	-	286
LI	-	0	-	-	74	-	-	-	-	-	-	9	83
TASK ACTION PLAN ITEMS (142)													
USI	-	-	-	-	27	0	-	-	-	-	-	-	27
GSI	-	20	0	1	32	-	0	3	3	11	0	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (246)													
GSI	-	54	4	1	57	0	3	4	17	78	9	-	227
RI	-	-	-	-	4	-	-	-	-	1	-	5	10
LI	-	-	-	-	3	-	-	-	-	-	-	6	9
HUMAN FACTORS (27)													
GSI	-	8	0	0	7	0	1	0	0	0	-	-	16
LI	-	-	-	-	3	-	-	-	-	-	-	8	11
CHERNOBYL ISSUES (32)													
LI	-	2	-	-	7	-	-	-	-	-	-	23	32
TOTAL:	87	130	5	2	373	0	5	8	32	99	9	66	816

TABLE IV

LISTING OF AEOD REPORTS AND RELATED GENERIC ISSUES

This listing shows all AEOD reports that have been addressed either as completely new safety issues or as part of existing safety issues. It should be noted that, in some cases, more than one AEOD report has been generated on a single topic. However, all AEOD reports related to the identified safety issues are listed alphanumerically including those that have been superseded by other AEOD reports. The following is a description of the types of AEOD reports:

C - Reactor Case Study
E - Reactor Engineering Evaluation
S - Special Study Report
T - Technical Review Report

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C001	Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980	41	-
C003	Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2	47	-
C004	AEOD Actions Concerning the Crystal River 3 Loss of Non-Nuclear Instrumentation and Integrated Control System Power on February 26, 1980	33	E122
C005	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	37, 42	-
C101	Report on the Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980	31	-
C102	H. B. Robinson Reactor Coolant System Leak on January 29, 1981	34	-
C103	AEOD Safety Concerns Associated with Pipe Breaks in the BWR Scram System	40	-
C104	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	46	-
C105	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	36	-
C201	Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors	50, 101	-

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C202	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	32	E016
C203	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	54	E305
C204	San Onofre Unit 1 Loss of Salt Water Cooling Event of March 10, 1980	54	-
C205	Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	56	-
C301	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	55	-
C401	Low Temperature Overpressure Events at Turkey Point Unit 4	94	E426
C403	Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982	85	E322
C404	Steam Binding of Auxiliary Feedwater Pumps	93	E325
C501	Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	106	E902
C503	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	99	-
C701	Air Systems Reliability	43	E123
E002	BWR Jet Pump Integrity	12	-
E005	Operational Restrictions for Class 1E 120 VAC Vital Instrument Buses	48	-
E007	Potential for Unacceptable Interaction Between the Control Rod Drive System and Non-Essential Control Air System at the Browns Ferry Plant	39	-
E010	Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant, Units 1 and 2	49	-
E011	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	38	-
E016	Flow Blockage in Essential Equipment at ANO Caused by _Corbicula_ sp. (Asiatic Clams)	32	C202
E101	Degradation of Internal Appurtenances in LWR Piping	35	-
E112	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E226
E122	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of ICS/NNI Power	33	C004
E123	Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines	43	C701

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
E204	Effects of Fire Protection System Actuation on Safety-Related Equipment	57	-
E209	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback 1 on 4/13/79)	30	-
E215	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	52	-
E226	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E112
E304	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	77	-
E305	Inoperable Motor-Operational Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	54	C203
E322	Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting	85	C403
E325	Vapor Binding of Auxiliary Feedwater Pumps at Robinson 2	93	C404
E414	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	105	-
E417	Loosening of Flange Bolts on RHR Heat Exchanger Leading to Primary to Secondary Side Leakge	C-9	-
E426	Single Failure Vulnerability of Power Operated Relief Valve (PORV) Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)	94	C401
E609	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	129	-
E804	Reliability of Non-Safety-Related Field Breakers During ATWS Events	151	-
E807	Pump Damage Due to Low Flow Cavitation	159	-
S401	Human Error in Events Involving Wrong Unit or Wrong Train	102	-
T302	Postulated Loss of Auxiliary Feedwater System Resulting from a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	68	-
T305	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Instrusion at Sequoyah 1	51	-
T420	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	87	-

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TABLE V

SUMMARY OF CONSOLIDATED GENERIC ISSUES

This table shows the consolidation of those issues whose technical concerns were found to be addressed either partially or completely in other (major issues). The table reflects the findings of the prioritization process that are summarized in Table II.

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues				
TMI ACTION PLAN ITEMS						
I.A.1.3	I	II.K.3(53)				
I.A.2.2	NOTE 3(b)	I.A.2.6(3) [II.K.3(56)]				
I.A.2.6(1)	NOTE 3(a)	I.B.1.1.(6),	I.B.1.1(7),	HF3.4		
I.A.3.1	I	II.K.3(56)				
I.A.4.1(2)	NOTE 3(a)	II.K.3(54)				
I.A.4.2(4)	NOTE 3(a)	HF3.3				
I.B.1.1 (1,2,3,4)	NOTE 3(b)	II.J.3.1,	II.J.3.2,	II.K.3(52),	HF6.1,	HF6.2
I.C.1	-	8, 67.4.3,	18,	31,	42,	67.3.1,
I.C.1(2)	I	37	67.9.0			
I.C.1(3)	I	II.K.2(12), II.K.3(37), II.K.3(47),	II.K.2(18), II.K.3(38), II.K.3(55),	II.K.3(6), II.K.3(39), 67.9.0	II.K.3(35), II.K.3(41),	II.K.3(36), II.K.3(42),
I.C.[illegible]	I	II.K.3(52)				
I.C.5	I	II.K.3(52)				
I.C.7	I	II.K.3(50)				
I.C.8	I	II.K.3(49)				
I.C.9	NOTE 3(b)	II.K.3(49),	II.K.3(50)	II.K.3(51)		
I.D.1	I	56,	67.4.2			
I.D.2	I	II.K.3(23),	II.K.3(55)			

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues			
I.D.3	MEDIUM	II.K.3(55)			
I.F.1	NOTE 3(b)	5			
II.B.8	NOTE 3(a)	II.B.7			
II.C.1	NOTE 3(b)	II.K.3(4),	II.K.3(8),	II.K.3(33),	II.K.3(48)
II.C.2	NOTE 3(b)	II.K.3(4),	II.K.3(48)		
II.E.1.1	I	II.K.2(8)			
II.E.1.2	I	II.K.2(8)			
II.E.2.2	NOTE 3(b)	II.K.3(32),	II.K.3(34),	II.K.3(47)	
II.E.6.1	NOTE 3(a)	54			
II.F.2	I	II.K.3(6),	67.3.4		
II.F.3	NOTE 3(a)	II.K.3(6),	A-34		
II.H.2	HIGH	II.H.3			
II.K.2(15)	I	II.K.3(43)			
II.K.2(16)	I	II.K.3(40)			
II.K.3(5)	I	9,	67.4.1		
II.K.3(17)	I	II.E.2.1[II.K.3(26)]			
III.A.1.2(1)	I	II.K.3(23)			
III.A.3.1	NOTE 3(b)	B-71			
III.A.3	-	67.6.0			
III.A.3.4	NOTE 3(b)	II.K.3(23)			
III.D.1.1(1)	I	B-69			
III.D.2.1	LOW	B-67			
III.D.2.5	NOTE 3(b)	III.D.2.2(2),	III.D.2.2(3),	III.D.2.2(4)	
III.D.3.1	NOTE 3(b)	B-34,	97		

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues
V.A.1	LI (NOTE 3)	II.A.2
TASK ACTION PLAN ITEMS		
A-2	NOTE 3(a)	B-52
A-12	NOTE 3(a)	60
A-17	NOTE 3(b)	II.C.3[II.K.3(4)], C-13, 77
A-18	DROP	B-16
A-37	DROP	A-32, 11
A-38	LOW	A-32
A-40	NOTE 3(a)	B-51
A-43	NOTE 3(a)	B-18, C-3
A-44	NOTE 3(a)	B-57
A-45	NOTE 3(b)	II.E.3.2[B-4], II.E.3.3[II.K.3(8)], II.E.3.5, 67.9.0, 125.I.2.d
A-46	NOTE 3(a)	B-24 114
A-47	NOTE 3(a)	19, 33, 37, 56, 67.3.1
A-48	NOTE 3(a)	B-14
A-49	NOTE 3(a)	28, 67.3.2
B-2	EI (NOTE 3)	B-45
B-17	MEDIUM	27, HF4.3
B-68	DROP	A-32
C-8	NOTE 3(b)	16
C-12	NOTE 3(b)	B-73

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues				
NEW GENERIC ISSUES						
17	DROP	26				
23	HIGH	65				
25	NOTE 3(a)	39				
29	HIGH	62				
43	NOTE 3(a)	44				
51	NOTE 3(a)	32,	52			
67.5.2	LI	36				
70	NOTE 3(a)	125.I.2.a,	125.I.2.b			
75	NOTE 3(a)	I.B.1.1(6),	I.B.1.1(7)			
76	NOTE 4	46				
83	NOTE 1	CH 2.3A				
105	HIGH	96				
119.1	RI(NOTE 3)	B-6				
124	NOTE 3(a)	68,	122.1.a,	122.1.b,	122.1.c,	122.II.1.b
128	HIGH	48,	49,	A-30		
135	NOTE 3(b)	67.2.1	67.7.0	67.8.0		
153	HIGH	B-32				
HUMAN FACTORS ISSUES						
HF5.2	HIGH	HF4.5,	HF5.3,	HF5.4		

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues
CHERNOBYL ISSUES		
CH1.4	LI(NOTE 5)	CH2.2

ITEM B-56: DIESEL RELIABILITY

DESCRIPTION

Historical Background

This issue was documented in NUREG-0471[3] and resulted from a review of LERs which indicated that onsite emergency diesel generators (EDGs) at operating plants were demonstrating an average starting reliability of about 0.94/demand. The goal for new plants, as expressed in Regulatory Guide 1.108,[216] was a starting reliability of 0.99/demand. The NRC awarded a contract to the University of Dayton Research Institute to identify the more significant causes of EDG unreliability. The Dayton University study was completed and the significant causes and recommended corrective actions were identified in NUREG/CR-0660.[217]

Safety Significance

Events (offsite and onsite) which result in a loss of offsite power necessitate reliance on the onsite EDGs for successful accident mitigation. Improvement of the starting reliability of onsite EDGs will reduce the probability of events which could escalate into a core-melt accident and thus could effect an overall reduction in public risk.

Possible Solution

The staff proposed a set of interim backfit requirements for operating plants that encompassed elements of Regulatory Guide 1.108[216] and the Dayton University recommendations.[217] These requirements were included in a proposed program[218] to establish a graded set of requirements based on the reliability actually exhibited by EDGs. This program adopted an EDG startup reliability of 0.95/demand as the minimum desired reliability and 0.9/demand as the minimum acceptable level of reliability. At or below the minimum desired level, licensees would be required to improve their EDG reliability and document their program for doing so. Below the minimum acceptable level, licensees would be required to improve or repair EDGs with reliability below the minimum acceptable level and perform a requalification program to demonstrate that the causes of the failures were corrected. The requalification program was intended to pass EDGs only if the reliability had been increased to 0.95/demand or greater.

The proposed interim program imposed a normal surveillance period of no more than 1 month. To increase assurance that a real change in reliability will be detected quick' , an increased test frequency was required when two or more failures had been experienced on an individual EDG in the previous 20 demands. However, the frequency of tests and the anticipated duration of the accelerated test frequency were not as restrictive as that recommended in Regulatory Guide 1.108.[216]

An extended out-of-service period could, in many cases, be necessary to allow sufficient time to correct the problems that caused low reliabilities. Therefore, the proposed program would allow out-of-service periods in excess of the existing 72-hour limit, when necessary, while at the same time placing a yearly limit on the cumulative time that a plant may operate in Modes 1 through 4 with one of the

EDGs of the power systems inoperable. The cumulative limit would vary depending upon the reliability of the in-service EDG with the lowest reliability.

PRIORITY DETERMINATION

A risk analysis was performed[64] using Oconee 3 and Grand Gulf Unit 1 as representative of PWRs and BWRs, respectively. Since the proposed position was expected to affect only those EDGs that had demonstrated a reliability of less than 0.95/demand, it was assumed that 25% of the EDG population would undergo a reliability improvement from 0.93 to 0.97/demand and 5% would undergo a reliability improvement from 0.9 to 0.97/demand (requalification).

Frequency Estimate

When the frequency of all core-melt scenarios (including EDG failure) was adjusted to include the above assumptions, it was found that the proposed solution would be expected to result in a significant core-melt frequency reduction for both the 25% EDG population and the 5% EDG population. The 25% EDG population, which was assumed to improve from 0.93/demand to 0.97/demand, would have core-melt frequency reductions of 1.7×10^{-5}/RY and 2.3×10^{-5}/RY for BWRs and PWRs, respectively. The 5% EDG population, which was assumed to improve from 0.9/demand to 0.97/demand, would have core-melt frequency reductions of 3.7×10^{-5}/RY and 7.5×10^{-5}/RY for BWRs and PWRs, respectively.

Consequence Estimate

Base case risk for both PWRs and BWRs was calculated by multiplying the expected frequency of each release category by the dose equivalent value for the category. Adjusted case risk was determined by the same technique using the core-melt frequency reduction calculated for the reliability improvement expected in the respective EDG populations (25% and 5%) for both PWRs and BWRs. The adjusted risk was subtracted from the base case risk and the public risk reduction obtained was multiplied by the appropriate number of PWRs and BWRs. The total public risk reduction calculated was 6.5×10^4 man-rem, with an average public risk reduction of about 1.5×10^3 man-rem/reactor.

Cost Estimate

Industry Cost: It was assumed that 30% of the 143 expected plants would institute a reliability improvement program. In addition, 5% of the plants were assumed to incur a major equipment (EDG) replacement and an associated loss of power production. Industry costs were estimated for revision of operating procedures and personnel training, installation of additional equipment (air dryers, dust-tight enclosures for electrical contacts, EDG room ventilation ducting, etc.) and ongoing increases in operation and maintenance costs. Thus, the total industry cost was estimated to be $46M.

NRC Cost: The cost to complete resolution of the issue, review and approve new requirements, and issue implementation orders was estimated to be $130,000. Review of plant responses to orders and periodic reports expected from plants which must develop and initiate EDG reliability improvement programs and long-term surveillance of the industry was estimated at $1M. Thus, total NRC cost was estimated to be $1.1M.

Total Cost: The total industry and NRC cost associated with the solution to this issue was \$(46 + 1.1)M or \$47.1M.

Value/Impact Assessment

Based on a potential public risk reduction of 6.5×10^4 man-rem and a cost of \$47.1M, the value/impact score was given by:

$$S = \frac{6.5 \times 10^4 \text{ man-rem}}{\$47.1\text{M}}$$

$$= 1{,}380 \text{ man-rem/\$M}$$

Other Considerations

An unusually significant avoided accident cost was calculated for the resolution of this issue. This cost represented the expected savings to the industry from lowering the core-melt probability by implementation of a specific improvement and was calculated by multiplying the expected cost of the loss of a plant (~\$3 Billion) by the expected total core-melt frequency reduction. In this instance, the avoided accident cost (savings to the industry) was estimated to be \$30M.

CONCLUSION

The calculated value/impact score was indicative of a medium priority assignment; however, other factors prevailed. The very large estimated total public risk reduction (6.5×10^4 man-rem) and high expected core-melt frequency reduction ($>10^{-5}$/RY) elevated the priority of this issue. In addition, if the averted accident cost (industry savings) were subtracted from the total resolution cost, a value/impact score of 3,800 man-rem/\$M would result. Therefore, the issue was given a high priority ranking.

The issue was resolved by the inclusion of guidance on EDG reliability in Regulatory Guide 1.160[1484] which was issued as part of the Maintenance Rule (10 CFR 50.65). This guide endorsed NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which addressed the optimization of EDG reliability and availability and contained an example of an acceptable means of establishing performance criteria and/or goals for EDGs. In addition, Regulatory Guide 1.9,[1483] Rev. 3 was issued to integrate into a single document pertinent guidance previously addressed in the following documents: Regulatory Guide 1.9,[1483] Rev. 2; Regulatory Guide 1.108,[216] Rev. 1; and Generic Letter 84-15.[1487] As a result, Regulatory Guide 1.108,[216] Rev. 1 was withdrawn.[1485] Thus, this issue was RESOLVED and new requirements were established.[1486]

REFERENCES

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

64. 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.

216. Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1976, (Rev. 1) August 1977.

217. NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel Generator Reliability," U.S. Nuclear Regulatory Commission, February 1979.

218. Memorandum for D. Eisenhut, et al., from S. Hanauer, "Diesel Generator Reliability of Operating Plants," May 6, 1982.

1483. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," March 1971, (Rev. 1) November 1978, (Rev. 2) December 1979, (Rev. 3) July 1993.

1484. Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1993.

1485. Federal Register Notice 58 FR 41813, "Regulatory Guide; Withdrawal," August 5, 1993.

1486. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue B-56, 'Diesel Generator Reliability,'" June 29, 1993.

1487. NRC Letter to All Licensees of Operating Reactors, Applicants for An Operating License, and Holders of Construction Permits, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability (Generic Letter 84-15)," July 2, 1984. [Accession# 8407020206]

ISSUE 76: INSTRUMENTATION AND CONTROL POWER INTERACTIONS

DESCRIPTION

Historical Background

This issue was identified[524] when a number of concerns regarding DC power systems were raised during the review of the proposed resolution of Issue A-30, "Adequacy of Safety-Related DC Power Supplies." The main concerns were:

(1) An instrumentation and control (I&C) power supply fault can cause a critical challenge to standby ESFs, i.e., cases including reactor trips, loss of main feedwater, loss of offsite power, and/or small LOCA through a failed-open PORV.

(2) The same I&C power supply fault could defeat some of the ESFs called upon to mitigate the initiating event, both core cooling systems and containment cooling systems.

(3) The same I&C power supply fault could blind or partially blind the operators to the status of the plant.

I&C electric power systems include AC and DC systems which provide control and motive power to several vital and non-vital components. These components include instrumentation and controls, emergency diesel generator controls, solenoid valves, and breaker controls. Many of these components are required to operate under abnormal and accident conditions. Large-capacity batteries are a typical electric power system component which provide electric power to the DC components. Battery chargers are provided to ensure that an adequate charge is maintained. Inverters are used to convert the DC to AC in order to provide continuous power to vital equipment during offsite AC power interruptions.

Operating experience[1428] has indicated that failures in these I&C systems have occurred at a significant frequency and a number of these failures have had potential safety implications. Potentially significant events include loss of DC power supplies for one hour, partial and total losses of normal and emergency AC power, loss of control room annunciators, control system malfunctions, reduction or loss of feedwater, and a variety of inadvertent valve actuations. The impact of these failures has ranged from minimal effects on plant operation to reactor trips with complications. Most notable is the event at Nine Mile Point in August 1991.[1429] The simultaneous loss of five uninterruptible power sources was unexpected and presented unique challenges to both equipment and personnel. Fixes that have been implemented to prevent recurrence of these events include modifications to operating procedures, changes to technical specifications, and repair or replacement of failed components. The evaluation of this issue included consideration of Issue 46, "Loss of 125 Volt DC Bus."

Safety Significance

The operating events that have occurred have been typically recoverable in a short period of time. However, the effects of the power failures may result in transients involving a series of multiple, propagating interactions that may lead to adverse conditions that are not readily reversible or correctable. This issue affected all operating and future plants.

Possible Solutions

Resolving this issue could require actions to increase the reliability of power systems. One method is to require additional sources and divisions of electric power which would involve a major hardware modification for some plants. For example, presently there are plants already equipped with four divisions of vital AC and DC power. Other possible solutions could include new testing, increase existing test frequencies, improve preventive maintenance and/or better operating procedures.

PRIORITY DETERMINATION

This issue affected 90 PWRs and 44 BWRs with average remaining lives of 28.8 and 27.4 years, respectively. This analysis was performed for Grand Gulf 1 (BWR) and scaled to Oconee 3 (PWR) using the scaling relationships given in NUREG/CR-2800.[64] The primary focus of the analysis was on DC power systems. Two situations involving DC power losses were analyzed separately and the results combined; one involved DC power failures as initiating events and the other involved DC power failures as contributing events.

Assumptions

The Grand Gulf 1 PRA includes DC power system failure as a contributing event. The analysis of this issue required added assumptions about DC power system failures as initiating events.

It was assumed that undervoltage and undercurrent events can have the same consequences as a sudden loss of power. This assumption was supported by LER data reviewed from the 1984 to 1990 time period which involved DC system failure. For example, an undervoltage can result in a main feedwater trip. The transient and resultant reactor trip are similar to a sudden loss of main feedwater. In analyzing the LER data, the undervoltage and undercurrent events were assumed to be failures of the affected equipment.

It was assumed that overvoltage and overcurrent events are recoverable because of the protective devices on the equipment. Unless the protective devices fail, the equipment will not be damaged and can be returned to service (if lost); the LER data from 1984 to 1990 supported this assumption.

The frequency of DC power system failures, using the above assumptions and the data from the LERs, was the basis for improving the adjusted case. The possible solution was assumed to increase the reliability of DC power systems, based on battery failure rate distributions given in the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR).[1327] The error factors given in NUCLARR for 7 battery failure rate data points ranged from 2 to about 8. The average of these data points was 4.76. The solution was conservatively estimated to reduce the frequency of battery failures by a factor of 3.[1327]

Using a station blackout analysis, an event tree was constructed with the loss of DC power as the initiating event. The loss of AC power was assumed to be independent of a loss of the DC power system. The emergency AC power reliability was assumed to be representative of a single, failed diesel in a two-diesel generator system; the probability of recovery of AC power within one hour was estimated to be 0.55.[89D]

If AC power is available, it was assumed that RCP seal cooling is available and an RCP seal LOCA is not likely to occur. However, the subsequent transient is likely to result in an increase in primary coolant system pressure and temperature. The potential exists for a LOCA to be caused by a stuck-open safety relief valve. The AC power recovery time to prevent core damage from a stuck-open relief valve is 1 to 2 hours. If AC power is not available, there is a significant probability that a RCP seal LOCA will occur. The AC power recovery time to prevent core damage from a RCP seal LOCA depends on the size of the LOCA. If RCP seal leakage is large (more than 100 gpm/pump), the core could be uncovered within a few hours. Smaller leak rates (a few gpm/pump) are not a limiting factor.[89D] Issue 23, "Reactor Coolant Pump Seal Failures," showed a probability of leak rates of 480 gpm/pump, which would reduce the recovery time significantly.

<u>Frequency Estimate</u>

<u>DC Power Failure - Initiating Events</u>: To estimate the reduction in core-melt frequency, a search of LERs from 1984 through 1990 was made using the key words "DC power" and "station battery." Only those LERs that had safety significance were considered. From this LER data, the base case value for the frequency of DC power failures and subsequent reactor trip as an initiating event was estimated to be 0.06/RY. Based on the Grand Gulf 1 PRA, the frequency of this initiating event leading to core-melt was calculated to be 6 x 10^{-7}/RY. The adjusted case was then calculated based on a factor of 3 reduction in initiating event frequency, resulting in a core-melt frequency of 2 x 10^{-7}/RY.

<u>DC Power Failure - Contributing Events</u>: DC power system failures as contributing events are represented in the Grand Gulf 1 PRA by events BATA and BATB. The base case failure probabilities for both these events are 0.001. The base case core damage frequency for Grand Gulf was 4.9 x 10^{-7}/RY and the adjusted case was calculated to be 1.6 x 10^{-7}/RY, based on a factor of 3 improvement in the unreliability of the batteries and DC system. Combining these 2 sets of events results in a base case core-melt frequency of 1.1 x 10^{-6}/RY and an adjusted case core-melt frequency of 3.6 x 10^{-7}/RY. Subtracting the adjusted case from the base case yields a reduction in core-melt frequency of 7.4 x 10^{-7}/RY for BWRs.

The PWR values of core-melt frequency were arrived at by scaling the BWR values and resulted in an estimated base case core-melt frequency of 2.4 x 10^{-6}/RY and an adjusted core-melt frequency of 8 x 10^{-7}/RY. The reduction in core-melt frequency then is 1.6 x 10^{-6}/RY for the PWR.

<u>Consequence Estimate</u>

For BWRs, the core-melt frequency reduction of 7.4 x 10^{-7}/RY translated to a public risk reduction of 2.1 man-rem/RY. For 44 BWRs with an average remaining life of 27.4 years, the estimated public risk reduction was 2,532 man-rem. For PWRs, the core-melt frequency reduction of 1.6 x 10^{-6}/RY translated to a public risk reduction of 1.7 man-rem/RY. For 90 PWRs with an average remaining life of 28.8 years, the estimated public risk reduction was 4,406 man-rem. Thus, the total potential public risk reduction associated with this issue was approximately 7,000 man-rem.

Cost Estimate

Industry Cost: All plants will need to prepare a FMEA of their power systems and will have to: (1) revise TS; (2) rewrite operating procedures; and (3) train operators. At a cost of $99,000/plant, the cost for these changes will be $13.3M. In addition, it was estimated that 27 plants with particularly unreliable DC power systems would require hardware modifications. These plant modifications were estimated to cost $275,000/plant for a total of $7.4M.

The TS changes were assumed to increase the power inspection/tests. The annual cost necessary for operating and maintaining the proposed solution was assumed to include approximately 48 man-hours/RY. This estimate included periodic retraining as well as additional time required to perform more surveillance tests on the batteries. This estimated annual cost was $2,724/RY. For all 134 plants with an average remaining life of 28.3 years, the cost was $10.3M.

NRC Cost: One man-year of contractor effort was estimated for reviewing and updating existing data, determining the feasibility of the possible solution, and developing a technical findings document. NRC technical oversight was estimated at 0.1 man-year. A value/impact and backfit analysis was estimated at $75,000. At a cost of $100,000/man-year, the total development costs were estimated at $0.185M.

NRC review of the FMEA and TS revisions was estimated at 0.5 man-week/plant. At a cost of $2270/man-week, the total estimated cost was $0.15M for all 134 plants. Reviewing the hardware modifications was estimated to require 2 man-weeks/plant. Since hardware modifications will be only required on the 27 plants with unreliable DC power systems, at a cost of $2270/man-week, these reviews will cost $0.123M. The total NRC cost to support implementation was estimated to be $0.273M.

The NRC support cost for operation and maintenance for plants requiring hardware modifications was estimated at 0.5 man-week/RY. Since the 27 plants had an estimated remaining life of 28.3 years, the total NRC operation and maintenance support cost was estimated to be $0.867M.

Total Cost: The total industry and NRC cost associated with the possible solution to this issue was $32.3M.

Value/Impact Assessment

Based on an estimated public risk reduction of 7,000 man-rem and a resolution cost of $32.2M, the value/impact score was given by:

$$S = \frac{7{,}000 \text{ man-rem}}{\$32.3\text{M}}$$

$$= 217 \text{ man-rem/\$M}$$

Other Considerations

Additional Public Risk Attributed to Vital AC Power Losses: The reduction in core-melt frequency and resultant risk was estimated while focusing on the DC portion of the issue. Inclusion of the vital AC portion would tend to raise the risk reduction and therefore the issue priority.

Other Related Actions: Issue 128, "Electrical Power Reliability," combined a number of electrical power issues and considered a number of related issues and actions. Three specific issues are A-30, "Adequacy of Safety-Related DC Power Supplies"; 48, "LCOs for Class 1E Vital Instrument Buses"; and 49, "Interlocks and LCOs for Class 1E Tie Breakers." With the resolution of Issue 128 and other issues, a number of actions have been taken or are underway that could have a significant impact (i.e., lower the assumed safety benefit) on the possible resolution of Issue 76 and, therefore, lower its priority.

IPE: One preliminary result[1430] from a plant IPE indicated that certain power system faults/failures can be a large contributor to a core-melt. In this instance, the unbalanced nature of the loads contributed to the significance of the postulated events. This would tend to increase the priority of the issue.

Life Extension: The remaining life of the plants used to calculate the value/impact score was based on the assumption that the total operating life of nuclear power plants was limited to 40 years. If the potential for license extension is considered, this would result in a higher score. For example, if it were assumed that 75% of the plants had their licenses extended for 20 years, the value/impact score would have increased to about 260 man-rem/$M.

CONCLUSION

The preliminary results[1430] from an IPE indicated that certain power system faults/failures can be a large contributor to core-melt probability. Although the potential risk reduction calculated above would place this issue in the medium priority category, it was concluded that the safety concern will be addressed more directly on a plant-specific basis in the IPE program. Therefore, this issue was DROPPED from further pursuit as a new and separate issue.

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524. Memorandum for R. Mattson from T. Speis, "Draft CRGR Package on A-30, DC Power," May 24, 1983.

890. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1988.

1327. NUREG/CR-4639, "Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)," U.S. Nuclear Regulatory Commission, (Volume 1) February 1988, (Volume 2) September 1988, (Volume 3) November 1988, (Volume 4) June 1988, (Volume 5) June 1988.

1428. NUREG/CR-4470, "Survey and Evaluation of Vital Instrumentation and Control Power Supply Events," U.S. Nuclear Regulatory Commission, August 1986.

1429. NUREG-1455, "Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1991," U.S. Nuclear Regulatory Commission, October 1991.

1430. Memorandum for T. Martin et al., from T. Murley, "Preliminary Results from Individual Plant Examinations (IPE)," April 22, 1991.

ISSUE 89: STIFF PIPE CLAMPS

DESCRIPTION

Historical Background

This issue was identified[1447] following a staff evaluation of allegations that improper consideration of "stiff" pipe clamps in Class 1 piping systems could result in unsafe plant operation. IE Information Notice No. 83-80[1448] was issued to alert OLs and CPs of this concern. In the staff's evaluation, it was found that piping designers often assumed that the clamp effects on piping systems were negligible and did not warrant any explicit consideration. This assumption was acceptable for most clamp applications. However, for some applications, certain piping system conditions coupled with specific stiff pipe clamp design requirements could result in interaction effects that should be evaluated in order to determine the significance of pipe stresses induced.

Safety Significance

Stiff pipe clamps were installed because of requirements for piping systems to withstand dynamic loads such as SRV discharges to suppression pools, LOCA-induced loads, and seismic loadings. A preloading of pipe clamp U-bolts or straps (which imposes a constant compressive load on the piping) is necessary to prevent stiff pipe clamps from lifting off piping under dynamic loading conditions. Since clamp-induced stresses are generally not significant with conventional pipe clamps, the pipe stresses induced by stiff pipe clamps generally were also not considered. Therefore, it was believed that further analyses of these stresses on piping systems were necessary before determining whether the stresses were significant.[1447]

In addition to the large preloading of the clamps, four other new design features were identified by the staff as requiring additional analyses because of their difference from conventional pipe clamps. These were: (1) use of high-strength or non-ASME approved materials; (2) local surface contact on the pipe; (3) uncommonly thick and/or wide design of clamp; and (4) clamp applications to piping components other than straight pipe, such as pipe elbows.

If neglect of the additional stress from stiff pipe clamps results in overestimating the pressure-retaining capabilities of piping systems, the probability of pipe breaks caused by dynamic loads may be higher than previously estimated. This increased probability could potentially result in an increased CDF that could lead to PRAs understating the public risk. This issue affected those operating and future plants that installed stiff pipe clamps.

Possible Solution

A possible solution could have the following elements:

(1) Evaluation of the local pipe stresses induced by stiff pipe clamps under all loading conditions;

(2) If the evaluation in (1) above indicated that clamp-induced pipe stresses were unacceptable, hardware modifications should be considered;

(3) As recommended[1447] by the staff, NRC could submit a request to ASME to revise Section III of the Code to include procedures for: (1) categorizing pipe stresses resulting from clamp-induced loads; and (2) evaluating those clamp applications where the ASME Code stress indices and flexibility factors do not apply;

(4) As recommended[1447] by the staff, a technical assistance program could be initiated to experimentally and analytically evaluate the interactions between piping and pipe clamps. The goal of this program would be to develop a simplified method to facilitate staff evaluations of clamp-induced pipe stresses.

PRIORITY DETERMINATION

Assumptions

It was assumed[1447] that the issue affected La Salle 1 and 2, Quad Cities 1 and 2, Dresden 2 and 3, and all plants whose operation or construction began in September 1983 or later. Thus, there were 44 operating plants affected by this issue: 27 PWRs and 17 BWRs, with average remaining lives of 33.4 and 28.9 years, respectively. These 44 plants included a few that were under construction at the time of the staff's evaluation.[1447] It was also assumed[64] that none of the 44 plants had upgraded their stiff pipe clamps as a result of IE Information Notice No. 83-80[1448] and all 44 plants had stiff pipe clamps that required some degree of hardware modification.

It was assumed that 20 future plants (10 PWRs and 10 BWRs) would be affected by this issue. The Oconee 3 and Grand Gulf 1 PRAs were used as the representative PWR and BWR, respectively.

Frequency Estimate

The risk associated with pipe breaks resulting from the use of stiff pipe clamps can be divided into the following two types: Type 1 seismic-induced pipe breaks, resulting in LOCA and/or reactor transients; and Type 2 pipe breaks in Class 1 piping, resulting from dynamic loads following LOCAs and transients.

Type 1 Pipe Break: The source of quantitative risk information was a study[1065] performed to identify risk-sensitive components in nuclear power plants during and after a seismic event. This study used PRA methodology to expand risk-sensitivity analyses by accounting for seismicity and component fragility data taken from existing nuclear power plant PRAs. To estimate the risk reduction achievable, the adjusted case assumed upgrades to various piping systems such that there would be an increase by a factor of 5 in the median peak ground acceleration (the level of peak ground acceleration at which a component has a 50% probability of failure) for these piping systems. The reduction in CDF due to this piping upgrade was 8% (0.08) for PWRs and 6% (0.06) for BWRs.

It was estimated that, for the base case, the affected annual CDF from seismic events was approximately 5.2×10^{-6}/RY for PWRs and 9.1×10^{-5}/RY for BWRs. The change in piping system reliability that could result from the possible solution would be less than the factor of 5 that was used in NUREG/CR-3357[1065] since pipe clamps are only one of the piping system components whose failure contribute to piping system failure probability; others are components such as welds, elbows, branch connections, and snubbers. Therefore, a factor needed to be developed to model the portion of the piping system reliability improvement that would result from improvements to pipe clamps. This factor was assumed to be the fractional difference between the upper and median bending moment capacity of a reference pipe segment. Using the results from NUREG/CR-2405,[1449] this factor was estimated[64] to be 0.145.

For operating plants, to calculate the reduction in CDF that could result from implementation of the possible solution, the product of the following three factors was calculated: piping component contribution; base case; and effects of pipe clamp improvement.

PWRs: CDF Reduction = $(0.08)(5.2 \times 10^{-6}/\mathrm{RY})(0.145) = 6.0 \times 10^{-8}/\mathrm{RY}$

BWRs: CDF Reduction = $(0.06)(9.1 \times 10^{-5}/\mathrm{RY})(0.145) = 7.9 \times 10^{-7}/\mathrm{RY}$

For future plants, the CDF reduction was assumed to be the same as that for operating plants.

Type 2 Pipe Break: The Oconee 3 and Grand Gulf 1 PRAs were reviewed[64] to identify those cut sets containing a hardware failure in an ECCS. For each element so identified, the largest hardware failure probability (typically associated with a valve or pump) was identified and its percentage contribution to the element's total failure probability was calculated. These percentage contributions were calculated for all the identified elements at Grand Gulf 1 and Oconee 3. These were averaged to yield values of 6.1% for Grand Gulf 1 and 6% for Oconee 3. These values were assumed to represent the failure contribution to the CDF resulting from Class 1 pipe breaks arising from dynamic loads induced following LOCAs and transients, and were used as a surrogate measure in estimating the risk contribution from Type 2 pipe breaks.

The same factor of 0.145 used above for a Type 1 pipe break was used to represent the portion of piping system reliability improvement that could result from improvements to pipe clamps. Based on the non-seismic total CDF reported in NUREG/CR-3357[1065] (6×10^{-5}/RY for PWRs and 2.9×10^{-5}/RY for BWRs), the changes in CDF resulting from implementation of the possible solution were:

PWRs: Reduction in CDF = $(0.060)(6.0 \times 10^{-5}/\mathrm{RY})(0.145) = 5.2 \times 10^{-7}/\mathrm{RY}$

BWRs: Reduction in CDF = $(0.061)(2.9 \times 10^{-5}/\mathrm{RY})(0.145) = 2.6 \times 10^{-7}/\mathrm{RY}$

Therefore, for operating plants, the total possible reduction in CDF, considering both Type 1 and Type 2 pipe breaks, was 5.8×10^{-7}/RY and 1.1×10^{-6}/RY for PWRs and BWRs, respectively. This reduction in CDF will be realized only if hardware modifications are made to the stiff pipe clamps.

For future plants, the CDF reduction was assumed to be the same as that for operating plants.

Consequence Estimate

Type 1 Pipe Break: The reduction in CDF, combined with the offsite consequences of the appropriate release categories[64] resulted in a potential public risk reduction of 0.3 man-rem/RY for PWRs and 2.1 man-rem/RY for BWRs. These values were used for all affected operating and future plants.

Type 2 Pipe Break: The reduction in CDF, combined with the offsite consequences of the appropriate release categories,[64] resulted in a potential public risk reduction of 0.6 man-rem/RY for PWRs and 1.4 man-rem/RY for BWRs. These values were used for all affected operating and future plants.

Assuming the 44 operating plants (27 PWRs and 17 BWRs) will need some degree of hardware improvements, the potential public risk reduction over their remaining lives was estimated to be:

$$[(0.3 + 0.6)(27)(33.4) + (2.1 + 1.4)(17)(28.9)] \text{ man-rem} = 2,500 \text{ man-rem.}$$

Assuming that there will be 20 future plants (10 PWRs and 10 BWRs) affected by this issue, the potential public risk reduction over their 40-year life would be:

$$[(0.3 + 0.6)(10)(40) + (2.1 + 1.4)(10)(40)] \text{ man-rem} = 1,760 \text{ man-rem.}$$

Cost Estimate

Industry Cost: Implementing the possible solution at each of the 44 operating plants would be done in two parts: (1) perform a piping analysis to assess the effects of pipe clamp to piping interaction; and (2) modify the stiff pipe clamps that produce significant pipe clamp to piping interaction.

A tota f 28 man-weeks were assumed for the pipe clamp and piping analyses. At a cost $2,270/man-week, this resulted in a cost of $63,560/plant and a total of $2 for all 44 affected plants.

For the 44 plants that require some degree of hardware modifications, the cost per plant was based on the estimate[961] of the cost of hangers for 1000 feet of 8-inch pipe; at $21/foot, this cost was $21,000/plant. Installation labor costs were estimated[64] based on $44/man-hour burdened labor rates. Based on an estimate[961] of 4.6 man-hours per linear foot, a total of 4600 man-hours/plant would be required. Applying a 10.08 adjustment factor for labor productivity effects for work in radiation zones and congested areas, manageability, and access/handling difficulties, labor costs were estimated to be $2.04M/plant. Summing over all plants yielded $0.92M for hardware and $89.8M for labor, for a total of $90.72M for 44 plants.

A total of 8 man-hours/RY were estimated for the inspection of the replacement pipe clamps at those plants requiring hardware madifications. For the 27 PWRs and 17 BWRs with average remaining lives of 33.4 and 28.9 years, respectively, and at a cost of $2,270/man-week, the total cost was $0.63M. Thus, the total industry backfit cost was $(2.8 + 90.72 + 0.63)M or $94.15M.

For the 20 future plants, the effect of stiff pipe clamps on piping can be evaluated and taken care of in the design and analysis stage, if required, and no backfit hardware modification will be necessary. Assuming that the cost/plant is also $63,560 to perform a piping analysis during the design and analysis stage to assess the effects of pipe clamp to piping interaction, the total cost for these plants will be $1.3M. Assuming a total of 8 man-hours/RY would also be required for inspection of stiff pipe clamps, the total industry operating and maintenance cost was estimated to be $[(20)(40)(8)(2270)/40]M or $0.4M. Therefore, the total industry cost for implementing the possible solution was $(1.3 + 0.4)M or $1.7M.

<u>NRC Cost</u>: NRC implementation of the possible solution at the 44 operating plants could be quite extensive. NRC would develop proposed procedures categorizing pipe stresses resulting from clamp-induced loads and procedures for evaluating those clamp applications where the ASME Code stress indices and flexibility factors are not applicable. Developing these procedures, a complicated problem, was estimated to require approximately 2 man-years of labor to develop, review, and approve. At $100,000/man-year, this cost would be $0.2M.

Implementation of the possible solution also included establishing a program to acquire experimental data to verify analytical techniques and results. The test equipment was estimated at $250,000 and preparation of test procedures, QA activities, and analysis of test results were estimated to require 1 man-year of labor at a cost of $100,000/year. Thus, the total cost of the program was $0.35M.

A generic letter directed to potentially affected plants would be required and this was estimated[961] to take 6 man-weeks. At a cost of $2,270/man-week, this cost was $0.01M. Review of licensee submittals in response to the generic letter was assumed to require 5 man-weeks/plant. At $2,270/man-week, the total cost for 44 plants was $0.5M.

The cost for reviewing operations and maintenance of the possible solution was estimated to be 0.5 man-day/RY. At $2,270/man-week, this cost will be $227/RY. Multiplying $227/RY by 44 plants over their average remaining lives resulted in a total operations and maintenance cost of $0.32M. Thus, the total NRC backfit cost was $(0.2 + 0.35 + 0.01 + 0.5 + 0.32)M or $1.38M.

Assuming the cost to develop procedures categorizing pipe stresses resulting from clamp-induced loads and procedures for evaluating those clamp applications was $0.2M, the cost for a program to acquire experimental data was $0.35M. Assuming also that the cost to update relevant Regulatory Guides and SRP[11] Sections was $0.5M and the cost to review operations and maintenance was $227/RY, for 20 plants with a 40-year plant life, the total cost was $0.2M. Therefore, the total NRC front-fit cost was $(0.2 + 0.35 + 0.5 + 0.2)M or $1.25M.

<u>Total Cost</u>: For the 44 operating plants, the total industry and NRC cost associated with the possible solution was $(94.15 + 1.38)M or $95.53M. For the 20 future plants, the total industry and NRC cost associated with the possible solution was $(1.7 + 1.25)M or $2.95M.

<u>Value/Impact Assessment</u>

Using the above estimates of total public risk reduction and implementation costs, separate value/impact scores were developed for the 44 operating plants and the 20 future plants.

(1) Operating Plants: Based on a risk reduction of 2,500 man-rem and a cost of \$95.53M for 44 plants, the value/impact score was given by:

$$S = \frac{2{,}500 \text{ man-rem}}{\$95.53\text{M}}$$

$$= 26 \text{ man-rem/\$M}$$

(2) Future Plants: Based on a public risk reduction of 1,760 man-rem and a cost of \$2.95M for 20 plants, the value/impact score was given by:

$$S = \frac{1{,}760 \text{ man-rem}}{\$2.95\text{M}}$$

$$= 597 \text{ man-rem/\$M}$$

Other Considerations

(1) Extensive work in radiation zones will be required at the 44 operating plants that need pipe clamp replacements and hardware changes. Using data from NUREG/CR-4627,[961] it was estimated that 46,000 man-hours/plant would be required. This work is in containment with an assumed radiation dose rate of 0.025 rem/hr.[64] The total occupational dose is (44 plants)(0.025 rem/hr)(4.6 x 10^4 man-hours/plant) or 51,000 man-rem.

(2) The occupational dose reduction due to accident avoidance was calculated from the reduction in CDF multiplied by the assumed accident dose of 19,860 man-rem.[64] The possible solution reduces the CDF in 44 plants which have an occupational dose rate reduction of 1.6 x 10^{-2} man-rem/RY: (19,860 man-rem x 5.8 x 10^{-7}/RY) for PWRs and (19,860 man-rem x 1.1 x 10^{-6}/RY) for BWRs. With the 27 PWRs having an average remaining life of 33.4 years and 17 BWRs having an average remaining life of 28.9 years, this resulted in a best estimate total occupational dose reduction due to accident avoidance of 22 man-rem.

(3) The accident avoidance cost savings for PWRs were estimated to be the CDF (5.8 x 10^{-7}/RY) multiplied by the estimated cost of a core-melt accident (\$1,650M) multiplied by the estimated remaining life of 33.4 years. The accident avoidance cost savings for BWRs were estimated to be the CDF (1.1 x 10^{-6}/RY) multiplied by the estimated cost of a core-melt accident (\$1,650M) multiplied by the estimated remaining life of 28.9 years. This resulted in a total cost of \$1.75M.

CONCLUSION

Based on the value/impact assessment and total reduction in public risk, the backfit actions described above were not be justified for the 44 operating plants considered. In addition, the accident avoidance cost savings and occupational dose reduction due to accident avoidance were not significant when compared to the cost and doses used in the value/impact score. However, the occupational dose increase is higher (51,000 man-rem) than the best estimate public risk reduction (2,500 man-rem). This occupational dose increase supported a LOW priority ranking for this group of plants, because the high occupational dose increase indicated that more dose would be taken modifying the stiff pipe clamps than the total estimated benefit realized from the solution.

For future plants, the value/impact consideration was more favorable since the effect of stiff pipe clamps on piping could be evaluated in the design and analysis stage. Furthermore, the occupational dosage for front-fitting future plants would be limited to operation and maintenance and should be minimal. Thus, this issue had a medium priority ranking for future plants only. RES recommended that a possible update of relevant Regulatory Guides and SRP[11] Sections be contemplated to ensure that interface design procedures are used by CPs to control the flow of design information from the support design group (which has the responsibility for the design of stiff clamps) to the pipe stress analysis group.[1476] Items 3 and 4 delineated in the Possible Solution should also be considered for future plants.

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961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

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ISSUE 105: INTERFACING SYSTEMS LOCA AT LWRs

DESCRIPTION

Historical Background

Issue B-63, which was resolved and implemented as MPA B-45, required leak-testing of the check valves that isolate those low pressure systems that are connected to the RCS outside the containment. However, except for Oyster Creek and Nine Mile Point, these low pressure systems in BWRs are isolated with check valves that have actuators that are used to test the operability of the valves. This operability test was considered sufficient to assure the integrity of the pressure isolation function and leak-testing of pressure isolation valves (PIVs) in BWRs was not required. However, beginning in 1980, the BWR STS Section 3.4.6.2 required the leak-testing of all RCS PIVs at least once every 18 months and after any work on a valve. This STS requirement was also applied to operating plants as they submitted their IST programs for review.

BWR operating experience indicated that the isolation valves between the RCS and low pressure interfacing systems (including related test and maintenance requirements) may not adequately protect against overpressurization of low pressure systems. There were three reported failures of the boundary between the RCS and low pressure injection systems in approximately 200 BWR-years of operation.[762] Two of the events (Vermont Yankee, 12/12/75, and Browns Ferry 1, 8/14/84) were the result of maintenance errors which left the testable isolation check valve in the open position. The third (Pilgrim, 9/29/83) was the result of personnel errors (improper combination of surveillance tests) and a stuck-open failure of an isolation check valve. In all three cases, there was a degradation of the PIVs due to personnel errors. None of these plants were required to leak test PIVs.

This issue, which is limited to PIVs in BWRs, is related to Issue 96 which addressed the failure of the PIVs between the RCS and the RHR system in PWRs.

Safety Significance

Overpressurization of low pressure piping systems due to RCS boundary isolation failure could result in rupture of the low pressure piping. This, if combined with failures in the ECI and/or the DHR systems, would result in a core-melt accident with an energetic release outside the containment building causing significant offsite radiation release. The STS require leak-testing of PIVs at least after every refueling and in some cases more frequently. Therefore, this issue applies to BWRs licensed before 1980.

Operating BWRs which have RCS/RHR system interface configurations similar to Hatch Unit 2 have been identified and include: Duane Arnold, Brunswick 1 and 2, Cooper, Dresden 2 and 3, Hatch 1, Fitzpatrick, Monticello, Peach Bottom 2 and 3, Pilgrim, and Quad Cities 1 and 2.[763] Browns Ferry 1 also experienced a similar isolation boundary problem. Therefore, the list of affected plants utilized in this analysis also includes BWR 3 and 4 operating plants (i.e., Millstone, Browns Ferry 1, 2 and 3, and Vermont Yankee). Therefore, the total number of potentially affected operating BWRs considered in this analysis is 20 with an average remaining life of 26 years.

Possible Solution

For the purpose of this evaluation, it was assumed that the frequency of low pressure system overpressurization events will be reduced by instigating a more rigorous revised inspection program (follow specific test and post-maintenance procedures, conduct surveillance tests one at a time, performing leak tests after operability demonstrations or flow tests) and making minor hardware modifications such as modifications to testable check valve air supply lines to precluding interchanging the lines (different threads, different size connectors, color coding, and labeling). Major system hardware changes were not anticipated.

Resolution of the issue was assumed to result in improved surveillance, maintenance, and test procedures, and minor modifications to make the air actuation system for testable check valves fool-proof.

PRIORITY DETERMINATION

Frequency Estimate

Since this issue affected only BWRs, the Browns Ferry, Unit 1, IREP[367] PRA was used in the estimation of public risk reduction.[64] The general approach was to use available historical data for failure of the high pressure/low pressure isolation boundary and a probability estimate for piping failure due to overpressurization to modify the appropriate LOCA sequences from the Browns Ferry PRA. These modified appropriate (affected) LOCA sequences are then assumed to represent the current (base case) level of plant risk associated with this issue. Specifically, the event Ls, large-break LOCA, from the Browns Ferry PRA was redefined as the product of the probability of failure of the high pressure/low pressure isolation boundary and the probability of failure of the low pressure piping as a result of overpressurization.

From the historical data (3 isolation boundary failures in about 200 BWR plant-years), a probability of failure of the isolation barrier of 1.5×10^{-2}/RY was estimated. Analysis of the low pressure piping revealed that the hoop stress in the low pressure piping would not be expected to exceed the yield value for the piping. Thus, failure of the low pressure piping was assumed to be likely only in the presence of a significant crack in the piping. Using data available on IGSCC, estimates of the number of piping welds in the low pressure piping systems, and estimates of the distribution of depth of cracks (percent of wall) from existing pipe crack data, PNL estimated the conditional probability of an intersystem LOCA, via the pipe cracking scenario, of 10^{-1}/event given an over-pressurization of the low pressure piping. This resulted in a new estimate of Ls of 1.5×10^{-3}/RY, as opposed to the value of Ls derived in the Browns Ferry PRA (3×10^{-3}/RY).

In NUREG-0677,[763] a probability of BWR intersystem LOCA (ISLOCA) of 6.2×10^{-4}/RY was calculated; no contribution from maintenance and operator errors was included. The BWR ISLOCA frequency derived for this analysis (1.5×10^{-3}/RY), which was based on previous LERs, was dominated by operator and maintenance errors and appeared to be an expected value when compared to the value derived in NUREG-0677.[763] When this new value of Ls (1.5×10^{-3}/RY) was inserted into the affected core-melt minimal cutsets in the Browns Ferry PRA, a base case core-melt frequency due to isolation boundary failures of 6.31 x 10^{-6}/RY was calculated.

Consequence Estimate

The effect of a core-melt accident resulting in direct releases outside containment was assumed to be equivalent to a BWR Release Category 2. When the dose conversion factor for BWR Category 2 events (7.1 x 10^6 man-rem/event) was multiplied by the base case core-melt frequency, a public risk of 44.7 man-rem/RY resulted.

Implementation of the possible solution to this issue was assumed to reduce the core-melt frequency and public risk due to overpressurization and failure of low pressure systems connecting to the RCS to those values calculated from the Browns Ferry PRA, i.e., 1.22 x 10^{-10} event/RY and 8.66 x 10^{-4} man-rem/RY, respectively. Therefore, implementation of the possible solution was estimated to result in a reduction in core-melt frequency of 6.3 x 10^{-6}/RY and a reduction of public risk of 44.7 man-rem/RY. The total public risk reduction for the 20 affected plants over their 26-year average remaining lifetime was calculated to be 2.3 x 10^4 man-rem.

Cost Estimate

Industry Cost: Implementation of the possible solution was estimated to require about 4 man-weeks/plant for revision of surveillance, maintenance, and test procedures, and installation of fool-proof features on the testable check valve actuation system, plus about $2,500/plant for materials (connectors, tags, etc.). Thus, an implementation cost of $220,000 was estimated. Increased surveillance testing, reduction of allowable concurrent testing and improved post-maintenance inspection procedures were estimated to increase plant maintenance and surveillance efforts by 40 man-hours/RY. Thus, the present worth of the increase in plant operation and maintenance costs for the 20 affected plants over their remaining lifetime was calculated to be about $650,000. Total industry cost for resolution (and implementation) of this issue was therefore estimated to be about $875,000.

NRC Cost: It was assumed that resolution of this issue will require 5 staff-months of technical effort and technical contract support for a more precise PRA, for a total resolution cost of about $100,000. It was assumed that NRC staff review of licensee implementation of the assumed solution would require 5 staff-weeks/plant for a cost of about $230,000. Resident inspector surveillance of site actions emanating from the resolution was estimated to require 0.5 staff-week/RY for a present worth of about $325,000 over the remaining lifetime of the 20 affected BWRs. The total present worth NRC cost for this issue was thus estimated to be about $650,000.

Total Cost: The total NRC and industry cost for resolution and implementation of the possible solution was estimated to be approximately $1.5M.

Value/Impact Assessment

Based on a potential public risk reduction of 2.3 x 10^4 man-rem and a total cost of $1.5M, the value/impact score was given by:

$$S = \frac{\underline{2.3 \times 10^{4} \text{ man-rem}}}{\$1.5\text{M}}$$

$$= 15{,}000 \text{ man-rem/\$M}$$

Other Considerations

The probability of ISLOCA may well be greater than that calculated above based on piping failure. Other components in low pressure systems, such as pump seals, heat exchanger tubes, thermocouple wells, etc., would also be subject to overpressure failures. Also, while not explicitly considered in calculating the estimated core-melt frequency and risk, the failure of all low pressure systems due to overpressure resulting from failure of PIVs contributes further to the risk. Although the risk from other interfaces was not calculated, the evaluation of Issue 96 showed that the risk from failures of the valves isolating the RHR system in a PWR was at least an order of magnitude less than the risk calculated for this issue. The failure of PIVs in a BWR RHR system would affect only part of the ECCS system, rather than all as in a PWR. Therefore, the risk in a BWR would be even less than in a PWR.

In addition, ISLOCA releases in the auxiliary building would also be expected to present an additional common mode failure mechanism for failure of redundant safety systems located in the auxiliary building. These considerations were not included in this analysis. However, had they been included, the estimates of frequency for ISLOCA and resultant core-melt would have been greater. For this reason, the priority reached on the basis of the simplified analysis performed for this issue was conservative.

A relatively small total increase in ORE (530 man-rem) was calculated due to assumed increases in surveillance and post-maintenance inspections. This calculation assumed 40 man-hours/RY for increased maintenance in a 25 millirem/hr field at the 20 affected BWRs for their remaining lifetime. Reduction in the estimated frequency of core-melt and non-core-melt intersystem LOCA which might be attained was calculated to result in a total averted ORE of 215 man-rem: 65 man-rem due to cleanup of a core-melt event and 150 man-rem due to cleanup of non-core-melt ISLOCAs. Both the increased ORE and the averted operator exposure were insignificant in comparison to the calculated public risk reduction of 2.3×10^{4} man-rem and did not alter the priority indicated by the value/impact assessment.

At an estimated industry cleanup and replacement power cost of \$1.65 Billion for a core-melt accident and \$720M for a successfully-mitigated LOCA, the frequency reduction of core-melt and non-core-melt ISLOCA estimated for resolution of this issue would result in an averted accident cost savings with a present worth of about \$2.7M. This exceeded the total expected NRC and industry cost and supported resolution of the issue.

CONCLUSION

This issue was given a high priority ranking and resolution was pursued. In resolving the issue, the staff conducted analyses of units representative of all NSSS vendors and considered: (1) human errors, both as initiators and during recovery operations; (2) component fragilities, to determine likely low pressure system break locations; and (3) the post-ISLOCA auxiliary building environment, to determine the survivability of recovery equipment. These analyses were documented in NUREG/CR-5604,[1494] NUREG/CR-5744,[1495] NUREG/CR-5745,[1496] NUREG/CR-5603,[1497] NUREG/CR-5862,[1498] and NUREG/CR-5928.[1499] It was concluded that the units studied posed little risk from ISLOCA.

In addition to the above analyses, previous PWR ISLOCA studies[1500] were reexamined with data updated to include the seven years of operating experience that had accrued since the initial analyses were undertaken. None of the studies supported generic requirements for PWRs, whether on absolute risk reduction or cost-beneficial bases.[1501] The study[1499] of ISLOCA at a BWR confirmed past PRA studies which generally indicated little risk contribution from ISLOCA sequences.

The staff found that ISLOCAs at PWRs were plant-specific in nature; however, the ongoing IPE program[1222] includes licensee analysis of ISLOCA sequences. With respect to future applicants, a draft SRP[11] Section covering design review of systems interfacing with the RCS in ALWRs was provided[1503] to NRR for information and use as appropriate. A supplement to Information Notice 92-36[1502] was also recommended to share insights from the ISLOCA program and to inform licensees of the availability of material useful for IPE ISLOCA analyses not yet completed, or as a check on analyses already completed. Thus, this issue was RESOLVED and no new requirements were established.[1504]

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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.

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1498. NUREG/CR-5862, "Screening Methods for Developing Internal Pressure Capacities for Components in Systems Interfacing With Nuclear Power Plant Reactor Coolant Systems," U.S. Nuclear Regulatory Commission, May 1992.

1499. NUREG/CR-5928, "ISLOCA Research Program Final Report," U.S. Nuclear Regulatory Commission, July 1993.

1500. NUREG/CR-5102, "Interfacing System LOCA: Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, February 1989.

1501. NUREG-1463. "Regulatory Analysis for the Resolution of Generic Safety Issue 105. Interfacing System Loss-of-Coolant Accident in Light-Water Reactors," U.S. Nuclear Regulatory Commission, July 1993.

1502. NRC Information Notice 92-36, "Intersystem LOCA Outside Containment," May 7, 1992. [Accession #9205010045]

1503. Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, 'Interfacing Systems LOCA in LWRs,'" April 2, 1993.

1504. Memorandum for J. Taylor from E. Beckjord, "Technical Resolution of Generic Issue 105 (GI-105) 'Interfacing Systems Loss of Coolant Accident (ISLOCA) in LWRs,'" June 3, 1993.

ISSUE 119: PIPING REVIEW COMMITTEE RECOMMENDATIONS

In an August 1983 memorandum,[834] the EDO requested a comprehensive review of NRC requirements in the area of nuclear power plant piping. In response to this request, the NRC Piping Review Committee (PRC) was formed to review and evaluate existing regulatory requirements to: (1) provide recommendations on where and how the NRC should modify requirements; and (2) identify areas requiring further action. The scope of the PRC review covered piping in safety-related systems and high energy lines important to safety in new and operating plants. With respect to postulated pipe breaks, the scope covered all high energy lines.

An NRC steering committee consisting of members from RES, NRR, OIE, and ELD was formed to review and develop a plan for implementing the changes recommended in the PRC report.[611] The steering committee agreed to focus its attention on the recommended research and regulatory changes designated in the PRC report[611] as Category A (high priority) recommendations. The PRC-recommended research and regulatory changes were restructured by the steering committee (combining of research and regulatory recommendations) to form 9 tasks to be addressed by the NRC implementation plan,[835] 5 of which are addressed below. These 5 tasks consist primarily of NRR regulatory actions and some closely-related research efforts. The remaining 4 tasks of the NRC implementation plan related only to research activities and were excluded from this issue.

The five parts of this issue primarily involve revisions to Regulatory Guides and the SRP.[11] No significant change in public safety was expected to result from resolution of this issue; however, resolution of the various tasks was expected to result in less complex and more realistic approaches to piping design and operation in nuclear power plants. The results were expected to yield more efficient regulatory practices, improve plant piping systems design, increase plant reliability, and decrease ORE associated with inspections and repairs. The NRC steering committee agreed that, based on the information provided in NUREG-1061,[611] this work should continue on a schedule consistent with high-priority issues. Therefore, this issue was classified as a Regulatory Impact issue. RES took the lead responsibility for resolution of this issue with assistance from other NRC Offices.[835] The following is an evaluation of the 5 parts of this issue.

ITEM 119.1: PIPING RUPTURE REQUIREMENTS AND DECOUPLING OF SEISMIC AND LOCA LOADS

DESCRIPTION

This task combined two PRC Category A regulatory recommendations with one PRC Category A research recommendation. The designations of the three PRC recommendations were: (1) leak-before-break (A-1); (2) decoupling of seismic and LOCA loads (A-5); and (3) completing research on decoupling (A-4).

One part of the task involved rulemaking changes to GDC-4 in Appendix A of 10 CFR 50 to redefine the need to consider the dynamic effects of pipe breaks. A proposed rule to modify GDC 4 was published[1339] in July 1985 and codified leak-before-break technology, but was limited only to the primary loop piping of PWRs; the final rule was published[1340] in April 1986. A proposed broad scope rule dealing with all high energy piping in LWRs was published[1341] in July 1986; the final rule was published[1342] in October 1987. With the issuance of these revised rules, revisions to SRP[11] Sections 3.6.1 and 3.6.2 were needed to eliminate the postulation of arbitrary intermediate breaks. The second part of this task involved relaxation of the requirement to consider LOCA and seismic loads simultaneously. A revision to SRP[11] Section 3.9.3 was to be pursued to decouple seismic and pipe rupture loads in the mechanical design of components and their supports.

The existing GDC-4 requirement and SRP[11] Section 3.6.2 pertaining to postulated double-ended guillotine breaks (DEGB) of the largest pipes and postulated arbitrary intermediate pipe breaks needed to be changed to include more realistic criteria and to allow consideration and acceptance of validated analysis methods. The requirements of GDC-4 led to a situation where protective devices were added to forestall events that are extremely unlikely. These protective devices that were designed for the extremely unlikely events could, however, reduce safety and increase worker radiation exposure under normal operations and design basis events.

SRP[11] Section 3.9.3 requires that piping systems and associated components be designed for the combined effects of an SSE and a LOCA. The evolution of seismic design requirements and the calculations of pipe rupture loads have significantly increased the resultant loads obtained by combining these effects. However, field evaluations of piping at conventional power plants and petrochemical facilities indicated that ruptures in piping of the type found in nuclear power plants do not occur during severe earthquakes. Therefore, the staff believed that relaxation of these requirements at all LWRs would not affect plant or public safety.

CONCLUSION

This task was classified as a Regulatory Impact issue that resulted in revisions[1343,1344] to SRP[11] Sections 3.6.1 and 3.6.2. In addition, Generic Letter No. 87-11[1345] was issued to licensees on the relaxation in arbitrary intermediate pipe rupture requirements (SRP Section 3.6.2). In 1986, the staff terminated[1346] all work on a proposed revision to SRP[11] Section 3.9.3. Thus, this issue was resolved.

ITEM 119.2: PIPING DAMPING VALUES

DESCRIPTION

Historical Background

This task combined PRC regulatory recommendation A-2 (modify seismic damping values used in seismic designs) and PRC research recommendation B-3 (complete research on damping tests). It constituted a two-level approach that could affect all LWRs: a short-term plan and a long-term plan. The short-term action called for a revision to Regulatory Guide 1.84[1347] as the vehicle for NRC endorsement of ASME Code Case N-411. The long-term action called for revisions to Regulatory Guide 1.61[1348] and SRP[11] Section 3.9.2 to incorporate, not only ASME Code Case

N-411, but also new positions on pipe damping for high-frequency loads and for time-history analyses.

The short-term endorsement of the ASME Code Case N-411 was to be restricted to seismic response analysis, but not time-history analysis. The long-term action was to result in extensive changes to SRP[11] Section 3.9.2 and Regulatory Guide 1.61[1348] to provide more comprehensive guidance on pipe damping for both seismic and BWR hydrodynamic loadings. Criteria for other non-seismic dynamic loads could also be addressed in the SRP[11] Section 3.9.2 revision.

In general, dynamic piping response could be more accurately predicted if use was made of higher piping damping values than those identified in the existing regulatory guide. The use of higher damping values would result in nuclear plant piping systems having significantly less snubbers and supports and an overall better balance of design, considering all piping loads. A decrease in the number of snubbers and supports could allow better inspection of equipment and components at significantly reduced ORE.

CONCLUSION

The staff originally planned to take the lead in developing improved pipe damping values and classified the task as a Regulatory Impact issue. However, with the cooperative effort of EPRI, ASME, and the NRC in pursuing the concern, the staff concluded that the most effective approach to the use of more realistic damping values for dynamic piping analysis was through ASME III, Appendix N. When this appendix is completed, the staff will make a decision on its endorsement. As a result, the issue was dropped from further pursuit.[1336]

ITEM 119.3: DECOUPLING THE OBE FROM THE SSE

DESCRIPTION

This task corresponds to PRC regulatory recommendation A-3 (decouple OBE from SSE). 10 CFR 100, Appendix A, Section V(a)(2), stipulates that "The maximum vibratory ground acceleration of the OBE shall be at least one-half the maximum vibratory ground acceleration of the SSE." Therefore, the current requirement implies the coupling of the two earthquake design levels: SSE and OBE. In developing the current regulations, it was assumed that the SSE would control the design in nearly all aspects and that the OBE would serve as a separate check of those systems where continued operation was desired at a lower level of ground motion. However, in practice, the assumed load factors, damping, stress levels, and service limits have caused the OBE, rather than the SSE, to control the design for many systems including concrete and steel structures and nuclear piping. In addition, seismic design for OBE accounts for certain safety-related factors such as fatigue and seismic anchor movement that are not considered in the design for the SSE.

Decoupling of the OBE from the SSE or modification of the associated load factors, etc., would impact the design of new plants and would extend well beyond piping considerations. The actions required to resolve this task include: (1) rulemaking to amend and revise Appendix A to 10 CFR 100 to permit decoupling of the OBE and SSE and to incorporate the use of probabilistic methodology in earthquake design; (2) revising and developing Regulatory Guides; (3) updating pertinent sections of the SRP[11]; and (4) advising various industry code committees to revise appropriate codes and guides to reflect changes in the regulations.

A complete listing of the Regulatory Guides and SRP Sections that may be affected by this task were to be identified during the review phase of this task and the related tasks contained in the NRC implementation plan[835] which is of much broader scope.

There is no technical basis for coupling the OBE with the SSE. Designing the piping systems to the SSE is the primary means of ensuring safety. Additional margin is provided by specifying the OBE and thus the level at which inspections will be required before continued operation would be permitted. The more realistic approach of using specific probabilities (return periods) for OBE and the decoupling of the OBE levels and frequencies from those of the SSE will allow assurance of public safety to be placed on a more rational basis.

CONCLUSION

This item is a Regulatory Impact issue that, in December 1991, was integrated into the revision to 10 CFR 100, Appendix A.

ITEM 119.4: BWR PIPING MATERIALS

DESCRIPTION

This task corresponds to PRC regulatory recommendation A-4 to replace regular grade 316SS and 304SS materials in BWR recirculation piping with an alloy resistant to IGSCC. The NRR action related to this task involved preparation of Revision 2 to NUREG-0313[750] and evaluation of each licensee's actions in compliance with this revision.

IGSCC in BWR piping has occurred in a range of piping sizes over the last 25 years and has resulted in major reactor outages. The risk studies reported[611] indicate that pipe failures, even assuming the higher rates due to IGSCC, would not be a major contributor to core-melt and public risk. However, use of materials more resistant to IGSCC should significantly reduce levels of ISI and reactor outage times. Therefore, plant outages and recurring ORE could be significantly reduced by resolution of this task.

CONCLUSION

This item is a Regulatory Impact issue that required[1506] updating of Regulatory Guide 1.44[1507] by RES to reflect the staff's findings in NUREG-0313,[750] Revision 2.

ITEM 119.5: LEAK DETECTION REQUIREMENTS

DESCRIPTION

This task corresponds to PRC regulatory recommendation A-6 (leak detection requirements). To accomplish this task, additional data are necessary to further validate and improve existing leak-rate prediction analyses. Of particular interest would be investigation and improvement of local leak detection systems such as acoustic emission monitors or moisture-sensitive tapes. These latter techniques may be important for establishing the validity of leak-before-break at specific locations in certain piping systems. The task requires a combination of two approaches: (1) the surveying of operating plants to determine the adequacy of existing leak detection systems; and (2) completion of the research recommended by the PRC and applying the results of the research to regulatory requirements. Subsequent to the completion of key elements of the research effort, the regulatory actions may include the following:

(1) Identify required TS changes such as: (a) unidentified leakage limits for BWRs and PWRs in the context of locating and detecting leakage from cracks with margin; (b) adequacy of surveillance requirements and calibration of systems; (c) alarms; (d) TS consistency; (e) new systems or different detection system combinations; and (f) forward-fit and backfit considerations.

(2) Revise SRP[11] Section 5.2.5 and Regulatory Guide 1.45.[603]

(3) Issue NUREG-0313,[750] Revision 2.

It was believed that resolution of this task could affect all LWRs to varying degrees.

No direct safety significance could be attributed to this task. However, knowledge of the leak rates associated with various postulated through-wall crack lengths and confidence in the ability to detect leakage in a timely manner are important elements of the leak-before-break concept that eliminates the postulated DEGB.

CONCLUSION

This item is a Regulatory Impact issue.

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835. Memorandum for W. Dircks from R. Minogue, "Plan to Implement Piping Review Committee Recommendations," July 30, 1985.

1336. Memorandum for B. Morris from L. Shao, "Resolution of Generic Issue 119.2," July 16, 1990.

1339. Federal Register Notice 50 FR 27006, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," July 1, 1985.

1340. Federal Register Notice 51 FR 12502, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," April 11, 1986.

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1342. Federal Register Notice 52 FR 41288, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," October 27, 1987.

1343. Federal Register Notice 53 FR 1968, "Standard Review Plan Revision," January 25, 1988.

1344. Federal Register Notice 52 FR 23376, "Standard Review Plan Issuance," June 19, 1987.

1345. NRC Letter to All Operating Licensees, Construction Permit Holders and Applicants for Construction Permits, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements (Generic Letter 87-11)," June 19, 1987.

1346. Memorandum for Distribution from G. Arlotto, "Termination of Proposed Revision to SRP 3.9.3," October 2, 1986.

1347. Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability -- ASME III, Division 1," U.S. Nuclear Regulatory Commission, Revision 26, July 1989.

1348. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1973.

1506. Memorandum for W. Minners from L. Shao, "Closeout of GSI 119.4," July 17, 1992.

1507. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," U.S. Nuclear Regulatory Commission, May 1973.

ISSUE 120: ON-LINE TESTABILITY OF PROTECTION SYSTEMS

DESCRIPTION

Historical Background

This issue was raised[1271] by the staff in 1985 during the review of several plant TS when it was found that the protection system designs of some older plants did not provide as complete a degree of on-line protection system surveillance testing capability as other plants undergoing staff review and evaluation at that time.

The requirements for at-power testability of components are included in GDC 21 of Appendix A to 10 CFR 50. Supplementary guidance is provided in Regulatory Guides 1.22 and 1.118 and IEEE Standard 338 to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is operating without adversely affecting the plant's operation. These requirements apply to both the RPS and the ESFAS. Existing STS indicate that it is desirable to test all protection systems through their sub-group relays every 6 months.

Safety Significance

This issue centered around the risk posed by those plants with lesser degrees of on-line testing capability and the value/impact effects of requiring modifications of the protection systems to allow for a greater degree of on-line testing. On-line testing increases the ability to detect existing failures of the protection system and could therefore result in improved reliability of the system; hence, a reduction in plant risk. In some older plants, a larger portion of the protection system hardware can only be tested through the sub-group relays during outages (i.e., shutdowns) which typically have an 18-month frequency. Therefore, modification of the protection system to allow for semiannual testing through the sub-group relays could result in risk reduction at those plants.

Possible Solution

The following two options were identified as potential solutions:

(1) Recognize that there are cases where there are no practical system design modifications that will permit at-power operation of the actuated equipment without adversely affecting the safety or operability of a plant. Exceptions could be taken that include not testing the automatic initiating logic and associated actuating devices. Actions could include: (1) submittal of information by licensees to describe and justify any deviations from regulatory requirements and to describe the revision of the plant TS stating the testing required; and (2) testing of those systems that can be tested without defeating the ESFAS train or RPS.

(2) Design and implement modifications to allow compliance with the requirements for on-line testing of all systems without defeating

the ESFAS train or RPS. Each channel of the reactor trip module (RTM) needs to be provided with two key-operated bypass switches, a channel bypass switch, and a shutdown bypass switch. The 2/4 system would then operate in the 2/3 mode during the testing.

It was believed that changing the testing frequency of the protection system components to 6-month intervals, instead of the existing 18-month intervals, would increase the reliability of these components and result in an overall enhancement of plant safety.

PRIORITY DETERMINATION

Assumptions

It was assumed that modifications would be made to allow for an increase in test frequency to 6 months (from 18 months) for 20% of the relays in the RPS. Changes in the test frequency for ESFAS relays were not considered because they could not be as readily incorporated into the representative plant PRAs.

Frequency Estimate

The Oconee 3 and Grand Gulf 1 PRAs were used as the representative PWR and BWR, respectively, to estimate the change in the reliability of RPS components due to revised testing frequency (from the current 18-month testing interval to 6-month interval) and the resultant change in the core-melt frequency.[64] Thus, the changes in core-melt frequency were estimated based on reductions in failure rates for relays in the RPS that would result from licensee implementation of potential solutions. It was assumed that the values in the Oconee 3 and Grand Gulf 1 PRAs were based on the 6-month test interval for all relays in the RPS and that these plants are in full compliance with on-line testing requirements. These values were then considered to be adjusted case values for the purposes of this analysis. Therefore, the base case represents the situation in which only a fraction of the relays can be tested during refueling outages or other extended shutdowns (an 18-month test interval for these relays is assumed).

The affected parameter in the Oconee 3 PRA was considered to be K, failure of RPS due primarily to test and maintenance faults (frequency = 2.6×10^{-5}/demand). The affected parameter for Grand Gulf 1 was considered to be C, failure to render the reactor subcritical (frequency = 7.7×10^{-7}/demand). These K and C estimates were then assumed to represent the adjusted case values. To calculate the base case values for a change in test frequency from 6 to 18 months, relay unavailability data from ANO-2 for the two testing frequencies were used. In addition, it was also assumed that the testing of all 100 relays, instead of the approximately 80 relays that are currently being tested, will increase the unavailability of 1 of 4 RTMs by 25%. The ANO-2 relay unavailability data for the 6-month and 18-month testing intervals were 7.2×10^{-4}/demand and 2.2×10^{-3}/demand, respectively.[1272] By using these values in the RPS fault tree given in NUREG/CR-2800,[64] base case values of 2.96×10^{-5}/demand and 9.2×10^{-7}/demand for K and C, respectively, were calculated. Note that these were the values relating to the 18-month testing intervals. Substituting these values for the affected parameters in the Oconee 3 and Grand Gulf 1 PRAs resulted in core-melt frequency reductions of 1.2×10^{-6} /RY and 10^{-6}/RY for a PWR and BWR, respectively. The generic release categories and containment failure modes associated with this issue were as follows:[64]

Release Category	Containment Failure Mode Probability	Whole Body Dose (Man-Rem)
PWR-3	0.5	5.4×10^6
PWR-5	0.0073	1.0×10^6
PWR-7	0.5	2.3×10^3
BWR-2	1.0	7.1×10^6

Accordingly, the associated public risk reduction was estimated to be 3.3 man-rem/RY and 4.8 man-rem/RY for PWRs and BWRs, respectively.

A total of 42 operating plants were affected by this issue: 8 PWRs with an average remaining life of 27.7 years and 34 BWRs with an average remaining life of 25.2 years. For the 8 affected PWRs, the estimated risk reduction was [(8)(27.7)(3.3)] man-rem or 731 man-rem. For the 34 affected BWRs, the estimated risk reduction was [(34)(25.2)(7.1)] man-rem or 6,083 man-rem. Thus, the average risk reduction was approximately 162 man-rem/reactor.

Cost Estimate

The plants affected by this issue were divided into two groups: Group 1, consisting of plants where no design modifications that would permit testing of the RPS at full power were possible; and Group 2, consisting of plants that could possibly implement design modifications that would permit this testing. It was assumed that the affected plants were divided equally into these two groups (21 plants each) and had an average remaining life of 26.9 years.

Industry Cost: The implementation of the possible solution for Group 1 plants would require 16 man-weeks/plant broken down as follows:

Inspection/review of current plant configuration = 1 man-week
Researching possible design modifications = 3 man-weeks
Analyze/justify deviations from regulatory requirements = 4 man-weeks
TS changes and associated technical/legal/administrative support = 8 man-weeks

At approximately \$2,270/man-week, the cost of implementation for Group 1 plants was estimated to be (16 man-weeks/plant)(\$2,270/man-week) or \$36,000/plant. The implementation cost for Group 2 plants was estimated to consist of about \$50,000/plant hardware costs and about 21 man-weeks/plant of labor itemized as follows:

Inspection/review of current plant configuration = 1 man-week
Design modifications = 3 man-weeks
Install and test design modifications = 16 man-weeks
Revise testing procedures = 2 man-weeks

Similarly, at \$2,270/man-week, the labor cost was estimated to be (21 man-weeks/plant)(\$2,270/man-week) or \$48,000/plant. Therefore, the total implementation cost/plant for Group 2 plants was (\$48,000 + \$50,000) or approximately \$100,000. Thus, the average implementation cost for the 42 affected reactors was \$68,000/plant.

It was assumed that Group 1 plants would require additional inspection activities during outages associated with assuring the operability of the relays in the RPS. It was estimated that an additional 4 man-hours/relay (i.e., those 20 relays that cannot be tested at power) would be required every 6 months for Group 1 plants for a total of 160 man-hours/RY. For Group 2 plants, it was estimated that an additional 2 man-hours/relay would be required every 6 months for a total of 80 man-hours/RY. Since most of the work would be in radiation zones, a 75% utilization factor for labor (210 man-hours/RY for Group 1 plants and 110 man-hours/RY for Group 2 plants) was assumed. At $2,270/man-week, maintenance and operation costs for Group 1 and Group 2 plants were estimated to be $12,000/RY and $6,200/RY, respectively. Using a 5% discount rate, the present worth of the recurring costs associated with plant maintenance and operation for Group 1 and 2 plants were $6,700/RY and $3,400/RY, respectively. Thus, the estimated operations and maintenance costs were $180,000/plant and $91,000/plant for Group 1 and Group 2 plants, respectively, and the average cost for all affected plants was $136,000/plant.

NRC Cost: NRC resource requirements consisted of preparation of a generic letter to the affected plants to inform them of the potential problems and requiring licensee inspection/review of the RPS testing capabilities, as well as the technical analyses and/or design modifications needed to implement the proposed resolutions. This effort was estimated to require 6 man-weeks of NRC labor or $14,000. For the 42 affected plants, this cost averaged $330/plant.

In addition, it was estimated that approximately 12 man-weeks (or $27,000/plant) of NRC labor were required for each Group 1 plant to review and approve licensee evaluations and TS changes. For each Group 2 plant, it was estimated that 10 man-weeks (or $23,000/plant) would be required for the review and approval of licensee evaluation, proposed design modifications, and TS changes. Thus, the average NRC cost for this effort was $25,000/plant for the 42 affected plants.

Inspection-related costs for each plant would be about $4,600/year for the remaining life of the affected plants. At a 5% discount rate, this translated to a present worth of $2,600/RY. This cost was $70,000/plant based on the average remaining life of the affected plants.

Total Cost: Based on the above estimates, the average cost for implementing the possible solutions was $[68,000 + 136,000 + 330 + 25,000 + 70,000]/plant or approximately $0.3M/plant.

Value/Impact Assessment

Based on a potential public risk reduction of 162 man-rem/reactor and an average cost of $0.3M/reactor, the value/impact score was given by:

$$S = \frac{162 \text{ man-rem/reactor}}{\$0.3\text{M/reactor}}$$

$$= 540 \text{ man-rem/\$M}$$

Other Considerations

(1) It was estimated that, for Group 1 plants, 1 man-week of utility labor in a radiation zone will be required to inspect the non-testable relays and

review the system design. Group 2 plants would be subjected to this review and would also require an additional 10 man-weeks to install the design modifications and 4 man-weeks to test the modified system. It was assumed that testing would be performed outside containment where the dose rate is 2.5 millirem/hr. It was further assumed that the work involved a 75% utilization factor. The implementation dose was, therefore, estimated to be about 1 man-rem/plant.

(2) It was estimated that, for Group 1 plants, operation and maintenance would require additional inspection activities during plant outages associated with assuring the operability of the relays in the RPS. It was estimated that a total of 160 man-hours/RY would be required for Group 1 plants. For Group 2 plants, it was estimated that the labor requirements were 110 man-hours/RY in a radiation zone. Assuming a 75% utilization factor, the total operation and maintenance dose was estimated to be about 12 man-rem/plant.

CONCLUSION

The estimated potential public risk reduction resulting from improvement in the on-line testability for the RPS at some older plants was significant and the value/impact score indicated a medium priority. Neglecting the ESFAS relays could result in an underprediction of the total potential risk reduction. Experience showed that testing of protection systems at power can have the potential for subtle interactions with other safety systems and/or plant operation that might result in negative effects on plant risk (i.e., an increase in plant risk). In addition, the negative aspects of increased testing (human error and reduced redundancy) could also produce a competing impact on plant risk. Based on these considerations and the value/impact score, this issue was given a medium priority ranking and RESOLVED with no new requirements.[1508]

REFERENCES

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1271. Memorandum for T. Speis from R. Bernero, "Request for Prioritization of Potential Generic Issue Per Office Letter No. 40," August 4, 1985.

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ISSUE 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS

DESCRIPTION

Historical Background

Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class 1E safety-related systems are transmitted to non-Class 1E control or display equipment.

There are a number of devices which may qualify as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible fault applied in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuit (Class 1E side) below an acceptable level.

This issue was identified[1270] by the staff in June 1987 and arose from observations made during SPDS evaluation tests that, for electrical transients below the maximum credible level, a relatively high level of noise could pass through certain types of isolation devices and be transmitted to safety-related circuitry.[1269] In some cases, the amount of energy that can pass through the isolator may be sufficient to damage or seriously degrade the performance of Class 1E components while, in other cases, electrically-generated noise on the circuit may cause the isolation device to give a false output.

Safety Significance

Recent observations have shown instances in which isolation devices subjected to failure voltages and/or currents less than maximum credible fault levels passed significant levels of voltage or current, but the same devices performed acceptably at maximum credible levels. The safety system on the Class 1E side of the isolation device may be affected by the passage of small levels of electrical energy, depending upon the design and function of the safety system.

In the event that safety systems are affected by less than maximum credible faults on the non-Class 1E side of isolators, the effects can range from degradation to failure of single or multiple trains of safety systems resulting in failure on demand or inadvertent operation. In one recorded incident, a voltage transient induced by a power line fault caused a false indication that the turbine-generator output breaker had tripped, resulting in a reactor scram.

Possible Solution

The assumed solution to this issue would require the staff to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. An NRC bulletin to all licensees to provide input on these questions would be necessary. Assuming that the staff determines from the licensee responses to the proposed bulletin that a potential problem exists, a research program consisting of two major objectives would have to be initiated to develop the solution to this issue. The first objective would be to develop test procedures and acceptance criteria for isolators that licensees could use to determine the adequacy of installed isolators. The second objective would involve development of appropriate hardware fixes that could resolve the issue.

Electrical hardware currently exists either to reduce the amount of energy that may leak through electrical barriers provided by various types of isolation devices, or to minimize the consequences of any unwanted signals that may leak through the isolator. Some of these devices are described below.

Surge arresters, also called lightning arresters, provide an effective means of eliminating high voltage transients from a circuit. These devices are simply connected from the conductor directly to ground, preferably as close as possible to the device to be protected. The arresters function by simply shunting to ground any voltage spikes above a certain level.

Filter chokes and capacitors can greatly attenuate high frequency electrical noise. These components create an impedance to the passage of electrical energy proportionate to the frequency of the signal and are especially effective against radio frequency noise. Filter chokes (or reactors) also function as current limiters in AC circuits and thus offer additional protection from overload currents.

At power frequencies, power conditioners can be employed to eliminate all unwanted signals. Power line conditioners function by rectifying an AC signal into DC and then reconverting power through an invertor into a clean, noise-free AC signal. These devices prevent notches, spikes, radio frequency, brownouts, and overload power at the input terminals from degrading the quality of power at the protected output.

The final step in the solution to this issue would be the issuance of a generic letter to licensees with the following guidelines for: (1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems; (2) repair/replacement of isolators that fail the tests, including description of acceptable hardware fixes to the isolators; and (3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

PRIORITY DETERMINATION

Assumptions

A total of 90 PWRs and 44 BWRs are potentially affected by this issue. The expected average remaining lives of these plants are 28.8 and 27.4 years for PWRs and BWRs, respectively.

Frequency Estimate

There are several sources of uncertainty associated with this issue, the most important of which are: (1) the extent to which potentially susceptible isolators are used at nuclear power plants; (2) the amount of electrical energy leakage through isolation devices that could compromise the function of Class 1E system components; and (3) the number of components in which such compromises would be critical. While a recent study[1269] indicated that a safety problem may exist due to energy leakage through electronic devices, no definitive research has been conducted to date to indicate the character and magnitude of the associated safety concerns. As a result, a sensitivity analysis was performed to bound the potential public risk reduction associated with this issue. Estimates of the upper and lower bounds were developed as well as a third case that represents the "best estimate" based on the available information.

The Oconee 3 and Grand Gulf 1 PRA studies were used as representative of PWRs and BWRs, respectively.[64] The parameters affected by this issue are those involving control circuitry failures and functional failure of ESF actuation systems. These components may be directly affected by energy leakage through isolation devices that are intended to protect them from signals originating in connected non-Class 1E systems. It is also possible that sensors in the Class 1E safety systems may be affected by the electrical energy leakage from the non-Class 1E system. These sensors may include valve position, temperature, and pressure sensors that alert plant operators to take a particular action. In this case, plant operators may be misled into not taking appropriate actions when required. For this reason, operator error terms are also included as potentially affected parameters. The affected parameters in the Oconee 3 and Grand Gulf 1 PRAs were identified and modified to model the three sensitivity cases.

Best Estimate: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of two (assumed) to account for the potential additional failures associated with electrical isolators. A factor of two was assumed based on engineering judgment and the findings of previous prioritization analyses.

Upper Bound: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of ten (assumed) to account for the potential additional failures associated with electrical isolators. A factor of 10 was likewise assumed based on judgment and previous analytical experience.

Lower Bound: The control circuitry and ESF actuation functional failures were multiplied by a factor of 1.4. This is based on an assumed factor of two increase in only the probability of fuse failures which are included in the control circuitry unavailability values. No effect on the operator error terms were assumed in this case.

It is noted that varying all the control circuitry, ESF function failure, and operator error terms is a conservative approach. Logic dictates that not all the terms would be affected at the same time and that a plant-specific detailed evaluation would probably result in a reduced sensitivity. After the failure terms were modified, they were combined with the remaining unaffected portions of the parameter unavailabilities to calculate the revised unavailabilities. The affected cut-set elements and their base case and adjusted case unavailability values are shown in Table 3.142-1.

TABLE 3.142-1

Base Case and Adjusted Case Values of Affected Parameters

Parameter	Adjusted Case[a]	Base b Case 1[a]	Base Case 2[c]	Base Case 3[d]
Grand Gulf				
H	0.0212	0.0225	0.0217	0.0329
HACT,RACT	0.00123	0.00223	0.00163	0.0102
R	0.0512	0.0530	0.0518	0.067
L	0.0213	0.0226	0.0218	0.033
LRACT,BCACT	0.00123	0.00223	0.00163	0.0102
LA2,LB2	0.0140	0.0151	0.0144	0.0240
LB1	0.0134	0.0138	0.0135	0.017
LC	0.0215	0.0230	0.0220	0.035
VGA1,VGB1	0.0148	0.0156	0.0150	0.022
VGA2,VGB2	0.0236	0.0273	0.0238	0.0553
SA,SB	0.0144	0.0150	0.0146	0.0198
SAACC, SBACC	0.00123	0.00223	0.00163	0.0102
SSA,SSB	0.0205	0.0223	0.0209	0.0361
SSC	0.0140	0.0151	0.0144	0.0239
SAC,SBC,SCC	0.00123	0.00223	0.00163	0.0102
V1, V2	0.00803	0.0091	0.00813	0.0173
V3	0.0033	0.0064	0.0033	0.0296
SCVA,SCVB	0.0315	0.0333	0.0321	0.0477
Oconee				
B,C	0.0033	0.0043	0.0037	0.0121
D,E	0.0231	0.0354	0.0249	0.1334
CONST1	0.0002	0.00048	0.0003	0.0007
CONST2	0.0006	0.00125	0.00083	0.0123
A1,C1	0.0098	0.0163	0.0124	0.0683
B1	0.0349	0.0502	0.0710	0.1718
G1	0.0136	0.0172	0.0150	0.046
RCSRBCM	0.00003	0.00007	0.00003	0.00032
WXCM	0.003	0.006	0.003	0.03
D.E	0.00049	0.00121	0.0006	0.0178
W.X	0.00009	0.00025	0.0001	0.00451
B.W, C.X	0.00003	0.00006	0.00004	0.00081
D.X, E.W	0.00021	0.0006	0.00029	0.00895
B.D, E.C	0.00006	0.0001	0.00008	0.0016

NOTES: (a) Original Oconee 3 and Grand Gulf 1 PRA values
(b) Best estimate
(c) Lower bound case
(d) Upper bound case

In performing the risk analysis, it was assumed that the isolator failures were not considered as potential causes of failure in the original Oconee and Grand Gulf PRAs. (This assumption may also introduce additional conservatism.)

Since the base case was intended to represent the situation in which isolator failures are considered as possible causes of safety system failures and the adjusted case represented the situation after the resolution is implemented, the modified parameter values were used in the base case and the adjusted case represent the original Oconee and Grand Gulf parameter values. The base case and adjusted case values of the affected parameters were then incorporated in the Oconee 3 and Grand Gulf 1 PRAs to derive the estimated core-melt frequency and the associated public risk reduction. Based on the data in Table 3.142-1, the following core-melt frequency reduction was estimated for the representative PWR and BWR.

Sensitivity Case	Core-Melt Frequency Reduction	
	PWR	BWR
Best Estimate	2.59×10^{-5}/RY	7.98×10^{-6}/RY
Lower Bound	5.37×10^{-6}/RY	2.07×10^{-6}/RY
Upper Bound	4.35×10^{-4}/RY	1.17×10^{-4}/RY

Utilizing generic release categories and containment failure modes, the public risk reduction was estimated to be as follows:

Sensitivity Case	Public Risk Reduction (man-rem/RY)	
	PWR	BWR
Best Estimate	57	53
Lower Bound	13	14
Upper Bound	1,016	789

Based on the public risk reduction estimates presented before for the representative PWR and BWR and the three sensitivity cases, the following public risk reduction was estimated (weighted average over all affected PWRs and BWRs and their remaining lives):

Best Estimate = 1,580 man-rem/plant
Lower Bound = 378 man-rem/plant
Upper Bound = 26,752 man-rem/plant

Cost Estimate

Industry Cost: It was assumed that the proposed generic letter would contain the following guidelines applicable to all affected plants: (1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems; (2) replacement of failed or unacceptable isolators, including descriptions of acceptable hardware fixes to the isolators; and (3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

The initial testing and inspection program at each plant was estimated to require approximately 4 man-weeks for planning and 8 man-weeks for review and evaluation of the data, preparation of the final response to the generic letter, and preparation of a safety analysis. The cost to conduct the initial test program was highly uncertain because there were unknown numbers of affected systems and susceptible isolators at each plant. For this analysis, the number of potentially affected isolators was estimated using the number of safety system components in the Oconee and Grand Gulf PRAs with functional and/or control circuitry failure terms. Accordingly, 46 isolators for BWRs and 78 isolators for PWRs were estimated. Assuming a two-man team can test 10 isolators per day, labor requirements for the initial test/inspection required by the generic letter were estimated at 10 man-days/plant for PWRs and 16 man-days/plant for BWRs.

Furthermore, isolators that fail the initial tests must be replaced or repaired. It was conservatively assumed that 25% of the tested isolators will fail the tests; this would result in 12 failures at PWRs and 20 failures at BWRs. The cost to purchase, install, test, and perform adequate QC of acceptable replacement isolators was estimated at $10,000/isolator and this included approximately 2 man-days/isolator for replacement. Thus, the total isolator replacement costs were estimated to be $120,000/plant and $200,000/plant for PWRs and BWRs, respectively. Assuming a cost of $2,270/man-week, the total implementation cost (including hardware) was estimated to be $156,000/plant and $239,00/plant for PWRs and BWRs, respectively.

The generic letter was assumed to include a requirement for annual testing and inspection of all electronic isolators. The industry labor requirements for this activity were estimated to be 1 man-wk/RY for test planning (this was significantly lower than the 4 man-wks for planning the initial test program), plus 10 man-days/RY to conduct the tests at PWRs and 16 man-days/RY to conduct the tests at BWRs. An additional 1 man-wk/RY at all plants to review the test results and prepare a report for the NRC was also included. This resulted in estimated labor requirements of 4 man-wks/RY and 5.2 man-wks/RY for PWRs and BWRs, respectively.

Furthermore, the annual testing program was likely to determine that there are additional failed or suspect isolators that require replacement. It was assumed that all the remaining isolators (i.e., other than those that were replaced as a result of the initial test program) will eventually be replaced with acceptable components. The number of remaining isolators to be replaced at PWRs was estimated to be 38 (i.e., 46 - 12) over a 28.8 year period or 1.2/RY. At BWRs, the annual replacement rate was 58 (i.e., 78 - 20) over a 27.4 year period, or 2.1/RY. The annual replacement costs at each plant were thus estimated to be $12,000/RY and $21,000/RY for PWRs and BWRs, respectively.

At $2270/man-week, the total cost of maintenance and operation (including hardware) of the possible solution at each plant was estimated to be $21,000/RY and $33,000/RY for PWRs and BWRs, respectively. Using a 5% discount rate, the present worth cost associated with plant maintenance and operation for PWRs and BWRs was estimated to be $11,600/RY and $18,300/RY, respectively.

NRC Cost: It was assumed that the first activity would involve issuance of a bulletin to determine the extent to which potentially susceptible isolators were used in nuclear power plants and to identify the systems in which they were used. It was estimated that 2 man-weeks ($4,000) would be required to prepare the bulletin. To perform the review and analysis of licensee responses to the bulletin, it was estimated that 6 man-months ($50,000) of technical support would be needed at a cost of $54,000.

Assuming that, after analyzing licensee responses, the staff concluded that the issue warranted further attention, the second activity would involve a research program that would develop the details of the final resolution to this issue. This program would involve two major objectives. First, test procedures and acceptance criteria for isolators would be developed for licensee use in determining the adequacy of their installed isolators. It was estimated that a $50,000 contract plus $10,000 for NRC contract support would be needed to accomplish this objective. Second, appropriate hardware fixes would be developed that could resolve the issue. Safety and cost analyses to determine the cost-effectiveness of the proposed hardware fixes would also be necessary. An estimated $150,000 contract plus $20,000 for NRC contract support would be needed to accomplish this activity. Thus, the total cost of this activity was estimated to be $230,000.

The next step was to prepare and issue a generic letter to all licensees. Approximately 4 man-weeks ($10,000) were estimated to prepare and issue the letter. It was estimated that 6 man-months of staff time would be required to review and evaluate each licensee response. (This was equivalent to a $55,000 contract and $10,000 for NRC contract support.) Thus, the total estimated cost for this effort was $75,000.

Based on the above estimates, the total NRC cost for development of the possible solution was $355,000. Averaging this cost over the 134 affected plants resulted in a cost of $2,650/plant for development.

It was assumed that the staff would review the implementation of the requirements in the generic letter, review the test procedures, review plant-specific implementation plans, and prepare a safety evaluation. The cost for this review was estimated to be 4 man-weeks/plant. At $2,270/man-week, this cost was $9,080/plant.

An additional 0.5 man-wk/RY of NRC effort would be required for an annual review of the operation and maintenance of the solution. Summing this cost over the remaining lives of the affected plants at $2,270/man-wk resulted in a cost of $32,200/plant. Using a 5% discount rate, the present worth of this review was $17,900/plant.

Therefore, the total NRC cost for the development and implementation of the possible solution was estimated to be approximately $30,000/plant.

Total Cost: The total cost of implementation of the proposed solution was estimated to be $0.6M/plant.

Value/Impact Assessment

Based on the above estimates, the following value/impact scores were calculated for the three cases considered.

Best Estimate:

$$S = \frac{1,580 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$$

$$= 2,633 \text{ man-rem/\$M}$$

Lower Bound: $S = \frac{378 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$

$= 630 \text{ man-rem/\$M}$

Upper Bound: $S = \frac{26{,}752 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$

$= 44{,}587 \text{ man-rem/\$M}$

Other Considerations

Implementation of the possible solution was assumed to include repair, replacement, and testing of potentially susceptible isolators. This resulted in labor estimates of 34 man-days/plant for PWRs and 56 man-days/plant for BWRs in radiation zones. Radiation fields of 25 millirem/hr were assumed to exist inside containment where most of the isolators were located. Utilizing a 75% efficiency factor for labor in radiation zones, the occupational dose increase for implementation of the possible solution was estimated to be 9.1 man-rem/plant and 14.9 man-rem/plant for PWRs and BWRs, respectively.

Licensee labor requirements in radiation zones for operation and maintenance of the possible solution included:

	PWRs (man-days/RY)	BWRs (man-days/RY)
Annual Test Program	10	16
Replacement of Isolators	2.4	4.2
Total:	12.4	20.2

Again, utilizing a 75% efficiency factor for labor in radiation zones and radiation fields of 25 millirem/hr resulted in an estimated increase in ORE of 3.3 man-rem/RY and 5.4 man-rem/RY for PWRs and BWRs, respectively. Summing these values over the remaining lives of the affected plants (28.8 years for PWRs and 27.4 years for BWRs) resulted in an increase in ORE of approximately 95 man-rem/plant and 148 man-rem/plant for PWRs and BWRs, respectively.

CONCLUSION

The best estimate of public risk reduction associated with preventing leakage through electrical isolators was significant and indicated a high priority ranking. However, the calculation of risk reduction included a number of conservative assumptions. Generally, use of conservative assumptions where real data does not exist will always result in overprediction of potential risk reduction. In acknowledgement of the conservatism in the analysis, a medium priority ranking was assigned to this issue. This ranking was consistent with the qualitative judgments of the staff and was further supported by NRR's stated intention to process a research request to initiate an electrical isolator testing program to improve the current state of knowledge concerning isolator characteristics at less than maximum credible fault levels. The resolution of the issue was expected to address the safety concern of Issue 156.4.1.

In resolving the issue, the staff determined from operating experience that isolation devices perform satisfactorily in the operating environment and have not been exposed to failure mechanisms that resulted in signal leakage. This determination was based in part on plants that predominantly use electromechanical controls and may not be applicable to control systems with digital or electronic components. Therefore, RES recommended the development of an SRP[11] Section to provide review guidance for future plants that use digital systems, and for OLs that convert safety-related systems from analog to digital. The regulatory analysis will be published in NUREG-1453. Thus, this issue was resolved and no new requirements were established.[1511]

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ISSUE 152: DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS

DESCRIPTION

Historical Background

This issue was identified[1416] by DSIR/RES following ACRS concerns raised during the review of the resolution of Issue 87, "Failure of HPCI Steam Line Without Isolation," which addressed the design bases for those MOVs that isolate the HPCI, RCIC, and RWCU systems in BWRs. These design bases required that the MOVs close against loads imposed by a double-ended pipe break at design basis flow conditions.

In resolving Issue 87, the staff issued Generic Letter No. 89-10[1217] which required licensees to identify safety-related valves that might not perform adequately under design basis conditions. However, the ACRS believed that the design basis for the HPCI steam line valves and other valves in some plants might not specify the type of heavy duty. Thus, it was possible that heavy duty loads might not be considered for these valves by licensees in response to Generic Letter No. 89-10.[1217] The ACRS recommended that the staff amend the generic letter to require licensees to examine their design bases to determine if safety-related valves, including but not limited to MOVs, were capable of operating against blowdown loads that might not have been considered (by licensees) in their original designs.

Safety Significance

The inability of valves that might be subjected to significant blowdown loads to meet their design bases is a compliance concern. Therefore, the safety significance of this issue lies in the environmental conditions that could result from the inability of containment isolation valves to close under accident conditions. The resulting environmental conditions could cause the malfunction of equipment required to cool the reactor. This issue affects all operating and future plants.

Possible Solution

A possible solution to this issue would include the following: (1) amendment of Generic Letter No. 89-10[1217] to ensure complete compliance with the original design bases; (2) licensee review of design bases for compliance; (3) licensee analyses to assess operability of valves; and (4) hardware modification of isolation valves and additional licensee analyses to bring the valves into compliance with the original design bases.

PRIORITY DETERMINATION

Assumptions

It was assumed that 50% of all 112 operating plants will find that they are in compliance with the amended generic letter. Of the remaining 50% that will have to perform analyses, 80% will demonstrate compliance. Thus, only 10% of all operating plants will make hardware modifications and perform additional analyses to comply with the amended generic letter. Therefore, the potential exists for a reduction in public risk and occupational dose at approximately 11 plants: 7 PWRs and 4 BWRs. Future plants would not require any modifications since their design would be based on the requirements of the amended generic letter. Oconee 3 and Grand Gulf 1 were selected as the representative PWR and BWR, respectively.

Frequency Estimate

For PWRs, a steam line break was assumed to correspond to an S_3 LOCA. If this LOCA is not isolated, the potential exists for introducing a harsh environment into the containment which may affect the operation of certain components needed to mitigate the LOCA. These components were assumed to be MOVs and pumps, specifically for failure modes designated as hardware or control circuitry, found in accident sequences initiated by an S_3 LOCA.

It was assumed[64] that the potential for increased failure under harsh environmental conditions was not factored into the failure probabilities of the affected parameters in the original plant evaluations. Therefore, the base case failure probabilities were assumed to be 10% higher than their original values. For Oconee 3, this resulted in a base case core-melt frequency of 1.18×10^{-6} /RY.[64]

For BWRs, a steam line break was assumed to correspond to an S LOCA. Assuming the same accident scenario and resultant effects described above for PWRs, the base case core-melt frequency for Grand Gulf 1 was estimated[64] to be 2.48×10^{-7}/RY.

It was assumed that resolution of the issue would return the failure probabilities to their original values in both PWRs and BWRs; this represented a 10% reduction in the base case values. Thus, the adjusted case core-melt frequencies were estimated to be 1.01×10^{-6}/RY and 2.09×10^{-7} for Oconee 3 and Grand Gulf 1, respectively. The potential core-melt frequency reduction associated with the possible solution was calculated to be 1.7×10^{-7}/RY and 3.9×10^{-8}/RY for the affected PWRs and BWRs, respectively.

Consequence Estimate

The affected release categories for Oconee 3 were PWR-2, -3, -4, -5, -6, and -7 and the base case and adjusted case public risk were estimated to be 3.14 man-rem/RY and 2.68 man-rem/RY, respectively, with a potential reduction of 0.46 man-rem/RY. For the 7 affected PWRS with an average remaining life of 25.8 years, the public risk reduction was estimated to be (0.46)(7)(25.8) man-rem or 83 man-rem.

Affected release categories for Grand Gulf 1 were BWR-1 and -2 and the base case and adjusted case public risk were estimated to be 1.76 man-rem/RY and 1.48 man-rem/RY, respectively, with a potential reduction of 0.28 man-rem/RY. For the 4 affected BWRs with an average remaining life of 24.1 years, the estimated public risk reduction was (0.28)(4)(24.1) man-rem or 27 man-rem.

Therefore, the total public risk reduction associated with the possible solution was estimated to be 110 man-rem.[64]

Cost Estimate

Industry Cost: The review of design bases was estimated to require 6 man-weeks/plant at all 112 operating plants affected by the amended generic letter. At $2,270/man-week, this cost was estimated to be $1.525M.

Additional analyses at 56 plants (50% of all affected plants) were estimated to require 12 man-weeks/plant for a total cost of $1.525M. Equipment costs were estimated to be $20,000/plant (10% of all affected plants) that will have to make valve modifications. These modifications were estimated to require 8 man-weeks of skilled labor and 16 man-weeks for additional engineering analyses. Thus, the total estimated cost for 11 plants that require modifications was $0.82M and the total industry cost associated with the possible solution was $3.87M.

NRC Cost: It was estimated that 8 man-weeks would be required to amend Generic Letter No. 89-10[1217] at a cost of $18,000. Review of licensee responses from all 112 plants was estimated to require 2 man-weeks/plant. Responding to the half of these plants that would have to submit analyses was estimated to require 6 man-weeks/plant. For the 11 plants that would have to be modified, NRC review of the additional analyses was estimated to require 12 man-weeks/plant. Thus, the total NRC review time was estimated to be 692 man-weeks. At $2,270 man-week, this translated to a cost of $1.57M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(3.87 + 1.57)M or $5.44M.

Value/Impact Assessment

Based on a potential public risk reduction of 110 man-rem and an estimated cost of $5.44M for a possible solution, the value/impact score is given by:

$$S = \frac{110 \text{ man-rem}}{\$5.44\text{M}}$$

$$\sim 20 \text{ man-rem/\$M}$$

CONCLUSION

Based on the potential public risk reduction, this issue has a LOW priority ranking. Additional concerns raised[1509] by the ACRS on the ability of safety-related MOVs to close under pipe break conditions were addressed[1510] by the staff but did not affect the priority ranking of the issue.

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1217. NRC Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10) - 10 CFR 50.54(f)," June 28, 1989, (Supplement 1) June 13, 1990, (Supplement 2)

August 3, 1990, (Supplement 3) October 25, 1990, (Supplement 4) February 12, 1992.

1416. Memorandum for B. Morris from W. Minners, "Prioritization of Proposed New Generic Issue," December 4, 1989.

1509. Letter to J. Taylor from P. Shewmon, "Prioritization of Generic Issue 152, 'Design Basis for Valves that Might Be Subjected to Significant Blowdown Loads,'" April 23, 1993.

1510. Letter to J. Wilkins (ACRS) from J. Taylor(EDO), June 8, 1993.

ISSUE 153: LOSS OF ESSENTIAL SERVICE WATER IN LWRs

DESCRIPTION

Historical Background

The reliability of essential service water (ESW) systems and related problems have been an ongoing staff concern which has been documented in NUREG/CR-2797,[1334] IE Bulletins 80-24[201] and 81-03,[207] Generic Letter No. 89-13,[1259] and Issues 51, 65, and 130. In a comprehensive NRC review and evaluation of operating experience related to service water systems (NUREG-1275,[1079] Volume 3), a total of 980 operational events involving the ESW system were identified, of which, 12 resulted in complete loss of the ESW system. The causes of failure and degradation included: (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion); (2) ice effects; (3) single failures and other design deficiencies; (4) flooding; (5) multiple equipment failures; and (6) personnel and procedural errors.

In the resolution of Issue 130, the staff surveyed seven multiplant sites and found that loss of the ESW system could be a significant contributor to core damage frequency (CDF). The generic safety insights gained from this study supported previous perceptions that ESW system configurations at other multiplant and single plant sites may also be significant contributors to plant risk and should also be evaluated. As a result, this issue was identified[1280] by DSIR/RES to address all potential causes of ESW system unavailability, except those that had been resolved by implementation of the requirements stated in Generic Letter No. 89-13.[1259]

Safety Significance

At each plant, the ESW system supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink. The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate the consequences of the accident. Under normal operating conditions, the ESW system provides component and room cooling (mainly via the component cooling water system). During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to fire protection systems, cooling towers, and water treatment systems at a plant.

The design and operational characteristics of the ESW system are different for PWRs and BWRs and also differ significantly from plant to plant within each of these reactor types. The success criteria associated with the functions of an ESW system are also plant-specific. A complete loss of the ESW system could potentially lead to a core-melt accident, posing a significant risk to the public. This issue affected all plants not covered in the resolution of Issue 130 and included consideration of Issue B-32.

Possible Solutions

The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the NSSS. As a result, generic solutions (if needed) are likely to be different for PWRs and BWRs. The possible solutions are: (1) installation of a redundant intake structure including a service water pump; (2) hardware changes of the ESW system; (3) installation of a dedicated RCP seal cooling system; or (4) changes to TS or operational procedures. These potential improvements were considered for the seven multiplant sites covered in the scope of Issue 130; however, these options will now be evaluated for the remaining LWRs (65 PWRs and 39 BWRs).

PRIORITY DETERMINATION

Frequency Estimate

The CDF resulting from the loss of service water system (LOSW) has been estimated in a number of PRAs and is listed in Table 3.153-1.

TABLE 3.153-1
Estimated CDF Contribution from LOSW

Plant	Frequency (RY^{-1})
Plant A(Old PWR),[1333] 3 SWP/unit, CT	1.2×10^{-6}
Plant B(New PWR),[1333] 3 SWP, CT	1.6×10^{-5}
Plant C(Old BWR),[1333] Multiple SWS	2.7×10^{-4}
Plant D(Old PWR),[1333] 2-3 SWP, CT	6.7×10^{-5}
Plant E(New PWR),[1333] Unique SWS	9.0×10^{-6}
Plant F(New BWR),[1333] Multiple SWS	3.0×10^{-5}
Plant G(Old PWR)[1081]	1.2×10^{-4}
Plant N-T(Old and New PWRs, Mean CDF)[1408]	1.5×10^{-4}

CT = cross-tie
SWP = service water pump
SWS = service water system

The mean value of the above frequencies was calculated to be 8.3×10^{-5}/RY.

Consequence Estimate

Dose consequence was estimated on the basis of the 15 release categories defined in WASH-1400.[16] Based on Issues 65 and 130, the release categories of PWR-2 and BWR-2 are dominant for LOSW and these were estimated[64] to be 4.8×10^6 man-rem and 7.1×10^6 man-rem, respectively. Assuming an average remaining lifetime of 30 years for the affected plants and using the calculated mean CDF of 8.3×10^{-5}/RY, the public risk for the base case was calculated to be:

(1) PWRs: $W = (30)(4.8 \times 10^6)(8.3 \times 10^{-5})$ man-rem/reactor
$= 12{,}000$ man-rem/reactor

(2) BWRs: $W = (30)(7.1 \times 10^6)(8.3 \times 10^{-5})$ man-rem/reactor
$= 18{,}000$ man-rem/reactor

The consequence estimate of 12,000 man-rem/reactor was used for this analysis and compared favorably with the estimate of 9,700 man-rem/reactor calculated for Issue 130.

Cost Estimate

Industry Cost: The cost of installing a redundant intake structure, including a pump, was estimated in Issue 130 and showed a range from $12M to $72M, with a best estimate of approximately $43M.

The cost estimate for the solution involving hardware changes could include: additional crosstie, additional valving and piping, or additional water source (fire water). The cost for these hardware changes was expected to be less than that for redundant intake structures, but higher than that for TS or procedure changes. The least expensive solution was estimated to cost $50,000/plant to change requirements for TS or procedures.

NRC Cost: The NRC costs were negligible in comparison to the industry costs.

Total Cost: The estimated total NRC and industry cost of the possible solution was $50,000/reactor.

Value/Impact Assessment

Separate value/impact scores (S) were calculated for the four possible solutions:

Possible Solution	Mean-Value Public Risk (Man-rem/R)	Estimated Reduction Coefficient	Risk Reduction (Man-rem/R)	Estimated Cost/Reactor ($M)	S (Man-rem/$M)
1	12,000	0.8	9,600	43.0	220
2 or 3	12,000	0.5	6,000	<43.0	>220
4	12,000	0.1	1,200	0.05	24,000

The reduction coefficient was defined as the estimated effectiveness of the possible solution after implementation and was based on operational experience and engineering judgment.

Other Considerations

(1) The mean CDF was derived from PRAs and studies of 20 operating plants some of which have multiple units. However, since the ESW system is highly plant-dependent and the key contributor to CDF varies, the uncertainty of the mean CDF could be a factor of 10; this did not affect the priority ranking.

(2) The possible TS changes should be applied to PWRs only because the ESW systems for BWRs are already required in the cold shutdown or refueling mode. For BWRs, possible changes to operational procedures to cope with a complete loss of service water systems would apply.

(3) Issue 23 identified the LOSW as one of the events that could cause failure of RCP seals. Among the options considered by the staff in the resolution of Issue 23 was the installation of an alternate AC source to provide seal cooling and performance of plant modifications to allow backup cooling from an existing plant water system other than the ESW system. The reduction in CDF for this proposed resolution could be substantial (10^{-5}). Should this proposed solution be implemented, the CDF resulting from a LOSW event could be reduced as much as 50%.[1408] This reduced CDF, however, would still place Issue 153 in the high priority category as discussed in the value/impact assessment above. However, in resolving Issue 153, the staff was expected to consider the proposed resolution of Issue 23.

(4) Issue 51 addressed service water system fouling and was considered to be resolved with the implementation of the baseline fouling program required by Generic Letter No. 89-13.[1259] Implementation of this program could result in a CDF reduction[1258] of approximately 2.6 x 10^{-6}. As indicated earlier, biofouling was to be excluded from the resolution of Issue 153.

(5) Issue 65 was integrated into Issue 23 and its impact on Issue 153 was discussed above.

(6) Issue 130 addressed the limited scope of multiplant configurations with 2 ESW pumps per plant; the possible solution was limited to 7 PWRs with a total of 14 units. The resolution of Issue 153 included all plants not covered in the resolution of Issue 130.

CONCLUSION

Based on the potential public risk reduction, this issue was given a high priority ranking. In resolving the issue, the staff found that the concerns involving ESW system reliability were being addressed on a plant-specific basis under various ongoing NRC and industry initiatives such as the Service Water System Operational Performance Inspection Program, Generic Letter 89-13,[1259] the IPE Program, and EPRI research programs. In addition, ESW system reliability concerns were to be addressed by the Maintenance Rule and in the resolution of Issue 23. The staff's technical findings were documented in NUREG/CR-5910[1514] and SEASF-LR-92-022, Revision 1[1519]; the regulatory analysis was documented in NUREG-1461.[1512] Thus, this issue was RESOLVED and no new requirements were established.[1513]

REFERENCES

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1079. NUREG-1275, "Operating Experience Feedback Report," U.S. Nuclear Regulatory Commission, (Vol. 1) July 1987, (Vol. 2) December 1987, (Vol. 3) November 1988.

1081. NUREG-1150, "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, December 1990.

1258. NUREG/CR-5234, "Value/Impact Analysis for Generic Issue 51: Improving the Reliability of Open-Cycle Service-Water Systems," U.S. Nuclear Regulatory Commission, February 1989.

1259. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," July 18, 1989.

1280. Memorandum for C. Ader from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'Loss of Essential Service Water in LWRs,'" May 2, 1990.

1333. NSAC-148, "Service Water Systems and Nuclear Plant Safety," Electric Power Research Institute, May 1990.

1334. NUREG/CR-2797, "Evaluation of Events Involving Service Water Systems in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1982.

1408. NUREG/CR-5526, "Regulatory Analysis for the Resolution of Generic Issue 130: Essential Service Water System Failuress at Multi-Plant Sites," U.S. Nuclear Regulatory Commission, June 1991.

1512. NUREG-1461, "Regulatory Analysis for the Resolution of Generic Issue 153: Loss of Essential Service Water in LWRs," U.S. Nuclear Regulatory Commission, August 1993.

1513. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 153 (GI-153), 'Loss of Essential Service Water in LWRs,'" June 14, 1993.

1514. NUREG/CR-5910, "Loss of Essential Service Water in LWRs," U.S. Nuclear Regulatory Commission, August 1992.

1519. SEASF-LR-92-022, "Supplemental Study of Generic Issue No. 153, 'Loss of Essential Service Water in LWRs,'" Science & Engineering Associates, Inc., (Rev.1) January 1993.

ISSUE 155: GENERIC CONCERNS ARISING FROM TMI-2 CLEANUP

The TMI-2 Safety Advisory Board was established to provide the licensee, General Public Utilities Nuclear Corporation, with a qualified, independent appraisal of the cleanup of TMI-2, with particular emphasis on the assurance of public and worker health and safety. As a result of this appraisal, seven recommendations[1362] were forwarded to the NRC for evaluation. These recommendations were treated as separate generic issues as outlined below.

ISSUE 155.1: MORE REALISTIC SOURCE TERM ASSUMPTIONS

DESCRIPTION

During the TMI-2 accident, fission products did not behave as predicted with the analytical methods and assumptions used in the licensing process at that time and delineated in Regulatory Guides 1.3[213] and 1.4[214] and TID-14844.[73] The earliest expert predictions were that major core damage had occurred. However, the NRC and the licensee believed that core damage was minimal and calculations were redone to confirm this view. Approximately 50% of the core was in a molten state, but there is evidence that only about 55% of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. There is also evidence that less than 5% of the medium and low volatile fission products were released from the reactor vessel.[1362] These observations were based on research conducted since the TMI-2 accident.

It is now generally accepted that the chemical conditions in the reactor vessel were "reducing" in nature as opposed to "oxidizing." The elemental iodine was driven (or converted) to the iodide ion which very readily combined with available metallic ions. The water soluble character of these chemical forms prevented a major release of iodine to the atmosphere of the containment or auxiliary buildings and only a few Curies were released to the environment. Throughout the TMI-2 accident sequence, the chemical state was maintained such that the water-soluble character was preserved.

With the completion of a large number of PRAs since the TMI-2 event, the Advisory Board believed that it should be possible to list accident sequences with chemical conditions similar to TMI-2. Such a listing could provide a guide as to which accidents might be regarded as hazardous, or less hazardous, relative to the possible escape of iodine and could be useful in the future design of safety features. Since some of the assumptions used for source term considerations at TMI-2 were flawed in this respect, the Board recommended that the source term be restated using current scientific knowledge.[1362]

CONCLUSION

This issue is being pursued by the staff as part of comprehensive revisions to 10 CFR Parts 50 and 100 to reflect a better understanding of accident source terms and severe accident insights, as well as evaluate the impact of these phenomena on plant engineered safety features. A replacement for TID-14844[73] is being formulated, based on recent severe accident research findings, to reflect the current understanding of fission product release timing, iodine chemistry, and source term magnitude and composition. Thus, a solution to this issue has been identified and the issue is considered nearly-resolved.

ISSUE 155.2: ESTABLISH LICENSING REQUIREMENTS FOR NON-OPERATING FACILITIES

DESCRIPTION

At the time the TMI-2 event occurred, 10 CFR 50 contained regulations primarily for the design, construction, and operation of nuclear facilities but did not provide adequate guidance for the post-accident condition. Much was learned while the unit was being defueled and prepared for the post-defueling, monitored storage phase. The decommissioning rule[1364] issued in 1988 addressed the safe removal of nuclear facilities from service and the reduction of residual radioactivity to a level that permits release of the property for unrestricted use and termination of the operating license. The options for compliance with this rule are described in NUREG-0586[173] and include DECON, SAFSTOR, and ENTOMB. Decommissioning activities do not include the removal and disposal of spent fuel; these are considered to be operational activities.

Once a reactor is permanently shut down and defueled, it enters a storage phase until the licensee begins implementation of a decommissioning plan approved by the NRC. During the storage phase, requirements for security plans, operator licensing, emergency planning, etc., that were in effect while the plant was operational, may become unnecessary and burdensome to the licensee. Once all nuclear fuel is removed from the reactor site, the risk of an extraordinary accident, as defined in 10 CFR 50.54(w) and 10 CFR 140.11, is essentially eliminated. The Board recommended that regulatory guidance be developed for use by non-operating and defueled facilities during the storage phase prior to decommissioning.[1362]

CONCLUSION

This issue addressed changes in existing regulatory guidance that could significantly reduce licensee costs without any substantial change in public risk. Thus, it was classified as a Regulatory Impact issue. Revisions to 10 CFR 50.54(w) and 10 CFR 140.11(a)(4) may be necessary to address insurance coverage for non-operating and defueled facilities during the storage phase prior to decommissioning.[1363]

ISSUE 155.3: IMPROVE DESIGN REQUIREMENTS FOR NUCLEAR FACILITIES

DESCRIPTION

The Board recommended[1362] that the NRC undertake an effort to evaluate lessons learned at TMI-2 and incorporate them into the design of future nuclear plants. The recommendations suggested by the Board focused on recovery from a severe accident and were as follows:

(1) Prohibit the use of cinder blocks inside the reactor building (because they absorb so much contamination and become a radiological hazard) or designing the facility to be "robot friendly."

(2) Utilize higher range radiation instrumentation in order to monitor the environment inside the reactor building during a severe reactor accident.

(3) Based on design criteria and clear evidence that the TMI-2 containment building was not challenged, a reduction in criteria might be prudent based upon actual accident conditions. The NRC had reviewed in some detail the capability of reactor containment structures to withstand accident environments, including significant pressure increases; a review of these studies might be helpful and may lead to a reduction in design criteria. A similar effort for reactor vessels has not been undertaken and should be, considering the condition of the lower head of the TMI-2 reactor vessel with the severity of the accident.

(4) TMI-2 has also demonstrated the need to provide access to the underside of a reactor vessel for remote inspections to determine the extent of possible damage in the aftermath of a severe reactor accident. The 52 instrument penetrations in the lower head of the TMI-2 reactor vessel have been a concern since the discovery of once-molten material on the lower head of the reactor vessel and thus lower head integrity has been a major concern during the recovery efforts. For future reactor vessel design, it was recommended that in-core instrumentation penetrate the head instead of the bottom.

PRIORITY DETERMINATION

The four concerns outlined in this issue were evaluated separately below:

(1) In accordance with 10 CFR 50, Appendix I, nuclear power plants are required to keep occupational risk exposure (ORE) as low as is reasonably achievable (ALARA). Cinder blocks constitute one of the materials that are used inside the reactor building of some operating plants as local shielding to meet this ALARA criterion. Prohibiting the use of cinder blocks inside the reactor building would have no impact on public risk in the event of a severe accident. The use of other shielding materials that do not absorb as much contamination has the potential for decreasing the decontamination time (and ORE) following a severe accident.

Designing future nuclear plants to be robot-friendly will require spatial considerations for the mobility of robots that could drastically increase design, engineering, and construction costs. However, as is the case above, the use of robots would have no impact on public risk in the event of a severe accident; only occupational risk would be affected.

From NUREG/CR-2800,[64] the occupational dose from cleanup, repair, and refurbishment following a severe accident was estimated to be 19,860 man-rem. Even assuming that 50% of this dose can be reduced with either the elimination of cinder blocks or the use of a robot for cleanup and assuming a core-melt frequency of 10^{-5}/RY and an average remaining reactor life of 28 years, the potential dose reduction is approximately 3 man-rem/reactor. Thus, this concern has negligible risk reduction potential and consideration of costs would only lower its priority ranking.

(2) The recommendation to utilize higher range radiation instrumentation in order to monitor the environment inside the reactor building during a severe accident was addressed by TMI Action Plan Item II.F.1. This item was clarified in NUREG-0737[98] and required implementation at all plants. Thus, this concern has been addressed by the staff.

(3) For future plants, the Commission's Severe Accident Policy Statement established the criteria and procedural steps under which new designs for nuclear power plants could be acceptable for meeting severe accident concerns. Rather than a reduction of criteria, it is expected that future plants would have to achieve a higher standard of severe accident safety performance, including clarification of containment performance. The staff's plan of action in this area was presented to the Commission in SECY-92-292.[1427] Operating plants were assessed under the Containment Performance Improvement Program (see Issue 157).

The mode of vessel failure, including investigation of the TMI-2 vessel, is being pursued by the staff as part of its severe accident research program.[1362] The results of this research will determine whether changes to future vessel design will be warranted. Thus, this concern is being addressed by the staff.

(4) The relocation of in-core instrumentation is being addressed by NSSS vendors in the design of future plants which is subject to review and approval by the staff. For example, the bottom-mounted instrumentation penetrations have been eliminated in the Westinghouse AP600 design to reduce building volume and costs significantly. Thus, this concern is being addressed by the staff.

CONCLUSION

Of the four recommendations contained in this issue, two were being addressed in other ongoing programs and one had been previously addressed by the staff. The remaining recommendation had negligible risk reduction potential and, therefore, was not considered to be safety-significant. Thus, this issue was DROPPED from further consideration as a new and separate issue.

ISSUE 155.4: IMPROVE CRITICALITY CALCULATIONS

DESCRIPTION

The Board believed that doubts still remained as to whether the TMI-2 core became critical, or was very close to critical, during the TMI-2 accident and recommended that the NRC establish guidelines that deal with criticality following a severe reactor accident.[1362] These guidelines should take into account abnormal geometries and possible core conditions that could result from the accident. The Board believed that the accident scenario developed by the TMI-2 licensee was sufficiently detailed that a series of geometric configurations could be simulated for criticality calculations. Variables that could be estimated reasonably well included the presence of water, oxidation of cladding, melting and movement of fuel, melting of poison rods, and movement of poison.

CONCLUSION

The safety concern was addressed by DSR/RES in SARP Task 4.3: Investigate the Possibility and Consequences of Recriticality in Degraded BWR Cores.[1362] The staff's study was documented in NUREG/CR-5653[1379] in which it was concluded that there is the potential for recriticality in BWRs, if core reflood occurs after control blade melting has begun but prior to significant fuel rod melting. However, a recriticality event would most likely not generate a pressure pulse significant enough to fail the vessel. Two strategies were identified that would aid in regaining control of the reactor and terminate the recriticality event before containment failure pressures are reached: (1) initiation of boron injection at or before the time of core reflood, if the potential for control blade melting exists; and (2) initiation of RHR suppression pool cooling to remove the heat load generated by the recriticality event and extend the time available for boration.

The issue was not considered to be a major concern for PWRs because of their design that includes a safety injection system for supplying borated water to the core. Furthermore, it was concluded in NUREG/CR-5856[1417] that, during a severe accident, an unmoderated recriticality of the molten, consolidated portion of a degrading core cannot occur at U_{235} enrichments characteristic of a PWR. Based on the staff's efforts in addressing the safety concerns in the SARP, this issue was DROPPED from further pursuit as a new and separate issue.

ISSUE 155.5: MORE REALISTIC SEVERE REACTOR ACCIDENT SCENARIO

DESCRIPTION

The TMI-2 event was a severe accident in which approximately 50% of the core was in a molten state at some point during the accident. Approximately 20 tons of the once-molten debris poured through the core support structure into the water-filled lower plenum and onto the lower head of the reactor vessel. Most codes in use at that time would have predicted a failure of the lower head under these conditions. The severity of the accident showed that the reactor vessel was more difficult to fail than was anticipated.

The Board recommended that in-vessel core-melt progression for severe accidents be studied further by the NRC and that the results be incorporated into existing codes and standards. The Board believed that codes should have the capability to reproduce the TMI-2 accident with reasonable accuracy before they can be accepted as predictive tools.[1362]

CONCLUSION

The safety concern is being addressed by DSR/RES in SARP Issue L2: In-Vessel Core Melt Progression and Hydrogen Generation.[1362] In considering core-melt progression, the staff will treat BWRs and PWRs separately because of their different fuel assembly, control element, and lower plenum structures. Concerns common to both BWRs and PWRs are: (1) the integrity of core structures; (2) the mode of core material relocation; (3) hydrogen generation; (4) the mode of bottom head failure; and (5) the effects of water injection. The answers to the above concerns will be different because of the physical differences of BWRs and PWRs. TMI-2 data and the results of new experiments and model development will be examined by the staff in its research. Based on the staff's efforts on SARP Issue L2, Issue 155.5 was DROPPED from further pursuit as a new and separate issue.

ISSUE 155.6: IMPROVE DECONTAMINATION REGULATIONS

DESCRIPTION

The Board believed that the decontamination techniques used throughout the nuclear industry for small activities were not applicable to large-scale activities and recommended that the NRC use the experience gained from the TMI-2 accident to prepare guidelines for decontamination and decommissioning of nuclear plants.[1362]

CONCLUSION

Traditionally, the NRC has not developed or approved decontamination techniques. Due to the many ways in which decontamination can be accomplished and the rapidly evolving technology in this area, it is not practical or beneficial for the NRC to establish guidelines for decontamination techniques. Rather, the NRC has focused on the development of criteria which set standards for exposure of workers and the public (e.g., 10 CFR 20), the levels of allowable residual contamination, and the handling and disposal of the radioactive waste generated. Efforts at establishing residual contamination criteria applicable to decommissioning were in progress as described below.

In June 1991, the Commission deferred[1412] implementation of the Below Regulatory Concern (BRC) policy but reaffirmed its intentions to carry out its responsibilities to address issues related to waste disposal, consumer products, recycling of materials, and decontamination and decommissioning, as necessary, on a case-by-case basis in the manner in which these issues were considered, prior to the development of the BRC policy statement. In this regard, the staff was directed to continue its accelerated efforts in completing the technical basis for rulemaking on residual contamination criteria.

In accordance with SECY-92-045,[1413] the staff is proceeding with an enhanced participative rulemaking process to develop radiological criteria for decommissioning. The staff's effort will be tracked in the NRC Regulatory Agenda (NUREG-0936). Based on the above considerations, Issue 155.6 was DROPPED from further pursuit as a new and separate issue.

ISSUE 155.7: IMPROVE DECOMMISSIONING REGULATIONS

DESCRIPTION

The Board raised concerns over the requirements for the disposal of highly contaminated components from a nuclear plant during decommissioning and recommended that regulations be developed.[1362]

CONCLUSION

The TMI-2 experience was considered by the staff in the development of the decommissioning rule[1364] in 1988. Industry options for complying with this rule are described in NUREG-0586[173] and include DECON, SAFSTOR, and ENTOMB. As part of its resolution of Issue B-64, "Decommissioning of Reactors," the staff is currently developing an SRP[11] Section for use in its review of licensee decommissioning plans. Concurrent with this effort is the development of two Regulatory Guides: DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors"; and DG-1006, "Records Important for Decommissioning of Nuclear Reactors." Thus, Issue 155.7 was DROPPED from further consideration as a new and separate issue. The related concern of decommissioning prematurely shutdown plants was addressed in Issue 155.2.

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73. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962.

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

173. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," U. S. Nuclear Regulatory Commission, August 1988.

213. 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, (Revision 1) June 1973, (Revision 2) June 1974.

214. 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, (Revision 1) June 1973, (Revision 2) June 1974.

1362. Memorandum for E. Beckjord from F. Gillespie, "Generic Concerns Arising from TMI-2 Cleanup," February 21, 1991.

1363. Memorandum for E. Beckjord from F. Gillespie, "Request for Generic Rulemaking Concerning Decommissioning Issues," January 7, 1992.

1364. Federal Register Notice 53 FR 24018, "10 CFR Parts 30, 40, 50, 51, 70, and 72, General Requirements for Decommissioning Nuclear Facilities," June 27, 1988.

1379. NUREG/CR-5653, "Recriticality in a BWR Following a Core Damage Event," U.S. Nuclear Regulatory Commission, November 1990.

1382. NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Revision 1) December 1992.

1412. Memorandum for J. Taylor, et al., from S. Chilk, "SECY-91-132 - Evaluation of the Feasibility of Initiating a Consensus Process to Address Issues Related to the Below Regulatory Concern Policy," June 28, 1991.

1413. SECY-92-045, "Enhanced Participatory Rulemaking Process," February 7, 1992.

1417. NUREG/CR-5856, "Identification and Evaluation of PWR In-Vessel Severe Accident Management Strategies," U.S. Nuclear Regulatory Commission, March 1992.

1427. SECY-92-392, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs," August 21, 1992.

ISSUE 156: SYSTEMATIC EVALUATION PROGRAM

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older, operating nuclear power plants. The SEP was divided into 2 phases. In Phase I, the staff defined 137 issues for which regulatory requirements had changed enough over time to warrant an evaluation of those plants licensed before the issuance of the SRP.[11] In Phase II, the staff compared the design of 10 of the 51 older plants to the SRP[11] issued in 1975. Based on these reviews, the staff identified 27 of the original 137 issues that required some corrective action at one or more of the 10 plants that were reviewed. The staff referred to the issues on this smaller list as the SEP "lessons learned" issues and concluded that they would generally apply to operating plants that received operating licenses before the SRP[11] was issued in 1975.

In SECY-84-133,[814] the staff presented the 27 SEP issues to the Commission as part of a proposal for an Integrated Safety Assessment Program (ISAP). The intent of the ISAP was to review safety issues for a specific plant in an integrated manner. Two SEP plants participated in the ISAP pilot efforts. Following the review of these two pilot plants, ISAP was discontinued.

In SECY-90-160,[1443] the staff forwarded for Commission approval a proposed license renewal rule and supporting regulatory documents. In this paper, the staff stated that certain unresolved safety issues could weaken the generic justification of the adequacy of the current licensing bases argument. These issues included SEP topics for 41 older plants that had not been explicitly reviewed under Phase II of the SEP. The Commission requested that the staff keep it informed of the status of the program to determine how the SEP "lessons learned" issues had been factored into the licensing bases of operating plants.

Resolution of the 27 SEP issues was deemed by the staff to be important to the development of the license renewal rulemaking. The key regulatory principle underlying the license renewal rule is that the current licensing bases (CLBs) at all operating nuclear power plants, with the exception of age-related degradation, provide adequate protection to the public health and safety. This principle is reflected in the provisions of the license renewal rule which limit the renewal decision to whether age-related degradation has been adequately addressed to assure continued compliance with a plant's CLB. In order to adopt this approach, the NRC must be able to provide a technical basis for the key principle of license renewal. Accordingly, the rulemaking included a technical discussion documenting the adequacy of the CLB for all nuclear power plants, in both the statement of considerations and in NUREG-1412.[1444] However, as discussed in SECY-90-160,[1443] the staff identified a potential weakness in the discussion of the adequacy of the CLB with regard to the 41 older, non-SEP plants. To address this potential weakness, the staff undertook an effort to determine whether or not each SEP issue either had been or was being addressed by other regulatory programs and activities.

The staff completed this effort and placed each SEP issue into one of the following categories: (1) issues that had been completely resolved (i.e., necessary corrective actions had been identified by the staff, transmitted to licensees, and implemented by licensees); (2) issues that were of such low safety significance so as to require no further regulatory action; (3) issues that were unresolved, but for which the staff had identified existing regulatory programs that cover the scope of the technical concerns and whose implementation would resolve the specific SEP issue (such as IPE and IPEEE); and (4) issues that were unresolved and regulatory actions to resolve the issues had not been identified. The 27 SEP issues and applicable regulatory programs were summarized and presented in SECY-90-343.[1351] The staff concluded that the 22 SEP issues in Categories 3 and 4 remained unresolved for purposes of justifying the adequacy of the CLB for some portion of the 41 older, non-SEP plants. The following is an evaluation of these 22 issues: nineteen from Category 3 and three from Category 4.

ISSUE 156.1.1: SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that safety-related structures, systems, and components were adequately protected against excessive settlement. The scope included the review of subsurface materials (soils or geologic) and foundations to assess the potential static and seismically-induced settlement of all safety-related structures and buried equipment.

Excessive settlement or collapse of foundations and buried equipment for structures, systems, and components under either static or seismic loading could result in failure of structures, interconnecting piping, control systems or cables, or other equipment (tanks, etc.) such that the capability to safely shut down a plant, or mitigate the consequences of an accident, could be compromised.

There are two specific concerns in this issue: (1) the potential impact of static soil settlements on foundations and buried equipment where the soil may not have been properly prepared; and (2) seismically-induced differential settlement and potential soil liquefaction following a postulated seismic event. These two concerns are limited only to plants that have soil-supported, safety-related structures (including vertical, field-erected tanks) and soil-buried piping and components (including tanks) that have the potential for excessive settlement but were not reviewed to the pertinent SRP[11] Sections 2.5.4 and 2.5.5.

For the 41 older, non-SEP plants with OLs issued before 1975, any impact of static settlement on structural foundations (including the foundations of buried components) should become noticeable in the first 5 to 10 years. Thus, any significant settlement would have already been revealed and warranted corrective action. In addition, the ongoing IPEEE program[1354] has elements in its seismic task which requires that, for plants on soil sites, potential seismically-induced settlement and soil liquefaction should be assessed during its implementation.

CONCLUSION

This issue is being addressed by the SRP[11] for future plants as well as for operating plants with OLs issued after 1975. For the 51 older, operating plants, this issue was considered resolved for the 10 SEP plants. For the remaining 41 non-SEP, operating plants, any significant static settlement would have been revealed already and warranted corrective action. The concern on the seismically-induced settlement and soil liquefaction for these 41 older, non-SEP operating plants will be addressed during the implementation of the IPEEE program. Therefore, Issue 156.1.1 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.1.2: DAM INTEGRITY AND SITE FLOODING

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The safety concern was the ability of a dam to prevent site flooding and ensure a cooling water supply. The safety features of a dam would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressure or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping. The objective of this issue was to ensure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained water are prevented. Plants must provide the basis for ensuring that all safety-related structures, systems, and components are adequately protected against flooding that might result from dam failures. Further, review of licensee procedures would determine whether an adequate supply of cooling water exists in the ultimate heat sink during normal and emergency operations. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

If a dam exists in the vicinity of a nuclear power plant, it will have to meet one of the following criteria:

(1) If the dam provides impoundment for an ultimate heat sink at a plant or provides flood protection, the dam is an essential part of the plant and the safety of the dam needs to be ensured throughout the life of the plant. The dam has to be designed and remain stable under both static and seismic conditions.[686,916]

(2) If the dam provides impoundment only for plant operation, but not as a part of the ultimate heat sink, there are no regulatory requirements for dam design. However, the flood conditions that could be caused by dam failures should be considered in establishing the design basis flood.[687] When upstream dams or other features that provide flood protection are present, in addition to the analyses of the most severe floods that may be induced by either hydrometeorological or seismic mechanisms, reasonable combinations of less severe flood conditions and seismic events should be considered in establishing the design basis flood.

Currently, the Dam Safety and IPEEE Programs address the safety and the flooding effects of dams. Under the Dam Safety Program,[1355] the NRC will request licensees to ascertain whether a dam or impoundment exists at their plant sites that is safety-related and integral to the operation of the plant. The NRC will also determine if any other dams exist for the facility that are not safety-related. The results of this effort will be used to update the NRC Dam Inventory to define those dams that should be considered under the federal guidelines,[1356] i.e., subject to a specific NRC evaluation in accordance with the federal guidelines.

The evaluation will address the design bases for the dam, the design, construction, testing, and inspection processes, as well as the operation, maintenance, and surveillance programs that must function during the life of the plant. If the federal guidelines are not met, the NRC will notify the affected licensees and set a timetable for implementation of the Dam Safety Program. The NRC will conduct inspections of licensee dams, related programs, and actions taken by the licensees, as well as review documents and data important to the safety of the dams. The criteria, frequency, and scope of the inspections shall, as a minimum, meet the federal guidelines. Where inspection findings and any subsequent analyses define inadequate margins of safety regarding dam failure, the NRC will require the affected licensees to undertake a rehabilitation program to upgrade the safety of the dams. The schedule for completion of such upgrades will be based on a case-by-case review.

Under the IPEEE, the safety of dams will be assessed by all licensees in the process of searching for severe accident vulnerabilities due to external events.[1222,1354] If the failure of these dams would have significant consequences, i.e., a breach of an ultimate heat sink which might lead to a severe accident, they would have to be evaluated and inspected to assess their existing condition and vulnerability to earthquakes. If the failure of an upstream dam could lead to significant flooding at a site, i.e., the postulated flood exceeded the design basis flood and might lead to a severe accident, the effect of flooding will have to be addressed in the IPEEE.

CONCLUSION

The safety concerns of dam integrity and site flooding will be addressed in the implementation of the IPEEE and the Dam Safety Programs at the 41 plants affected by this issue. Therefore, Issue 156.1.2 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.1.3: SITE HYDROLOGY AND ABILITY TO WITHSTAND FLOODS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The concerns of this issue included identifying the site hydrologic characteristics, the capability of structures important to safety to withstand flooding, the determination of the adequacy of the cooling water supply, and the ISI of water control structures. Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design, or operating limitations with regard to floods, and water supply requirements. The specific items to be reviewed in this issue were:

(1) Hydrologic Description - To ensure that plant design reflects appropriate hydrologic conditions.

(2) Flooding Potential and Protection - To ensure that the plant is adequately protected against floods.

(3) Ultimate Heat Sink - To ensure an appropriate supply of cooling water is available during normal and emergency shutdowns.

(4) ISI of Water Control Structures - To ensure an adequate inspection program is in place to prevent water control structure deterioration or failure which could result in flooding or loss of the ultimate heat sink.

The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

At a nuclear plant, the safety-related structures, systems, and components, identified in accordance with Regulatory Guide 1.29,[916] must be designed to withstand the conditions resulting from the worst probable site-related flood and retain the capability for shutdown and maintenance.[687] Alternatively, NRC permits licensees not to design against the worst flood conditions for safety-related structures, systems, and components if sufficient warning time is shown to be available to shut down the plant and implement adequate emergency procedures. However, the safety-related structures, systems, and components must be designed to withstand the conditions resulting from a Standard Project Flood (with a flowrate about 40 to 60% of the probable maximum flood).[687]

On June 28, 1991, the NRC requested all licensees to conduct an IPEEE to search for severe accident vulnerabilities due to external events.[1222] External flooding is one of the events that will be addressed in the IPEEE.[1354] All licensees will have to examine the flood designs and associated flood protection measures at their sites to determine if severe accident vulnerabilities due to external floods exist. Therefore, the above Items 1 and 2 have been addressed in the external flood portion of the IPEEE program.

Item 3 is related to maintaining the functionality of the service water system and the decay heat removal system of the plant. The severe accident vulnerability resulting either from failure or unavailability of the ultimate heat sink is one of the important items to be examined in the IPE and IPEEE programs.

Item 4 is related to the Dam Safety Program[1355] to be implemented by the NRC. Under this program, the NRC will evaluate licensee inspection procedures and surveillance programs in accordance with federal guidelines.[1356] If these guidelines are not met, the NRC will notify the affected licensees and set a timetable for the upgrading of their inspection programs. The NRC will conduct inspections of the licensee facilities and review related programs and actions taken by the licensees, as well as review documents and data important to the safety of the plants. The inspection criteria, frequency, and scope of the inspections shall, as a minimum, meet the federal guidelines. Where inspection findings and any subsequent analyses reveal inadequacies, the NRC will require the affected licensees to upgrade their ISI programs.

CONCLUSION

The safety concerns of site hydrologic characteristics and the capability of plants to withstand flooding will be addressed in the implementation of the IPE, IPEEE, and Dam Safety Programs at the 41 plants affected by this issue. Therefore, Issue 156.1.3 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.1.4: INDUSTRIAL HAZARDS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that the integrity of safety-related structures, components, and systems will not be damaged by potential hazards from nearby transportation, storage, or industrial facilities. Such hazards include: (1) shock waves and thermal flux from nearby explosions of munitions or explosive gases or chemicals; (2) drifting toxic/explosive vapor clouds; (3) aircraft; and (4) missiles that can result from nearby explosions, such as a rocketing chemical tank car. In a few past licensing cases, reactor containment and intake structure hardening and pipeline relocation have been required to ensure safety of the plants. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Regulatory Guide 4.7[1372] and SRP[11] Sections 2.2.1, 2.2.2, and 2.2.3 have been used since 1975 in the design of nuclear power plants for protection against industrial hazards. In addition, Regulatory Guides 1.78,[1373] 1.91,[1374] and 1.95[1375] were issued to provide further regulatory guidance in this area. Prior to the issuance of these criteria, offsite hazards had been an area of long-standing concern and were reviewed on a case-by-case basis.

Supplement 4 to Generic Letter No. 88-20[1222] required all licensees to conduct an Independent Plant Examination of External Events (IPEEE) to search for severe accident vulnerabilities due to external events. Industrial hazards comprise one of the external events that will be addressed in the IPEEE.[1354]

CONCLUSION

Based on past staff reviews, existing review criteria and guidance, and the implementation of the IPEEE program for all plants, the concern for industrial hazards has been adequately addressed. Therefore, Issue 156.1.4 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.1.5: TORNADO MISSILES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] All plants licensed after 1972 were designed for protection against tornadoes. The concern existed, however, that plants constructed prior to 1972 may not be adequately protected, in particular, those reviewed before 1968 when criteria on tornado protection were first developed. The objective of this issue was to ensure that safety structures, systems, and components can withstand the impact of an appropriate postulated spectrum of tornado-generated missiles. The failure of safety-related structures, systems, or components due to a tornado-induced missile could compromise the ability of a plant to safely shut down. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A plant must be designed to remain in a safe condition in the event that the most severe tornado that can be reasonably predicted occurs at the plant site as a result of severe meteorological conditions. All safety-related structures, systems, and components must be designed to withstand the effects of the design basis tornado, tornado-generated missiles, and other tornado-induced effects.[42,816]

Under the IPEEE program, all licensees are required to examine their plants to determine if severe accident vulnerabilities due to high winds/tornadoes exist.[1222,1354] The criteria used for plant design (such as the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) will be examined. The reporting criterion, 10^{-6}/year CDF, specified for the IPEEE, however, is considered to be less stringent compared to the CDF associated with tornado missiles design criteria (a product of combining the probability of exceedance associated with the design basis tornado and the conditional failure probability associated with engineering design and construction against tornado missiles). Therefore, meeting the objectives of the IPEEE does not mean, in this situation, that current NRC guidelines for tornado design have been met. Thus, the staff believes that any vulnerability associated with tornado missiles will be evaluated and reported in the IPEEE submittals.

CONCLUSION

The safety concern for tornado missiles will be addressed in the implementation of the IPEEE Program at the 41 plants affected by this issue. Therefore, Issue 156.1.5 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.1.6: TURBINE MISSILES

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern was the potential damage from turbine missiles in nuclear plants licensed before 1973.

As a result of turbine disc failures at two nuclear plants and a number of non-nuclear plants prior to 1973, the staff believed that high energy missiles could be generated from steam turbines with the potential for causing failures in safety-related systems. The two areas of concern were: (1) failures at design overspeed because of degraded disc material, poor ISI of flaws, or chemistry conditions leading to stress corrosion cracking (SCC); and (2) destructive overspeed failures that would bring into question the reliability of electrical overspeed protection systems, the reliability and testing programs for stop and control valves, and the ISI of valves. For plants licensed after 1973, the safety concerns of this issue were reviewed by the staff as part of its OL activities; turbine overspeed protection designs were found acceptable and the magnitude of the potential damage from turbine missiles was determined to be plant-specific.

CONCLUSION

The safety concerns of this issue were addressed in the evaluation of Issue A-37, "Turbine Missiles," which focused primarily on plants licensed prior to November 1976; SRP[11] requirements for turbine design were issued for use by CP applicants after this date. Based on the historical failure rate of turbines used in the evaluation, Issue A-37 was determined to have little safety significance. No new data were provided in SECY-90-343[1351] that changed this conclusion. Therefore, this issue was DROPPED from further consideration as a new and separate issue.

ISSUE 156.2.1: SEVERE WEATHER EFFECTS ON STRUCTURES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include straight winds, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site. The objective of this issue was to identify those meteorological conditions which should be considered in the structural reviews to determine the ability of structures to withstand conditions such as flooding, wind, tornadoes, hurricanes, tsunamis, and seiches. The dynamic effects of waves, tornado pressure drop loading, and possible in-leakage due to floods were to be considered. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A nuclear power plant must be designed to remain in a safe condition in the event that the most severe weather conditions that can reasonably be predicted at the site occurs. All the safety-related structures must be designed to withstand the effects of the design basis flood, wind, hurricane, tornado, wind/tornado-generated missiles, and other wind/tornado-induced effects.[916]

Under the IPEEE program, all licensees were requested to examine their plants to determine if severe accident vulnerabilities due to floods or high winds/tornadoes exist.[1222,1354] Licensees were expected to examine their design criteria (such as the design flood level, the hydrostatic pressures against the structures, the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) used for plant structures to determine if the 1975 SRP[11] criteria are satisfied. If a plant conforms to these criteria, it will be judged that the contribution to CDF from the effects of severe weather is less than 10^{-6}/year and the IPEEE screening criterion would be met. Otherwise, additional evaluation will have to be made to establish severe accident vulnerabilities due to the effects of severe weather. The reporting criterion of 10^{-6}/year CDF specified for the IPEEE will provide a means by which the ability of a nuclear power plant to withstand severe weather conditions can be reviewed and examined for severe weather-induced vulnerabilities.

Snow and ice loads, when accompanied by strong winds, have caused several complete and partial losses of offsite power and the potential of causing severe accidents at a particular site will be evaluated in the IPE program. Snow and ice loads alone, are judged, based on limited PRA experience, to be unlikely to cause significant structural failure that might lead to severe accidents at nuclear power plants.

CONCLUSION

The safety concern of severe weather effects on structures will be addressed in the implementation of the IPEEE program. Therefore, Issue 155.2.1 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.2.2: DESIGN CODES, CRITERIA, AND LOAD COMBINATIONS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] With the development of nuclear power, provisions addressing nuclear power plants were progressively introduced into codes and standards to which plant buildings and structures are constructed. Because of this evolutionary development, older nuclear power plants conform to a number of different versions of codes and standards, some of which have since undergone considerable revision. There has likewise been a corresponding development of other licensing criteria, resulting in similar non-uniformity in many of the requirements to which plants have been licensed.

Individual SEP plant reviews identified specific areas of structural design code changes for which the previous codes used in the SEP review required greater safety margins than earlier versions of the codes, or for which no original code provision existed. Most plants demonstrated that safety margins in building structures were not significantly lower than those required by the codes and standards used in the SEP review. A few SEP plants required certain modifications to plant structures.

The concern of this issue was to provide assurance that building structures that house systems and components important to safety are capable of withstanding the effects of natural phenomena such as earthquakes.[916] tornadoes, (See Issue 156.1.5), hurricanes, and floods without loss of capability to perform their safety function. These events could cause walls or roofs to collapse damaging equipment that perform a safety function, thereby increasing the likelihood of a transient or LOCA.

CONCLUSION

On June 28, 1991, Supplement 4 to Generic Letter 88-20[1220] was issued requesting all licensees to perform an IPEEE. Under the IPEEE program, all licensees were requested to perform a plant-specific evaluation to determine if vulnerabilities to severe accidents initiated by natural phenomena exist.[1222,1354] The as-built structures, systems, and components in conjunction with operating plant conditions will be used to assess the adequacy of plant safety. Although this program does not directly address the effects of specific structural design code changes, it does in part focus on evaluating the capability of building structures to withstand natural phenomena and to search for cost-effective improvements that can be made to either prevent or reduce the impact of severe accidents. Thus, the staff believed that any severe accident vulnerabilities associated with the effects of natura phenomena on building structures will be evaluated and reported in the IPEEE submittals.

The safety concern with respect to the capability of building structures to withstand the effects of natural phenomena will be sufficiently addressed in the implementation of the IPEEE program at the 53 operating plants (34 PWRs and 19 BWRs) affected by this issue. Therefore, Issue 156.2.2 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.2.3: CONTAINMENT DESIGN AND INSPECTION

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review the inspection program for tendons in prestressed concrete containment structures to determine whether the inspection programs included testing of prestressed tendons, checking for corrosion or relaxation and possible deterioration of prestressed containments, and whether the concrete in the containment dome or walls degraded due to shrinkage or creep. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

The concerns about the tendons were addressed in Issue 118, "Tendon Anchor Head Failure," which was identified when a dented and leaking tendon grease cap was found during inspection at Farley Unit 2. The generic implications of tendon anchor head failures were studied under Issue 118 and tendon inspection and surveillance programs were developed that could be followed by licensees to mitigate or reduce such problems. The guidance for inspection and surveillance are contained in Regulatory Guides 1.35[481] and 1.35.1.[1360]

The containment dome or wall degradation due to shrinkage or creep is an age-related factor and is also addressed in Regulatory Guide 1.35.1. For license renewal applications, this concern was addressed in Draft Regulatory Guide DE-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," which will resolve the concern when issued in final form.

10 CFR 50 Appendix A (GDC 53), as implemented by Regulatory Guide 1.35,[481] requires that measured tendon forces (guidance provided in Regulatory Guide 1.35.1[1360]) be compared with acceptance criteria. This issue was reviewed by the staff for all SEP plants and accepted on a case-by-case basis, as documented in SERs; some of these plants also developed ISI programs.

CONCLUSION

The safety concerns of containment design and inspection at the 41 plants affected by this issue were addressed in the resolution of Issue 118. Beyond the normal life of the plants, the age-related concrete degradation concern will be addressed in the License Renewal Program. Therefore, 156.2.3 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.2.4: SEISMIC DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

DESCRIPTION

This issue is of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review and evaluate the original seismic design (seismic input, analysis methods, design criteria, seismic instrumentation, seismic classification) of safety-related plant structures, systems, and components to ensure the capability of plants to withstand the effects of an earthquake. Further, this issue would verify whether the free field ground motion specified for plant design adequately represents the vibratory ground motion associated with a postulated SSE at each plant. The free field ground motion will be utilized as the input to analyses to verify the design adequacy of structures, piping, and equipment. This review and evaluation will address the SSE only, since it represents the most severe event that must be considered in plant design. The scope of the review includes three major areas: (1) the integrity of the reactor coolant pressure boundary; (2) the integrity of fluid and electrical distribution systems related to safe shutdown; and (3) the integrity of mechanical and electrical equipment and engineered safety features systems (including containment). This issue did not call for a detailed review of all safety-related structures, systems, and components; rather, a sampling approach supported by a set of confirmatory analyses were to be performed. The sample size and confirmatory analyses were to be increased, if necessary. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

GDC 2 of Appendix A to 10 CFR 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. An earthquake is one of the natural phenomena whose effects nuclear power plants must be designed to withstand and remain in a safe condition.

In Supplement 4 to Generic Letter No. 88-20,[1222] licensees were required to conduct an IPEEE to search for severe accident vulnerabilities due to external events. A seismic event is one of the external events that should be addressed in the IPEEE.[1371] All licensees will have to review and evaluate the seismic capabilities of their plants (the as-built, as-operated plants) to withstand the earthquake effects well beyond the design basis and to determine if severe accident vulnerabilities due to seismic events exist at their plants. The seismic input has been evaluated by the staff in the Eastern United States Probabilistic Seismic Hazard Program and the results have been factored into the process of determining the seismic review scope in the IPEEE.

The seismic qualification of mechanical and electrical equipment is being resolved by the implementation of the resolution of Issue A-46. A seismic IPEEE can be accomplished by performing either a seismic PRA with enhancements or a seismic evaluation using a seismic margins method with enhancements. The review scope may vary from plant to plant depending on the selected method and the prescribed seismic hazard condition at the site. Even with the minimum effort under the IPEEE seismic program, at least two success paths (a preferred and an alternative) to shut down and maintain a plant in a safe shutdown condition will be evaluated.[1371] This process, when using the seismic margins approach, might not provide a detailed review of all safety-related structures, systems, and components, but it will represent a sampling approach, thus fulfilling the objective of Issue 156.2.4. Furthermore, if warranted as a result of staff review, additional analyses on selected safety-related structures, systems, and components can be performed.

CONCLUSION

The safety concerns for the seismic design of structures, systems, and components will be addressed in the implementation of the IPEEE. Therefore, Issue 156.2.4 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.3.1.1: SHUTDOWN SYSTEMS

DESCRIPTION

Issues 156.3.1.1 and 156.3.1.2 were combined and evaluated together. These issues are two of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by these issues.

Issue 156.3.1.1 addressed the capability of plants to ensure reliable shutdown using safety-grade equipment. Systems and components important to safety should be designed, fabricated, installed, and tested to quality standards commensurate with the safety function to be performed. Also, systems and components that are required to withstand the effects of an SSE and remain functional should be classified as Seismic Category I. Due to the evolutionary nature of design codes and standards, the staff believed that operating plants may have been designed to requirements that are not as conservative as those currently required. Systems needed to remove decay heat and reach safe shutdown should have sufficient redundancy to ensure that their function can be accomplished with a loss of offsite power and a single failure. Systems needed to shut down must also remain functional following external events. In addition, the plant operating procedures which direct the use of these systems during normal and abnormal events were to be evaluated.

Issue 156.3.1.2 addressed the review of electrical instrumentation and control features of systems required for safe shutdown, including support systems, to determine whether they met existing licensing requirements. This review was to include the capability and methods of bringing the plant from a high pressure to a low pressure cooling condition, assuming the use of only safety equipment.

The intent of these issues have been met by a number of NRC requirements and initiatives that are already in place to secure reliable plant shutdown capability. These are as follows:

(1) The fire protection rule (10 CFR 50, Appendix R) requires that the capability for shutdown be maintained, in the event of a fire in any location;

(2) The station blackout rule (10 CFR 50.63) requires the capability to cope with a complete loss of AC power and maintain safe shutdown at the same time;

(3) A number of initiatives under the TMI Action Plan[48] enhance auxiliary feedwater capability, including emergency power provisions;

(4) Improved capability for natural circulation cooldown was required by Generic Letter No. 81-21 and improved TS that enhance RHR operability in all modes were required by Generic Letter Nos. 80-42 and 80-53;

(5) TMI Action Plan[48] Item I.C.1 requires upgraded procedures for emergency conditions, including alternate means of providing a heat sink;

(6) The TMI Action Plan,[48] as clarified by NUREG-0737,[98] resulted in the issuance of requirements to licensees to implement Regulatory Guide 1.97[55]

which specifies instrumentation for monitoring important parameters such as pressure, flow, and temperature. (Continuing improvements in emergency procedures and training also address these issues.)

(7) The resolution of Issue A-46, "Seismic Qualification of Equipment in Operating Plants," and the imposition of Generic Letter Nos. 87-02 and 87-03 require licensees to address the seismic adequacy of equipment needed to bring a plant to hot shutdown and maintain that condition for a minimum of 72 hours.

(8) The resolution of Issue 99, "Loss of RHR Capability in PWRs," addressed corrective actions to reduce risk during shutdown with requirements issued in Generic Letter No. 88-17.[1146] The program described in this letter was included in a broader program described in SECY-91-283 to evaluate the risk associated with shutdown and low power.

The resolution of Issue A-45, "Shutdown Decay Heat Removal Requirements," spanned the period from March 1981 to September 1988. During that time, extensive, PRA-based determinations of the risk resulting from shutdown cooling system failures at six representative operating plants were made. These studies included (but were not limited to) the concerns of Issues 156.3.1.1 and 156.3.1.2. The technical resolution of Issue A-45 was described in SECY-88-260[1143] in which the following conclusions were presented:

(1) The risk due to loss of decay heat removal (DHR) systems could be unduly high for some plants;

(2) DHR failure vulnerabilities and the optimum corrective actions for those vulnerabilities are strongly plant-specific;

(3) Detailed plant-specific analyses under the IPE program, including extension of the IPE program to require consideration of externally-initiated events (anticipated at the time of the resolution of Issue A-45 but since accomplished), will be needed to impose and implement the resolution of this issue.

The staff concluded from the PRA studies that the risk from DHR-related failures might be too high at some plants, but a generic corrective action or a set of actions could not be identified that would both reduce that risk to an acceptable level and be cost-effective at all plants. It was believed, however, that cost-effective plant-specific actions might be possible that would reduce DHR-failure-related risk and it was concluded that the most efficient method to identify any such actions would be through the IPE program.

Appendix 5 of Generic Letter No. 88-20[1222] provided a specific description of those topics addressed in Issue A-45 and related to internally-initiated events (including those raised in Issues 156.3.1.1 and 156.3.1.2) that are to be considered in the IPE program. The IPE process was extended to include externally-initiated events (IPEEE) upon issuance of Supplement 4 to Generic Letter No. 88-20.[1222] Section 5 of this supplement specifically described how the IPEEE program was to be used to implement the technical resolution of those topics in Issue A-45 that are related to externally-initiated events.

The studies performed in the resolution of Issue A-45 included the analysis of events that initiate at full power conditions. Although the final results (total risk resulting from DHR-related failures) were increased by 20% for PWRs and 30% for BWRs to account for risk from DHR-related failures, during events that initiate when a plant is not at full power (such as hot standby and cold shutdown), such events were not investigated in detail. The IPE process was lconsistent with the analyses completed for Issue A-45 in that it only required consideration of events that initiate at full power conditions.

However, detailed attention is currently being paid to DHR failure-related events that initiate at conditions other than full power by an extensive NRC program initiated with the issuance of Generic Letter No. 88-17[1145] which resulted from an Augmented Inspection Team (AIT) investigation of a 1987 loss-of-DHR event at Diablo Canyon.[1369] This letter required licensees to investigate and, if necessary, improve procedures involving containment isolation and cooling and DHR-related equipment operation methods and training during non-power operations, when the reactor primary coolant inventory is reduced. This work received additional impetus since the issuance of Generic Letter No. 88-17 by a loss-of-DHR event at the Vogtle nuclear plant. The Vogtle event resulted in the issuance of SECY-91-283[1370] which described all aspects of the extensive program including, but not limited to, the program outlined in Generic Letter No. 88-17. Some aspects of the program described in SECY-91-283 will contribute to the imposition and implementation of the resolution of Issue A-45. This program now includes the NRC-sponsored Low Power and Shutdown (LP&S) Program which was originally formulated as part of the NRC response to the Chernobyl event.[1195] The LP&S work is being performed by BNL and SNL with additional work regarding seismically-initiated events being performed by Future Resources Associates (FRA), Inc. The objectives of the LP&S program were to: (1) assess the frequency and risk of accidents initiated during LP&S modes of operation for two nuclear power plants; (2) compare the assessed frequency and risk with those of accidents initiated during full power operations; and (3) develop new methods for assessing LP&S accident frequency and risk, as necessary.

CONCLUSION

The safety concerns of Issues 156.3.1.1 and 156.3.1.2 were addressed in the resolution of Issue A-45 and in the IPE and IPEEE programs which were supplemented by the Evaluation of Shutdown and Low Power Risk Issues Program described in SECY-91-283.[1370] Therefore, Issues 156.3.1.1 and 156.3.1.2 were DROPPED from further consideration as new and separate issues.

ISSUE 156.3.1.2: ELECTRICAL INSTRUMENTATION AND CONTROLS

This issue was evaluated with Issue 156.3.1.1 above and DROPPED from further consideration as a new and separate issue.

ISSUE 156.3.2: SERVICE AND COOLING WATER SYSTEMS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The safety concern was the capability of service and cooling water systems to meet their design objective with adequate margin. This issue was raised to provide assurance that service and cooling water systems are: (1) capable of transferring heat from structures, systems, and components important to safety to the ultimate heat sink; (2) provided with adequate physical separation such that there are no adverse interactions among the systems under any mode of operation; and (3) provided with sufficient cooling water inventory or that adequate provisions for makeup are available. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Concerns for the potential unavailability of service water systems (SWS) have been addressed in the following issues: 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems"; 130, "Essential Service Water Pump Failures at Multiplant Sites"; and 153, "Loss of Essential Service Water in LWRs." Issue 51 was resolved and implemented at operating plants in accordance with Generic Letter No. 89-13.[1259] The resolution identified a recommended improvement in the reliability of open cycle SWS that could result from reducing the potential for flow blockage in safety-related components caused by bivalves, sediment, and corrosion products. This improvement is in the form of an integrated, baseline fouling surveillance and control program for all nuclear power plant open cycle SWS.

Issue 130 was resolved and is being implemented at certain specific plants in accordance with Generic Letter 91-13.[136] This issue addressed the concerns regarding the SWS reliability of 14 PWRs at multi-unit sites with two SWS trains per unit and a crosstie capability. The resolution identified several cost-effective options that were considered for reducing the risk from loss of SWS (due to causes other than fouling), including a backup means of RCP seal cooling plus additional SWS TS and emergency procedures.

Issue 153 affected all LWRs except those that were addressed in Issue 130. All potential causes of SWS unavailability were to be considered, except those that were resolved and implemented in accordance with Generic Letter No. 89-13.[1259] The resolution plan for Issue 153 was divided into two phases: Phase I, a pilot study; and Phase II, a generic evaluation. The results of Phase I were to be used to determine if an interim resolution was viable and how to proceed with Phase II. Issue B-32, "Ice Effects on Safety-Related Water Supplies," was to be addressed in the resolution of Issue 153.

Concerns for the availability of cooling water systems were addressed in the resolution of Issue 143, "Availability of Chilled Water Systems and Room Cooling." This issue addressed the potential unavailability of chilled water systems which provide room cooling to maintain adequate environmental temperature for non-safety-related and safety-related equipment. The potential loss of room cooling could affect the operability of the safety-related systems including the SWS system.

CONCLUSION

All of the concerns regarding the performance capability and reliability of service and cooling water systems at the 41 affected plants either have been addressed or are being addressed in the issues discussed above. Additionally, a staff action plan was developed that established NRR as the focal point to ensure that all existing and future SWS issues are adequately addressed.[1367] Therefore, Issue 156.3.2 was DROPPED from further consideration as a new and separate issue.

ISSUE 156.3.3: VENTILATION SYSTEMS

DESCRIPTION

This issue is one of nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] At issue was the adequacy of ventilation systems to provide a safe environment for plant personnel and ESF systems under normal, anticipated transient, and design basis operational conditions. A safe environment is one that is effectively controlled with respect to radiation, heat, humidity, smoke, and toxic gases. Five ventilation systems were identified in SRP[11] Section 9.4 to effect ESF equipment and plant personnel: the control room area, spent fuel area, auxiliary and radwaste area, turbine area, and ESF area.

With respect to plant personnel, the concerns about ventilation are grouped under radiation exposure as the first, and exposure to excessive levels of environmental pollutants such as smoke, toxic gases, heat, and humidity as the second. These concerns may be considered for both normal operating and abnormal conditions. For normal conditions, the first concern is addressed by existing regulations in 10 CFR 20 which is quite clear and comprehensive concerning monitoring of restricted and unrestricted areas and radiation limits in each. In particular, 10 CFR 20.106 applies to radioactivity in effluent between restricted and unrestricted areas. Coverage includes limits of concentrations of radioactive material in air as well as water. For applications filed after January 2, 1971, 10 CFR 50.34a requires ALARA programs which are elaborated upon in 10 CFR 50, Appendix I. In addition, 10 CFR 50.34a requires design and installation of equipment "to maintain control over radioactive materials in gaseous and liquid effluent" not only during normal operations but also during expected operational occurrences. 10 CFR 50.36a requires TS on effluent from nuclear power reactors.

For normal operating conditions, the second concern is the responsibility of OSHA whenever the safety of licensed radioactive materials is not involved. This responsibility was outlined in an MOU between OSHA and the NRC issued on October 25, 1988. For abnormal conditions, the second concern comprises potentially unpleasant plant nuisance factors with the exception of the control room and turbine area. One potentially serious atmospheric contaminant in the turbine building and the auxiliary building of PWRs is hydrogen with its potential for deflagration or detonation. Issue 106 addressed the role of ventilation systems in the prevention of H_2 deflagration from leaks in the H_2 distribution piping.

Issue 136 addressed the issue of vapor clouds from liquified combustible gases drifting into safety-related air intakes.

Abnormal control room environmental conditions could exist that adversely affect operator performance to a degree sufficient to cause operator-initiated transients. These conditions are within the NRC scope as defined in the above MOU. Conditions affecting mitigation of accidents are also clearly NRC responsibility. The resolution of Issue 83, "Control Room Habitability," will address the limits of plant personnel functioning from radiation and toxic gas exposure. The scope of Issue 83 includes "...provisions for personnel to remain in the control room as needed to manage accidents which have the potential for offsite and onsite radiological consequences, and protection of control room occupants to the degree necessary to prevent an accident occurring as a result of operator incapacitation." SRP[11] Section 6.4, Rev. 2, describes review of the control room ventilation system with the objective of assuring protection for plant operators from the effects of accidental releases of toxic and radioactive gases. A third revision draft is under consideration as part of the resolution of Issue 83. Thus, accident initiation and mitigation capabilities of control room personnel are being addressed with respect to radiation and toxic gas exposure. Control room concerns remaining are high temperature and humidity and smoke.

With respect to high temperature and humidity, the ACRS has recommended that "Temperature limits should be revised taking into account low air exchange rate, operation of ESF filter system heaters and perspiration." The ACRS considers a temperature limit of 120°F for the control room as unacceptable; this is a TS limit derived for control room equipment.[678] Under accident conditions, no NRC requirement exists for temperature limits for reliable performance of control room personnel. However, documentation exists that supports a maximum effective temperature of 85°F for reliable human performance. (A defined effective temperature includes some combination of dry bulb temperature, relative humidity, and air velocity). Although no accident condition temperature limit has been formalized, SRP[11] Section 9.4.1, "Control Room Area Ventilation System," concerns itself in part with "...the comfort of control room personnel during normal operating, anticipated operational transient, and design basis accident conditions." The control room area ventilation system (CRAVS) is reviewed, among other things, with respect to ability to maintain a suitable ambient temperature for control room personnel. The single failure criterion is applied in the CRAVS review. In addition, the CRAVS must function unaffected by loss of equipment that is not seismic Category 1 and the integrated system design must satisfy GDC 2 with respect to earthquakes. The designs are reviewed for protection from floods, hurricanes, tornadoes, internally- or externally-generated missiles, fires, and loss of offsite power. At some plants, the CRAVS is capable of functioning in an internal-filtered recirculation mode of operation.

A survey of twelve plants reported some problems with adequacy and demonstration of adequacy of control room cooling for a postulated 30-day accident period.[1371] The plants surveyed were a mix of ages, ranging from some of the oldest to some of the newest. While the problems identified produced no added industry requirements, a recommendation was made for more [staff] attention to detail in evaluations of control room cooling systems design and operations that rely on two separate cooling systems, i.e., a non-safety-related system for normal operations and a safety-related system for emergency operations only. In sum, no additional regulatory requirements or guidance are warranted for investigation with respect to high temperature and humidity vis-a-vis control room personnel under accident conditions.

Issue 143 is to be resolved and will address the importance of ventilation systems on cooling for the operation of ESF equipment. Activities in support of the resolution of Issue 143 will identify the vulnerabilities of safety-related systems and their support systems to the effects of HVAC and chilled water system failures and adverse temperature fluctuations. An evaluation will be made of equipment environmental qualification, equipment room heat load and heat-up rate to identify areas in which a reduction in the dependence of equipment operability on HVAC and room cooling may be required. The control of smoke in plants is being addressed in Issue 148, "Smoke Control and Manual Fire Fighting Effectiveness."

CONCLUSION

The safety concerns of Issue 156.3.3 were either being addressed in ongoing staff actions on Issues 83, 106, 136, 143, and 148, or were covered by existing regulations. Therefore, Issue 156.3.3 was DROPPED from further pursuit as a new and separate issue.

ISSUE 156.3.4: ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

DESCRIPTION

This issue is one of nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] At issue were low pressure systems (such as the RHR systems) that interface with the reactor coolant system through isolation valves. The concern was that systems with low design pressure, in comparison with reactor coolant pressure, will incur damage due to valve failure or inadvertent valve opening.

Issue 105, "Interfacing Systems LOCA in LWRs," addressed the possible breach of those interfacing boundaries that are created by a series of pressure isolation valves (PIVs) and the consequences of failure of a boundary by mechanical failure, human error, or external event. Thus, Issue 105 covered all interfacing systems, including those identified in Issue 156.3.4. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of Issue 156.3.4 was addressed in the resolution of Issue 105, "Interfacing Systems LOCA in LWRs." Therefore, Issue 156.3.4 was DROPPED from further pursuit as a new and separate issue.

ISSUE 156.3.5: AUTOMATIC ECCS SWITCHOVER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Most PWRs require operator action to realign the ECCS for the recirculation mode following a LOCA. Existing guidelines state that automatic transfer to the recirculation mode is preferable to manual transfer. However, a design that provides manual switchover is sufficient provided that adequate instrumentation and information displays are available for the operator to manually transfer from the injection mode to the recirculation mode at the correct time. Automatic in lieu of manual switchover could possibly provide an improvement of ECCS reliability at a cost that could result in a worthwhile safety enhancement. This issue addressed the procedures for manual switchover, the adequacy of available instrumentation, and the possible operator errors associated with the switchover process. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

All 41 plants affected by this issue were to be considered in the resolution of Issue 24, "Automatic Switchover to Recirculation," which was directed at studying the merits of manual, automatic, and semi-automatic ECCS switchover to recirculation. Thus, Issue 156.3.5 was covered in the resolution of Issue 24.

ISSUE 156.3.6.1: EMERGENCY AC POWER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The electrical independence and redundancy of safety-related onsite power sources must meet the single failure criterion. Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures over the years that have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air system. The objective of this issue was to review the reliability of protection interlocks and testing of diesel generators to assure that diesel generator systems meet the availability requirements for providing emergency standby power to the engineered safety features, as well as the independence of onsite power distribution systems and features, such as automatic bus transfers and breaker connections, that could affect the independence of redundant trains. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of this issue was addressed in the resolution of Issues A-44, "Station Blackout," and 128, "Electrical Power Reliability." The concern was also addressed in the resolution of Issue B-56, "Diesel Reliability." The requirements that result from the resolution of these three issues will affect the 41 non-SEP plants. In addition, MPAs B-23, "Degraded Grid Voltage," and B-48, "Adequacy of Station Electric Distribution Voltage," have been implemented at several of the 41 plants affected by this issue and will not have to be repeated in the implementation of the resolution of Issue A-44.[1108] Based on the above considerations, Issue 156.3.6.1 was DROPPED from further pursuit as a new and separate issue.

ISSUE 156.3.6.2: EMERGENCY DC POWER

DESCRIPTION

Historical Background

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343[1351] following its study of how the lessons learned from the SEP have been factored into the licensing bases of operating plants. The issue addresses the concern that safety-related DC power system bus voltage monitoring and annunciation may not adequately notify operators of DC bus status. Responses to Generic Letter 91-06[1399] indicated that a significant number of licensees could be affected by the concerns of this issue. Based upon a PRA analysis of the DC power system at six plants, it was concluded that additional DC power system bus voltage monitoring and annunciation for licensed facilities would not have a significant impact on safety and would not be a cost-effective means of increasing plant safety.

This issue addressed the criteria in 10 CFR 50.55a(h) and 10 CFR 50 (GDC 2, 4, 5, 17, 18, and 19) which require that the control room operator be given timely indication of the status of the safety-related DC power system batteries and their availability. The current staff position is that the following separate and independent control room indications and alarms for the Class 1E DC power system status are recommended in order to meet these criteria:

(1) battery disconnect or circuit breaker open alarm
(2) battery charger disconnect or circuit breaker open alarm (both input AC and output DC)
(3) DC system ground alarm
(4) DC bus undervoltage alarm
(5) DC bus overvoltage alarm
(6) battery charger failure alarm
(7) battery discharge alarm
(8) battery float charge current ammeter
(9) battery circuit output current ammeter
(10) battery discharge indicator
(11) bus voltage voltmeter

These annunciators and alarms are needed in order to ensure that the control room operators are alerted in the event of DC power system or battery failure. If a less extensive configuration of equipment is used, it is possible that a DC power system or battery failure mode could exist which would not result in the actuation of any alarms or annunciators. In this event, the DC power supply would remain in the degraded condition until a periodic surveillance test or maintenance was performed to identify the condition of the batteries.

Safety Significance

Based upon the SEP reviews, it was apparent that some licensees had received operating licenses without providing the above recommended alarms and annunciators. However, in most cases the licensees in the SEP reviews were able to demonstrate to the staff that modifications were unnecessary. The concern in this issue is that some licensees that were not reviewed in the SEP program might have insufficient annunciators and alarms in the control room to alert the operators to some safety-related DC power supply or battery failure modes, which would increase the likelihood that a DC power supply is unavailable when needed.

PRIORITY DETERMINATION

The issue of control room annunciation and alarms for the safety-related DC power supplies was also addressed in Issue A-30, "Adequacy of Safety-Related DC Power Supplies," which was combined with other generic issues involving safety-related power supplies to form Issue 128, "Electric Power Reliability." Generic letters 91-06[1399] and 91-11[1400] were issued in the resolution of Issue 128; Generic Letter 91-06 addressed the concerns of Issue A-30. Industry organizations such as NUMARC and INPO asserted that most licensees already have alarm and annunciator configurations that are equivalent to the current staff recommendations which were based in part on industry standards. Therefore, the questions in Generic Letter 91-06[1399] which addressed available alarms and annunciators did not represent a minimum acceptable configuration, but were formulated to provide sufficient information to the staff to determine if licensees had met or adequately addressed the current recommendations.

An INEL review[1457] of the responses to Generic Letter 91-06[1399] showed that 42 licensees do not have any separate and independent alarms in the control room for their DC power system. However, these licensees typically had local alarms which were separate and independent, and a single battery condition monitor which alarms in the control room in the event that one or more of the local battery alarms actuate. In addition, the INEL review indicated that 15 licensees have not performed a human factors review of their testing and maintenance procedures, and 5 licensees do not have procedures that specifically prevent simultaneous testing or maintenance of redundant safety-related DC power sources. In most cases, the licensees supplied justification for the discrepancies between their licensed configuration and the current staff position. INEL did not evaluate licensee responses to determine what modifications would be required to adequately resolve the concerns of Issue A-30, and recommended that the staff perform a PRA study to determine the impact on plant safety of existing configurations of safety-related DC power supply annunciation and alarms.

Frequency Estimate

The concern in this issue was that the safety-related DC power supplies might be unavailable because of inadequate control room annunciators and alarms. This concern correlates with the results of NUREG-0666,[164] which included a FMEA and a PRA of a model DC power system. This model system consisted of two independent DC buses each of which were supplied by a single battery charger and had a single battery back-up. In addition, this system had the following alarms and annunciators in the control room: (1) battery charger ground alarm; (2) battery charger AC power supply failure alarm; (3) DC bus undervoltage alarm; (4) battery charger DC ammeter; and (5) battery charger DC voltmeter.

NUREG-0666[164] concluded that battery unavailability is dominated by inadequate maintenance practices and failure to detect battery unavailability due to bus connection faults. By improving battery surveillance, DC power system unreliability could be decreased by a factor of two, and improving maintenance and testing practices could decrease DC power system unavailability by a factor of 10. The report does not quantify a safety benefit which would result from additional alarms or annunciators in the control room, but additional alarms and annunciators would result in the enhancement of surveillance, maintenance and testing capabilities. Additional recommendations were made in NUREG-0666,[164] but these relate to aspects of the DC system which would not be enhanced by the addition of alarms or annunciators, such as the addition of a third DC power train.

In addition to the concerns relating to alarms and annunciators, the responses to Generic Letter 91-06[1399] also identified concerns with the probability of CCF of the DC power supplies. In order to evaluate these two concerns, the PRAs for 6 licensees were reviewed and found to include basic events which modeled the probability of battery unavailability and common cause battery failure. A study was performed to determine the effect on the CDF of decreasing battery unavailability and common cause battery failure probability. This study was performed by the staff using the SARA[1456] software. The results are described below.

The assumption was made that improved alarms and annunciators would result in continuous battery condition indication and would essentially result in an undetected battery failure probability of zero, since the operators would be notified of a DC power system failure immediately. However, this approximation would give a greater estimate of the effectiveness of modifications of alarms and annunciators than could actually be obtained. A better estimate of the effect on DC power system reliability resulting from an increase in the number of alarms and annunciators in the control room was obtained by decreasing the battery unavailability from the base case value to a test case value of 10^{-6}. For the plants considered in this analysis, the base case values ranged from 6.12×10^{-3} to 7.2×10^{-4}, which reflects an hourly failure rate of approximately 10^{-6}/hour, and an interval between tests which are capable of detecting a failed battery ranging from 6,120 to 720 hours.

This modification in battery unavailability will also account for any decrease in the battery charger unavailability resulting from the additional hardware. Because the battery must be instantaneously available to supply power if the battery charger fails, the battery unavailability terms in a PRA model are always multiplied by the battery charger unavailability terms. This analysis is conservative because it overestimates the effectiveness of additional alarms and annunciators, which will improve DC power system reliability by a much smaller factor. In addition, this approximation is made under the assumption that the DC power systems have been accurately modelled by PRA analysts for the existing PRAs and is only valid if the configuration of alarms and annunciators modelled by the existing PRAs is less effective than the currently recommended configuration.

Common cause failure (CCF) of the DC power system can be caused by maintenance activity, the most significant of which is inadvertent connection of redundant trains. Generic Letter 91-11[1400] addressed the use of interconnections between Class 1E vital instrument buses and LCOs for Class 1E vital instrument buses. The purpose of this generic letter was to decrease the probability and sources of CCF of redundant Class 1E AC and DC buses and inverters. It was assumed that CCF of the Class 1E buses and inverters has been adequately addressed and the scope of this issue was limited to the batteries and battery chargers.

The SARA[1456] software was used to model the effect of decreasing battery unavailability. There are currently nine operating plants which have PRA models which can be used with SARA. These are listed below, in addition to the configuration of the DC power system at the plant.

Plant	Number of 125 V DC Batteries	Number of Battery Chargers
Grand Gulf 1[1318]	3	6
Brunswick 1 & 2*	4 (each)	4 (each)
Peach Bottom 2*	4	4
Surry 1[1318]	2 + diesel	2
Sequoyah 1[1318]	2 + diesel + 1 common	2 + 1 common
Oconee 3[889]	2	3
Zion[1318]	2 + 1 common	2 + 1 common
Indian Point 2*	4	4

* Based on IPE Submittal

Peach Bottom Unit 2: This unit has two independent divisions of safety-related 125 V DC power, one of which is required to safely shut down the plant. Each division is comprised of two batteries, each with it's own charger. The control room has 3 of 7 recommended alarms and 1 of 4 recommended annunciators. The Peach Bottom PRA included probability terms for battery unavailability due to common mode failure and unavailability of the individual Unit 2B and 3C battery banks. The terms for the remaining battery banks (2A, 2C, 2D, and 3D) were not included in any significant minimal cutsets, and decreasing these basic event probabilities would have a negligible effect on the CDF. The probability of battery unavailability was estimated in the original PRA to be 0.001.

Peach Bottom 2 - Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.4×10^{-6}	-2.0×10^{-7}

Peach Bottom 2 - Battery 2B and 3C Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.6×10^{-6}	-

Decreasing the probability of common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 2.0×10^{-7}/year, whereas decreasing the probability of the unavailability of batteries 2B and 3C would result in less than a 10^{-7} decrease in CDF.

Grand Gulf Unit 1: This unit has three independent divisions of safety-related 125 V DC power, two of which are required to safely shut down the plant. The control room has 1 of 7 recommended alarms and 1 of 4 recommended annunciators. The Grand Gulf PRA included terms for the probability of battery common mode failure and failure of the individual Unit 1A3, 1B3, and 1C3 battery banks. All battery banks were included in significant minimal cutsets.

Grand Gulf 1 - Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.6×10^{-6}	-5.0×10^{-7}

Grand Gulf 1 - Loss of Power from Batteries 1A3, 1B3, 1C3

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.9×10^{-6}	-2.0×10^{-7}

Decreasing common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 5×10^{-7}/RY, whereas decreasing the unavailability of battery 1A3, 1B3 and 1C3 would result in a decrease of 2×10^{-7} in CDF.

Brunswick Units 1 and 2: These units each have two independent divisions of safety-related 125 V DC power, one of which is required to safely shut down the plant. Each division is comprised of two independent batteries, each with its own charger. The control room has 5 of 7 recommended alarms and 2 of 4 recommended annunciators. The Brunswick Units 1 and 2 PRAs included terms for the probability of individual battery bank unavailability but not for common cause unavailability. The terms for failure of three of the four batteries were included in some minimal cutsets.

Brunswick 1 - Battery Bank 1A1, 1A2, and 1B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.47×10^{-5}	base case
0.000001	2.46×10^{-5}	-1.0×10^{-7}

Brunswick 2 - Battery Bank 2A1, 2A2, and 2B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.08×10^{-5}	base case
0.000001	2.06×10^{-5}	-2.0×10^{-7}

Units 1 and 2 differed slightly in their response to battery failure rate changes. However, decreasing the unavailability of battery 2A1, 2A2, and 2B1 would result in a decrease of 10^{-7}/RY and 2×10^{-7}/RY in CDF for Unit 1 and 2, respectively.

Surry Unit 1: This unit has two independent divisions of safety-related 125 V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has 4 of 7 recommended alarms and 1 of 4 recommended annunciators. The Surry PRA included terms for the probability of battery common mode failure and failure of the individual I and II battery banks. Neither the common mode battery failure term or individual battery failure terms were included in any significant minimal cutsets. The assumed battery unavailability was 7.2×10^{-4}, which suggests a 2-month interval between tests that would detect battery problems for the typical failure rate. Because the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8}, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Sequoyah Unit 1: This unit has two independent divisions of safety-related 125 V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has zero of 7 recommended alarms and 3 of 4 recommended annunciators. The Sequoyah PRA included probabilities for battery common mode unavailability and unavailability of the individual I and II battery banks. Battery unavailability was initially estimated to be 7.2×10^{-4}, which suggests a two-month surveillance test or maintenance interval for a failure rate of 10^{-6}/hour. The common mode unavailability was estimated to be 5.8×10^{-6}. Neither the common mode unavailability or individual battery unavailability were included in any significant minimal cutsets. The unavailabilities used in this analysis were slightly lower than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Oconee Unit 3: This unit has two independent divisions of safety-related DC power, one of which is required to safely shut down the plant. The control room has 1 of 7 recommended alarms and none of 4 recommended annunciators. The Oconee PRA[889] included terms for unavailability of the individual 1CA, 1CB, 3CA, and 3CB battery banks. The probability of battery unavailability was estimated to be 6.12 x 10^{-3}, which is based on a one-year surveillance test or maintenance interval and a failure rate of 1.4 x 10^{-6}/hour. Common mode unavailability was not included in the PRA model. The individual battery unavailability terms were not included in any significant minimal cutsets. The probabilities used in this analysis were significantly greater than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

The average decrease in CDF from the proposed modifications was estimated to be approximately 10^{-7}/RY.

Consequence Estimate

It was assumed that all affected operating plants had an average remaining life of 20 years, based on their original licenses. It was also assumed that each of these plants would be granted a life extension of 20 years. Thus, the average remaining life for all affected plants was 40 years.

The public risk associated with the event considered in this issue was estimated[64] to be 6.76 x 10^6 man-rem and 2.52 x 10^6 man-rem for BWRs and PWRs, respectively. For BWRs, the total potential risk reduction was estimated to be (6.76 x 10^6)(10^{-7})(40) man-rem/reactor or 27 man-rem/reactor. For PWRs, the total potential risk reduction was estimated to be (2.52 x 10^6)(10^{-7})(40) man-rem/reactor or 10 man-rem/reactor.

Cost Estimate

Improving the control room annunciators and alarms for all safety-related DC power systems at each plant would involve a different amount of effort for each licensee, depending upon the amount of instrumentation currently installed, available space for additional annunciators and alarms, and whether existing raceway could hold additional cables. In addition, new procedures and operator training would be required. This additional hardware would include the following:

(1)	Data transmitters at each battery room. Design, installation and testing assumed to be $100,000/battery room, with 3 battery rooms per facility	$300,000
(2)	Raceway and cable from each battery room to the control room. Design, installation and testing costs assumed to be $100 per linear foot, with 1000 linear feet of raceway per battery room and 3 battery rooms per facility	$300,000
(3)	Control room modifications to add annunciators and alarms. Design, installation and testing assumed to be $100,000/battery, 3 batteries per facility	$300,000

(4)	Procedure changes, drawing changes, training, and administrative costs	$100,000
	TOTAL:	$1,000,000

Value/Impact Assessment

Separate value/impact scores were calculated for PWRs and BWRs.

BWRs: Based on a potential public risk reduction of 27 man-rem/reactor and a cost of $1M/reactor, the value/impact score was given by:

$$S = \frac{27 \text{ man-rem/reactor}}{\$1\text{M/plant}}$$

$$= 27 \text{ man-rem/\$M}$$

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

$$S = \frac{10 \text{ man-rem/reactor}}{\$1\text{M/reactor}}$$

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

Other Considerations

It is important to monitor the condition of the safety-related DC power system, including the condition of batteries which may be needed in the event of a station blackout. In addition, it is also necessary to have procedures which minimize the probability of a common cause fault of the safety-related DC power systems. Operating experience so far does not indicate that significant problems exist in this area.

Based upon the results of this study, it could be asserted that the control room alarms and annunciators recommended by the staff in current licensing guidelines do not result in a significant increase in plant safety beyond that realized by existing alarm and annunciator configurations and weekly or quarterly maintenance programs. It should be noted that the empirical battery failure rate of approximately 10^{-6}/hour, which is used to determine battery unavailability, is dependent upon the frequency of battery failures for systems with existing configurations of control room annunciators and alarms. Therefore, it might not be accurate to conclude that the existing recommendations for annunciators and alarms should be relaxed.

Battery unavailability and CCF are recognized by some licensees to be sufficiently probable so as to require modelling in PRAs. Based upon these PRA models, decreasing the unavailability of the batteries and safety-related DC power supplies by several orders of magnitude over that used in the base case does not result in a significant decrease in CDF for these licensees. This observation must be tempered with the knowledge that licensees currently monitor important DC bus parameters, and that other DC power system design features, such as the number of batteries, have a greater impact on DC power system reliability than the number of alarms and annunciators.

CONCLUSION

Based on the potential public risk reduction, this issue had a low priority ranking for BWRs and was in the drop category for PWRs. Overall, the issue was given a LOW priority ranking.

ISSUE 156.3.8: SHARED SYSTEMS

DESCRIPTION

This issue is one of the nineteen category 3 issues identified by NRR in SECY-90-343.[1351] The sharing of the ESFS for a multi-unit plant, including onsite emergency power systems and service systems, can result in a reduction of the number and capacity of onsite systems to below that which is needed to bring either unit to a safe shutdown condition, or to mitigate the consequences of an accident. Shared systems for multiple unit stations should include equipment powered from each of the units involved. There were 13 multi-unit sites that could be affected by this issue among the 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976.

CONCLUSION

The safety concerns associated with systems that are shared by two or more units at multi-unit sites have been previously identified by the staff. The most important contributors to core damage probability at these sites have been determined to be air, cooling water, and electric power systems. These systems have been adequately addressed in the following issues: 43, "Reliability of Air Systems"; 130,"Essential Service Water Pump Failures at Multiplant Sites"; 153,"Loss of Essential Service Water in LWRs"; and A-44, "Station Blackout." Based on these considerations, this issue was DROPPED from further pursuit as a new and separate issue.

ISSUE 156.4.1: RPS AND ESFS ISOLATION

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern was that, in the event of non-safety system failures, the lack of isolation devices could result in the propagation of faults to safety systems and common cause failures may result. In its study, the staff found that approximately 39 plants at 28 sites were not required to meet IEEE 279-1971[397] and have not been reviewed for this safety concern since the time of their licensing.

Non-safety systems generally receive control signals from the RPS and ESF sensor current loops. The non-safety circuits are required to be isolated to ensure the independence of the RPS and ESF channels. Requirements for the design and qualification of isolation devices are quite specific. Evaluation of the quality of isolation devices is not the safety issue of concern; rather, the issue is the existence of isolation devices which will preclude the propagation of non-safety system faults to safety systems.

CONCLUSION

The safety concerns of leakage through electrical isolators in instrumentation circuits and electrical isolation in plants not required to meet IEEE 279-1971[397] were addressed in the resolution of Issue 142, "Leakage Through Electrical Isolators in Instrumentation Circuits." Therefore, Issue 156.4.1 was covered in the resolution of Issue 142.

ISSUE 156.4.2: TESTING OF THE RPS AND ESFS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review plant designs to ensure that: (1) all ECCS components, including the pumps and valves, are included in the component and system test; (2) the frequency and scope of periodic testing are identified; and (3) the test programs will provide adequate assurance that the systems will function when needed. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

A portion of this issue was covered by existing requirements; specifically, ECCS pumps and valves are required to be tested quarterly by the ASME Code in accordance with 10 CFR 50.55(a), unless the NRC grants relief to defer testing until refueling outages. The remainder of this issue was covered in the resolution of Issue 120, "On-Line Testability of Protection Systems," which addressed the concern regarding on-line (at-power) testability af protection systems (both the RPS and the ESFS) and the possibility that some plants may not provide complete testing capability at power.

ISSUE 156.6.1: PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS

This issue is being prioritized.

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an Accident," U.S. Nuclear Regulatory Commission, December 1975, (Revision 1) August 1977, (Revision 2) December 1980, (Revision 3) May 1983.

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688. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1976.

814. SECY-84-133, "Integrated Safety Assessment Program (ISAP)," March 23, 1984.

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1108. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," U.S. Nuclear Regulatory Commission, June 1988.

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1259. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," July 18, 1989.

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1360. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," U.S. Nuclear Regulatory Commission, July 1990.

1367. Memorandum for W. Russell from A. Thadani, "Task Action Plan for Resolution of Service Water System Problems," June 27, 1991.

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1375. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," U.S. Nuclear Regulatory Commission, February 1975, (Revision 1) January 1977.

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ISSUE 161: USE OF NON-SAFETY-RELATED POWER SUPPLIES IN SAFETY-RELATED CIRCUITS

DESCRIPTION

This issue was identified[1481] by NRR to address the concerns raised during the licensing of Nine Mile Point Unit 2.

On February 7, 1985, Niagara Mohawk submitted to the NRC a report on "Non-Class 1E Devices Connected to Class 1E Power Supplies" which stated that "... the non-Class 1E devices to be analyzed were identified by a study of Elementary Diagrams and Elementary Diagram Device Lists for all safety systems in the GE scope of supply for NMP2." These devices were asserted to be acceptable by the licensee and the vendor (GE) based on an FMEA; however, this FMEA was not accepted by the staff. Ultimately, of the 239 identified components, 35 were isolated with qualified isolation devices and 76 were upgraded to Class 1E by the licensee in order to meet the regulatory requirements imposed by the staff.

The fundamental concern in this issue was whether the staff's actions in requiring the isolation or replacement of 111 out of 239 components during the licensing review was a change in regulatory position and, if so, whether the position should be backfit to all licensed facilities. The position taken by the staff in 1985 during the OL review of Nine Mile Point 2 was necessary to meet the regulations in effect at that time. It was GE's contention that the design approach at Nine Mile Point 2 was similar to that used at other plants and was accepted by the NRC during the licensing review of BWR plants.

The determination of whether a component or system is an associated circuit is based on the application of IEEE 384-1977 and IEEE 279-1971. Any non-Class 1E component or system which interconnects with a Class 1E component or system is an associated circuit, unless it is adequately isolated. IEEE 279-1971, which was incorporated by reference into 10 CFR 50.55a(h), does not identify associated circuits but only states that "... the transmission of signals from protection system [Class 1E] equipment for control system [non-Class 1E] use shall be through isolation devices which shall be classified as part of the protection system..." This restriction was relaxed in IEEE 384-1977 to define an associated circuit as any circuit comprised of non-Class 1E components that is not isolated from a Class 1E circuit by an isolation device. This configuration is acceptable if the associated circuit is "... analyzed or tested to demonstrate that Class 1E circuits are not degraded below an acceptable level." IEEE 384-1977 was adopted in Regulatory Guide 1.75, Revision 3 in September 1978.

CONCLUSION

Prior to the publication of IEEE 384, the classification of an associated circuit did not exist and any non-Class 1E component should have been isolated from a Class 1E component or system by a qualified isolation device. Therefore, it is unlikely that the staff accepted a practice in the past which they found unacceptable in 1985, since the requirements in 1985 were less strict than the requirements before 1977. GE was unable to document previous acceptance by the staff of the design practice under contention. In addition, the criteria used by the staff in 1985 to determine whether a circuit is an associated circuit and whether the associated circuit should be isolated would also be applied currently to OL applicants.

The components at Nine Mile Point 2 were analyzed by GE and resulted in 128 of the 239 identified components being accepted as associated circuits without modification; however, the remaining 111 components were upgraded to Class 1E or were isolated. It appears reasonable to assume that the determination made by NRR regarding the need for components to be Class 1E is correct. Thus, this issue was determined to be a matter of compliance with existing regulations[1482] and was DROPPED from further consideration as a new and separate issue.

REFERENCES

1481. Memorandum for E. Beckjord from T. Murley, "Potential New Generic Issues," September 25, 1991.

1482. Memorandum for T. Murley from E. Beckjord, "Prioritization of Generic Issue 161, 'Associated Circuits,'" March 12, 1993.

ISSUE 164: NEUTRON FLUENCE IN REACTOR VESSEL

DESCRIPTION

To calculate the value of RT_{PTS} as required in 10 CFR 50.61 and 10 CFR 50, Appendix G, licensees must determine the value of the fast neutron fluence on the inside surface of their pressure vessels. Through a number of reviews, NRR found[1515] a non-conservative computational bias in the Westinghouse methodology which, in one instance (Yankee Rowe), was determined to be 13% while in WCAP-11815 (Indian Point 3 Surveillance Capsule Z Report) was reported to be 20%.

Several publications suggest that the iron inelastic scattering cross-sections in ENDF/B-VI yield higher fluence values for transmissions through iron; thus, ENDF/B-IV (currently in use) may not be conservative.

Licensees are required to determine applicable uncertainties in the measurements and calculations for reactor cavity dosimetry. In Regulatory Guide 1.99, Revision 2, the staff assumed a fluence uncertainty of 20% in determining trend curves. Thus, this level of uncertainty at the inside surface of the pressure vessel should be supported. The bias in the Westinghouse calculations may exist in other vendor or licensee methodologies, or it may pertain to the iron cross-sections. Thus, the staff believed that this issue could affect all PWRs.

CONCLUSION

The safety concern of this issue is being addressed in the Surveillance Data Base, Analysis, and Standardization Program (FIN B04152). Concurrent with this program, the staff developed Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Thus, the issue was DROPPED from further consideration as a new and separate issue.[1516]

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1515. Memorandum for E. Beckjord from T. Murley, "Proposed New Generic Issue: 'Determination of Neutron Fluence to Pressure Vessels,'" October 8, 1992.

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ISSUE 166: ADEQUACY OF FATIGUE LIFE OF METAL COMPONENTS

DESCRIPTION

Select portions of the STS contain requirements to monitor cumulative fatigue usage for critical components associated with the Safety Injection systems. In addition, STS Section 5.0, "Design Features," requires the tracking of certain transients to ensure that design bases are not exceeded. However, many facilities do not have the STS or any requirement to monitor for fatigue limits. The resolution of Issue 78 was expected to determine the degree to which fatigue limit monitoring was necessary, address fatigue adequacy in general, and recommend actions, it any, to be taken by the staff. However, during the Commission meetings on promulgating requirements for license renewal, it became increasingly apparent that the adequacy of fatigue life of metal components should not be conducted solely for license renewal, but should also be conducted for current operating plants. Therefore, NRR took[1517] the lead responsibility for addressing this issue for operating plants. The resolution of Issue 78 was to be integrated into the resolution of the Issue 166.

CONCLUSION

This issue is considered nearly-resolved based on the NRR decision to pursue resolution of the issue.[1517]

REFERENCE

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.

ISSUE 168: ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

DESCRIPTION

As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases, particularly for older plants whose licensing bases differ from newer plants, should be reassessed or enhanced in connection with license renewal or whether they should be reassessed for the current license term. The staff concluded that differences in EQ requirements constituted a potential generic issue which should be evaluated for backfit independent of license renewal.[1518]

During the staff's development of an interoffice action plan to address upgrading EQ requirements for older plants during the current licensing term, the staff evaluated the technical adequacy of EQ requirements. As part of this evaluation, the staff reviewed recent tests of qualified cables performed by SNL, under contract with the NRC. The purpose of these tests was to determine the effects of aging on cable products used in nuclear power plants. After accelerated aging, some of the environmentally-qualified cables either failed or exhibited marginal insulation resistance during accident testing, indicating that qualification of some electric cables may have been non-conservative. Although the SNL tests may have been more severe than required by NRC regulations, the test results raised questions with respect to the EQ and accident performance capability of certain artificially-aged cables. Depending on the application failure of these cables during or following design basis events could affect the performance of safety functions in nuclear power plants.

CONCLUSION

This issue is considered nearly-resolved based on the NRR decision to pursue resolution of the issue.[1517]

REFERENCES

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.

1518. Memorandum for The Chairman et al., from J. Taylor, "Environmental Qualification of Electric Equipment," May 27, 1993.

TASK HF5: MAN-MACHINE INTERFACE

The objective of this task was to ensure that the man-machine interface (MMI) is adequate for the safe operation and maintenance of nuclear power plants. This objective was to be attained by developing: (1) human factors engineering guidelines for correcting MMI problems; and (2) regulatory guidance for integrating human factors engineering into new designs and into advanced technological improvements incorporated into existing designs. This task was also to provide for the preparation of evaluation tools for: (1) the next generation of nuclear power plants; and (2) expected changes or upgrading to designed plants in the area of data and information management and improved annunciator systems. These efforts were expected to improve the staff's capability to evaluate reactor incidents involving MMI errors. This task was identified as four distinct items in Table 7 of the NRC 1985 Annual Report (Items 5.1, 5.2, 5.3, and 5.4). The following is a discussion of these four items.

ITEM HF5.1: LOCAL CONTROL STATIONS

DESCRIPTION

Previous regulatory efforts dealing with the MMI were limited to the control room and remote shutdown panel. It was believed that further guidance regarding local control stations and auxiliary operator interfaces was necessary as well as additional guidance regarding improvements to existing annunciator systems.

Information was to be developed to determine if guidance on local control station design and auxiliary operator interfaces with these stations was required. To accomplish this task, job/task analyses of control room crew activities were to be conducted to identify and describe communication and control links between the control room and the auxiliary control stations. In addition, the functions of the auxiliary personnel were to be analyzed from the task analyses to estimate the potential impact of auxiliary personnel job errors on plant safety.

CONCLUSION

The issue was given a high priority ranking and a survey of safety-significant local control stations was conducted at 4 plants. This survey included remote shutdown panels, local diesel generator panels, and local ECCS panels. Deficiencies found were poor lighting, poor labeling, obstructed view of instrumentation, and unavailable communication equipment. The survey was documented in NUREG/CR-3696.[1521]

A preliminary value/impact analysis that considered various combinations of upgrades involving panel re-design as well as functional centralization was completed and documented in NUREG/CR-5572.[1524] However, with the publication of NUREG-1150,[1081] the potential risk reduction was found to be considerably lower than previously anticipated and work was curtailed. The staff's studies were to be published together with a "good practices" discussion on local control station design. Thus, this issue was RESOLVED and no new requirements were established.[1525]

ITEM HF5.2: REVIEW CRITERIA FOR HUMAN FACTORS ASPECTS OF ADVANCED CONTROLS AND INSTRUMENTATION

DESCRIPTION

The existing human engineering guidelines for nuclear power plant control rooms primarily addressed the control, display, and information concepts and technologies that were being used in process control systems. While these guidelines were adequate for the existing generation of nuclear power plants, the staff did not believe that they were sufficient for advanced and developing technologies that could be introduced into existing and future designs. Improved annunciator systems utilizing advanced technologies were expected to become available and guidelines for the utilization and evaluation of these longer-term annunciator improvements were to be developed, based on evaluations of results from advanced concept activities performed by governmental and commercially-sponsored research activities.

Thus, this issue focused on the potential risk that could result from the human error in the use of control room annunciators and included consideration of Items HF4.5 (automation and artificial intelligence), HF5.3 (operational aids), and HF5.4 (computers and computer displays). Proposed solutions to this combined issue were to be changes to the SRP,[11] industry guidance such as a Regulatory Guide, and development of the necessary staff expertise to evaluate proposed designs for the MMI based on advanced technology.

CONCLUSION

This issue was given a high priority ranking and work was undertaken to determine the potential public risk from human error in the use of information from control room annunciators and to assess the safety significance of upgrades identified in studies documented in NUREG/CR-3217[1522] and NUREG/CR-3987.[1523] However, work on this issue was terminated[1526] when the development of review guidance for advanced annunciators was integrated into an existing RES program to develop an "Advanced Human-Interface Design Review Guideline."

ITEM HF5.3: EVALUATION OF OPERATIONAL AID SYSTEMS

DESCRIPTION

Staff guidance pertinent to MMI involving new control and display techniques were to be prepared to include: (1) identification of new and developing display and control technologies having a potential application in nuclear power plant control rooms; (2) development of evaluating methods and design criteria related to visual displays; and (3) establishment of the criteria needed for regulatory assessment of advanced control room concepts. In addition, the control and display requirements for crew response needs following a seismic event were to be identified.

Based on the results of an investigation of means for monitoring and verifying operations, test, and maintenance activities, the staff was to make determinations concerning: (1) the comparative adequacy of status monitoring in plants that did not have automatic monitoring systems; (2) the adequacy of operational systems designed to be in conformance with Regulatory Guide 1.47[150];

and (3) the development of long-term improvement guidance addressing the feasibility and value/impact of instrumentation backfits.

CONCLUSION

This issue was covered in Item HF5.2.

ITEM HF5.4: COMPUTERS AND COMPUTER DISPLAYS

DESCRIPTION

A program plan will be developed to evaluate the safety significance and problems relating to the management of data and information in the nuclear power plant control room during abnormal events. Products may include the development of guidelines on control room information management during severe transients and accidents. These guidelines may be in the form of NUREG reports and Regulatory Guides.

CONCLUSION

This issue was covered in Item HF5.2.

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APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING AND FUTURE PLANTS

This appendix contains a listing of those residual GSIs that are applicable to operating and future plants and includes: issues that have been resolved with requirements [NOTE 3(a), I]; 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. Also included in this listing are those GSIs that were either prioritized or resolved with no impact on operating plants, but contain recommendations for future plants (NOTE 6).

Legend

NOTES: 1 - Possible Resolution Identified for Evaluation
2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER or equivalent)
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

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			
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	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
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

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.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.3	Safety System Status Monitoring	MEDIUM	All	All		-	-
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.D.5(3)	On-Line Reactor Surveillance System	NOTE 1	All	All			
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
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

Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants - Effective Date	Future Plants- Effective Date
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
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

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.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.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	NA	B&W		05/--/80	NA
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
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

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(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, W		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, W			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, W			NA
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	NOTE 3(a)	NA	W			
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	03/31/30
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	

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
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
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

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(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	I	GE	NA	F-4,	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
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	I	All	All			
III.A.2.1(2)	Conduct Public Regional Meetings	I	All	All			

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			
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	I	All	All			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		
III.A.2.2	Development of Guidance and Criteria	I	All	All	F-68		
<u>III.A.3</u>	<u>Improving NRC Emergency Preparedness</u>						
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			
<u>III.D</u>	<u>RADIATION PROTECTION</u>						
<u>III.D.1</u>	<u>Radiation Source Control</u>						
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
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
III.D.3.3	Inplant Radiation Monitoring	-					
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	--	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		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		09/13/79	[illegible]/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
	<u>TASK ACTION PLAN ITEMS</u>						
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	<u>W</u>		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
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

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-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		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		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-17	Criteria for Safety-Related Operator Actions	MEDIUM	All	All		TBD	TBD
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-55	Improved Reliability of Target Rock Safety Relief Valves	MEDIUM	All	NA		TBD	TBD
B-56	Diesel Reliability	NOTE 3(a)	All	All	D-19	06/--/93	06/--/93
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	All	All		TBD	TBD
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	NOTE 3(a)	All	All		04/20/81	
B-64	Decommissioning of Reactors	NOTE 2	All	All		TBD	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
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

Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants - Effective Date	Future Plants- Effective Date
	NEW GENERIC ISSUES						
15.	Radiation Effects on Reactor Vessel Supports	HIGH	All	All		TBD	TBD
23.	Reactor Coolant Pump Seal Failures	HIGH	NA	All		TBD	TBD
24.	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	NA	All		TBD	TBD
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		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		07/18/89	07/18/89
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	All	All		TBD	TBD
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
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM	All	All		TBD	TBD
83.	Control Room Habitability	NOTE 1	All	All		TBD	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		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		10/17/88	10/17/88
103.	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All		10/19/89	10/19/89
106.	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	All	All		TBD	TBD
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

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			
128.	Electrical Power Reliability	NOTE 3(a)	All	All		04/29/91	04/29/91
130.	Essential Service Water Pump Failures at Multiplant Sites	NOTE 3(a)	NA	All		09/19/91	09/19/91
143.	Availability of Chilled Water Systems	HIGH	All	All		TBD	TBD
146.	Support Flexibility of Equipment and Components	NOTE 4	All	All		TBD	TBD
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-		-	-
155.1	More Realistic Source Term Assumptions	NOTE 2	All	All		TBD	TBD
156.	Systematic Evaluation Program	-	-	-		-	-
156.6.1	Pipe Break Effects on Systems and Components	NOTE 4	All	All		TBD	TBD
158.	Performance of Power-Operated Valves Under Design Basis Conditions	NOTE 4	All	All		TBD	TBD
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	NOTE 4	All	All		TBD	TBD
160.	Spurious Actions of Instrumentation Upon Restoration of Power	NOTE 4	All	All		TBD	TBD
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	NOTE 4	All	All		TBD	TBD
163.	Multiple Steam Generator Tube Leakage	NOTE 4	NA	All		TBD	TBD
165.	Safety and Safety/Relief Valve Reliability	NOTE 4	All	All		TBD	TBD
166.	Adequacy of Fatigue Life of Metal Components	NOTE 1	All	All		TBD	TBD
167.	Combustible Gas Storage Facilities	NOTE 4	All	All		TBD	TBD
168.	Environmental Qualification of Electrical Equipment	NOTE 1	All	All		TBD	TBD
		HUMAN FACTORS ISSUES					
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84
HF4	PROCEDURES						
HF4.4	Guidelines for Upgrading Other Procedures	HIGH	All	All		TBD	TBD

APPENDIX C

PRIORITY RANKING NUMERICAL THRESHOLDS

USED IN PRIORITIZATIONS COMPLETED BEFORE JUNE 30, 1993

TABLE 1

RISK THRESHOLDS

(a) The priority rank is always HIGH when any of the following risk (or risk related) thresholds are estimated to be exceeded (or when extraordinary uncertainty suggests that they may well be exceeded):

(1) 1,000 person-rem estimated public dose per remaining reactor lifetime
(2) 50,000 person-rem total estimated for all affected reactors for their remaining lifetime (e.g., 500 person-rem/reactor for 100 reactors)

(3) 10^{-5}/reactor-year large-scale core-melt

(4) 5×10^{-4}/year large-scale core-melt (total for all affected reactors)

(b) Always at least MEDIUM priority:
10 or more percent of the always-HIGH criteria

(c) Always at least LOW priority:
1 or more percent of the always-HIGH Criteria

(d) Never higher than MEDIUM priority:
Less than 10% of the always-HIGH criteria

(e) Never higher than LOW priority:
Less than 1% of the always-HIGH criteria

(f) Aways DROP category:
Less than 0.1% of the always-HIGH criteria

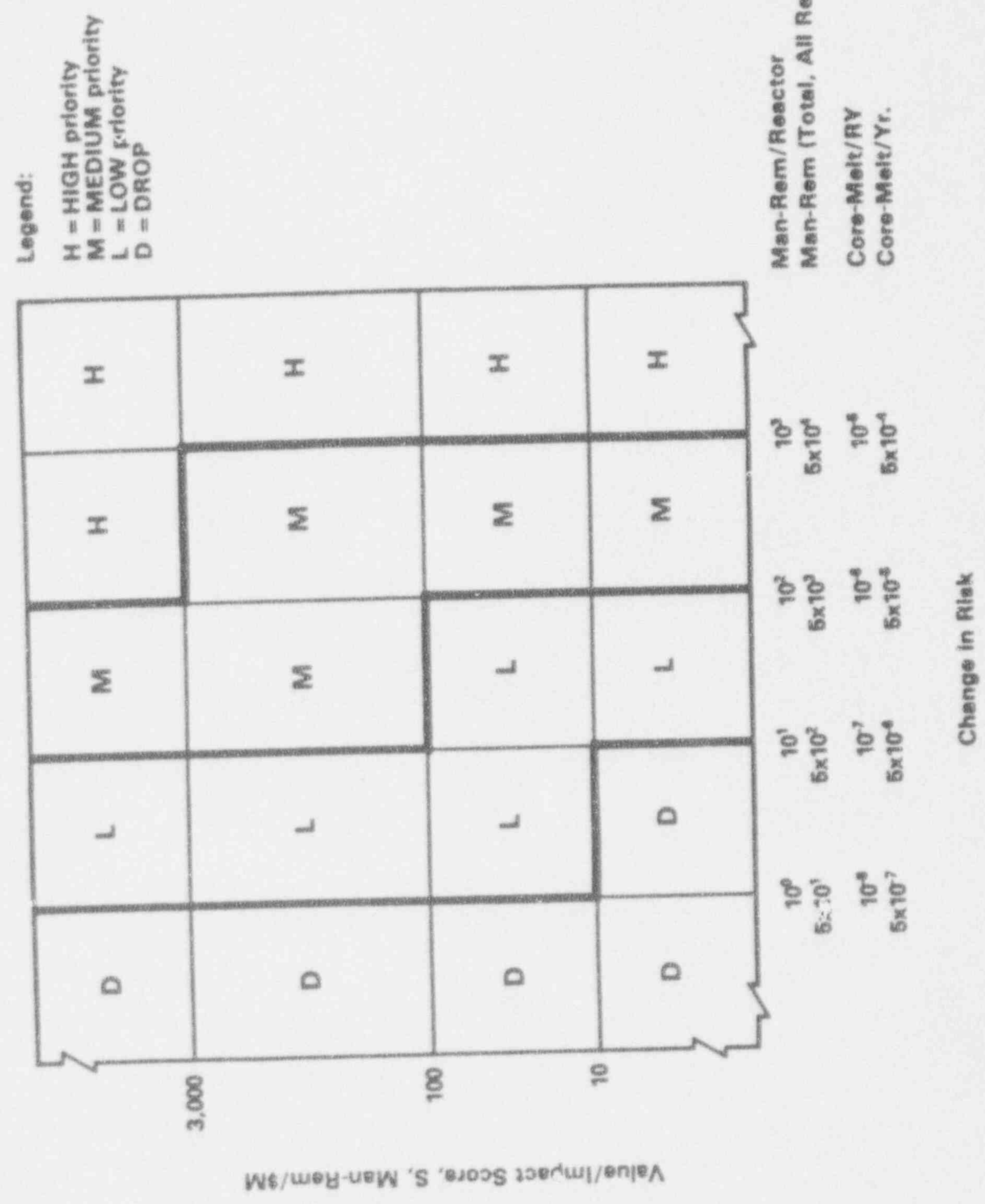

Figure 1-Priority Ranking

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The report presents the priority rankings 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 resolutions 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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