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SUPPLEMENT 3 TO NUREG-0933

"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

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## INTRODUCTION

### I. BACKGROUND

#### A. NRR Operating Plan

The NRC Policy and Planning Guidance, 1983 (NUREG-0885,<sup>210</sup> Issue 2), in addressing the area of Coordinating Regulatory Requirements (Planning Guidance, Item 5, p.6), states that "...a priority list of generic safety issues including TMI-related issues based on the potential safety significance and cost of implementation of each issue" should be submitted to the Commission for approval. This guidance is reflected in the NRR Operating Plan which assigns to the Division of Safety Technology (DST) lead responsibility for preparing a list of generic safety issues and their priority.

This report contains a recommended priority list with a documented basis for the priority of each issue, submitted in response to the assignment made by the NRR Operating Plan. These "final" priority rankings can, of course, be reconsidered in those cases where developments in the course of resolution efforts or other new information suggest cause for review.

#### B. Purpose and Scope

The primary purpose of the priority rankings 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 not included in this prioritization program, because of the compressed time scale on which decisions for such issues must be made.

The prioritization focuses on generic safety issues, i.e., possible deficiencies in the design, construction, or operation of several or a class of nuclear power plants such that the protection of the public from radiation may be inadequate. 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 have been identified as Regulatory Impact issues to clearly differentiate them as not being potential deficiencies in the safety of nuclear power plants but, nevertheless, possibly worthwhile.

In order to identify generic safety issues, all issues are reviewed to determine their safety significance. Where the list includes issues that concern primarily the licensing process or environmental protection and do not involve significant safety-improvement elements, they are identified accordingly and noted for separate consideration

outside the safety-issue priority ranking scheme. Environmental protection issues are issues involving impacts on the human environment and the values sought to be protected by the National Environmental Policy Act (NEPA). Licensing issues are issues not directly related to protecting public health and safety or the environment. These include issues related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety; improving or maintaining the NRC capability to make independent assessments of safety; establishing, revising, and carrying out programs to identify and resolve safety issues; documenting, clarifying, or correcting current requirements and guidance; and improving the effectiveness or efficiency of the review of applications.

The risk estimates developed for safety issues are useful as rough approximations for comparative purposes but are not necessarily applicable to assessment of absolute levels of risk attributable to particular issues. Similarly, the value/impact scores provide, for the limited purpose of prioritization, tentative assessments of relative potential for cost-effective resolution. They are not intended to be applied as value/impact 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 do prioritizations.

The list of issues includes pending issues in the following groups:

1. TMI Action Plan items under development (NUREG-0660).<sup>48</sup> These issues are covered in Section 1. The priority recommendations in the present report exclude the issues that are being implemented under NUREG-0737.<sup>98</sup>
2. Task Action Plan items, previously-proposed issues in NUREG-0371<sup>2</sup> and NUREG-0471<sup>3</sup>, plus the subsequently added issues A-42 through A-49. These issues are covered in Section 2. However, issues designated as USIs are excluded from this current prioritization because they are already receiving high-priority attention on the basis of priority decisions previously made. In the future, USIs will come from the list of newly-proposed issues and will have been prioritized.
3. New Generic issues, originated in NRR or identified by the ACRS, AEOD, or others. These issues are covered in Section 3. As issues identified by AEOD and others are prioritized, they will be included in Section 3 and published in future supplements to this report. A cross-reference listing of AEOD reports and corresponding generic issues is provided in Table IV.
4. Human Factors Program Plan (HFPP) items under development in NRR and outlined in NUREG-0985.<sup>603</sup> These items will be prioritized in future supplements to this report and included in Section 4.

A listing of all issues and their priority rankings appears in Table II. A summary of the number of issues in each category is shown in Table III.

C. How the Work Was Done

The work was done, in accordance with the criteria described in Paragraph II, by the Safety Program Evaluation Branch (SPEB), D3T, in consultation with others in NRR and elsewhere in NRC with knowledge of the issues or expertise in the technical disciplines involved. In a number of instances, technical or cost information was 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 for specific issues and provided safety-benefit analyses and cost information for many of the issues. SPEB, with internal consultations as necessary, reviewed and applied the PNL-supplied technical factors, in conjunction with additional factors, in actually developing the proposed priority rankings and recommendations.

Systematic peer review of each prioritization analysis within NRC contributed to the assurance that analyses were complete and accurate and that the judgments were soundly based. This review was done in two stages. First, the analysis for each issue was reviewed by the NRC organization unit or units whose area of responsibility or specialized knowledge was substantially involved. These reviews were usually made by the cognizant Branch Chiefs and concurred in by NRC Division Directors. Second, comments were either resolved or, in a few instances, identified as differences that could not be resolved.

After publication of this report, comments from the ACRS, the industry, and the public will be considered in any further reassessment of priority.

D. 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 the nature, extent, and availability of manpower and material resources estimated to be required; length of time needed to resolve; conflicts in resource allocation and scheduling among items of comparable priority; status of affected reactors; and budget constraints).

An issue is considered resolved, indicated by NOTE 3 in Table II, when its resolution has resulted in the establishment of regulatory requirements or guidance (by rule, Standard Review Plan change, or equivalent) or a documented authoritative decision that no change in requirements is warranted. For those issues that result in new requirements, the next step is implementation which is considered complete when the licensees have committed to, and the staff agrees with, a scope and schedule for the modification of hardware or operations at the affected plants. Verification that licensee commitments have been met is done by the Office of Inspection and Enforcement (OIE). Generally, 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 for future use.

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, or equivalent. 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. Priority rankings were not assigned to issues that are nearly resolved (denoted by NOTES 1 and 2) 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 prioritize an issue should be devoted to completing the final evaluation of the issue rather than making a tentative judgment as to the importance and value of the issue.

Assignment of a HIGH priority means that strong efforts to achieve an earliest practical resolution are appropriate. This is because: (a) an important safety deficiency is involved (though generally the deficiency is not severe enough to require prompt plant shutdown), (b) a substantial safety improvement is likely to be attainable at a low enough cost to make the improvement very worthwhile, or (c) the uncertainty of the safety assessment is unusually large and an upper-bound risk assessment would indicate an important safety deficiency. Issues in this category are candidates for possible designation as USIs.

A MEDIUM priority means that no safety deficiency 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, though less so than for items assigned a HIGH priority. Efforts at resolution should be planned, perhaps over the next several years, but on a basis of not interfering with pursuit of HIGH-priority generic issues or other high-priority work.

A LOW priority means that no safety deficiencies 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 NRR Director signifies that the issue has been eliminated from further pursuit.

The DROP category covers proposed issues that are without merit or whose significance is clearly negligible. When the prioritization process results in a DROP priority ranking for an issue, approval of this ranking by the NRR Director signifies that the issue has been eliminated from further pursuit.



## II. CRITERIA FOR ASSIGNING PRIORITIES

### A. Basic Approach

The method of assigning priority rank involves two primary elements: (1) the estimated safety importance of the issue, and (2) the estimated cost of developing and implementing a resolution. Special considerations may influence the proper use of those estimates. These elements are applied as follows:

1. 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.
2. A quantitative estimate is made of the safety importance of the issue, measured in terms of the risk (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.
3. A quantitative estimate is made of the cost of resolution.
4. A numerical value/impact score is calculated by dividing the estimated potential risk reduction by the estimated cost entailed. This score denotes a value-impact relation, i.e., an estimated ratio of safety-improvement value to cost impact.
5. A priority rank (HIGH, MEDIUM, LOW, or DROP) is obtained by application of criteria in which both the safety importance of the issue and the value/impact-based numerical score are taken into account. The score is not always directly applied to determine the priority rankings. In some cases the safety importance 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 value/impact assessment.
6. The priority ranking is reviewed and modified if appropriate in light of any special factors (discussed later in this section) that might (a) bring into question the applicability of the necessarily simplified calculation technique, (b) call for special consideration of often large uncertainties in the quantitative estimates, or (c) should for some other reason influence the ranking.

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.

## B. Safety Importance

The safety importance 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 man-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 "value/impact score" as part of the simplified value/impact analysis, discussed in Paragraph II.C below. It is also used directly as a measure of safety importance, as discussed in Paragraph II.D below, in arriving at a priority rank that is influenced by the safety importance of an issue as well as by the estimated value/impact relation of a projected solution.

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

## C. Value/Impact Relation

### 1. The Value/Impact Score Formula

To the extent reasonably possible, quantitative estimates are made of the projected worthwhileness of resolving a generic safety issue, by calculating a "priority score" that reflects the relation between the risk reduction value expected to be achieved and the associated cost impact. The concept is the same as that presented in a Commission information paper in the summer of 1981 (SECY-81-513,<sup>1</sup> Enclosure 3), but there have been subsequent modifications to the detailed method of calculation.

The basic formula is:

$$\text{Value/Impact Score, } S = \frac{\text{Safety Benefit}}{\text{Cost}},$$

where the safety benefit is the estimated risk reduction (event frequency x public dose averted) that is 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:

$$S = \frac{\text{NFTD}}{C},$$

where N is the number of reactors involved; T is the average remaining life of the affected plants, stated in years; F is the accident frequency reduction, stated in events/reactor-year; D is the public dose from the radioactive material released from containment, stated in man-rem; and C is the total cost of developing and implementing the resolution of the issue for all plants affected, stated in millions of dollars. The total cost, C, includes both the costs of developing the generic solution, which are typically NRC costs, and the costs of implementation of the solution in all affected plants, which include design, equipment, installation, test, operation, and maintenance, and are typically industry costs. The priority score, S, has the units of man-rem per million dollars.

## 2. Rationale for the Formula

The qualitative diversity of factors entering value/impact analyses in support of safety-issue prioritization, together with inevitable quantitative uncertainties, makes any of various possible value/impact score formulas necessarily imperfect. Provisions are, accordingly, made to compensate for those imperfections to the extent practical (as discussed in Paragraph II.E below).

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

- (a) The numerator is designed as a direct measure of the safety values that it is NRC's primary mission to protect. The denominator 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 ratio (the value/impact score) should, subject to the stated caveats, reasonably approximate measuring the overall public interest in safety value received for total resources expended.
- (b) Optimizes the allocation of national resources, which in most cases are mostly industry sources.

## 3. Risk Estimates

The basis of frequency estimates generally involves the following:

- (a) 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,



- (b) Use of event sequence diagrams, fault trees, or decision trees, if possible,
- (c) Identified references and calculations, or stated assumptions for the numbers used,
- (d) Consideration of the probability of common mode as well as random independent failures.

Where possible numerical estimates are made based on operating experience (usually 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 man-rem generally based on the radioactive release categories described in the Reactor Safety Study (WASH 1400,<sup>16</sup> Appendix VI, pp. 2-1 to 2-5, reproduced as Appendix A to this report). The table below gives estimated curies released and approximate population doses for each release category. The computer program CRAC2, applied to a typical mid-west 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.<sup>64</sup> Assumptions and parameters used for the calculations were as follows:

- Dose consequences are represented by the whole body population dose commitment (man-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. [That is the mean 50-mile-radius population density projected for the year 2000 (NUREG-0348,<sup>70</sup> p. T52).]
- Evacuation of people was not considered because calculations suggest that, important though it may sometimes be for people directly affected, the effect of evacuation on the total population dose is likely to be small.
- 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 man-rem factors for each release category are given in the table below. Although generally used, consequence estimates were not solely based on these factors. Other factors were used in some cases when more appropriate.

Estimated occupational doses in postaccident cleanup, repair, and refurbishment are added to the public dose. Generally, 20,000 man-rem for PWR-1 to 7 and BWR-1 to 4 releases and 6,000 man-rem for PWR-8 and 9 and BWR-5 releases were assumed, based on the PNL estimates.<sup>64</sup>

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

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

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

No separate estimates were 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 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 is not 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 total human-factors contribution to total risk. However, most issues not related to human factors are much less strongly interrelated. 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.

#### 4. 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. Dependability, in terms of guarding against omission of important or even dominant cost elements, is more important than precision of the estimates.

The expected technical solution on which the 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:

- (a) Issue identification, analysis, resolution, and report issuance,
- (b) Research to establish proposed specific changes to licensing requirements (or to determine that no change is required); technical assistance contracts (including associated NRC effort),
- (c) Discussions, correspondence with industry owners' groups,
- (d) Plant reviews,

- (e) Preparing SERs and requirement documents and review of these.

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:

- (a) Licensing,
- (b) Design,
- (c) Equipment procurement,
- (d) Installation,
- (e) Testing, inspection, monitoring, and periodic maintenance,
- (f) Plant downtime to effect a change, taken as the cost of replacement power, at \$300,000/day.

Industry manpower costs are taken as \$100,000 per person-year.

In some cases, averted plant-damage costs can substantially affect the priority. Estimates for such averted costs are developed and used in separately stated calculations, so that the priority scores both with and without adjustment for averted plant-damage costs are readily apparent. 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 use of the plant, estimated at approximately \$1 billion present worth, and plant-wide cleanup, estimated (on a basis consistent with TMI estimates<sup>393</sup>) at a present worth of about \$400 million, both based on a 5% real discount rate and multiplied in each case by the reduction in frequency of such events that would be brought about by resolution of the generic safety issue. 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. Thus, for complete plant loss, the \$1 billion and \$400 million are added. 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 5% discount rate. Where costs that are continuous (or periodically recurring) throughout the plant's remaining life are involved, a figure of 10 times the annual cost is taken as

a reasonable approximation of the present worth of the continuing (or repetitive) costs for plants with remaining operating lives of 20 years or longer.

#### 5. 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 the PNL report (NUREG/CR-2800,<sup>64</sup> 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 E.1.). Most often, however, a rigorous uncertainty analysis has not been 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. 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 the uncertainty has a special character or importance, this is discussed and considered in the final conclusion for an issue.

#### D. Priority Ranking

##### 1. Priority Ranking Chart

A chart showing how the tentative priority rankings are derived from the safety importance of an issue and its value-impact priority score is presented in Figure 1. The thresholds on the chart are explained in Paragraphs D.2 and D.3 below.

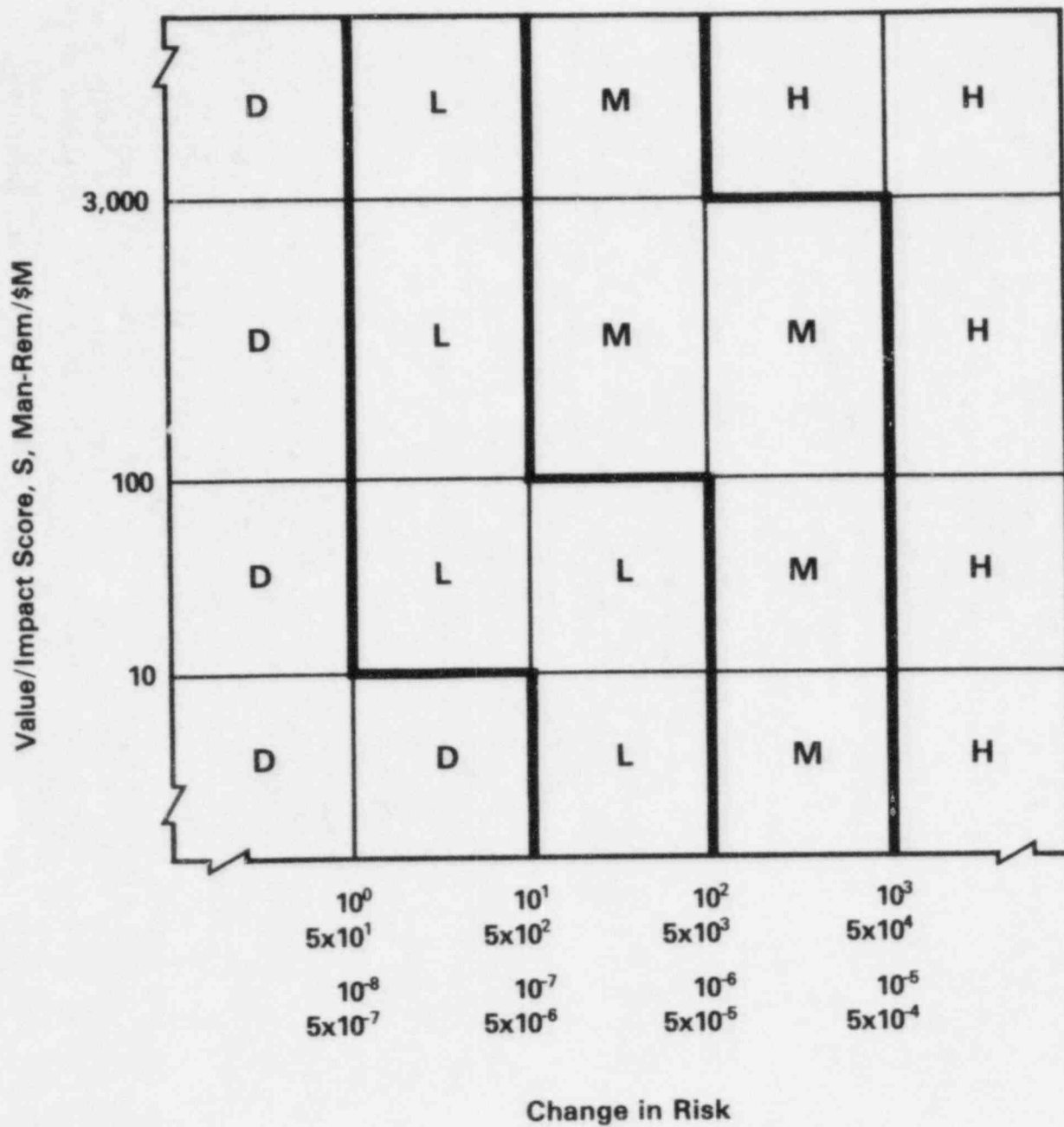
##### 2. Preliminary Screening for Safety Importance

The value/impact-based priority score is applied after a preliminary screening on the basis of safety importance, i.e., the incremental risk associated with the issue.

The safety importance of an issue may be so great that it should be accorded a HIGH priority regardless of other considerations, such as an initially estimated high cost, which might result in a low priority score: when a generic safety issue is very important from the safety viewpoint, the assignment of a HIGH priority to its resolution should not be deterred by the initial absence of an identified solution that could be implemented with a moderate cost.



Figure 1-Priority Ranking



Legend:

H = HIGH priority  
 M = MEDIUM priority  
 L = LOW priority  
 D = DROP

Man-Rem/Reactor  
 Man-Rem (Total, All Reactors)  
 Core-Melt/Ry  
 Core-Melt/Yr.

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 high priority score because an inexpensive solution is believed to be available. Below a minimal safety importance 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 that demand an at least MEDIUM (or at least LOW) priority or warrant an at most MEDIUM (or at most LOW) priority.

The risk-based priority ranking thresholds are shown in Table I. Thresholds a(2) and a(4) in Table I reflect the view that an issue affecting a large number of reactors may warrant as high a priority as an issue that involves somewhat greater per-reactor risk but affects only a few reactors.

### 3. Value/Impact Score Thresholds

To the extent consistent with the safety-importance screening criteria just discussed, the value-impact priority score, S, is translated into priority rankings in accordance with the following thresholds:

- a. If at least 3,000 man-rem/\$million, an issue that is above 10% of the HIGH risk threshold would warrant a HIGH priority rather than a MEDIUM priority.
- b. If less than 100 man-rem/\$million, an issue that is below 10% of the HIGH risk threshold would only warrant a LOW priority rather than a MEDIUM priority.
- c. If less than 10 man-rem/\$million, an issue that is below 1% of the HIGH risk threshold would only warrant a DROP priority rather than a LOW priority.

### E. Other Considerations

The formula-based rankings represent the primary concerns of the NRC: public safety and the impact on licensees. However, these tentative priority rankings are subject to the limitations of an often incomplete and quite imprecise data base and to possible distortions due to the nature of the necessarily highly simplified quantitative formula underlying them. (This is the principal reason for establishing such low threshold values for the LOW and DROP categories.) Special situations with respect to some issues may cause added difficulty in priority assignment. While the formula-based tentative rankings must 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 estimate, whether the effect



TABLE I  
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 man-rem estimated public dose per remaining reactor lifetime
  - (2) 50,000 man-rem total estimated for all affected reactors for their remaining lifetime (e.g., 500 man-rem/reactor for 100 reactors)
  - (3)  $10^{-5}$ /reactor-year large-scale core melt
  - (4)  $5 \times 10^{-4}$ /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) Always DROP category:  
Less than 0.1% of the always-HIGH criteria
-

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, averted plant-damage costs, and uncertainty bounds [factors 1(a), 1(b), 1(c), and 2(a) 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 priority score and its associated uncertainties. Sometimes, there are special considerations that are quite specific to an issue or some aspect of it. The partial list of other factors is listed below.

1. Special risk and cost aspects not included in or potentially masked by the numerical formulas:
  - (a) The net change in occupational doses implicit in implementing the current versus the proposed requirements; also, non-radiological occupational hazards inherent in, or affected by, the proposed resolutions,
  - (b) Any significant non-radiation related occupational risk,
  - (c) Averted cost of plant damage from the postulated accident,
  - (d) Loss or severe degradation of a layer in the defense-in-depth concept (e.g., one mode of core cooling or containment cooling),
  - (e) 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.
2. Factors related to uncertainties stemming from an incomplete or imprecise data base for the priority formula:
  - (a) 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,
  - (b) Situations where uncertainty is extraordinarily large (in accident probability or consequences or in cost, or any or all of these),
  - (c) 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,
  - (d) 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 current 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,

- (e) 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,
  - (f) Potential for human intervention, using available equipment.
3. Perceptions and judgments that cannot (or cannot readily) be quantified:
- (a) Public concern about a particular issue, or special Commission or Congressional concern,
  - (b) Acute knowledgeable professional controversy concerning the importance of an issue or modes of dealing with it.
4. Change with passage of time:
- (a) Potential substantial deterioration of the value/impact 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),
  - (b) 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,
  - (c) The span of time predicted to resolve an issue and implement the resolution,
  - (d) 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,
  - (e) Change of perceptions (of safety importance or value/impact 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 the issue as currently understood is assigned. Thus, where no solution is evident, assignment of the highest priority consistent with the safety importance of the issue may lead to search for resolution or mitigation at 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 continued 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.

#### F. 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.

### III. LISTING OF ALL ISSUES EVALUATED

The classification, lead responsibility, priority ranking, and status of each issue evaluated in this report are listed in Table II.

### IV. RESULTS OF PRIORITIZATION

The results of the prioritization of all issues contained in this report are summarized and tabulated by issue type and priority category 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 the prioritization process and only includes: (1) issues that have been resolved with new requirements [NOTE 3(a)]; (2) USI, HIGH and MEDIUM priority issues that are under development; (3) nearly-resolved issues (NOTES 1 and 2) whose impact is not yet known; (4) issues that are scheduled for prioritization (NOTE 4); and (5) issues that are covered in other issues that fall in any of the above categories.

### REFERENCES

1. SECY-81-513, "Plan for Early Resolution of Safety Issues," August 25, 1981.
2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.
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.

- 48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.
- 64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.
- 70. NUREG-0348, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," U.S. Nuclear Regulatory Commission, November 1979.
- 98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.
- 210. NUREG-0885, Issue 2, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission, January 1983.
- 393. "TMI-2 Recovery Program Estimate," Rev. 1, General Public Utilities Corp., July 1981.
- 603. NUREG-0985, "U.S. NRC Human Factors Program Plan," U.S. Nuclear Regulatory Commission, August 1983.

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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, 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 resolved issues that have resulted in new requirements for operating plants, the appropriate multi-plant 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
- HIGH - High Safety Priority
- MEDIUM - Medium Safety Priority
- LOW - Low Safety Priority
- DROP - Issue Dropped as a Generic Issue
- E - Environmental Issue
- HFPP - Human Factors Program Plan
- I - TMI Action Plan Item With Implementation of Resolution Mandated by NUREG-0737<sup>98</sup>
- LI - Licensing Issue
- MPA - Multi-Plant Action (See Status in NUREG-0748)<sup>578</sup>
- NA - Not Applicable
- RI - Regulatory Impact Issue
- USI - Unresolved Safety Issue (See Status in NUREG-0606)<sup>60</sup>



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TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>TMI ACTION PLAN ITEMS</u>							
<u>I.A.</u>	<u>OPERATING PERSONNEL</u>						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I			F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I			
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I			F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFQ/HFBR	NOTE 3(a)	1	6/30/84	
<u>I.A.2</u>	<u>Training and Qualifications of Operating Personnel</u>						
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			F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I			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			F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	NOTE 3(b)	2	6/30/85	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I			
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI	2	6/30/85	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	LOW	2	6/30/85	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	Colmar	NRR/DHFS/LQB	HFPP	2	6/30/85	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	2	6/30/85	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	2	6/30/85	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	MEDIUM	2	6/30/85	
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	2	6/30/85	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	2	6/30/85	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	NOTE 3(b)	2	6/30/85	NA
<u>I.A.3</u>	<u>Licensing and Requalification of Operating Personnel</u>						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	3	6/30/85	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	3	6/30/85	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DFQ/HFBR	HFPP	3	6/30/85	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	NOTE 3(b)	3	6/30/85	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	3	6/30/85	NA
<u>I.A.4</u>	<u>Simulator Use and Development</u>						
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)	1	12/31/84	NA

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TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	1	12/31/84	
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	RES/DHFS/OLB	HFPP	1	12/31/84	NA
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFQ/HFBR	NOTE 3(a)	1	12/31/84	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFQ/HFBR	NOTE 3(a)	1	12/31/84	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DHFS/OLB	HFPP	1	12/31/84	NA
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	1	12/31/84	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI	1	12/31/84	NA
<u>I.B.</u>	<u>SUPPORT PERSONNEL</u>						
<u>I.B.1</u>	<u>Management for Operations</u>						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	1	12/31/84	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	75, HFPP	1	12/31/84	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	75, HFPP	1	12/31/84	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	I			
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	I			
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	I			
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)	1	12/31/84	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	1	12/31/84	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	1	12/31/84	NA
<u>I.B.2</u>	<u>Inspection of Operating Reactors</u>						
<u>I.B.2.1</u>	<u>Revise OIE Inspection Program</u>	-	-	-			
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

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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
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
<u>I.C</u>	<u>OPERATING PROCEDURES</u>						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I			
I.C.1(2)	Inadequate Core Cooling	-	NRR	I			
I.C.1(3)	Transients and Accidents	-	NRR	I			
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I			
I.C.3	Shift Supervisor Responsibilities	-	NRR	I			
I.C.4	Control Room Access	-	NRR	I			
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I			F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I			F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I			
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I			
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	HFPP	1	12/31/84	NA
<u>I.D</u>	<u>CONTROL ROOM DESIGN</u>						
I.D.1	Control Room Design Reviews	-	NRR/DL	I			F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I			F-09
I.D.3	Safety System Status Monitoring	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.4	Control Room Design Standard	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	1	12/31/84	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	1	12/31/84	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DET/EEIGB	NOTE 1	1	12/31/84	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	1	12/31/84	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	1	12/31/84	NA
<u>I.E</u>	<u>ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE</u>						
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
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
<u>I.F</u>	<u>QUALITY ASSURANCE</u>						
I.F.1	Expand QA List	Pittman	OIE/DQASIP/QUAB	HIGH		11/30/83	
I.F.2	Develop More Detailed QA Criteria	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	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		11/30/83	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>I.G</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I			
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PSRB	NOTE 3(a)	1	12/31/84	NA
<u>II.A</u>	<u>SITING</u>						
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
<u>II.B</u>	<u>CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW</u>						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I			F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I			F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I			F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I			F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	V'Molen	RES/DAE/FBRB	HIGH		11/30/83	
II.B.5(2)	Behavior of Core Melt	V'Molen	RES/DAE/CSRB	HIGH		11/30/83	
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DAE/CSRB	MEDIUM		11/30/83	
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	HIGH		11/30/83	
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8		11/30/83	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/ASTOP	HIGH		11/30/83	
<u>II.C</u>	<u>RELIABILITY ENGINEERING AND RISK ASSESSMENT</u>						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRA/RRBR	HIGH		11/30/83	
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	HIG		11/30/83	
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17		11/30/83	
II.C.4	Reliability Engineering	Pittman	RES/DRA/RRBR	HIGH		11/30/83	
<u>II.D</u>	<u>REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES</u>						
II.D.1	Testing Requirements	-	NRR/DL	I			F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW		11/30/83	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>II.E</u>	<u>SYSTEM DESIGN</u>						
II.E.1	Auxiliary Feedwater System	-	NRR/DL	I			F-15
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I			F-16, F-17
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication						
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)		11/30/83	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)		11/30/83	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	MEDIUM		11/30/83	
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW		11/30/83	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR	I			
II.E.3.2	Systems Reliability	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)		11/30/83	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45		11/30/83	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I			F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I			F-19
II.E.4.3	Integrity Check	Milstead	NRR/DSI/CSB	HIGH		11/30/83	
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.5	Design Sensitivity of B&W Reactors						
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	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	Thatcher	NRR/DE/EQB	MEDIUM		11/30/83	



TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I			F-20, F-21, F-22, F-23, F-24, F-25 F-26
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I			
II.F.3	Instruments for Monitoring Accident Conditions	V'Molen	RES/DFO/ICBR	NOTE 3(a)		11/30/83	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP		11/30/83	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DFO/ICBR	MEDIUM		11/30/83	
<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/DAE/FBRB	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>						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
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



TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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	RES/DRA/RABR	NOTE 2		11/30/83	
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 PN's 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 Phases of, the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	Emrit	NRR	NOTE 3(a)		12/31/84	-
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		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		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
II.K.3(6)	Instrumentation to Verify Natural Circulation	Emrit	NRR/DSI	I.C.1, 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	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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
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, 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

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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)	Kelap-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		12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	Emrit	NRR	I.C.1		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		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		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		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensable Gases	Emrit	NRR	I.C.1		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, 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		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	Emrit	NRR	I.C.1, 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, 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

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>III.A</u>	<u>EMERGENCY PREPAREDNESS AND RADIATION EFFECTS</u>						
<u>III.A.1</u>	<u>Improve Licensee Emergency Preparedness - Short Term</u>						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I			
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	I			
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-			
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I			F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB	I			F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB	I			F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-			
III.A.1.3(1)	Workers	Riggs	OIE/DEPER/EPB	NOTE 3(b)		11/30/83	NA
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 1		11/30/83	
<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	6/30/85	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	6/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	6/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	6/30/85	NA
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	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	6/30/85	NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	6/30/85	NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	6/30/85	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<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
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
<u>III.C</u>	<u>PUBLIC INFORMATION</u>						
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
<u>III.D</u>	<u>RADIATION PROTECTION</u>						
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			
III.D.1.1(2)	Review Information on Provisions for Leak Detection	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP		11/30/83	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		11/30/83	NA
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)		11/30/83	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP		11/30/83	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	1	6/30/84	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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	1	6/30/84	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	1	6/30/84	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)	1	6/30/84	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	1	6/30/84	NA
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	1	6/30/84	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	1	6/30/84	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 1	1	6/30/84	-
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	-
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	-
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	-
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)	1	6/30/84	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/DRPB	LI (NOTE 3)	1	6/30/84	NA
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3(b)	1	6/30/84	-
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/DRPB	LI (NOTE 3)	1	6/30/84	NA
III.D.3	Worker Radiation Protection Improvement	-	-	-	-	-	-
III.D.3.1	Radiation Protection Plans	V'Molen	NRR/DSI/RAB	HIGH	-	11/30/83	-
III.D.3.2	Health Physics Improvements	-	-	-	-	-	-
III.D.3.2(1)	Amend 10 CFR 20	V'Molen	RES/DFQ/DRPBR	LI (NOTE 2)	-	11/30/83	NA
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFQ/DRPBR	LI (NOTE 3)	-	11/30/83	NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	RES/DFQ/DRPBR	LI (NOTE 2)	-	11/30/83	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	V'Molen	RES/DFQ/DRPBR	LI (NOTE 2)	-	11/30/83	NA
III.D.3.3	Implant Radiation Monitoring	-	-	-	-	-	-
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I	-	-	F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	I	-	-	-
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	I	-	-	-
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	I	-	-	-
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/DFQ/DRPBR	LI	-	11/30/83	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	V'Molen	RES/DFQ/DRPBR	LI (NOTE 3)	-	11/30/83	NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/DFQ/DRPBR	LI (NOTE 3)	-	11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>IV.A</u>	<u>STRENGTHEN ENFORCEMENT PROCESS</u>						
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
<u>IV.B</u>	<u>ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES</u>						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
<u>IV.C</u>	<u>EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS</u>						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
<u>IV.D</u>	<u>NRC STAFF TRAINING</u>						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA
<u>IV.E</u>	<u>SAFETY DECISION-MAKING</u>						
IV.E.1	Expand Research on Quantification of Safety Decision-Making	Colmar	RES/DRA/RABR	LI		11/30/83	NA
IV.E.2	Plan for Early Resolution of Safety Issues	Emrit	NRR/DST/SPEB	LI (NOTE 3)		11/30/83	N/A
IV.E.3	Plan for Resolving Issues at the CP Stage	Colmar	RES/DRA/RABR	LI (NOTE 2)		11/30/83	NA
IV.E.4	Resolve Generic Issues by Rulemaking	Colmar	RES/DRA/RABR	LI		11/30/83	NA
IV.E.5	Assess Currently Operating Reactors	Matthews	NRR/DL/SEPB	HIGH		11/30/83	
<u>IV.F</u>	<u>FINANCIAL DISINCENTIVES TO SAFETY</u>						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	Thatcher	OIE/DQASIP	NOTE 3(b)		11/30/83	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	Matthews	SP	NOTE 3(b)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>IV.G</u>	<u>IMPROVE SAFETY RULEMAKING PROCEDURES</u>						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)		11/30/83	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI		11/30/83	NA
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)		11/30/83	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)		11/30/83	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)		11/30/83	NA
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TASK ACTION PLAN ITEMS

A-1	Water Hammer	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	D-10
A-3	Westinghouse Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-4	CE Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-5	B&W Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-6	Mark I Short-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-7	Mark I Long-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	D-01
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	NA
A-9	ATWS	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-10	BWR Feedwater Nozzle Cracking	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-25
A-11	Reactor Vessel Materials Toughness	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	Emrit	NRR/DST/GIB	USI [NOTE 2]	1	6/30/85	NA
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
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 Interaction	-	NRR/DST/GIB	USI		11/30/83	
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	Thatcher	NRR/DSI/ICSB	NOTE 4		11/30/83	
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI		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



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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		11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	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	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI		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	NRR/DSI/ASB	MEDIUM		11/30/83	
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	HIGH		11/30/83	
A-31	RHR Shutdown Requirements	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
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	E(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	Emrit	NRR/DSI/GIB	USI [NOTE 3(a)]	1	6/30/85	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	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-40	Seismic Design Criteria - Short Term Program	-	NRR/DST/GIB	USI		11/30/83	
A-41	Long Term Seismic Program	Colmar	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-05
A-43	Containment Emergency Sump Performance	-	NRR/DST/GIB	USI		11/30/83	
A-44	Station Blackout	-	NRR/DST/GIB	USI		11/30/83	
A-45	Shutdown Decay Heat Removal Requirements	-	NRR/DST/GIB	USI		11/30/83	
A-46	Seismic Qualification of Equipment in Operating Plants	-	NRR/DST/GIB	USI		11/30/83	
A-47	Safety Implications of Control Systems	-	NRR/DST/GIB	USI		11/30/83	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	-	NRR/DST/GIB	USI		11/30/83	
A-49	Pressurized Thermal Shock	-	NRR/DST/GIB	USI		11/30/83	
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	E (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (DROP)		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	NRR/DE/SGEB	MEDIUM		11/30/83	
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DE/MEB	HIGH		11/30/83	
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (DROP)		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		11/30/83	NA

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B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI		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 (DROP)		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	NRR/DHFS/LQB	MEDIUM		11/30/83	
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)	1	6/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI		11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-22	LWR Fuel	V'Molen	NRR/DSI/CPB	NOTE 4		11/30/83	
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Components	Emrit	NRR	A-46		11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI		11/30/83	
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI		11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	NOTE 4		11/30/83	
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI		11/30/83	
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	NOTE 4		11/30/83	
B-32	Ice Effects on Safety Related Water Supplies	Pittman	NRR/DE/EHEB	NOTE 4		11/30/83	
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
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		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	E		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	E		11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	E (B-2)		11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	E (DROP)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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 CRD Mechanical Failure (Collet Housing)	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI		11/30/83	
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	RI (LOW)	1	6/30/85	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	NRR/DE/MEB	MEDIUM		11/30/83	
B-56	Diesel Reliability	Milstead	NRR/DSI/PSB	HIGH		11/30/83	
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	Colmar	NRR/DE/eqb	MEDIUM		11/30/83	
B-59	N-1 Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	RI (NOTE 3)	1	6/30/85	NA
B-60	Loose Parts Monitoring System	Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	Pittman	NRR/DST/RRAB	MEDIUM		11/30/83	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (DROP)		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	NRR/DE/CHEB	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		11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
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		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		11/30/83	NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSI/RSB	NOTE 4		11/30/83	
C-5	Decay Heat Update	Riggs	NRR/DSI/CPB	NOTE 4		11/30/83	

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
C-6	LOCA Heat Sources	Riggs	NRR/DSI/CPB	NOTE 4		11/30/83	
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	NRR/DSI/ASB	HIGH		11/30/83	
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	Matthews	NRR/DE/MEB	MEDIUM		11/30/83	
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		11/30/83	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	NOTE 4		11/30/83	
C-15	NUREG Report for Liquids Tank Failure Analysis	-	NRR/DE/EHEB	LI (DROP)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	E (DROP)		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	NRR/DSI/RSB	NOTE 4		11/30/83	
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
<u>NEW GENERIC ISSUES</u>							
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	Colmar	NRR/DSI/ICSB	NOTE 4		11/30/83	
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSI/ICSB	NOTE 2		11/30/83	
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		11/30/83	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		11/30/83	NA
10.	Surveillance and Maintenance of T1P 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	Matthews	NRR/DE/MTEB	NOTE 2		11/30/83	
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	LOW		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
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 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 Plant Systems	Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	6/30/84	NA
21.	Vibration Qualification of Equipment	Thatcher	NRR/DE/eqB	NOTE 4		11/30/83	
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	NRR/DSI/ASB	HIGH		11/30/83	
24.	Automatic Emergency Core Cooling System Switch to Recirculation	V'Molen	NRR/DSI/RSB	NOTE 4		11/30/83	
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	NRR/DE/MTEB	HIGH		11/30/83	
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR	NOTE 4		11/30/83	
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	6/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	V'Molen	NRR/DSI/CPB, RSB	LOW	1	6/30/85	NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, AEB, RSB	NOTE 1	1	6/30/84	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Colmar	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C. 1	1	6/30/85	NA
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	Milstead	NRR	NOTE 4		11/30/83	
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	6/30/84	B-65
41.	BWR Scram Discharge Volume Systems	V'Molen	NRR/DSI/RSB	NOTE 3(a)		11/30/83	B-58
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	I.C. 1	1	6/30/85	NA
43.	Contamination of Instrument Air Lines	Milstead	NRR/DSI/ASB	DROP		11/30/83	NA
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43		11/30/83	NA



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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3(a)	1	6/30/84	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76		11/30/83	NA
47.	Loss of Off-Site 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	NOTE 2		11/30/83	
49.	Interlocks and LCOs for Redundant Class 1E Tie Breakers	Sege	NRR/DSI/PSB	MEDIUM	1	12/31/84	
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	NRR/DSI/ASB	MEDIUM		11/30/83	
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	6/30/85	NA
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Emrit	NRR/DSI/PSB	NOTE 4		11/30/83	
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	V'Molen	NRR	NOTE 4		11/30/83	
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	1	6/30/85	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	MEDIUM		11/30/83	
62.	Reactor Systems Bolting Applications	V'Molen	NRR	NOTE 4		11/30/83	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	V'Molen	NRR	NOTE 4		11/30/83	
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	HIGH		11/30/83	
66.	Steam Generator Requirements	Riggs	NRR/DL/DRAB	NOTE 2	1	6/30/85	
67.	Steam Generator Staff Actions	Riggs	-	-		-	
67.2.1	Integrity of Steam Generator Tube Sleeves	Riggs	NRR/DE/MEB	RI	1	6/30/85	NA
67.3.1	Steam Generator Overfill	Riggs	NRR/DST/GIB	A-47, I.C.1	1	6/30/85	NA
67.3.2	Pressurized Thermal Shock	Riggs	NRR/DST/GIB	A-49	1	6/30/85	NA
67.3.3	Improved Accident Monitoring	Riggs	NRR/DSI/ICSB	I	1	6/30/85	A-17

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Action Plan Item/ Issue No.	Title	Lead SPEB 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	1	6/30/85	F-26
67.4.1	RCP Trip	Riggs	NRR/DSI/RSB	II.K.3(5)	1	6/30/85	G-1
67.4.2	Control Room Design Review	Riggs	NRR/DHFS/HFEB	I.D.1	1	6/30/85	F-08
67.4.3	Emergency Operating Procedures	Riggs	NRC/DHFS/PSRB	I.C.1	1	6/30/85	F-05
67.5.1	Reassessment of Radiological Consequences	Riggs	NRC/DSI/AEB	LI	1	6/30/85	NA
67.5.2	Reevaluation of SGTR Design Basis	Riggs	NRR/DSI/RSB	LI	1	6/30/85	NA
67.5.3	Secondary System Isolation	Riggs	NRR/DSI/RSB	DROP	1	6/30/85	NA
67.6.0	Organizational Responses	Riggs	OIE/DEPER/IRDB	III.A.3	1	6/30/85	NA
67.7.0	Improved Eddy Current Tests	Riggs	NRR/DE/MTEB	MEDIUM	1	6/30/85	
67.8.0	Denting Criteria	Riggs	NRR/DE/MTEB	RI	1	6/30/85	NA
67.9.0	Reactor Coolant System Pressure Control	Riggs	NRR/DSI/GIB	A-45,	1	6/30/85	
			NRR/DSI/RSB	I.C.1			
67.10.0	Supplemental Tube Inspections	Riggs	NRR/DL/ORAB	LI	1	6/30/85	NA
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	Pittman	NRR/DSI/ASB	HIGH	1	6/30/84	
69.	Make-up Nozzle Cracking in B&W Plants	Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	(later)
70.	PORV and Block Valve Reliability	Riggs	NRR/DSI/RSB	MEDIUM	1	6/30/84	
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Emrit	NRR	NOTE 4		11/30/83	
72.	Control Rod Drive Guide Tube Support Pin Failures	V'Molen	NRR	NOTE 4		11/30/83	
73.	Detached Thermal Sleeves	Colmar	NRR	NOTE 4		11/30/83	
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR	NOTE 4		11/30/83	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Thatcher	NRR/DSI	NOTE 1		11/30/83	
76.	Instrumentation and Control Power Interactions	Colmar	NRR	NOTE 4		11/30/83	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	NRR/DSI/ASB	HIGH		11/30/83	
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Riggs	NRR	NOTE 4		11/30/83	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	NRR/DE/MEB, NRR/DSI/RSB	MEDIUM	1	12/31/84	
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		11/30/83	NA
81.	Impact of Locked Doors and Barriers on Plant Personnel and Safety	Colmar	NRR/DHFS/PSRB	DROP	1	12/31/84	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen	NRR/DSI/AEB	MEDIUM		11/30/83	
83.	Control Room Habitability	Matthews	NRR	NOTE 4		11/30/83	
84.	CE PORVs	Riggs	NRR/DSI/RSB	NOTE 1	1	6/30/85	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR	NOTE 4		11/30/83	

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DE/MTEB	NOTE 2		12/31/84	
87.	Failure of HPCI Steam Line Without Isolation	Pittman	NRR	NOTE 4		(later)	
88.	Earthquakes and Emergency Planning	Emrit	NRR	NOTE 4		(later)	
89.	Stiff Pipe Clamps	Riggs	NRR	NOTE 4		(later)	
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	NRR	NOTE 4		(later)	
92.	Fuel Crumbling During LOCA	V'Molen	NRR/DSI/RSB, CPB	LOW		12/31/84	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	NRR/DSI/ASB	HIGH		12/31/84	
94.	Additional Low Temperature Overpressure Protection Issues for Light Water Reactors	Pittman	NRR	NOTE 4		(later)	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	NRR	NOTE 4		(later)	
96.	RHR Suction Valve Testing	V'Molen	NRR	NOTE 4		(later)	
97.	PWR Reactor Cavity Uncontrolled Exposures	V'Molen	NRR/DSI/RAB	III.D.3.1		6/30/85	NA
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR/DSI/ASB	DROP		6/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	NRR	NOTE 4		(later)	
100.	OTSG Level	Pittman	NRR	NOTE 4		(later)	
101.	Break Plus Single Failure in BWR Water Level Instrumentation	V'Molen	NRR/DSI/RSB	HIGH		6/30/85	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	Emrit	NRR/DHFS/LQB	HF02		6/30/85	NA
103.	Design for Probable Maximum Precipitation	Emrit	NRR	NOTE 4		(later)	
104.	Reduction of Boron Dilution Requirements	V'Molen	NRR	NOTE 4		(later)	
105.	Interfacing Systems LOCA at BWRs	Milstead	NRR/DSI/RSB	HIGH		6/30/85	
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Colmar	NRR	NOTE 4		(later)	
107.	Generic Implications of Main Transformer Failures	Colmar	NRR	NOTE 4		(later)	
108.	BWR Suppression Pool Temperature Limits	Colmar	NRR/DSI/CSB	RI (LOW)		6/30/85	NA
109.	Reactor Vessel Closure Failure	Pittman	NRR	NOTE 4		(later)	
110.	Equipment Protective Devices on Engineered Safety Features	Pittman	NRR	NOTE 4		(later)	
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	Riggs	NRR	NOTE 4		(later)	
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	Pittman	NRR	NOTE 4		(later)	
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Riggs	NRR	NOTE 4		(later)	
114.	Seismic-Induced Relay Chatter	Pittman	NRR	NOTE 4		(later)	
115.	Reliability of Westinghouse Solid State Protection System	Milstead	NRR	NOTE 4		(later)	
116.	Accident Management	Pittman	NRR	NOTE 4		(later)	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
<u>HUMAN FACTORS ISSUES</u>							
<u>HF01</u>	<u>HUMAN FACTORS PROGRAM PLAN (HFPP)</u>						
HF01.1.0	Staffing and Qualifications	-	-	-			
HF01.1.1	NPP Staffing Requirements	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.2	NPP Personnel Qualifications Requirements	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.4	Fitness for Duty	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.2.0	Training	-	-	-			
HF01.2.1	Development of Training Regulation and Guidance	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.2.2	NRC Training Evaluation Program	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.3.0	Licensing Examination	-	-	-			
HF01.3.1	The Examination Content	Pittman	NRR/DHFS/OLB	HIGH		12/31/84	
HF01.3.2	The Examination Process	Pittman	NRR/DHFS/OLB	HIGH		12/31/84	
HF01.4.0	Procedures	-	-	-			
HF01.4.1	Procedures Guidance and Criteria	Pittman	NRR/DHFS/PSRB	HIGH		12/31/84	
HF01.5.0	Man-Machine Interface (MMI)	-	-	-			
HF01.5.1	MMI Guidance for Existing Designs	Pittman	NRR/DHFS/HFEB	HIGH		12/31/84	
HF01.5.2	MMI Guidance for Designs Based on Advanced Technologies	Pittman	NRR/DHFS/HFEB	HIGH		12/31/84	
HF01.6.0	Management and Organization	-	-	-			
HF01.6.1	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.6.2	NRC Management and Organization Guidelines and Assessment Procedures for Operating License Reviews	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.7.0	Human Reliability	-	-	-			
HF01.7.1	Human Error Data Acquisition	Pittman	RES	LI		12/31/84	
HF01.7.2	Human Error Data Storage and Retrieval	Pittman	RES	LI		12/31/84	
HF01.7.3	Reliability Evaluation Specialist Aids	Pittman	RES	LI		12/31/84	
HF01.7.4	Safety Event Analysis Results Application	Pittman	RES	LI		12/31/84	
HF02	Maintenance and Surveillance Program	Pittman	NRR/DHFS/LQB	HIGH		6/30/85	

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TABLE III

SUMMARY OF THE PRIORITIZATION OF ALL TMI ACTION PLAN ITEMS,  
TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS 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 - 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  
USI - Unresolved Safety Issue  
I - TMI Action Plan Item with Implementation  
of Resolution Mandated by NUREG-0737

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								
1. <u>TMI ACTION PLAN ITEMS (352)</u>													
(a) <u>Safety</u>													
(i) Generic Safety	94	58	6	1	93	0	12	5	13	7	2	-	291
(b) <u>Non-Safety</u>													
(i) Licensing	-	0	0	4	51	-	-	-	-	0	0	6	61
2. <u>TASK ACTION PLAN ITEMS (142)</u>													
(a) <u>Safety</u>													
(i) USI	-	-	0	1	14	12	-	-	-	-	-	-	27
(ii) Generic Safety	-	17	0	1	26	-	4	7	3	9	10	-	77
(iii) Regulatory Impact	-	0	0	0	2	-	-	-	1	0	0	1	4
(b) <u>Non-Safety</u>													
(i) Licensing	-	0	0	0	1	-	-	-	-	7	0	11	19
(ii) Environmental	-	1	0	0	6	-	-	-	-	6	0	2	15
3. <u>NEW GENERIC ISSUES (131)</u>													
(a) <u>Safety</u>													
(i) Generic Safety	1	32	3	5	13	0	8	7	5	12	38	-	124
(ii) Regulatory Impact	-	0	0	0	0	-	-	-	1	0	0	3	4
(b) <u>Non-Safety</u>													
(i) Licensing	-	0	0	0	0	-	-	-	-	0	0	3	3
4. <u>HUMAN FACTORS ISSUES (18)</u>													
(a) <u>Safety</u>													
(i) Generic Safety	-	0	0	0	0	0	14	0	0	0	0	-	14
(b) <u>Non-Safety</u>													
(i) Licensing	-	0	0	0	0	-	-	-	-	0	0	4	4
TOTAL:	95	108	9	12	206	12	38	19	23	41	50	30	643

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 new or 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	44	-
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
C404	Steam Binding of Auxiliary Feedwater Pumps	93	E325
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 <i>Corbicula</i> 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 Closing 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	-
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

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
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-Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	54	C203
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 Leakage	C-9	-
E426	Single Failure Vulnerability of Power Operated Relief Valve (PORV) Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)	94	C401
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 Intrusion at Sequoyah 1	51	-

TASK I.A.2: TRAINING AND QUALIFICATIONS OF OPERATING PERSONNEL

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

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

Under the TMI Action Plan,<sup>48</sup> the NRC may require reactor licensees to review their training and qualification programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel and supervisors. The review is to examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper safety-related staff conduct, then documentation of the justification for this determination is required. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualification practices to ensure adequate performance of operations personnel. The evaluation of this issue includes the consideration of Item I.A.2.6(3).

PRIORITY DETERMINATION

The first step in estimating the effect of training reviews on operator-error contributions to plant risk was to assemble a panel of experts from the PNL staff. This panel represented considerable experience in reactor operations, utility training programs, and reactor plant systems. The panel included members with utility field experience and reactor operator licensing examiners.

The judgments of the panel, as detailed below, are based on the two following considerations:<sup>64</sup>

- (1) The potential effect of this issue is limited by its semi-voluntary nature. That is, the judgment of adequacy is in the hands of the individual utilities. Furthermore, the current INPO and NRC research work in task analysis deals with generic routine operations. Plant-specific operation and operation under upset conditions are left to the individual utilities. This dilutes the effectiveness of the task analysis efforts in providing the basis for the training and qualification review.

Related issues which are supported by and in turn support this issue are the conducting of plant drills and accreditation of training programs.



While neither of these is directly required by the training and qualifications review, both could be a part of the response and both would have a positive effect on personnel performance.

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

### Assumptions

In estimating the benefit and costs, the PNL panel divided licensees into three groups.

- (1) Minimally affected group: These utilities currently have a good effective training and qualification program and good operations personnel performance. They should be minimally affected by this safety issue. The fractional population of this group is estimated to be 15% of the reactor licensees.
- (2) Intermediately affected group: These utilities' training and qualification programs and/or operations performance have room for improvement. This group, estimated to be 50% of the population, would undergo improvements and therefore be affected by the issue.
- (3) Maximally affected group: These utilities have deficiencies in their training and qualification programs and in operations personnel performance. They would be significantly affected by this safety issue and major restructuring of programs would be expected. This group is estimated to contain 25% of reactor licensees.

From the estimates for these groups, weighted composite estimates can be derived. NUREG/CR-2800<sup>64</sup> shows the safety benefit estimates from the panel for each of the groups and also gives the weighted averages.

The values given in NUREG/CR-2800<sup>64</sup> are in terms of percent changes. For inclusion into the value/impact score formulation they must be converted to other measures. The reduction in human error must be transformed into the resulting reduction in risk as measured by change in probabilistic exposure (man-rem/reactor-year). The change in annual occupational exposure must be transformed from percent improvement into man-rem per reactor-year.

The reduction in risk will be developed by examining the quantitative impact on accident event frequencies of human error rates in key scenarios. The reduction in human error will thereby be translated into a reduction in accident frequency. No additional reduction due to accident mitigation will be assumed. The values given in NUREG/CR-2800,<sup>64</sup> for the best estimate of improvement will be used, or 17% for operator error and 28% for maintenance.

## Frequency/Consequence Estimate

This issue centers around operator and maintenance training programs to improve personnel performance. This issue relates generically to both BWRs and PWRs, and ideally the risk reduction attributable to its resolution would be estimated by selecting a representative plant of each type. However, maintenance and operator performance impact essentially accident sequences in the risk equations. To save time, the calculations were performed for one representative PWR and inferences drawn for all reactors. The Oconee 3 (a RSSMAP PWR) plant risk equations developed in NUREG/CR-1659,<sup>54</sup> Vol. 4 (Hatch 1981) were used for this analysis.

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

To calculate the total public risk reduction it was assumed that issue resolution would apply to all plants existing and planned as given in NUREG/CR-2800, Appendix C.<sup>64</sup> This would represent a grand total of 4,000 plant-years of operation (143 plants with an average life expectancy of 28 years). Implementation of the solution would provide a reduction of 9 man-rem/plant-year. For all plants, assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the total public risk reduction totals 122,400 man-rem.

## Cost Estimate

Industry Cost: In estimating the costs to industry of implementing and operating under the resolution of this issue the PNL panel divided the industry once again into three categories. These groups and their estimates are shown in NUREG/CR-2800.<sup>64</sup> The total costs to industry for implementation is the product of the number of plants and the per-plant cost,  $(143)(\$0.335\text{M}) = \$48\text{M}$ . The total operation cost is the product of the number of plants, the average remaining life, and the plant annual cost,  $(143)(28)(\$0.16\text{M}) = \$640\text{M}$ . The overall cost to industry is the sum of the total implementation and operational cost,  $[\$640 + 48]\text{M} = \$688\text{M}$ .

NRC Cost: The cost for the NRC to implement the safety issue resolution was taken from NUREG-0560.<sup>48</sup> This called for 1.1 person-years of NRC effort which is equivalent to \$110,000. The annual NRC effort through OIE to review the justification documentation and new training programs is estimated to be one person-year. This is \$100,000 per year. Over the lifetime of the completed and planned reactors this is \$2.8M. Therefore, the overall cost to the NRC is the sum of the implementation and operation costs,  $[\$0.11 + 2.8]\text{M}$  or \$2.9M.

According to PNL estimates and calculations, the total cost for the implementation and operation of this safety issue is then [\$688M + \$2.9M] or approximately \$691M.

#### Value/Impact Assessment

The public risk reduction estimated for this issue is 122,400 man-rem. The value/impact score based on this result is

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

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

#### Other Considerations

Including the occupational dose reduction ( $2.4 \times 10^5$  man-rem) in the value/impact equation gives a score of 524 man-rem/\$M. PNL calculated<sup>64</sup> the occupational risk reduction for accident-related occupational exposure to be 880 man-rem. However, it was estimated that with improved training the operational doses could be reduced by  $2.4 \times 10^5$  man-rem for 143 plants over the average remaining plant lifetime.

#### CONCLUSION

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

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

##### DESCRIPTION

##### Historical Background

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

## Safety Significance

The principal safety benefit to be derived from NRR participation in OIE inspector training is in the improvements those inspectors will bring about because of that enhanced training. The training will increase inspector awareness in human factors and personnel-related problems. In areas such as emergency procedures reviews, routine operational practices and hardware-to-human interface deficiencies may be found by inspectors and corrected. A panel of PNL experts explored the potential significance of this issue.<sup>64</sup> This panel included three reactor operator license examiners, members with utility field experience, experience in training as well as general reactor safety experience.

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

PNL estimated<sup>64</sup> a human-error rate reduction of 2% for operators and maintenance personnel (operations staff assumed most likely to affect plant safety). It is important to note that this is an overall industry-wide estimate. Some isolated actions could be highly significant. The PNL estimated cost for this additional training is about \$1,000.

## CONCLUSION

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

## ITEM I.A.2.5: PLANT DRILLS

### DESCRIPTION

The intent of this TMI Action Plan item is to upgrade operator training by requiring operating personnel to conduct plant drills during shifts. Normal and off-normal operating maneuvers would be simulated for walk-through drills on a plant-wide basis. Drills would also be required to test the adequacy of reactor and plant operating procedures.



This is an effort to reduce the risk of off-normal operating conditions by improving the capability of operators and supervisors to understand and control complex reactor transients and accidents, and also to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions.

## PRIORITY DETERMINATION

### Assumptions

Assume that the frequency of core-melt incidents is  $5 \times 10^{-5}$ /plant-year, based on WASH-1400.<sup>16</sup> Also, assume that operator error accounts for 50% of these events, but that the plant drills will improve operator performance by 2%. In addition, assume that the release associated with core-melt is the value averaged over the probabilities of the WASH-1400<sup>16</sup> accident categories for PWRs and BWRs and weighted by the number of PWRs (95) and BWRs (48). This results in a total of  $2.4 \times 10^6$  man-rem per accident. The remaining average plant lifetime is assumed to be 28 years.

### Frequency/Consequence Estimate

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

Risk Reduction =  $(5 \times 10^{-7})(2.4 \times 10^6)(28)(143)$  man-rem = 4,805 man-rem

### Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the utility staff time for attendance at the drill, preparation by staff and management, and staff time dedicated to the dissemination of insights gained from the drills. At a cost of \$100,000/man-year and with 4.33 weeks per month, this yields a per-plant cost of \$8,333. Across the industry, i.e., 143 plants, this would be \$1.2M.

The industry resources required annually to participate in the plant drills are estimated to be two person-months per plant, which includes drill attendance, preparation before the drill, and dissemination of information afterward. This would be equivalent to \$16,660/plant-year. For the total industry (143 plants), this works out to an estimated 143 person-months/year or \$2.38M/year. Given the average remaining lifetime for the plants (28 years), this gives a total operational cost of \$67M.

The total cost to industry is then the sum of the implementation and operational costs,  $$(1.19 + 67)$ M or approximately \$68.2M.

NRC Cost: The total costs to the NRC to implement the resolution of this issue includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates so that the total implementation cost is estimated to be \$300,000. The annual cost to the NRC was also estimated to be \$300,000. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life (28 years), the operational

cost comes to \$8.4M. Therefore, the total cost to the NRC is the sum of implementation and operation costs, \$(8.4 + 0.3)M or \$8.7M.

Hence, the total costs associated with this issue are \$(68.2 + 8.7)M or \$76.9M.

#### Value/Impact Assessment

Based on a public risk reduction of 4,805 man-rem, the value/impact score is given by:

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

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

#### CONCLUSION

Based on the above value/impact score, the ranking of this issue would be low to medium. However, because the risk may have been estimated to be well on the conservative side, our judgment is that the issue of plant drills should receive a LOW priority ranking.

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

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

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

#### DESCRIPTION

##### Historical Background

Item I.A.2.6 of the TMI Action Plan<sup>48</sup> calls for the long-term upgrading of training and qualifications for operations personnel. The specific paragraphs of this item in NUREG-0660<sup>48</sup> call for a revision of "Regulatory Guide 1.8<sup>226</sup> (ANSI/ANS 3.1),"<sup>253</sup> in order to incorporate short-term requirements into this issue and any other changes resulting from a national standards effort. Also, it is stated that more explicit guidance regarding exercises in simulator requalification programs will be included in the regulatory guide (Recommendation 8 of SECY-79-330E<sup>251</sup>) as will qualifications of shift supervisors and senior reactor operators [NUREG-0585,<sup>174</sup> Recommendations 1.6(1) and (2)]. In addition, based on the NRC staff review of NRR-80-117,<sup>252</sup> recommendations will be made to the Commission and Commission decisions will be factored into the regulatory guide or regulation changes. Moreover, appropriate revisions to 10 CFR 55, Operator Licenses, are to be recommended for action by the Commission in order to incorporate the applicable short-term changes plus requirements based on Commission action on SECY-79-330E<sup>251</sup> for mandatory simulator training for applicants for licenses (Recommendation 4); mandatory simulator training in requalification programs (Recommendation 7); NRC administration of requalification examinations (Recommendation 9 as modified by the Commission); and mandatory operating tests at simulators (Recommendation 11). Finally, it is noted that the Nuclear Waste Policy Act of 1982, Public Law 97-425, Section 306



authorized and directed NRC to promulgate regulations or guidance for the training and qualifications of civilian nuclear power plant personnel. A task force has been formed within NRC as a result of this bill. As part of the task force objectives, Items I.A.2.6 (1, 2, and 3) are to be addressed.

The numerical assessment of this safety issue was conducted by the PNL staff<sup>64</sup> with experience in reactor operator licensing, reactor operation, and general reactor safety in consultation with General Physics Corporation. General Physics Corporation provides utility training services and has significant experience in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

### Safety Significance

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

### Possible Solutions

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

### PRIORITY DETERMINATION

#### Assumptions

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

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

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

For the analysis performed by PNL,<sup>64</sup> Oconee-3 is taken as the representative PWR plant. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR but the fractional risk and core-melt frequency reductions are also applied to the BWR. The dose calculations are based on a reactor site population density of 340 people per square mile and a typical mid-west meteorology is assumed.

### Frequency/Consequence Estimate

Based on the affected accident sequences and the parameters affected by this safety issue resolution (SIR), the original core-melt frequencies of  $8.2 \times 10^{-5}$  per plant-yr for PWRs and  $3.71 \times 10^{-5}$ /plant-yr for BWRs are calculated to be reduced by about 16%. The associated reduction in public risk is 31 man-rem/plant-yr for PWRs and 37.4 man-rem/plant-yr for BWRs resulting in a total public risk reduction of 132,600 man-rem.

### Cost Estimate

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

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

#### (1) Implementation of the SIR

$$(12 \text{ man-yrs/plant}) (49 + 95) \text{ plants } (\$100,000/\text{man-yr}) = \$173\text{M}$$

#### (2) Operation and Maintenance of the SIR

##### (a) Labor

$$\text{Training Staff} = \frac{(3 \text{ man-yr})}{\text{plant-yr}} \left( \frac{52 \text{ man-wk}}{\text{man-yr}} \right) = 156 \frac{\text{man-wk}}{\text{plant-yr}}$$

$$\text{Operators} = \frac{(800 \text{ man-hr})}{\text{plant-yr}} \div \frac{(40 \text{ man-hr})}{\text{man-wk}} = \frac{20 \text{ man-wk}}{\text{plant-yr}}$$

$$\text{Total Labor} = \frac{176 \text{ man-wk}}{\text{plant-yr}}$$

(b) Simulator Time (Operators)

$$\frac{(800 \text{ man-hr})}{\text{plant-yr}} \div \frac{(3 \text{ man-hr})}{\text{simulator-hr}} = \frac{267 \text{ simulator-hr}}{\text{plant-yr}}$$

The industry cost per plant-year for operation and maintenance is given by:

$$\left( \frac{176 \text{ man-wk}}{\text{plant-yr}} \right) \times \left( \frac{\$100,000/\text{man-yr}}{52 \text{ man-wk/man-yr}} \right) + \left( \frac{267 \text{ simulator-hr}}{\text{plant-yr}} \right) \left( \frac{\$600}{\text{simulator-hr}} \right) \\ = 500,000/\text{plant-year}$$

Therefore, for all affected plants, the total industry cost for operation and maintenance is given by:

$$(\$500,000/\text{plant-yr}) [(49)(27) + (95)(28.5)] \text{ plant-yr} = \$2,000\text{M}$$

The total industry cost for implementation, operation, and maintenance of the solution is then [\$173M + \$2,000M] or \$2,173M.

NRC Cost: The NRC effort to implement the resolution of this issue would be significant. It is estimated in NUREG-0660<sup>48</sup> that 5.4 man-years plus \$259,000 would be required. Some of these development activities have been completed. However, much work remains to be done. The remaining effort is estimated to be 4.5 man-years and \$100,000.

The operational activities of the NRC would include reviews of training programs, increase inspection and additional examination. The annual labor for reviews and inspections is estimated to be equivalent to 3 person-years. The principal addition in examinations is assumed to be NRC conduct of a portion of requalification examinations. It is assumed the NRC will conduct 25% of the requalification examinations and the 20 operators are requalified at each plant every year. It is estimated that one person-month is required for each plant. This assumes the five (25% of 20) operators selected for NRC examination at each plant are tested at the same time. NRC costs are estimated as follows:

(1) Implementation of the SIR

$$\text{Staff Labor + Other Costs} \\ = (1.4 \text{ man-wk/plant})(\$1,600/\text{man-wk}) + (\$100,000)/144 \text{ plants} \\ = \$3,386/\text{plant}$$

Total cost for all affected plants is (\$3,386/plant)(144 plants) or \$488,000.

(2) Review of Maintenance and Operation of SIR

$$(a) \text{ Review and Inspection} = \left(\frac{3 \text{ man-yr}}{\text{yr}}\right) \left(\frac{52 \text{ man-wk}}{\text{man-yr}}\right) / 144 \text{ plants}$$

$$= 1.08 \text{ man-wk/plant-yr}$$

$$(b) \text{ Examination} = \left(1 \frac{\text{man-month}}{\text{plant-yr}}\right) \left(\frac{3.7 \text{ man-wk}}{\text{man-month}}\right)$$

$$= 3.7 \text{ man-wk/plant-yr}$$

Total time spent is 4.78 man-wk/plant-yr.

The NRC cost per plant-yr due to review of operation and maintenance is  
 $(4.78 \text{ man-wk/plant-yr})(\$1,900/\text{man-wk}) = \$9,088/\text{plant-yr}.$

The total NRC cost for operation and maintenance of the SIR is then  
 $(\$9,088)[(49)(27) + (95)(28.5)] = (\$9,088)(4,030) = \$36.6\text{M}$

Therefore, the total industry and NRC costs are estimated to be  
 $\$[2,173 + 0.488 + 36.6]\text{M}$  or  $\$2,210\text{M}$

Value/Impact Assessment

Based on the estimated reduction in public risk of 132,600 man-rem, the value/impact score is given by:

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

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

Other Considerations

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

CONCLUSION

Although the value/impact score was low, this issue was determined to be high priority because of the large potential public risk reduction. However, with the publication of NUREG-0985, Revision 1,<sup>651</sup> Item I.A.2.6(1) is now covered in Sections 1.2 and 2.1 of the HFPP.

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

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

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

This item was evaluated in Item I.A.2.6(1) above and, as a result of the Nuclear Waste Policy Act of 1982 (Public Law 97-425), the scope of this item is now covered under Item I.A.2.2.<sup>428</sup>

ITEM I.A.2.6(4): OPERATOR WORKSHOPSDESCRIPTIONHistorical Background

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

Safety Significance

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

PRIORITY DETERMINATIONAssumptions

PNL has conducted and is conducting a series of these workshops for NRR. In the assessment of this issue, PNL staff responsible for these workshops were consulted. Their judgments form the basis of our analysis.

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

A panel of PNL experts was assembled and included staff that conduct operator licensing examinations, staff with experience in reactor operations, reactor safety and risk assessment, and the staff responsible for the conduct of the current operator feedback workshops. This panel produced the estimates that form the basis of this analysis.



The analysis is based on the following additional assumptions:

1. Applicable Plants: 95 PWRs and 48 BWRs
2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.
3. Affected Accident Sequences and Base-Case Frequencies: Most sequences are affected. The affected sequences and the base-case frequencies are shown in NUREG/CR-2800.<sup>64</sup>
4. Affected Release Categories and Base-Case Frequencies: All release categories are affected by issue resolution. The original base-case frequencies are used as given below.

<u>Oconee</u>	<u>Grand Gulf</u>
PWR-1 = $1.10 \times 10^{-7}$ /plant-yr	BWR-1 = $1.09 \times 10^{-7}$ /plant-yr
PWR-2 = $1.0 \times 10^{-5}$ /plant-yr	BWR-2 = $3.35 \times 10^{-5}$ /plant-yr
PWR-3 = $2.86 \times 10^{-5}$ /plant-yr	BWR-3 = $1.44 \times 10^{-6}$ /plant-yr

#### Frequency/Consequence Estimate

The PNL panel estimated<sup>64</sup> the most likely reduction in human error rates for operators due to the conduct of the proposed workshops would be 3%. This is assuming the workshops are conducted in the manner now perceived. That is, to focus on data gathering for the NRC. This reduces the amount of time that could be devoted to inter-licensee sharing of operational experiences which would have a more direct effect on safety-related operational performance in the plants. The possible range of reduction stretched from 1% to 10%. If the focus could be shifted toward the inter-licensee exchange of operational experiences, the most likely reduction in error rate would shift upward. However, it is not expected to exceed 10%.

Based on the PNL estimates and calculations,<sup>64</sup> and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the public risk reduction is 7,140 man-rem for 143 plants with an average existing lifespan of 28 years. The occupational dose reduction is minor at a calculated value of 46 man-rem.

#### Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the trial workshops currently being conducted. It includes utility staff time for attendance of the workshop, preparation by staff and management, and staff time dedicated to the dissemination of insights gained at



the workshop. At a cost of \$100,000/person-year and with 4.33 weeks per month, this yields a per-plant cost of \$8,333. Across the industry, i.e., 143 plants, this would be \$1.19M.

The industry resources required annually to participate in the training workshops are estimated to be the same as those for implementation. That is, one person-month per plant, which includes workshop attendance, preparation before the workshop, and dissemination of information afterward, would be needed. This would be equivalent to \$8,333/plant-year. For the total industry (143 plants), this works out to an estimated 143 person-months per year or \$1.19M per year. Given the average remaining lifetime for the plants, this gives a total operational cost of \$33.3M. Therefore, the total industry cost associated with this issue is \$34.5M.

NRC Cost: The total cost to the NRC to implement the resolution of this issue was estimated to be \$0.3M. This includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates. The annual cost to the NRC was also estimated to be \$0.3M. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life, the operational cost comes to \$8.4M. While not specific, these estimates for implementation and operation are firmly based on the experience of conducting the present trial workshops. Therefore, the total cost to the NRC is the sum of implementation and operation costs which amounts to \$8.7M.

#### Value/Impact Assessment

Based on the estimated public risk reduction of 7,140 man-rem, the value/impact score is given by:

$$S = \frac{7,140 \text{ man-rem}}{\$(34.5 + 8.7)\text{M}}$$

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

#### Other Considerations

The accident avoidance cost is the product of the change in accident frequency ( $\Delta F$ ) and the estimated cost to the utility of a major accident (A). This latter term is estimated<sup>64</sup> to be \$1.65 Billion. The cost per plant-year is then estimated to be:

$$\begin{aligned} \text{PWRs: } (\Delta F)(A) &= (7 \times 10^{-7})(\$1,650\text{M})/\text{plant-yr} = \$1,200/\text{plant-yr} \\ \text{BWRs: } (\Delta F)(A) &= (3.2 \times 10^{-7})(\$1,650\text{M})/\text{plant-yr} = \$530/\text{plant-yr} \end{aligned}$$

The total cost for all plants is the per-plant-year cost multiplied by the number of plants (N) and the average remaining lifetime (T) for each type of plant:

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

#### CONCLUSION

Because of the extensive number of sequences considered by PNL to be affected by this issue, the base-case risk is high at a calculated range of from 60 to

73 man-rem/plant-year. With a value/impact score of 165 man-rem/\$M and an estimated risk reduction of 7,140 man-rem, this issue should have a MEDIUM priority ranking.

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

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

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

##### DESCRIPTION

This TMI Action Plan item calls for NRR to develop requirements for the inclusion of nuclear power fundamentals within the instruction given to reactor operators. This arose out of a concern<sup>174</sup> that the 12 weeks of fundamentals training given to operators at that time was insufficient.

##### PRIORITY DETERMINATION

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

##### Assumptions

The panel felt there had been significant progress across the industry in the area of instruction in nuclear power fundamentals since the issuance of NUREG-0585<sup>174</sup> in 1979. Further increase in emphasis on fundamentals was felt to be unlikely to improve operator performance. The current trend in operator licensing examinations is to stress operational knowledge and de-emphasize fundamentals. This supports the view that further fundamental training would not add to plant safety.

It was assumed that, if implemented, the additional nuclear power fundamentals training would add 4 weeks to the training period. Also, it was assumed that 20 operators complete the training course each year at every plant. In addition, one full-time instructor was assumed to be required. This yields 80-person-weeks for the operators, 44 person-weeks for the instructors, or 124 person-weeks overall per plant each year. To implement this practice an effort equivalent to one year of operation (124 person-weeks) was estimated to be required.

##### Frequency/Consequence Estimate

Safety issues which deal with operator training can affect the public risk by improvements in the operator safety-related performance. This can lead to a reduction in core-melt frequency and a reduced probabilistic risk. For this safety issue the PNL panel felt that the current level of instruction in nuclear power fundamentals was adequate. Further emphasis of fundamentals was viewed as not likely to improve operator safety performance. Therefore, there

would be no measurable public risk reduction associated with the implementation of this issue. The PNL panel also saw no reduction in occupational dose associated with the implementation of the solution.

#### Cost Estimate

NRC effort to implement the solution is estimated<sup>48</sup> to be 0.4 person-year or approximately 18 person-weeks. No added costs are estimated for operation for the NRC. The review of the additional instruction could be contained in the current routine function thereby causing no added expense.

#### Value/Impact Assessment

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

#### CONCLUSION

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

### ITEM I.A.2.7: ACCREDITATION OF TRAINING INSTITUTIONS

#### DESCRIPTION

##### Historical Background

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

##### Safety Significance

There are two aspects to the safety benefit for this issue. One is the reduction of public risk through the improvement of operator performance, which is expected from the improved training accreditation. The second is a reduction in occupational exposure. This will primarily be for operators who often supervise maintenance or perform other duties in radiation zones. However, some reduction in routine occupational exposure can also be expected for other operations personnel as a result of the increased awareness by the operators.

##### Possible Solution

In order to assess this safety issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The panel envisioned the resolution of this safety issue as the formation of an accreditation board consisting of representatives from the NRC, industry, and academia. This board would develop and apply criteria for accreditation. This would include training programs of utilities, university-related programs, and independent training institutions. While theoretically applying to training for all operations staff, the PNL panel felt the current thrust was focused on reactor operators. Therefore, the assessment was made assuming only operators would be affected.<sup>64</sup>

## PRIORITY DETERMINATION

### Assumptions

The views of the panel include an awareness of the fact that some training programs are very near to accreditation already. Either through association with the universities or through other means of providing high quality instruction, these programs would be likely to acquire accreditation from the board easily. Other training programs are not so well prepared for accreditation and may require significant effort and expense to upgrade them. Some savings may be gained for multi-unit sites in sharing costs.

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

1. Applicable Plants: BWRs and PWRs - 90% of total plants; 43 BWRs, 86 PWRs, or 129 plants in all.
2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

### Frequency/Consequence Estimate

Based on the PNL analysis,<sup>64</sup> and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the anticipated public risk reduction is calculated to be 26,180 man-rem.

### Cost Estimate

The PNL panel estimated<sup>64</sup> the costs associated with implementation and operation of the resolution to this safety issue. The one-time costs to industry to implement the change initially was estimated to be in the range of \$0.1M to \$1M per

reactor. Those with training programs closer to accreditable status would enjoy the smaller costs. The best estimate for the average plant was taken to be \$0.3M. Operation under the accreditation program was estimated to cost between \$0.05M and \$0.25M per plant annually for additional funding to maintain an accredited training program. The best estimate was \$0.1M per plant annually.

The cost to the NRC to implement the accreditation was estimated to be \$0.635M which is equivalent to 330 person-weeks. The annual operational cost to the NRC is estimated<sup>64</sup> to be \$100,000 or one person-year.

The detailed breakdown of these costs are as follows:

\$300,000/Plant Industry Implementation (approximately 3 man-yrs):

- to review accreditation standards
- to compare the present utility practices with the developed standards
- plan the necessary upgrades
- implement the program upgrades to fulfill the accreditation requirements.

\$100,000/Plant-yr Industry Operation and Maintenance:

- time invested by the staff in upgraded training (increased course time, quality, etc.)
- instruction upgrade (time, quality, etc.)

\$500,000 NRC Implementation (approximately 5 man-yrs)

- predicated on the possibility that INPO accreditation will not be forthcoming; NRC may have to do
- NRC to develop accreditation standards, regulations, and implement to adoption by the industry.

\$100,000 NRC Operation and Maintenance (approximately 1 man-yr/yr)

- additional OIE efforts to assure industry maintenance of standards (all plants).

The total costs for this safety issue are, therefore, estimated<sup>64</sup> by PNL as follows:

1.	Implementation of the Safety Issue Resolution (SIR) by industry	\$ 39,000,000
2.	Operation and Maintenance of the SIR by the industry	360,000,000
3.	NRC Implementation of the SIR	635,000
4.	NRC Operation and Maintenance of SIR	2,800,000
	Total:	<u>\$402,435,000</u>



Value/Impact Assessment

Based on the estimated public risk reduction of 26,180 man-rem, the value/impact score is given by:

$$S = \frac{26,180 \text{ man-rem}}{\$402.4\text{M}}$$

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

Other Considerations

The industry accident avoidance cost was estimated by PNL<sup>64</sup> to be \$14M. The occupational risk reduction is estimated to be 22,170 man-rem resulting from accident avoidance (170 man-rem) and from operation and maintenance of the safety issue resolution (22,000 man-rem).

CONCLUSION

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

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438. Memorandum to Office Directors from W. Dircks, "NRC Actions Required by Enactment of the Nuclear Waste Policy Act of 1982," January 19, 1983.
651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.
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TASK I.A.3: LICENSING AND REQUALIFICATION OF OPERATING PERSONNEL

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

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

This NUREG-0660<sup>48</sup> item called for NRR to notify all operator license holders and applicants of the new scope of examinations and criteria for issuance of reactor operator (RO) and senior reactor operator (SRO) licenses and renewal of licenses. Simulator examinations were to be included as part of the license examination. Clarifications to this item were issued in NUREG-0737.<sup>98</sup>

CONCLUSION

This item was resolved and requirements were issued. However, as a result of P.L. 97-425, it was determined that additional staff work on the issue was required and a proposed rule for operator licensing was presented to the Commission in SECY-84-76.<sup>593</sup> Approval of this rule would effectively close out this item.

ITEM I.A.3.2: OPERATOR LICENSING PROGRAM CHANGESDESCRIPTION

This TMI Action Plan item<sup>48</sup> called for NRR to take the following actions:

- (1) Develop and implement a plan to relocate Operator Licensing Branch (OLB) examiners at Nuclear Power Plant Simulator Training Centers or in Inspection and Enforcement Regions.
- (2) Conduct a study of the staffing of the operator licensing program and the qualifications and training of examiners.
- (3) Develop and implement a plan to report operator errors and to act on operator errors with respect to continuation of licensing.

As a result of the above actions, the following accomplishments were made:

- (1) "The administering of examinations and issuance/renewal of operator licensing will be transferred to Region III in FY 1982 and to Region II in FY 1983. All regions will have operator licensing authority in FY 1984. NRR will provide oversight and guidance, including examination procedures and criteria."<sup>88</sup>

- (2) A study of the staffing of the operator licensing program and the qualifications and training of examiners was completed in November 1980 and documented in NUREG/CR-1750.<sup>89</sup>
- (3) A plan for reporting operator errors and for acting on operator errors with respect to continuation of licensing was developed in NUREG/CR-1750.<sup>89</sup> However, after review of this recommended plan, DHFS concluded that no further action was required.<sup>440</sup>

### CONCLUSION

This item has been RESOLVED and no new requirements were established.

### ITEM I.A.3.3: REQUIREMENTS FOR OPERATOR FITNESS

#### DESCRIPTION

##### Historical Background

This safety issue as described in NUREG-0660<sup>48</sup> calls for the NRC to develop a regulatory approach to: (1) provide assurance that applicants for RO and SRO licenses are psychologically fit, and (2) prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds. The regulations will be applied to all current and future operating power plants.

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

This issue was assessed by PNL<sup>64</sup> in consultation with a number of engineers who have expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

##### Safety Significance

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

### Possible Solutions

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

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

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

### PRIORITY DETERMINATION

#### Assumptions

The major result of this safety issue was assumed to be a reduction in operator error. For some utilities, this new system may result in some reduction in operator error whereas in others the system it may have no discernible effect. Based on engineering judgment, an average of about 2% was arrived at by PNL to apply to all currently operating and future plants. Thus, this issue assumes the implementation of the access authorization system at all 134 plants either under construction (63) or already in operation (71), with average lifetimes of 28.8 yrs for 90 PWRs and 27.4 yrs for 44 BWRs. Thus, the total remaining life of the affected plants is  $[(28.8)(90) + (27.4)(44)]RY$  or 3,798 RY.

Neither the implementation, operation, or maintenance of this SIR would involve any changes in occupational dose accrued by any personnel.

For the analysis performed by PNL,<sup>64</sup> Oconee 3 is taken as the representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

### Frequency/Consequence Estimate

All release categories are affected by this safety issue but the principal release categories affected by the SIR are 3, 5, and 7. The numerical calculations are based on these categories. The dose calculations are based on a reactor site population density of 340 people per square mile and a typical midwest meteorology is assumed.

The calculated reduction in core-melt frequencies are  $4 \times 10^{-7}/\text{RY}$  for PWRs and  $1.8 \times 10^{-7}/\text{RY}$  for BWRs. Based on this, the total estimated public risk reduction is 16,000 man-rem. The occupational risk reduction for implementation, operation, and maintenance is zero.

### Cost Estimate

Industry Cost: A value/impact analysis in support of the anticipated rule of access authorization has been prepared by the NRC staff and cost estimates for industry have been developed. These cost estimates, which have been reviewed and accepted by AIF, are as follows:

- (1) For all existing plants, the implementation cost is \$140,000/plant and includes the preparation of the plant and associated procedures (\$33,000), licensee management and clerical staff (\$63,000), training to implement the behavioral observation program (\$34,000), and storage for files (\$10,000). The total industry implementation cost for existing plants is  $\$(140,000)(71) = \$9.94\text{M}$ .
- (2) For all future plants (in which none of the employees will be grandfathered), the implementation costs are estimated to be \$590,000 per plant. In addition to the costs noted above for existing plants, this implementation includes the cost of background investigations (\$375,000), review process and appeals procedures (\$36,000), increased file storage requirements (\$30,000), and miscellaneous criminal checks with the FBI, etc. (\$9,000). The total industry cost for future plants is  $\$(590,000)(63) = \$37.2\text{M}$ .
- (3) The cost of operation of the access authorization system at each plant is estimated to be \$300,000/year. This operating cost includes background investigations for new people as a result of employee turnover (\$94,000), professional management and clerical staff (\$63,000), review and appeal process (\$67,000), refresher training for old supervisors (\$19,000), training of new supervisors (\$9,000), plan maintenance and updates (\$8,000), file storage (\$39,000), and criminal history checks with the FBI for new people (\$2,000). The total industry cost for operation and maintenance of the access authorization system is  $\$(0.3\text{M}/\text{RY})(3,798 \text{ RY})$  or \$1,140M.

The total industry cost for the SIR is  $\$[1,140 + 9.94 + 37.2]\text{M}$  or \$1,187M.

NRC Cost: The NRC costs for the SIR are estimated as follows:

- (1) The NRC time for further development and issuance of the proposed plan is estimated to be 1.5 man-years. At a rate of \$100,000/man-year, the estimated cost for this effort is \$150,000.



- (2) For implementation of the plan, which includes the review and modification of the utilities' plans, the NRC effort was estimated to be 1.5 man-years. For the 134 affected plants, this amounts to 0.6 man-week/plant. At a cost of \$2,270/man-week, the NRC implementation cost is \$182,500.
- (3) NRC review of the operation and maintenance of the SIR is estimated to require 1 man-week/RV for all plants. At a cost of \$2,270/man-week, the total NRC cost for operation and maintenance of the SIR is \$8.6M.

The total NRC cost for the SIR is  $[\$0.15 + 0.1825 + 8.6]M = \$8.9M$ .

#### Value/Impact Assessment

Based on a public risk reduction of 16,000 man-rem, the value/impact score is given by:

$$S = \frac{16,000 \text{ man-rem}}{\$(1,187 + 8.9)M}$$

$$= 13.4 \text{ man-rem}/\$M$$

#### Other Considerations

It has been estimated by cognizant personnel at the NRC that the Fitness for Duty Rule will have a negative cost impact on operating licensees in the long run. The NRC estimates that initial licensee burden to develop written procedures required by the rule will be approximately 1,200 man-hours over a six-month period at a total cost between \$50,000 and \$75,000, if no fitness for duty program exists at the licensee's facility. While utilities such as TVA claim that alcohol abuse alone costs them approximately \$18.5M annually, fitness for duty programs of the type envisioned by the Fitness for Duty Rule are expected to save costs through quicker identification of employees not fit for duty and through assisting these employees, in whom considerable resources have been invested, so that they might return to high levels of productivity. Absenteeism due to alcohol-drug abuse costs U.S. industry an average of \$300 annually for every worker nationwide. Alcohol drug-abusers lose an additional 25% of their productive time when on the job, at an average annual cost to U.S. industry of approximately \$2,900 per abuser. The total annual cost to U.S. industry is between \$12 billion to \$15 billion. Wrich, in "The Employee Assistance Program; Updated for the 1980's," Hazelden, 1980, reports that U.S. industry receives a return of \$10 in decreased absenteeism, accidents, and increased productivity for every dollar it spends on fitness for duty.

#### CONCLUSION

Although the estimated risk reduction was 16,000 man-rem and the value/impact score only 13.4 man-rem/\$M, this issue was given a high priority because of its advanced state of completion. However, with the publication of NUREG-0985, Revision 1, <sup>651</sup> this item is now covered in Section 1.4 of the HFPP.



ITEM I.A.3.4: LICENSING OF ADDITIONAL OPERATIONS PERSONNELDESCRIPTIONHistorical Background

This TMI Action Plan item<sup>48</sup> seeks to upgrade the operations performance in nuclear power plants by imposing licensing requirements upon other operations personnel in addition to ROs and SROs.

Safety Significance

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

Possible Solution

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

PRIORITY DETERMINATIONAssumptions

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

Frequency Estimate

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

Consequence Estimate

The corresponding reduction in public risk for Oconee 3 was calculated to be 2.4 man-rem/RY and the public risk reduction for Grand Gulf 1 was calculated to be 2.7 man-rem/RY.

The risk reduction for each type of plant is given as follows:

$$\begin{aligned}\text{PWRs: } (28.5 \text{ yrs})(95 \text{ reactors})(2.4 \text{ man-rem/RY}) &= 6.5 \times 10^3 \text{ man-rem} \\ \text{BWRs: } (27 \text{ yrs})(49 \text{ reactors})(2.7 \text{ man-rem/RY}) &= 3.6 \times 10^3 \text{ man-rem}\end{aligned}$$

Therefore, the total risk reduction for this issue is  $1.01 \times 10^4$  man-rem.

#### Cost Estimate

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

NRC Cost: To implement this requirement, the NRC would have to prepare qualification criteria, licensing exams, and procedures. This would be a major undertaking. The NRC costs for implementation were estimated to be in the range of \$20M to \$50M. For analysis purposes, \$35M was used. To operate with the new licensing requirements, it was estimated that the NRC would need 50 additional staff members at a total cost of \$5M/year. To perform the annual operational needs of the program, funds would be needed for travel, publications, etc. This was estimated to be an additional \$2M/year. Therefore, the total NRC cost is approximately \$240M.

#### Value/Impact Assessment

Based on a total public risk reduction of 10,100 man-rem, the value/impact score is given by:

$$\begin{aligned}S &= \frac{10,100 \text{ man-rem}}{(\$240 + 250)\text{M}} \\ &= 20 \text{ man-rem}/\$M\end{aligned}$$

#### Uncertainty

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

#### Other Considerations

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

#### CONCLUSION

Although the value/impact score was low, the potential for risk reduction was considered and this issue was given a medium priority. However, in February 1985, the staff determined that there was insufficient evidence to support the licensing of additional plant personnel.<sup>778</sup> Thus, this item was RESOLVED and no new requirements were established.

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

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

CONCLUSION

A Memorandum of Agreement<sup>148</sup> between INPO and NRC was issued in April, 1982. However, it did not specifically address training and certification. Following this, the EDO agreed with a revision<sup>594</sup> of Appendix Four to the Memorandum of Agreement (Coordination Plan for NRC/INPO Training-Related Activities) in November 1983. As a result, this Licensing Issue has been resolved.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.
64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.
88. Memorandum for All Employees from H. Denton, "Regionalization of Selected NRR Functions," June 15, 1982.
89. NUREG/CR-1750, "Analysis, Conclusions, and Recommendations Concerning Operator Licensing," U.S. Nuclear Regulatory Commission, January 1981.
651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.
148. "Memorandum of Agreement Between the Institute of Nuclear Power Operations and the U.S. Nuclear Regulatory Commission," Rev. 1, April 1, 1982.
181. SECY-82-155, "Public Law 96-295, Section 307(B), Study of the Feasibility and Value of Licensing Nuclear Plant Managers and Senior Licensee Officers," April 12, 1982.
289. NUREG/CR-2075, "Standards for Psychological Assessment of Nuclear Facility Personnel," U.S. Nuclear Regulatory Commission, July 1981.
290. NUREG/CR-2076, "Behavioral Reliability Program for the Nuclear Industry," U.S. Nuclear Regulatory Commission, July 1981.

- 440. Memorandum for W. Minners from D. Ziemann, "Schedules for Resolving and Completing Generic Issues," April 5, 1983.
- 593. SECY-84-76, "Proposed Rulemaking for Operator Licensing and for Training and Qualifications of Civilian Nuclear Power Plant Personnel," February 13, 1984.
- 594. Letter to E. Wilkinson (INPO) from W. Dircks (NRC), November 23, 1983.
- 651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.
- 778. Memorandum for W. Dircks from H. Denton, "TMI Action Item I.A.3.4," February 12, 1985.

TASK III.A.3: IMPROVING NRC EMERGENCY PREPAREDNESS

The objective of this task is to enable NRC, in the event of a nuclear accident at a licensed reactor facility, to: (1) monitor and evaluate the situation and potential hazards, (2) advise the licensee's operating staff as needed, and (3) in an extreme case, be able to issue orders governing such operations.

ITEM III.A.3.1: NRC ROLE IN RESPONDING TO NUCLEAR EMERGENCIES

The five parts of this item have been combined and evaluated together.

DESCRIPTION

This TMI Action Plan<sup>48</sup> item was to define the NRC role in emergency situations involving NRC licensees. The definition of the NRC emergency response role will be used by OIE in revising and upgrading plans and procedures for the NRC emergency operations center. OIE, with input from other NRC offices, will revise NRC Manual Chapter 0502 and NUREG-0610<sup>488</sup> to describe and implement the NRC emergency response program.

NUREG-0610<sup>488</sup> was revised as Appendix I to NUREG-0654<sup>224</sup> in November 1980. NUREG-0728,<sup>257</sup> published in September of 1980, provided the basis for continued upgrading of the NRC Incident Response Program and information to be included in the revised NRC Manual Chapter 0502. In the interim, until NRC Manual Chapter 0502 was approved by the Commission, NUREG-0845<sup>258</sup> written for trial use in March 1982, provided detailed procedures for the NRC Incident Response Plan. When the Commission approves the proposed revisions to NRC Manual Chapter 0502, NUREG-0845<sup>258</sup> will be issued for final publication.

The proposed revision to NRC Manual Chapter 0502 was approved by the Commission on January 5, 1983. Resolution of Item III.A.3.1 also resolved Item B-71, "Incident Response," which was essentially superseded by Item III.A.3.1. All required action on this item has been completed.<sup>408,548</sup>

CONCLUSION

This item has been RESOLVED.

ITEM III.A.3.1(1): DEFINE NRC ROLE IN EMERGENCY SITUATIONS

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.1(2): REVISE AND UPGRADE PLANS AND PROCEDURES FOR THE NRC EMERGENCY OPERATIONS CENTER

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.



ITEM III.A.3.1(3): REVISE MANUAL CHAPTER 0502, OTHER AGENCY PROCEDURES, AND NUREG-0610

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.1(4): PREPARE COMMISSION PAPER

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.1(5): REVISE IMPLEMENTING PROCEDURES AND INSTRUCTIONS FOR REGIONAL OFFICES

This item was evaluated in Item III.A.3.1 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.2: IMPROVE OPERATIONS CENTERS

DESCRIPTION

This TMI Action Plan<sup>48</sup> item called for the NRC Operations Center (OC) in Bethesda, Maryland to be upgraded to support activities in response to a major accident.

Near-term improvements<sup>235</sup> made to the OC during 1979-1981 included improved physical space, rearrangement, better facilities (such as status systems and weather information), and improved telecommunications equipment including the possible use of HF radios. A study has recently been completed on long-term improvements in the OC. This study addressed a complete redesign of the OC taking into account specifically human factors considerations and improved communications.

OIE considers implementation of this item important and is working toward its completion. Taking into account the problems in logistics of construction relocation, equipment purchase, and budget constraints, implementation should be completed by December 1983.<sup>235,379</sup>

CONCLUSION

This item was RESOLVED and no new requirements were established.

ITEM III.A.3.3: COMMUNICATIONS

Both parts of this item have been combined and evaluated together.

DESCRIPTION

The TMI Action Plan<sup>48</sup> included communications in the required improvements for NRC emergency preparedness. Included in communications are two items:



(1) direct and dedicated telephone lines (OPX) between the licensee facilities and NRC, and (2) the use of the dedicated short-range radio communication system (FIRS).

OPX and HPN telephone systems were installed at all operating reactors by August 1980 and are being installed at newer plants prior to operation. FIRS has been obtained for use by NRC field personnel during emergencies. All required action on this item has been completed (see References 235, 248, 379, and 406).

### CONCLUSION

This item has been RESOLVED.

### ITEM III.A.3.3(1): INSTALL DIRECT DEDICATED TELEPHONE LINES

This item was evaluated in Item III.A.3.3 above and was determined to be RESOLVED. New requirements were established.

### ITEM III.A.3.3(2): OBTAIN DEDICATED, SHORT-RANGE RADIO COMMUNICATION SYSTEMS

This item was evaluated in Item III.A.3.3 above and was determined to be RESOLVED. New requirements were established.

### ITEM III.A.3.4: NUCLEAR DATA LINK

#### DESCRIPTION

#### Historical Background

After the TMI event, the NRC concluded that the NRC Operations Center (Incident Response Center) should be upgraded to allow NRC personnel to analyze and evaluate plant conditions based on directly transmitted information, as opposed to a voice link. The term "Nuclear Data Link" (NDL) was given to a conceptual system that would access plant data and directly transmit the information to the OC.

#### Safety Significance

It was believed that, with more current and reliable plant data available to the NRC, the staff could help develop and evaluate accident mitigating actions.

#### Possible Solution

It was determined that a phased approach would be utilized. The first phase was to have Sandia study the available options and to report their findings. Sandia completed their report which will not be published. The Sandia options were evaluated and it was determined that an elaborate NDL configuration which was interactive with the licensees' system was inappropriate to the NRC's

role. The second phase is to be implementation of a prototype which will be evaluated to help the Commission decide whether an NDL is needed and, if so, what it should look like.<sup>240</sup>

## PRIORITY DETERMINATION

### Assumptions

PNL did an assessment of this issue.<sup>64</sup> To assess the impact of this issue, we needed to consider all the other related issues which are involved with the OC. (See Items III.A.3.1, III.A.3.2, III.A.3.3, III.A.3.5, and III.A.3.6.) Many of these issues have been completed or almost completed. In addition, we considered the utilities' emergency response facilities (ERFs). The ERFs are planned to be completed [along with the Safety Parameter Display System (SPDS) and the Data Acquisition System (DAS)] according to requirements outlined in SECY-82-111<sup>151</sup> and a letter<sup>376</sup> issued to all licensees of operating reactors.

### Frequency/Consequence Estimate

We constructed an event tree which assumed a base case core-melt based on the Oconee and Grand Gulf risk studies. We then analyzed certain event tree branches based on a risk reduction with results from the possibility that NRC personnel at the OC could: (1) detect and correct an error by the plant operators during accident recovery, or (2) provide optimum approaches to the operators for the mitigation of particular evolving sequences.

It was first assumed that the base case core-melt frequencies are  $8.15 \times 10^{-5}/RY$  for PWRs and  $3.67 \times 10^{-5}/RY$  for BWRs. We assumed that 90% of the core-melt scenarios would proceed slowly enough to allow input from observers at the OC or ERFs. Next, it was assumed that the operator's judgment was not optimum in about 50% of the cases. This includes consideration of the fact that he is not able to take a step back and completely evaluate the accident sequence or evaluate and/or anticipate ahead in the scenario. We then assumed that, given the above, the utilities' ERFs would be manned and available in 90% of the cases and that the utilities' ERF personnel could provide successful input in 75% of the cases.

Of the remaining 25% of the cases, we assumed that the OC would be available 90% of the time and that the NRC personnel could provide the successful input about 50% of the time. This number was assumed smaller than the utility's ERF success rate because of the data available at the OC, i.e., it is not complete and available only by voice communications. This would somewhat hinder the NRC staff's performance.

For this calculation, we ignored the smaller contribution of the event tree branch which is due to the 10% unavailability of the utility's ERF and the success of the OC staff.

Therefore, with the assumption that Items III.A.3.1, III.A.3.2, III.A.3.3, III.A.3.5, and III.A.3.6 are completed and the ERFs are in place, we estimated a potential core-melt frequency reduction for the present OC of about 4.5%.

This was then considered the base-case value for the overall OC as it is completed to date. We then estimated that the incorporation of an NDL could improve the success of the OC staff by about 50% due to the availability of more complete, more accurate, and more timely information. This would then equal an additional core-melt frequency reduction of about 2%.

From the reduction in core-melt frequency, the per plant reduction in public risk was then calculated (based on a population density of 340 people per square mile) to be 4.5 man-rem/Ry for PWRs and 5.5 man-rem/Ry for BWRs. With 95 PWRs, 49 BWRs, and an average remaining life of 28.5 years for PWRs and 22 years for BWRs, the total public risk reduction is then 18,000 man-rem.

#### Cost Estimate

Industry Cost: Licensees are not implementing standard data sets, formats, or equipment and the NRC will have to electronically process each of the data outputs that it receives from licensees. Relatively simple equipment at each site, costing perhaps \$20,000 for hardware and \$15,000 for labor to install, will transmit data in the licensee format to the NRC. There are 50 sites with operating reactors (counting Indian Point as two sites because of the mixed ownership) and 35 additional sites with reactors under construction. New reactors at six existing sites might also be built with new (separate) DAS. Rounding off to be conservative, an estimated 100 sites will require data-transmitting equipment at a total initial cost of \$3.5M.

NRC Cost: It would be expected that the NRC would incur the majority of the cost of the overall data link. It was assumed that the OC will have been improved (Item III.A.3.2) before the NDL is implemented. With respect to NRC equipment costs, it was assumed that the ERF at individual utilities would be completed. Based on this, the DAS necessary for support of the facility will already be implemented.

At the OC, the NRC will need a unit for receiving and processing the data. The unit may cost up to \$500,000 and software as much as \$30,000 for each site, since processing instructions will be different for each different licensee output. Therefore, the estimated initial cost at the OC is \$3.5M. System maintenance is estimated at 2% of equipment costs per year for 30 years, or \$1.5M.

The total estimated NDL system cost, regardless of who pays it, is \$8.5M for concepts currently envisioned. The planned Prototype Program will develop more refined evaluations and cost estimates to permit the Commission to decide what is really needed.

#### Value/Impact Assessment

Based on the estimated public risk reduction of 18,000 man-rem, the value/impact score is given by

$$S = \frac{18,000 \text{ man-rem}}{\$8.5\text{M}}$$

$$= 2,100 \text{ man-rem}/\$M.$$

Other Considerations

- (1) Present plans are to implement a prototype system.<sup>254,255</sup>
- (2) More accurate cost estimates are difficult without clearer system definition which is to be provided by evaluation of the prototypes.
- (3) The estimate of the potential reduction in core-melt frequency is subject to large uncertainty because of the sequences of assumptions which went into the event tree.
- (4) OIE believes that this issue should receive high priority.

CONCLUSION

Based on the value/impact score and the total risk reduction potential, this issue was given a MEDIUM priority ranking. However, in June 1985, it was determined by the staff that the design that met NRC requirements was one that utilized electronic data transmission systems that were already being developed by licensees for their own ERFs. This concept, Emergency Response Data System (ERDS), was approved by the Commission in March 1985.<sup>779</sup> Licensees will not be required to backfit their systems to include additional parameters to provide data on NRC's parameter list. Data that is not available from the electronic data stream can be provided by voice over existing phone lines. Thus, this item was RESOLVED and no new requirements were established.

ITEM III.A.3.5: TRAINING, DRILLS, AND TESTSDESCRIPTION

The TMI Action Plan<sup>48</sup> identified a need to improve the capability to respond to emergencies by continuing the headquarters and regional drills and exercises. The scope is envisioned to be slowly expanded to include joint exercises with State and local agencies and other Federal response capabilities. A schedule involving various levels of participation by the various parties is to be prepared.

Exercises, scheduling, and training are being conducted with gradually increasing scope and continuing programs related to this item have been incorporated into routine ongoing NRC operations. (See References 235, 248, 379, and 406.)

CONCLUSION

This item was RESOLVED and no new requirements were established.

ITEM III.A.3.6: INTERACTION OF NRC AND OTHER AGENCIES

The three parts of this item have been combined and evaluated together.

DESCRIPTION

The TMI Action Plan<sup>48</sup> identified the requirement to establish interaction agreements between NRC and other agencies for cooperation, communication, and assistance during emergency situations. Agencies involved include other international governments, i.e., Mexico and Canada, other Federal agencies, and State and local governmental bodies.

In September 1980, the NRC published NUREG-0728<sup>257</sup> which described in general the NRC's responsibilities and plans for responding to emergencies at nuclear power reactors. This report further described the coordination/liaison with other agencies and organizations. In March 1982, the NRC published NUREG-0845,<sup>258</sup> which contains detailed agency procedures for the NRC incident response plan. It also includes the details for providing the interaction between NRC and other involved Federal agencies and other organizations.

All work required by this item has been completed and the NRC Incident Response Plan is being implemented.<sup>235,256,379</sup>

CONCLUSION

This item has been RESOLVED with changes in the NRC procedures that address the interaction with other agencies during emergency situations.

ITEM III.A.3.6(1): INTERNATIONAL

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.6(2): FEDERAL

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

ITEM III.A.3.6(3): STATE AND LOCAL

This item was evaluated in Item III.A.3.6 above and was determined to be RESOLVED. No new requirements were established.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.
64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.
151. SECY-82-111, "Requirements for Emergency Response Capability," March 11, 1982.



- 224. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1978.
- 235. Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982.
- 240. SECY-81-153, "Nuclear Data Link," March 11, 1981.
- 248. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Items," December 28, 1981.
- 254. Memorandum to N. Palladino from M. Udall, Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives, June 4, 1982.
- 255. Memorandum to M. Udall, Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives, from N. Palladino, June 30, 1982.
- 256. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Items," June 2, 1982.
- 257. NUREG-0728, "Report to Congress-NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, September 1980.
- 258. NUREG-0845, "Agency Procedure for the NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, March 1982.
- 376. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.
- 379. Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983.
- 406. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Status Report," March 14, 1982.
- 408. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Item," May 11, 1982.
- 488. NUREG-0610, "Draft Emergency Action Level Guidelines for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1979.
- 548. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan Completed Items," January 26, 1983.
- 779. Memorandum for W. Dircks from J. Taylor, "TMI Action Plan - Completed Item," June 26, 1985.



ITEM A-1: WATER HAMMERDESCRIPTION

The issue was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping, emergency core cooling systems, RHR systems, containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage has been relatively minor and involved pipe hangers and restraints. However, there have been several incidents which have resulted in piping and valve damage. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

No water hammer incident has resulted in the release of radioactivity outside of plants. However, because of the continuing incidence of water hammer events, the number of phenomena, and the potential safety significance of the systems involved, the staff believed that systematic review procedures should be developed to ensure that water hammer is given appropriate consideration in CP and OL reviews and in the review of operating reactors.

CONCLUSION

This USI was RESOLVED<sup>60</sup> on March 15, 1984 with the publication of NUREG-0927, Rev. 1<sup>698</sup> and the following SRP<sup>11</sup> Sections: 3.9.3, Rev. 1; 3.9.4, Rev. 2; 5.4.6, Rev. 3; 5.4.7, Rev. 3; 6.3, Rev. 2; 9.2.1, Rev. 3; 9.2.2, Rev. 2; 10.3, Rev. 3; and 10.4.7, Rev. 3. The revised SRP Sections will be used only for the review of "custom plant" CP applications and for standard plant applications docketed after the issuance of these revised SRP Sections (which are intended for referencing in CP applications). Thus, this USI affects all future plants only.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
698. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

ITEM A-2: ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMSDESCRIPTION

On May 7, 1975, the NRC was informed by VEPCO that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by W or S&W in the original design of the reactor vessel support systems for North Anna Units 1 and 2. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

In a postulated event at the vessel nozzle, asymmetric LOCA loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more-detailed analytical models, it became apparent to W that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports, thereby affecting their integrity. This issue was determined by the NRC to have generic implications for all PWRs.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in January 1981 with the publication of NUREG-0609<sup>699</sup> and affected all operating and future PWRs. For operating PWRs, MPA D-10 was established by DL for implementation purposes. Generic Letter 84-04<sup>700</sup> was also issued by the staff.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
699. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," U.S. Nuclear Regulatory Commission, January 1981.
700. NRC Letter to all Operating PWR Licensees, Construction Permit Holders, and Applicants for Construction Permits, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04)," February 1, 1984.

ITEM A-6: MARK I SHORT-TERM PROGRAMDESCRIPTION

During the conduct of a large scale testing program for an advanced design BWR pressure suppression containment system (MARK III), new suppression pool hydrodynamic loads associated with a postulated LOCA were identified which had not been explicitly included in the original design of the MARK I containment systems. These additional loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool (Torus) during a postulated LOCA event. Consequently, it was determined that a reassessment of the MARK I containment system design would be required. This item was originally identified in NUREG 0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in December 1977 with the publication of NUREG-0408.<sup>701</sup> All plant-unique analyses and required equipment modifications were reviewed and accepted by the staff and appropriate TS changes were made by the affected licensees.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission.
701. NUREG-0408, "MARK I Containment Short-Term Program Safety Evaluation Report," U.S. Nuclear Regulatory Commission, December 1977.

ITEM A-7: MARK I LONG-TERM PROGRAMDESCRIPTION

During testing for an advanced BWR containment system design (MARK III), suppression pool hydrodynamic loads were identified which had not been considered in the original design of the MARK I containment system. To address this issue, a MARK I Owners Group was formed and the assessment was divided into a short-term and long-term program. The results of the NRC staff's review of the MARK I Containment Short-Term Program are described in NUREG-0408.<sup>701</sup> The long-term program was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria, such that a comprehensive reassessment of each MARK I containment system would be performed. A series of experimental and analytical programs were conducted by the MARK I Owners Group to provide the necessary bases for the generic load definition and structural assessment techniques. The generic methods proposed by the MARK I Owners Group, as modified by the NRC staff's requirements, will be used to perform plant-unique analyses, which will identify the plant modifications, if any, that will be needed to restore the originally intended margin of safety in the MARK I containment designs. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in August 1982 with the issuance of Supplement 1 to NUREG-0661<sup>702</sup> and SRP<sup>11</sup> Section 6.2.1.1C. For operating BWRs, MPA D-01 was established by DL for implementation purposes.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
701. NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report," U.S. Nuclear Regulatory Commission, December 1977.
702. NUREG-0661, Supplement 1, "Safety Evaluation Report for the MARK I Containment Long-Term Program," U.S. Nuclear Regulatory Commission, August 1982.

ITEM A-8: MARK II CONTAINMENT POOL DYNAMIC LOADS LONG-TERM PROGRAMDESCRIPTION

As a result of the GE testing program for the MARK III pressure-suppression containment program, new containment loads associated with a postulated LOCA were identified in 1975 which had not been explicitly included in the original design of MARK I and MARK II containments. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. Other pool dynamic loads previously unaccounted for result from the actuation of SRVs in the MARK II containment. The review and evaluation of the MARK I loads were addressed in USI A-7 and SRV loads for all suppression-type containments were addressed in USI A-39. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in August 1981 with the issuance of NUREG-0808<sup>703</sup> and SRP<sup>11</sup> Section 6.2.1.1C.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
703. NUREG-0808, "MARK II Containment Program Evaluation and Acceptance Criteria," U.S. Nuclear Regulatory Commission, August 1981.

ITEM A-9: ATWSDESCRIPTION

The technical report on ATWS for water-cooled reactors (WASH-1270)<sup>751</sup> discussed the probability of an ATWS event as well as an appropriate safety objective for the event. After several years of discussions with vendors and evaluations of vendor models and analyses, the staff published in 1975 a status report on each vendor analysis. This report included detailed guidelines on analysis models and ATWS safety objectives. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

The staff's technical findings on the issue were published in Volume 4 of NUREG-0460.<sup>704</sup> The USI was RESOLVED<sup>60</sup> on June 26, 1984 with the publication of a final rule.<sup>724,725</sup>

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, March 1980.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
704. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," U.S. Nuclear Regulatory Commission, March 1980.
724. Memorandum for W. Dircks, et al., from S. Chilk, "Staff Requirements - Affirmation/Discussion and Vote, 11:30 a.m., Friday, June 1, 1984, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," June 1, 1984.
725. Federal Register, Vol. 49, No. 124, pp. 26036-26045, "10 CFR Part 50, Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," June 26, 1984.
751. WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, September 1973.



ITEM A-10: BWR FEEDWATER NOZZLE CRACKINGDESCRIPTION

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. Most of these BWRs contained 4 nozzles with diameters ranging from 10 in. to 12 in. Although most cracks ranged from 1/2 in. to 3/4 in. in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 1.5 in. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This issue was RESOLVED<sup>60</sup> in November 1980 with the issuance of NUREG-0619.<sup>749</sup> MPA B-25 was established by DL for implementation purposes.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
742. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, November 1980.

ITEM A-11: REACTOR VESSEL MATERIALS TOUGHNESSDESCRIPTION

Because of the remote possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code, the design of nuclear facilities does not provide protection against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. At service times and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in October 1982 with the issuance of NUREG-0744, Revision 1<sup>743</sup> which was later transmitted to all licensees with Generic Letter 82-26.<sup>744</sup>

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
743. NUREG-0744, Revision 1, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," U.S. Nuclear Regulatory Commission, October 1982.
744. NRC Letter to All Power Reactor Licensees (Except Ft. St. Vrain), "NUREG-0744, Rev. 1; Generic Letter No. 82-26) - Pressure Vessel Material Fracture Toughness," November 12, 1982.

ITEM A-12: FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT  
PUMP SUPPORTS

DESCRIPTION

During the course of the licensing action for North Anna Units 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and RCP support materials for these facilities. Two different steel specifications (ASTM A36 and ASTM A572) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made at various temperatures. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at a temperature of 80°F. In the case of North Anna Units 1 and 2, the applicant agreed to raise the temperature of the A572 beams in the steam generator supports to a minimum temperature of 225°F, prior to reactor coolant system pressurization to levels above 1,000 psig. Auxiliary electrical heat was supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials. Concerns regarding the supports at North Anna were applicable to all PWRs. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This solution to this USI was made available in October 1983 with the publication of NUREG-0577,<sup>388</sup> Revision 1. This resolution contains no backfit requirements and will apply to new construction only when SRP<sup>11</sup> Section 5.3.4 is issued.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U. S. Nuclear Regulatory Commission (Latest Edition).
388. NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U.S. Nuclear Regulatory Commission, October 1983.

ITEM A-24: QUALIFICATION OF CLASS 1E SAFETY-RELATED EQUIPMENTDESCRIPTION

CP applicants for which SERs were issued after July 1, 1974 were required by the NRC to qualify all safety-related equipment to IEEE 323.<sup>90</sup> From the time this standard was originated, the industry developed methods that were used to qualify equipment in accordance with the standard. Some of these methods had not been resolved to the satisfaction of the NRC.

In order to expedite the review and assess the adequacy of the equipment qualification methods and acceptance criteria used by NSSS and BOP vendors, the NRC determined that a generic approach was required. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> with the publication of NUREG-0588.<sup>19</sup> A new rule affecting future plants was also issued.<sup>60</sup> MPA B-60 was established by DL for implementation of the solution at operating plants.

REFERENCES

2. NUREG-(371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
19. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," U.S. Nuclear Regulatory Commission, July 1981.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
90. IEEE-323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

ITEM A-26: REACTOR VESSEL PRESSURE TRANSIENT PROTECTIONDESCRIPTION

Since 1972, there have been numerous reported incidents of pressure transients in PWRs where TS pressure and temperature limits have been exceeded. The majority of these events occurred while the reactors were in a water solid condition during startup or shutdown and at relatively low reactor vessel temperatures. Since the reactor vessels have less toughness at lower temperatures, they are more susceptible to brittle fracture under these conditions than at normal operating temperatures, they are more susceptible to brittle fracture under these conditions than at normal operating temperatures. In light of the frequency of the reported transients and the associated potential for vessel damage, the NRC concluded that measures should be taken to minimize the number of future transients and reduce their severity. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> with the publication of NUREG-0224<sup>746</sup> and SRP<sup>11</sup> Section 5.2. All operating PWRs were requested to provide an overpressure prevention system that could be used whenever the plants, were in a cold shutdown condition. The issue affected all operating and future plants, and MPA B-04 was established by DL for implementation of the solution at operating PWRs.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
746. NUREG-0224, "Final Report on Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, September 1978.



ITEM A-31: RHR SHUTDOWN REQUIREMENTSDESCRIPTION

The safe shutdown of a nuclear power plant following an accident not related to a LOCA has been typically interpreted as achieving a "hot-standby" condition (i.e., the reactor is shut down, but system temperature and pressure are still at or near normal operating values). Considerable emphasis has been placed on the hot-standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling, which is typically achieved by the RHR system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. It is essential that a power plant be able to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under any accident conditions.

This issue was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in May 1978 with the issuance of SRP<sup>11</sup> Section 5.4.7. Only those plants expected to receive operating licenses after January 1, 1979 were affected by the resolution.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).

ITEM A-36: CONTROL OF HEAVY LOADS NEAR SPENT FUELDESCRIPTION

At all nuclear plants, overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If a heavy object such as a spent fuel shipping cask or shielding block were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, there could be a release of radioactivity to the environment. Such an occurrence also has the potential for overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there was a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary. This item was originally identified in NUREG-0371<sup>2</sup> and was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> with the publication of NUREG-0612<sup>747</sup> and SRP<sup>11</sup> Section 9.1.5. MPAs C-10 and C-15 were established by DL for the implementation of Phases I and II, respectively, of the resolution at operating plants.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-066, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition)
747. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," U.S. Nuclear Regulatory Commission, July 1980.

ITEM A-39: DETERMINATION OF SAFETY RELIEF VALVE POOL DYNAMIC LOADS AND  
TEMPERATURE LIMITS

DESCRIPTION

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. The item addresses GE MARK I, II, and III containments and was originally identified in NUREG-0371<sup>2</sup> but was later determined to be a USI.

CONCLUSION

This USI was RESOLVED<sup>60</sup> with the issuance of SRP<sup>11</sup> Section 6.2.1.1.C. NUREG-0763,<sup>748</sup> NUREG-0783,<sup>734</sup> and NUREG-0802<sup>749</sup> were also issued for Mark I, II, and III containments, respectively.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.
11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.
60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
734. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," U.S. Nuclear Regulatory Commission, November 1981.
748. NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety Relief Valve Discharges for BWR Plants," U.S. Nuclear Regulatory Commission, May 1981.
749. NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," U.S. Nuclear Regulatory Commission, October 1982.

ITEM A-42: PIPE CRACKS IN BOILING WATER REACTORSDESCRIPTION

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in BWRs since mid-1960. These cracks have occurred mainly in Type 304 stainless steel which is the type used in most operating BWRs. The major problem is recognized to be IGSCC of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized," either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were very early (late 1960's) found to be susceptible to IGSCC. Because of this, the AEC took the position in 1969 that furnace-sensitized safe ends should not be used on new applications. Most of the furnace-sensitized safe ends in older plants have been removed or clad with a protective material and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop bypass lines and in 10-inch diameter core spray lines. Cracking is most often detected during ISI using UT techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

CONCLUSION

This USI was RESOLVED<sup>60</sup> in February 1981 when NUREG-0313, Revision 1<sup>750</sup> was issued to all holders of BWR operating licenses or construction permits and to all applicants for BWR operating licenses. MPA B-05 was established by DL for implementation of the resolution at operating plants.

REFERENCES

60. NUREG-0606, Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission (Latest Edition).
750. NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," U.S. Nuclear Regulatory Commission, July 1980.

ISSUE B-19: THERMAL-HYDRAULIC STABILITYDESCRIPTIONHistorical Background

The possibility of thermal-hydraulic instability in a BWR had been investigated by GE since the start-up of early BWRs. Analytical methods and codes were formulated on the basis of these early investigations to predict the stability characteristics of BWRs. Eventually, the licensing basis and analytical methods used by GE to evaluate the stability of BWRs were documented and presented in January 1977 in NEDO-21506.<sup>726</sup>

Since 1977, significant effort has been expended on developing an understanding of BWR instability. Testing at operating BWRs has added to the information obtained from single channel and control rod oscillator tests in the early BWRs.<sup>727</sup> In addition, improved state-of-the-art thermal-hydraulic methods and fuel rod performance studies have permitted greater definition of the stability phenomenon and criteria for prevention of instability. Developments along these lines have resulted in updated methods and models for the assessment and evaluation of BWR stability limits for licensing purposes.<sup>728</sup> However, recent data from a high-power-density foreign BWR unexpectedly indicated that scram protection based on the APRM signals would not necessarily prevent violation of the critical heat flux limits if local instabilities occur. As a result of these findings, the staff proposed the issuance of a Board Notification.<sup>729</sup> This item was identified in NUREG-0471.<sup>3</sup>

At the request of the licensees, the NRC staff has reviewed two submittals and has recently approved TS changes for two BWRs to resolve the concerns related to the thermal-hydraulic stability in these plants.<sup>730,731</sup>

Safety Significance

Hydrodynamic flow instabilities may occur in a BWR when two-phase flow exists in a channel with critical dimensions and particular flow parameters. The instability can cause power oscillations and lead to local violation of the critical heat flux.

Possible Solution

The proposed resolution is technical specifications that will restrict operation of the reactor in regions of potential thermal-hydraulic instability and/or provide for surveillance and corrective measures under conditions of marginal stability.

CONCLUSION

Updated analytical methods and analyses based on the recent experimental results have been made available to address thermal-hydraulic instability concerns. These methods are being reviewed by the NRC staff to determine their acceptability for evaluating the stability of core designs and for delineating the power/



flow regions of potential instability for which reactor operation will be restricted by appropriate modification of the plant TS.<sup>730,731</sup> Based on a study performed by the staff, it was concluded that thermal-hydraulic stability does not pose an immediate safety concern for continued BWR operation prior to orderly examination and possible TS changes. In response to GE and GE Owners' Group recommendations, most licensees have either submitted revised stability TS or plan to do so. A generic letter will be issued by DL to the affected licensees.<sup>769</sup> Thus, this issue has been RESOLVED and no new requirements were established.

#### REFERENCES

3. NUREG-0471. "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.
726. NEDO-21506, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Company, January 1977.
727. Memorandum for D. Crutchfield from L. Rubenstein, "Staff Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8," April 17, 1985.
728. XN-NF-691(P)(A) & Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores Sensitivity Analyses & Benchmark Analysis," Exxon Nuclear Company, Inc., August 22, 1984.
729. Memorandum for D. Eisenhut from R. Mattson, "Board Notification - BWR Core Thermal Hydraulic Stability," February 27, 1984.
730. Memorandum for T. Novak from L. Rubenstein, "Susquehanna 1 and 2 - Thermal Hydraulic Stability Technical Specification Change (TACS 55021 and 55022)," July 11, 1984.
731. Memorandum for G. Lainas from L. Rubenstein, "SER Input for Peach Bottom-3 Technical Specification Changes for Cycle 6 Operation with Increased Core Flows and Decreased Feedwater Temperatures (TACS #55123)," October 23, 1984.
769. Memorandum for V. Stello from H. Denton, "Close Out Generic Issue #B-19 - Thermal-Hydraulic Stability," May 21, 1985.

ITEM B-50: POST-OPERATING BASIS EARTHQUAKE INSPECTIONDESCRIPTIONHistorical Background

Appendix A of CFR 10 Part 100 specifies that the operating basis earthquake (OBE) shall be defined by response spectra and that the maximum vibratory ground acceleration of the OBE shall be at least one-half of the maximum vibratory ground acceleration of the safe shutdown earthquake (SSE). Suitable instrumentation is required at the reactor site, so that the seismic response of the nuclear power plant components that are important to safety can be determined promptly in order to permit a comparison of such response with that used as a design basis. Some detailed guidance on the nature and extent of this seismic instrumentation is provided in Regulatory Guide 1.12, Rev. 1, "Instrumentation for Earthquakes," April 1984. Shutdown of the nuclear plant is required in the event that vibratory ground motion exceeds that of the OBE. Prior to resuming operation, the licensee is required to demonstrate that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. This item was originally identified in NUREG-0471.<sup>3</sup>

A demonstration that all of the required equipment is functional following an earthquake greater than the OBE might be done by analysis, inspection, and/or test in some appropriate combination. At present, there are no specific requirements for demonstrating the safety of the plant in the event of an earthquake that exceeds the OBE. This issue is intended to improve this matter by providing guidance on the nature of the analysis, inspection, and/or tests that would be required, thereby permitting for a more rapid restart of the plant.

Safety Significance

Even though it would be expected that a plan would tend to provide systematic guidance, there is no safety significance for this issue inasmuch as it is likely that the NRC requirements for a reactor inspection following an earthquake that exceeds the OBE would contain the same elements and cover the same plant details whether it was planned in advance or developed following the earthquake. The intention in this issue is to establish a methodical approach to the inspection that might be necessary based on a systematic inspection plan, which would not result in any change in the potential core-melt frequency but would enable the licensee to conduct a site inspection expeditiously.

Possible Solution

Establish a post-OBE inspection plan that is comprehensive and detailed enough so that the inspection can be completed in a thorough, yet expeditious manner.

PRIORITY DETERMINATION

The prioritization of this issue was formulated with the technical assistance of PNL.<sup>64</sup>

### Frequency/Consequence Estimate

There is no reduction in core-melt frequency as a result of the implementation of the resolution of this issue.

### Cost Estimate

An earthquake that exceeds the OBE has occurred at only one nuclear site - Humboldt Bay (1975). This earthquake occurred during a refueling outage and has been reported by the licensee, PG&E.<sup>770</sup> For this plant, an emergency operating procedure covering earthquakes and tsunamis was already in place prior to the earthquake and covered power operation inspection procedures. The licensee decided to have an engineering inspection performed by Bechtel Power Corporation who had accumulated a large amount of detailed knowledge of the plant in connection with their ongoing seismic re-analysis of the plant. The Bechtel inspection was completed in 2 days.

For this analysis it is assumed that, without an inspection and test procedure in place before the earthquake (the base case), the inspection would require 2 weeks by the licensee using 10 of the licensee's personnel. Moreover, it is assumed that, with a comprehensive and systematic inspection procedure specified in advance (adjusted case), the inspection process could be completed more expeditiously and require only 4 days, with the same number of licensee personnel with no special seismic knowledge of the plant. On this basis the costs are estimated as follows.

- (1) The incremental cost of the post-OBE inspection based on 10 workers on a 20 hrs/day shift is:  

$$(10 \text{ workers/plant})(20 \text{ hrs/day})(14 - 4) \text{ days} = 2,000 \text{ man-hrs/plant.}$$
- (2) The occupational dose increment associated with the improved inspection procedures, based on an assumed dose rate of 175 mr/hr (Reference: Virgil Summer PWR FSAR Amendment 7, August 1978, Table 12.2-22b) is calculated to be:  

$$(-2,000 \text{ man-hrs/plant})(0.075 \text{ R/hr}) = -150 \text{ man-rem/plant.}$$
- (3) Based on the reduced inspection time of 2,000 man-hrs/plant noted above, the labor cost is calculated to be:  

$$(-2,000 \text{ man-hrs/plant})(40 \text{ man-hrs/man-wk})(\$2,270/\text{man-wk}) \\ = -\$114,000/\text{plant.}$$
- (4) The cost of the plant's downtime, based on the improved inspection procedures, is estimated as follows:  

$$(4 - 14) \text{ days } (\$300,000/\text{day}) = -\$3\text{M/plant (net savings).}$$
- (5) There are no operation and maintenance costs associated with this issue, nor are there any accident avoidance costs involved. The development of this inspection program is assumed to be borne by the

industry, presumably through ANSI. The cost to develop these inspection procedures is assumed to require technical assistance for a total cost of approximately \$100,000.

- (6) The total future cost to the industry is, therefore, estimated to be:
- $$(-\$114,000) + (-\$3M) = -\$3.114M \text{ (net savings)}$$
- (7) It is expected that the cost to the NRC in the development of the new inspection procedure would be limited to a review of the final inspection procedure submitted by the industry. It is assumed that the NRC review would require 20 man-weeks for a total cost of approximately (20 man-weeks)(\$2,270/week) or \$45,400.
- (8) It is anticipated that the NRC staff would participate in some way at the plant site during an inspection of the facility following an OBE. It is assumed that this would require 3 full-time staff members for the 2 weeks of inspection as well as an additional 2 weeks after the completion of the inspection as a follow-up measure. It is expected that, with the improved inspection procedures, only 2 full-time NRC staff members would be required for the 4 days of the inspection and for the 2 weeks following the completion of inspection and the resumption of normal operations. The incremental cost to the NRC for the implementation of the improved procedures is estimated as follows:

$$(\$2,270/\text{man-wk})[(112 \text{ man-hrs/man})(2 \text{ men/plant}) - (160 \text{ man-hrs/man})(3 \text{ men/plant})] = \$581,000$$

- (9) The total future costs to the industry and the NRC for the development and implementation of the resolution of this issue (C) is estimated from the results above to be:

$$C = -\$3,114,000 - \$581,000 = -\$3,695,000$$

- (10) The present value of the net future costs per plant (PW) is given by:

$$PW = (SP)(r)^{-1}(e^{-rt_1} - e^{-rt_2})$$

where  $S = \$3,695,000$

$P$  = frequency of exceeding the OBE per year.

$t_i$  and  $t_f$  are the initial and final times over which the savings are realized, with  $t_i = 0$  and  $t_f = T$  = average plant life = 28 yrs.

$r$  = the real discount rate = 5%/yr.

Based on these values the present value is calculated to be:

$$\begin{aligned} PW &= (\$3.695M)(P)(0.05)^{-1}(1 - \exp(-0.05t)) \\ &= (\$3.695M)(P)(15) = \$55.4M(P) \text{ per plant.} \end{aligned}$$

The frequency of exceeding the OBE is site-dependent but, on the basis of general studies, this frequency is in the order of  $10^{-3}$  to  $10^{-2}$ /year.<sup>771</sup>

These values represent an expected frequency range for exceedance of the OBE for a given plant. For this analysis, however, it is necessary to estimate the expected number of plants affected by any one OBE occurrence. In view of the existence of multiple plant sites, for purposes of this analysis it will be assumed that a twin plant site will be affected with a frequency of OBE exceedance of  $P_c = 10^{-2}/\text{yr}$ . The present value (PW) of the cost savings per site on this basis is estimated to be (2 plants)  $[(\$55.4\text{M})(10^{-2}/\text{plant})] = \$1.11\text{M}$ .

It is to be noted that the actual number of sites that would be affected is uncertain. In the West, only one site would likely be affected by any one OBE whereas, in the East, more sites might be affected but the probability for the latter would be lower by a factor of 10. Therefore, the present worth will be assumed as the total industry cost.

### CONCLUSION

It is estimated that the implementation of this issue requires a total combined cost of approximately \$145,000 for the industry and NRC in order to formulate a plan and procedure for expeditious inspection of a plant site following exceedance of an OBE. The advanced planning of this inspection procedure is also estimated to provide net future savings to the NRC and the industry of approximately \$3.7M at the time of the OBE, which is calculated to have a present worth of about \$1.1M.

In view of these costs and benefits it would seem that this is a Regulatory Impact issue that should be completed because it appears to have merit. However, it is to be noted that a considerable amount of uncertainty exists in the results on which to base realistic estimates of inspection requirements, costs, and the time involved. Moreover, it is not entirely clear that a detailed inspection plan attempting to anticipate the effects of an OBE on a complicated facility such as a nuclear reactor site can be adequately and meaningfully established in advance to the degree that the formulated plan and procedure will result in the estimated reduction of inspection time. It is also noted that this issue primarily concerns economic benefits rather than safety and that there is no burden on the licensees concerning this matter at this time. In view of these considerations as well as the uncertainties noted above, it is concluded that this issue should be left to be developed by industry initiatives.

Therefore, the NRC should assign a low priority to this regulatory impact issue unless requested by the industry to endorse a proposal submitted to NRC for this purpose.

### 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.
770. "Report on June 7, 1975 Ferndale Earthquake," Pacific Gas & Electric Company, August 2, 1975.



771. Memorandum for W. Minners from L. Reiter, "Generic Issue No. B-50 Post Operating Basis Earthquake Inspection," June 7, 1985.

ITEM B-59: (N-1) LOOP OPERATION IN BWRs AND PWRsDESCRIPTIONHistorical Background

The majority of the presently operating BWRs and PWRs are designed to operate with less than full reactor coolant flow. If a PWR RCP or a BWR recirculation pump becomes inoperative, the flow provided by the remaining (N-1) loops is sufficient for steady state operation at a power level less than full power. Although the FSARs for the licensed BWRs and PWRs present (N-1) loop calculations showing allowable power and protective system trip set-points, the NRC staff has disallowed this mode of operation for most plants primarily due to insufficient analyses. At present, BWR and PWR licensees have TS which require shutdown within a fairly short time if one of the reactor coolant loops becomes inoperable. This issue was originally identified in NUREG-0471.<sup>3</sup>

Allowing (N-1) loop operation gives utility operators more flexibility in deciding whether to shut down a plant or let it operate at a reduced power level. In this issue, (N-1) loop operation is restricted to operation during a single RCP failure. When fixing an out-of-service pump becomes a major task, it is not expected that the pumps will be repaired while the plant is on-line. By continuing operation in the (N-1) mode, the repair work may be postponed until a scheduled refueling time.

In connection with MPA E-05, a SER<sup>773</sup> was completed in July 1984 for the request by Beaver Valley Unit No. 1 (BV-1) for (N-1) loop operation. Based on this SER, it is expected that BV-1 will be authorized to operate with (N-1) loops when the TS are revised and updated appropriately in the near future. The SER for BV-1 represents the resolution of this issue for PWRs. For PWRs, DL is expected to close out MPA E-05 on the basis that there are no other active PWR applications for (N-1) operation and, further, that none is expected in the foreseeable future.<sup>774</sup> On the other hand, MPA E-04 covers 10 licensing actions on BWR submittals for (N-1) or single loop operation (SLO) for 7 licensees. The staff has reviewed the requests and submittals from the BWR licensees and has approved them such that (N-1) loop operation for BWRs would be authorized if the licensees submit the appropriate TS changes. The question of potential thermal-hydraulic instability problems during SLO for BWRs and how restrictive the TS changes would have to be was raised<sup>775,776</sup> by the staff, but this issue was resolved in Item B-19. However, in an effort to resolve certain plant-specific concerns about thermal-hydraulic instability in the Browns Ferry plant, TVA completed tests at Browns Ferry on February 9, 1985, and those concerns have been resolved. The tests demonstrated that TS based on GE SIL-380, which have been proposed for some BWRs and approved by the staff, are unlikely to result in any limitation on the achievable power level in SLO. They also indicated (pending verification by data analyses) that SLO is not significantly less stable than two-loop operation under similar power/flow operating conditions.

Permanent SLO has been approved for Peach Bottom Unit 3, Quad Cities Units 1 and 2, and Dresden Units 2 and 3 and will soon be approved for Duane Arnold.

The staff expects to approve permanent SLO for the SLO applicants when appropriate TS changes have been submitted.

### Safety Significance

In the event that a loop becomes inoperative in an operating plant, it is not always feasible to place it back in service by the repair of the failed pump while the plant is on-line. The plant operation with the (N-1) loops, however, will not differ from operation with all loops, except for the requirement to operate at a decreased power level for the lower flow condition and with corresponding instrument/control set-point limitations. The accident sequences would be essentially the same as with all loops in operation and there will be no change in accident initiator frequencies. Moreover, the loss of a loop because of pump malfunction would not impair the function of the ECCS and the other on-demand systems should an accident initiator arise. There had been some concern that operation with one loop out of service could result in thermal-hydraulic instabilities and possibly core damage at low flow conditions as well as with jet pump vibration at high flow conditions that could lead to damage to the reactor internals; but this matter has been adequately resolved. Therefore, the resolution would affect public risk or ORE only slightly and might reduce risk because power and fission product levels would be smaller than at full power. The purpose of this change is to reduce the impact on licensees.

### Possible Solution

The purpose of this task is to develop a set of acceptance criteria, review guidelines, and TS changes for the (N-1) loop authorization requests. This set of criteria, guidelines, and TS changes will encompass accident scenarios (both LOCAs and non-LOCAs) to be analyzed by the licensees, computer models acceptable to NRC for these analyses and acceptable input parameters in terms of reactor operating conditions (such as allowance for uncertainties in power level and fluid measurement). This has already been accomplished for PWRs by virtue of the completion of the SER<sup>773</sup> for BV-1. In addition, the BWR analyses have been reviewed and accepted by the staff and the appropriate generic TS changes to allow SLO for BWRs have been identified.

### PRIORITY DETERMINATION

The analysis will be limited here to (N-1) loop operation for BWRs since the issue is essentially inactive for PWRs. The estimates provided below were based on calculations performed by PNL.<sup>64</sup>

### Frequency/Consequence Estimate

When operating a nuclear plant at a power level proportional to a reduced number of loops, the safety margins are somewhat increased from those at full power, but this increased margin is not regarded as contributing to a significant reduction in risk. Therefore, any potential risk reduction associated with this issue is perceived to be negligible. Moreover, no additional ORE is anticipated for this issue inasmuch as major loop repair is likely to be done during scheduled downtimes.

## Cost Estimate

Industry Cost: To estimate the cost to industry, it is assumed that the amount of work performed on BV-1 by the licensee to analyze plant performance will be comparable to that required for BWR plants.<sup>772</sup> BV-1 analyzed a (N-1) loop large break LOCA and 12 non-LOCAs. Accidents involving the partial loss of forced reactor coolant flow, startup of an inactive reactor coolant loop, single RCP locked rotor, and complete loss of forced reactor coolant flow were analyzed in the original FSAR. Therefore, they were not reanalyzed. This leads to 13 transient scenarios to be analyzed.

Using a resource requirement of 5 man-wks and 15 computer hours for each case leads to a total of 65 man-wks and 195 computer hours. Another 30 man-wks/plant are allowed for preparing TS changes, modifying and upgrading procedures and/or systems, and familiarizing operations staff with upgrades. Using the industry rate of \$2,270/man-wk and an estimated computer cost of \$1,000/hr, the total implementation cost is estimated to be approximately \$420,000/plant for BWRs and PWRs. In addition, the plant-specific tests run by TVA at the Browns Ferry plant on the weekend of February 9, 1985, required operation at reduced power ranging from 50% to 65% for about 6 hours. On this basis it is estimated that the cost to conduct the test, obtain replacement power, and reduce the data will not exceed \$150,000.

The labor and analysis required for operation and maintenance of the resolution of this issue by the licensees is estimated to be negligible.

With the implementation of (N-1) loop operation, plant downtime can be reduced. The results of EPRI NP-1138<sup>431</sup> and EPRI NP-2094<sup>114</sup> indicate that the main contributor to (N-1) loop operation are pump seal failures. For Oconee 1, 88% of pump failure events are due to pump seal problems and 99% of pump maintenance time is on seal fixes. Since the non-seal failures only contribute 1% of the total maintenance time in the Oconee case, we ignore them for this analysis and use pump seal failure probability as the probability of losing one loop and operating under (N-1) loop conditions. While the failure frequency of PWR pump seals that contribute significantly to core-melt frequency is only 0.02/RY,<sup>366</sup> seal failures that result in the loss of one loop are estimated to be at a rate of 0.5/RY for both PWRs and BWRs. Some of these are LOCAs or would become LOCAs, but these can be isolated by the BWR recirculation loop valves.

If a plant is base-loaded, it is more economical to shut down and repair a seal that fails more than 20 days before the end of a 540-day cycle than continue with one-out-of-two loop reduced power operation. But all plants are not run at full power. Also one loop operation allows flexibility in shutting down to make repairs. Therefore, it is assumed that out of the 0.50/RY events, at best only one-third of the events will be continued as (N-1) loop operation, i.e., 0.17/RY.

The savings in terms of the avoided outage is estimated <sup>113</sup> to be 10 days (average extra outage time per pump seal failure). Therefore, the savings from avoidance of outage per reactor-year is given by:

$$(10 \text{ days})(1 \text{ loop}/2 \text{ loops})(\$300,000/\text{day})(0.17/\text{RY}) = \$255,000/\text{RY}$$

It is noted that the licensee implementation phase covering the analysis and evaluations of the potential accident sequences have already been submitted

to the NRC in many cases for BWRs. Therefore, it will be assumed that the remaining plant analyses and/or re-analysis that may be required by the NRC staff will affect one-half of the total number of reactors. Also, it will be assumed that the cost savings resulting from the avoidance of outages with (N-1) loop operation is the same each year for the industry. Further, assuming an average reactor lifetime of 28 years and 44 BWRs, the total industry costs are estimated to be as follows:

Implementation [(1/2)(\$420,000)(44 BWRs)]	= \$ 9,240,000
TVA Test at Browns Ferry	= 150,000
Operation and Maintenance	= 0
Outage Avoidance [(\$255,000/Ry)(44 BWRs)]	= 11,220,000/yr

For BWRs, the present worth (PW) of the annual savings from outage avoidance over the average reactor lifetime of 28 years at a real discount rate of 5% is:

$$PW = (\$11.22M)(0.05)^{-1}[1 - (1 + 0.05)^{-28}] = \$168.3M$$

NRC Cost: The cost to NRC of developing a set of acceptance criteria and review guidelines and TS changes for the issue are negligible inasmuch as these have already been identified. Some additional effort will be required to revise SRP Chapter 15 to reflect the criteria needed to review (N-1) loop operation. The revision to the SRP may require approximately 4 man-weeks. In addition, NRC labor to support SER implementation should be minimal at about 1 man-wk/plant. The total costs to the NRC are:

Revision to the SRP [(4 man-wk)(\$2,270/man-wk)]	= \$ 9,080
Implementation	
[(1 man-wk/plant)(\$2,270/man-wk)(44 plants)]	= 99,800
Development of Resolution	= 0

Thus, the total NRC costs are approximately \$110,000.

## CONCLUSION

It is concluded that this is a Regulatory Impact issue which has been resolved for BWRs and PWRs. For PWRs, the issue has been resolved on the basis of the BV-1 SER (MPA E-05) and, for BWRs, the issue has been resolved on the basis of Item B-19, the plant-specific tests at Browns Ferry, and the review of licensee submittals under MPA E-04.

## REFERENCES

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64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission.
113. EPRI NP-1194, "Operation and Design Evaluation of Main Coolant Pumps for PWR and BWR Service," Electric Power Research Institute, September 1979.



114. EPRI NP-2092, "Nuclear Unit Operating Experience, 1978 and 1979 Update," Electric Power Research Institute, October 1981.
366. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit One Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1982.
431. EPRI NP-1138, "Limiting Factor Analysis of High Availability Nuclear Plants," Electric Power Research Institute, September 1979.
772. Letter to A. Schwencer (NRC) from C. Dunn (Duquesne Light Company), "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, Request for Amendment to the Operating License-No. 35," October 27, 1978.
773. Letter to J. Carey (Duquesne Light Company) from S. Varga (NRC), "Beaver Valley Unit No. 1 - Operation With Two Out of Three Reactor Coolant Loops - Safety Evaluation," July 20, 1984.
744. Memorandum for D. Eisenhut from D. Wigginton, "Closeout of MPA E-05; Westinghouse N-1 Loop Operation," January 11, 1985.
775. Note to G. Lainas from R. Clark, "Status of Single Loop Operation for BWRs," October 2, 1984.
776. Memorandum for R. Bernero from D. Eisenhut, "BWR Thermal-Hydraulic Stability Technical Specifications," November 16, 1984.

ISSUE 35: DEGRADATION OF INTERNAL APPURTENANCES IN LWRsDESCRIPTIONHistorical Background

This item was started by an AEOD case study on internal appurtenances in LWRs.<sup>47,493</sup> This case study was begun because of a relatively high number of LERs which described events in which internal appurtenances (flow straighteners, orifices, diffusers, etc.) in the secondary system piping became loose or dislodged.<sup>368</sup>

Safety Significance

Originally, the safety concern was that, if a steam line break were to occur in a PWR, any loose objects in the secondary piping could become missiles during the steam generator blowdown and rupture one or more steam generator tubes.<sup>429</sup> A combined SGTR and MSLB was not a design basis accident.

The issue was, however, broadened to include loose objects in all LWRs, presumably in all locations. With such a broad definition, it automatically follows that there is a relatively large number of safety aspects. In general, a loose object causes problems either by causing impact damage or by blocking flow. In addition, the presence of a loose object of internal origin automatically implies that the system or component from which the loose object originated is now degraded.

Concern with loose objects is by no means new. Item B-60, "Loose Parts Monitoring System," extensively studied the occurrence and safety significance of loose objects within the primary system. Therefore, this study of Issue 35 will not include the primary system since the primary system is already addressed by Item B-60.

In addition, degradation of the ESFs will not be considered here, since the general issue of ESF reliability is addressed by other issues, e.g., Items A-45, B-4, and II.E.3.2. Moreover, since ESFs are not operating during normal operation of the plant, degradation of internal appurtenances in service should not be a problem.

We are left with the secondary system, which was where this issue began. In addition to the MSLB/SGTR scenario described earlier, a loose or disengaged object in the secondary system can have additional safety significance such as:

A loose object in the feedwater system can cause a loss-of-feedwater transient.

A sufficiently massive object could cause a feedwater line break.

A loose object in the main steam system could cause a transient. In many plants, plugging or isolating the main steam lines will also cause loss of main feedwater.

A massive object could cause a steam line break.

A loose object in a PWR steam generator could cause a steam generator tube rupture.

A loose object could prevent containment isolation valves from closing in the event of an accident.

Finally, a loose object could conceivably cause a small LOCA in non-safety systems connected to the primary system (e.g., RWC System). These are primary rather than secondary systems which are not covered by Item B-60 and thus remain within the scope of Issue 35.

### Possible Solutions

The systems under consideration are comprised primarily of piping rather than plena and thus are not amenable to the loose part detectors of Item B-60. About all that can be done is more frequent inspection and/or greater care in design and assembly, each where appropriate.

### PRIORITY DETERMINATION

#### Frequency/Consequence Estimate

Each of the scenarios described under Safety Significance above was examined.

#### (1) Steam Line Break with Steam Generator Tube Rupture

The frequency of a steam line break is estimated to be  $\leq 10^{-3}/\text{RY}$ , of which about 10% are expected to occur within containment. (See Item A-22). Assuming this frequency, we need the probability of a loose part being present at the time of the break. References 352 and 433 list 12 such events as of June 15, 1982, which corresponds to 360 PWR-years. We will assume that these reported events constitute 20% of the total events. In addition, we will assume that a loose object would go unnoticed for 2 years, i.e., at least a complete re-load cycle. The probability of a loose object being present during such a two-year period can then be estimated using the Poisson formula:

$$P = 1 - \exp\left[-\left(\frac{12 \text{ recorded events}}{360 \text{ years}}\right)\left(\frac{1 \text{ actual event}}{0.20 \text{ recorded events}}\right)(2 \text{ years})\right]$$

$$= 0.28$$

We will further assume that, given a MSLB event and a loose part present somewhere in the steam generator in feedwater lines, there is a 10% chance of the loose object rupturing a steam generator tube. Using these numbers, an event tree was constructed. Branches accounted for whether the break was inside or outside containment, whether the MSIV closed or not, whether feedwater to the affected steam generator was shut off, and whether HPSI operated or failed. The result was:

$$\begin{aligned}\text{Core-melt/PWR-year} &= 8.7 \times 10^{-8} \\ \text{Man-rem/PWR-year} &= 0.32\end{aligned}$$

The dominant sequence was as follows:

Steam line break occurs	$10^{-3}/\text{PWR-year}$
Loose object is present	0.28
Steam generator tube(s) rupture	0.10
Break is inside containment	0.10
MSIV is closed	0.90
Feedwater to affected steam generator continues (containment fails due to overpressure).	0.10
HPSI successfully cools core, but noble gases and some iodine are released	0.98
Net probability	$2.5 \times 10^{-7}/\text{PWR-year}$
Release is $\sim \text{PWR-8}$	$7.5 \times 10^4 \text{ man-rem}$
Net risk for this sequence	$0.20 \text{ man-rem/PWR-year}$

(2) Loss of Feedwater Transients and Transients Induced by Loose Objects in the Steam Lines

References 352 and 433 list 17 events of this nature in a period covering about 600 RY. Thus, the frequency of reported events is  $17/600$  per reactor year, or  $2.8 \times 10^{-2}/\text{RY}$ . Again, there are probably a sizeable number of loose objects which were not reported. However, not all loose objects cause transients and those that do probably will be reported. Thus, the frequency of transients is estimated to be the same as the frequency of reported events,  $2.8 \times 10^{-2}/\text{RY}$ .

Such transients normally have no safety consequences. However, they can initiate an accident if other equipment fails. To estimate risk, we simply scale the frequencies of the transient-initiated sequences in Tables V 3-14 and V 3-16 of WASH-1400<sup>16</sup> to match the frequency estimated above. The results are shown below in Table 3.35-1.

(3) Feedwater Line Break and Steam Line Break

The frequency of line breaks due to loose objects is very small, since these lines are quite massive. Nevertheless, steam flows through steam lines at roughly 300 miles per hour and a loose object traveling with the flow will have considerable impact when it encounters a  $90^\circ$  bend. The result, if there is damage, is likely to be a hole punched in the piping rather than a complete circumferential break.

We will assume that the frequency of line breaks is 1% of the frequency of transients estimated above. This results in an estimated break frequency on the order of  $2.8 \times 10^{-4}/\text{RY}$ . Although this number is judgmental in nature, if the actual frequency were an order of magnitude higher, two steam line breaks should have occurred in the 630 RY accumulated to date.

In PWRs, steam line breaks outside of containment are not particularly significant from the point of view of the reactor. The licensing analysis of such an event concentrates on the cooldown reactivity

Table 3.35-1

Release Category	Frequency (RY) <sup>-1</sup>	Consequences (Man-rem)
PWR-1	$8.1 \times 10^{-10}$	$5.4 \times 10^6$
PWR-2	$8.1 \times 10^{-9}$	$4.8 \times 10^6$
PWR-3	$1.1 \times 10^{-9}$	$5.4 \times 10^6$
PWR-4	$1.9 \times 10^{-10}$	$2.7 \times 10^6$
PWR-5	$5.4 \times 10^{-10}$	$1.0 \times 10^6$
PWR-6	$5.4 \times 10^{-9}$	$1.5 \times 10^5$
PWR-7	$2.7 \times 10^{-8}$	$2.3 \times 10^3$
BWR-1	$2.7 \times 10^{-9}$	$5.4 \times 10^6$
BWR-2	$1.6 \times 10^{-8}$	$7.1 \times 10^6$
BWR-3	$5.4 \times 10^{-8}$	$5.1 \times 10^6$
BWR-4	$5.4 \times 10^{-9}$	$6.1 \times 10^5$

Core-melt Frequency:  $4.3 \times 10^{-8}$ /PWR-year  
 $7.8 \times 10^{-8}$ /BWR-year

Public Risk: 0.051 man-rem/PWR-year  
0.41 man-rem/BWR-year

transient (and associated radiological consequences), assuming a complete circumferential break, end-of-cycle moderator temperature coefficients, and failure of the highest worth control rod to insert. It is most unlikely that a loose object would cause a complete circumferential break, regardless of the probabilities of the other assumptions of the licensing basis. It should also be noted that opening an ADV is equivalent to a steam line break passing 10% of rated steam generator flow. Feedwater line breaks outside containment are still more innocuous, since check valves will prevent blowdown of the steam generator.

Inside containment, breaking a steam line will dump the mass and energy content of a steam generator to the containment. If feedwater to this steam generator is not cut off, continued steam production could endanger the containment. (Feedwater line breaks are less troublesome, since breaking the feedwater line is guaranteed to shut off feedwater to the steam generator which is blowing down.)

This is exactly the safety concern of Item A-22, "PWR Main Steam Line Break - Core, Reactor Vessel, and Containment Response." If we scale the priority parameters calculated for Item A-22 to the frequency of  $2.8 \times 10^{-5}$ /PWR-year estimated for a break inside containment, the result is:

Core-melt Frequency: zero

Public Risk:  $\leq 0.00038$  man-rem/PWR-year

In BWRs, steam line breaks and feedwater line breaks are small LOCAs. In view of the presence of two MSIVs in each steam line and two check



valves in each feedwater line in a BWR, we will consider only breaks inside containment. Again, the estimated frequency is  $2.8 \times 10^{-5}$ /BWR-year, just as in the PWR case. In addition, we will assume that the break is an "S1" LOCA (equivalent diameter of 2 to 6 inches). Scaling Table V 3-16 of WASH-1400<sup>16</sup> to  $2.8 \times 10^{-5}$  "S1s" per BWR-year:

<u>Release Category</u>	<u>Frequency (RY)<sup>-1</sup></u>	<u>Consequences (Man-Rem)</u>
BWR-1	$9.3 \times 10^{-10}$	$5.4 \times 10^6$
BWR-2	$8.4 \times 10^{-9}$	$7.1 \times 10^6$
BWR-3	$1.9 \times 10^{-8}$	$5.1 \times 10^6$
BWR-4	$1.9 \times 10^{-9}$	$6.1 \times 10^5$

Core-melt Frequency:  $3.0 \times 10^{-8}$ /BWR-year

Public Risk: 0.16 man-rem/BWR-year

(4) Steam Generator Tube Rupture

This particular scenario is already addressed in the steam generator tube integrity Issues 66, 67, and USIs A-3, A-4, and A-5. Thus, it will not be considered further here.

(5) Loss of Containment Isolation Capability

Loose objects can interfere with valve operation, particularly since valve seats are natural collection points for debris. Most valves in the secondary system are not safety-related; interference with these valves will at most cause a transient, as discussed earlier. The safety-related valves include steam safety valves and isolation valves. Of these, the steam safeties are not susceptible to damage by loose objects under normal circumstances, since there is no flow to carry objects into them nor are most loose objects likely to float upwards in steam. Even if a loose object were carried into a safety valve during a safety valve actuation, the overpressurization analysis assumes one failed valve. Thus, a loose object plugging one safety valve will not result in overpressurization of the secondary side.

Interference with isolation valves (preventing complete closure) is more plausible, since these valves are usually passing flow during normal operation. References 352 and 433 list only one event of this nature in a period of 600 RY. Since isolation valves are tested periodically and problems are reportable, it is unlikely that the actual number of events is significantly larger than the number of reported events. Thus, we expect the frequency of occurrence of an inoperable isolation valve due to a loose object to be on the order of  $1.7 \times 10^{-3}$ /RY.

The longest interval between isolation valve tests is a full 18-month fuel cycle. A simple application of the Poisson formula gives us a probability estimate of  $2.5 \times 10^{-3}$  for the failure of an isolation valve somewhere in the plant to close on demand.

Failure of an isolation valve does not automatically mean that the containment fails to isolate, since isolation valves on primary systems are double and even secondary system isolation valves are usually backed up by check valves, turbine stop valves, etc. However, the potential for a common mode failure is quite high for loose part events. We will assume that failure of one isolation valve does result in failure of the containment to isolate.

The effect of a containment isolation failure of this nature is to change accident scenarios which otherwise would have resulted in Release Category PWR-7 or PWR-9 into Release Categories PWR-5 and PWR-8, respectively. Using the WASH-1400<sup>16</sup> frequencies for PWR-7 and PWR-9, a simple multiplication by the estimated containment isolation failure probability gives the change in the PWR-5 and PWR-8 frequencies:

Frequency PWR-7 and PWR-9 (PWR-years) <sup>-1</sup>	Containment Failure Probability	Frequency PWR-5 and PWR-8 (PWR-year) <sup>-1</sup>	Consequences PWR-5 and PWR-9 (man-rem)	Public Risk (man-rem/ PWR-year)
$4 \times 10^{-5}$	$2.5 \times 10^{-3}$	$1.0 \times 10^{-7}$	$1.0 \times 10^6$	0.100
$4 \times 10^{-4}$	$2.5 \times 10^{-3}$	$1.0 \times 10^{-6}$	$7.5 \times 10^4$	0.075

Core-melt Frequency: zero

Public Risk: 0.175 man-rem/PWR-year

The corresponding calculation for BWRs is more difficult. The analog of the PWR-7 to PWR-5 case, which corresponds to a jammed isolation valve releasing core-melt activity which would otherwise be trapped within containment, does not exist for a BWR, since BWR core-melts are expected to cause containment overpressurization and failure anyway. The analog of the PWR-9 to PWR-8 case, which corresponds to a jammed isolation valve releasing contaminated containment atmosphere during and after a successfully mitigated LOCA, does exist in theory but no appropriate BWR release category has been calculated. We will make the pragmatic assumption that the BWR analog of the PWR-9 to PWR-8 case results in roughly the same radiological release.

Public Risk: 0.075 man-rem/BWR-year

#### (6) Small LOCA in System Connected to Primary

References 352 and 433 list one loose object in a system connected to a PWR primary loop. In 350 PWR-years, this gives an estimated frequency of reported loose object events of  $2.8 \times 10^{-3}$ /PWR-year. If these represent 10% of the actual events, the actual event frequency is  $2.8 \times 10^{-2}$ /PWR-year.

In BWRs, all systems are connected to the primary side and our definition needs some modification. Here, we mean systems connected to the reactor, exclusive of the main steam, condensate and feedwater,

and normally idle safety systems. No such loose objects have been reported. Therefore, we will assume that the rate of occurrence is the same as for PWRs,  $2.8 \times 10^{-2}/\text{RY}$ .

Flow rates for such systems are low. Moreover, they are equipped with various types of leak detection coupled with automatic isolation. Therefore, it is very unlikely that a loose object will cause an unisolated leak. We will assume that the probability of such a leak is on the order of  $10^{-3}$ , given the presence of a loose object. Thus, the overall frequency of the leak is estimated to be on the order of  $2.8 \times 10^{-5}/\text{RY}$ .

The size of such a LOCA would be in the "S2" class. Scaling WASH-1400<sup>16</sup> Tables V 3-14 and V 3-16 to this frequency, the results are shown below in Table 3.35-2.

TABLE 3.35-2

Release Category	Frequency (RY) <sup>-1</sup>	Consequences (Man-Rem)
PWR-1	$2.8 \times 10^{-9}$	$5.4 \times 10^6$
PWR-2	$8.4 \times 10^{-9}$	$4.8 \times 10^6$
PWR-3	$8.4 \times 10^{-8}$	$5.4 \times 10^6$
PWR-4	$8.4 \times 10^{-9}$	$2.7 \times 10^6$
PWR-5	$8.4 \times 10^{-9}$	$1.0 \times 10^6$
PWR-6	$5.6 \times 10^{-8}$	$1.5 \times 10^5$
PWR-7	$5.6 \times 10^{-7}$	$2.3 \times 10^3$
BWR-1	$5.6 \times 10^{-10}$	$5.4 \times 10^6$
BWR-2	$2.8 \times 10^{-9}$	$7.1 \times 10^6$
BWR-3	$1.1 \times 10^{-8}$	$5.1 \times 10^6$
BWR-4	$1.1 \times 10^{-9}$	$6.1 \times 10^5$

Core-melt Frequency:  $7.3 \times 10^{-7}/\text{PWR-year}$   
 $1.5 \times 10^{-8}/\text{BWR-year}$

Public Risk:  $0.63 \text{ man-rem}/\text{PWR-year}$   
 $0.080 \text{ man-rem}/\text{BWR-year}$

The scenarios above add up to the following:

Core-melt Frequency:  $8.6 \times 10^{-7}/\text{PWR-year}$   
 $1.2 \times 10^{-7}/\text{BWR-year}$

Public Risk:  $1.18 \text{ man-rem}/\text{PWR-year}$   
 $0.73 \text{ man-rem}/\text{BWR-year}$

Currently, with 95 PWRs and 47 BWRs operating, planned, or under construction, we estimate an aggregate remaining lifetime of 3,400 PWR-years and 1,600 BWR-years. This allows us to make the following estimates:

Man-rem/Reactor = 40  
 Man-rem, Total = 5,000

$$\begin{aligned}\text{Core-melt/RY} &= 6 \times 10^{-7} \\ \text{Core-melt/Year} &= 9 \times 10^{-5}\end{aligned}$$

### Cost Estimate

Industry Cost: We will postulate that 10 man-weeks spent on inspections every refueling outage will be about 90% effective in revealing degraded appurtenances. If refuelings occur every 18 months, this works out to \$13,300/RY. The cost of actual repair, replacement, or upgrading is not included since this would eventually have to be done anyway. It should also be noted that avoiding even one unnecessary reactor scram would pay for 20 years of inspections. Thus, there is some actual financial advantage to the utility. Based on a remaining lifetime of 5,000 RY for all reactors, the total industry cost for the solution to this issue is \$66.5M.

NRC Cost: NRC costs are estimated to be on the order of \$500,000 (5 man-years). This estimate is higher than usual since licensing effort currently does not emphasize the BOP systems and opposition and delay are quite likely.

### Value/Impact Assessment

Based on a total risk reduction of 5,000 man-rem, the value/impact score is given by:

$$\begin{aligned}S &= \frac{5,000 \text{ man-rem}}{\$(66.5 + 0.5)\text{M}} \\ &= 70 \text{ man-rem}/\$M\end{aligned}$$

### Uncertainties

None of the scenarios described above is dominant. However, all follow much the same pattern. The estimated frequency of loose part occurrences is unlikely to be more than a factor of 10 too low or too high. The probability of a loose part causing an accident (purely judgmental estimates) might be uncertain by as much as a factor of 20 in some cases. The estimates of consequences should be within a factor of five. Finally, the estimates of cost could well be off by a factor of 10. If log normal distributions are assumed, the man-rem and core-melt figures should be within a factor of about 60 and the priority score within a factor of 100.

### CONCLUSION

The core-melt/year estimate is barely in the medium priority range (because of the large number of plants affected); all others factors are in the low priority range. Therefore, it is recommended that this issue be given a LOW priority.

### REFERENCES

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.
47. Memorandum for H. Denton from C. Michelson, "Degradation of Internal Appurtenances in LWR Piping," January 19, 1981.

- 352. Memorandum for C. Michelson from E. Brown, "Internal Appurtenances in LWRs," December 24, 1980.
- 368. Memorandum for ACRS Members from C. Michelson, "Failure of a Feedwater Flow Straightener at San Onofre Nuclear Station, Unit 1," June 13, 1979.
- 429. Memorandum for J. Knight from E. Sullivan, "Review ACRS Consultant Report," January 10, 1980.
- 433. Memorandum for C. Michelson from E. Brown, "Degradation of Internal Appurtenances and/or Loose Parts in LWRs," June 15, 1982.
- 493. Memorandum for C. Michelson from H. Denton, "January 19, 1981, Memorandum on Degradation of Internal Appurtenances in LWR," April 30, 1981.



ISSUE 37: STEAM GENERATOR OVERFILL AND COMBINED PRIMARY AND SECONDARY  
BLOWDOWN

DESCRIPTION

In AEOD/C005,<sup>759</sup> AEOD identified potential safety problems concerning steam generator overfill due to control system failures and combined primary and secondary blowdown. As a result of discussions with the Commissioners and the EDO, NRR reported<sup>495</sup> that the steam generator overfill problem was to be integrated as a sub-task of USI A-47. In addition, separate work was scheduled on the matter of combined primary and secondary blowdown to establish its safety significance.

With regard to the issue concerning steam generator overfill, USI A-47 deals with control system failures leading to overfill. A related issue, (Issue 67) includes actions which address steam generator overfill following a SGTR and are not part of USI A-47.

With regard to the safety issue concerning combined primary and secondary blowdown, this issue has been addressed in NUREG-0737, Item I.C.1, which is being implemented under MPAs F-04 and F-05.

CONCLUSION

The safety concerns identified in this issue are addressed in USI A-47 and Item I.C.1.

REFERENCES

- 495. Memorandum for C. Michelson from H. Denton, "Steam Generator Overfill and Combined Primary and Secondary Blowdown," May 27, 1981.
- 759. AEOD/C005, "AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Blowdown," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 17, 1980.

ISSUE 42: COMBINATION PRIMARY/SECONDARY SYSTEM LOCADESCRIPTION

AEOD issued a memorandum<sup>446</sup> in which a potential safety issue involving combined primary and secondary system LOCAs was raised. The issue was discussed at Commission meetings on October 16, 1980 and on November 10, 1980. NRR informed AEOD of the results of the then current status of the staff's evaluation.<sup>36</sup> In SECY-82-296,<sup>432</sup> the staff recommended to the Commission that this issue not be classified as an USI and that the issue could be adequately handled by on-going reviews. This issue is identical to Issue 18 which is being implemented as part of TMI Action Plan Item I.C.1 clarified in NUREG-0737.<sup>98</sup>

CONCLUSION

This issue is covered in TMI Action Plan Item I.C.1.

REFERENCES

- 36. Memorandum for C. Michelson from H. Denton, "Combination Primary/Secondary System LOCA," December 8, 1981.
- 98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.
- 432. SECY-82-296, "Resolution of AEOD Combination LOCA Concern," July 13, 1982.
- 446. Memorandum for Chairman Ahearne from C. Michelson, "New Unresolved Safety Issues," August 4, 1980.

## ISSUE 54: SURVEY OF VALVE OPERATOR-RELATED EVENTS OCCURRING DURING 1978, 1979, AND 1980

### DESCRIPTION

#### Historical Background

In December 1981, AEOD completed a survey of valve operator-related events that occurred during 1978, 1979, and 1980 and submitted a draft report<sup>707</sup> to NRR for comment. In this study, AEOD established that motor-operator failures could be grouped into three principal categories: (1) torque switches, (2) limit switches, and (3) motor burn-out. On March 5, 1982, NRR comments<sup>708</sup> on this draft report were forwarded to AEOD. Following issuance of the final AEOD report (AEOD/C203),<sup>709</sup> additional comments were provided by NRR.<sup>509</sup>

Following receipt of comments from the various NRC Offices, AEOD provided an assessment of these comments.<sup>710,711</sup> NRR then provided a response<sup>712</sup> to this assessment in an effort to clarify certain matters related to this issue in terms of an expansion of an RES program (FIN A6367) to address features of this issue. Subsequently, AEOD issued a report (AEOD/E305)<sup>713</sup> covering 8 LERs on the degradation of MOV assemblies and presented results that augmented the findings of AEOD/C203.<sup>709</sup> In March 1984, the status of this issue was reported in an MEB memorandum.<sup>714</sup> This issue is related to Item C-11 and TMI Action Plan Item II.E.6.

#### Safety Significance

Degradation of motor-driven valve motors from thermal overloads may be occurring in repeated high-temperature cycles that is not detected in the present in-service testing programs. This may lead to premature motor burnout and subsequent unexpected unavailability of the valve during accident conditions. Moreover, torque switch settings and adjustments may be inadequate to ensure valve operability and qualification under accident conditions.

This issue would affect the design and operation of all existing and future nuclear plants for the lifetime of the plants.

#### Possible Solutions

The possible solutions are: (1) reassessment of the present guidance in Regulatory Guide 1.106 on bypassing thermal overload-protection devices for motors, (2) improvement of the methods and procedures for the setting of torque switches, and (3) the development of signature tracing technique(s) to be incorporated into the in-service testing programs in order to detect valve deterioration, aging effects, improper maintenance, and improper valve adjustments that are not presently detectable.

### CONCLUSION

The principal concern in this issue the operability of the motor-driven valves under accident, faulted, or emergency conditions. It is noted from the Action

Plan<sup>715</sup> for TMI Action Plan Item II.E.6.1 that the work scope includes verification of safety-related valve function under system emergency or faulted conditions, torque and limit switches settings and adjustments, motor burn-out resulting from improper cycling, and MOV operator signature tracing by evaluation, test, and analysis. Therefore, it is concluded that the objectives of Issue 54 will be met completely in TMI Action Plan Item II.E.6.1.

#### REFERENCES

509. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Final Report: Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," August 19, 1982.
707. Memorandum for H. Denton et al., from C. Michelson, "Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," December 23, 1981.
708. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Draft Report: Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," May 5, 1982.
709. AEOD/C203, "Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, May 1982.
710. Memorandum for C. Michelson from E. Brown and F. Ashe, "Assessment of Program Office Responses to the Report AEOD/C203, 'Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980,'" December 23, 1982.
711. Memorandum for H. Denton from C. Michelson, "AEOD Assessment of Program Office Responses to AEOD Case Study (C-203), 'Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" January 13, 1983.
712. Memorandum for C. Michelson from H. Denton, "AEOD Assessment of Program Office Responses to AEOD Case Study (C-203), 'Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" February 23, 1983.
713. Memorandum for K. Seyfrit from E. Brown and F. Ashe, "Engineering Evaluation Report AEOD/E305 Inoperable Motor Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment," April 13, 1983.
714. Memorandum for W. Minners from R. Bosnak, "Status of Potential Generic Issue 54, 'Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" March 26, 1984.
715. Memorandum for R. Vollmer from R. Bosnak, "MEB Task Action Plan for Resolution of Generic Issue II.E.6.1, 'In Situ Testing of Valves,'" July 30, 1984.

## ISSUE 59: TECHNICAL SPECIFICATION REQUIREMENTS FOR PLANT SHUTDOWN WHEN EQUIPMENT FOR SAFE SHUTDOWN IS DEGRADED OR INOPERABLE

### DESCRIPTION

#### Historical Background

As a result of the loss of high head injection capability at McGuire Unit 1 on February 12, 1982, this issue was raised by Region II because plant TS require (somewhat rapid) plant shutdown if certain safety equipment is inoperable.<sup>553</sup> The main concern is that the TS requirements may not adequately consider the potential for placing the plant in a "less safe" condition by requiring shutdown of an otherwise normally functioning unit or by requiring a plant to proceed to cold shutdown when "hot shutdown" may be the more desirable condition.<sup>767</sup>

#### Safety Significance

Plant TS LCOs are typically written to require proceeding to various stages of shutdown if certain systems are inoperable. If some systems are inoperable and a plant is required by the TS to go to some stage of shutdown, this action may increase the probability of needing the inoperable systems as a line of defense. In some cases, the shutdown process itself may require operation of the inoperable equipment.

#### Possible Solution

A resolution could require TS modifications to acknowledge when continued power operation or other mode of operation is preferable. Because of the wide range of possible system failures, operating conditions, and plant configurations, a systematic quantification of all the alternatives could be a fairly large task and would probably result in a number of decisions based on very close calls.

### PRIORITY DETERMINATION

#### Assumptions

In order to provide some indication of what a quantitative analysis may involve, an assessment of this issue was done by PNL.<sup>64</sup> A number of assumptions were made in this analysis in an attempt to quantify a potential core-melt frequency reduction for a case in which a plant was left operating, as opposed to rapidly shutting it down, given some safety equipment is inoperable.

The ORNL Precursor Study (NUREG/CR-2497)<sup>76</sup> was used along with data from an EPRI ATWS study (NP-2230)<sup>307</sup> to calculate a base case and an adjusted-case core-melt frequency for this issue. The techniques and data presented in NUREG/CR-2497<sup>76</sup> were modified to allow a comparison of the risk of core damage with a safety system inoperable for continued reactor operation versus immediate shutdown. To accomplish this, specific systems were chosen for failure and appropriate event trees developed. Data on system failure were then adapted to fit the need for: (1) failure frequency, (2) failure on demand, or (3) failure over a specified time interval. For this analysis, a BWR was chosen and it



was assumed that the HPCI and RCIC are redundant safety systems. Failure of both the RCIC and HPCI was then postulated.

To examine this issue, generic event trees were developed based on the flow logic developed in the Precursor Study<sup>76</sup> for BWR transients. The first event tree<sup>64</sup> depicted a failure of safety systems followed by a shutdown by the operator. The transient which could then follow was shown as "loss of feedwater given shutdown," chosen here as representative of transients which would challenge the ECCS. The second event tree depicted the case where operation continues. Another initiating event is then required, taken here as loss of feedwater given an ECCS subsystem failure. The following data were taken from NUREG/CR-2497:<sup>76</sup>

Event Description	Occurrences	Plant-Years (RY)	Demands	Failure Frequency (RY) <sup>-1</sup>	Failure Probability on Demand
Loss of Feedwater	39	66	-	0.58	-
Reactor Subcritical	-	-	-	-	$1.3 \times 10^{-6}$
RCIC/HPCI Failure	4.9	99	-	0.049	0.0039
Long-Term Core Cooling Failure	-	-	-	-	$1.1 \times 10^{-4}$

The analysis results hinge on the probability of inducing a feedwater transient on shutdown vs. a feedwater transient occurring at power during the time the systems remain inoperable. Data for these values are lacking at this time so values are estimated based on the ATWS report.<sup>307</sup> For BWR Transient Category 26 (decreasing feedwater flow during startup or shutdown), the frequency reported is 0.07/RY. It is assumed here that the plant is shutdown about 12 times per year resulting in a probability of about 0.01 for a feedwater transient on shutdown. It is further assumed that 50% of these transients are decreases in feedwater during shutdown with 50% of these resulting in complete loss of function. The probability ( $p_1$ ) of loss of feedwater on scram is therefore assumed to be  $(0.01)(0.5)(0.5) = 0.0025$ .

To estimate the probability of feedwater failure during an ECCS subsystem outage, a one-day failure duration is assumed with the plant remaining at power for that 1 day. The probability ( $p_2$ ) of independent loss of feedwater over the one-day ECCS subsystem outage is approximated by  $p_2 = \lambda t = 0.0016$ , where  $\lambda = 0.58/\text{RY}$  and  $t = (1 \text{ day})/(365 \text{ day/RY}) = 0.0027 \text{ RY}$ .

These data were entered in the event trees by PNL,<sup>64</sup> Sequences 5, 6, 7, and 8 were summed and then Sequences 12, 13, 14, and 15 were summed. The core-melt frequency was calculated to be  $3.5 \times 10^{-6}/\text{RY}$  for the base case, i.e., the rapid shutdown of the plant. The core-melt frequency was calculated to be  $2.2 \times 10^{-6}/\text{RY}$  for the adjusted case, i.e., where the plant continued to operate.

The event trees developed for the Precursor Study<sup>76</sup> give a measure of core damage only. To equate this with the core-melt frequency used in other risk

studies, the above core-damage frequencies were divided by a factor of 30 for the reasons given below.

An analysis of the ORNL Precursor Study by INPO claims that the chances of a severe nuclear accident were estimated 30 times too high.<sup>64</sup> Furthermore, severe core damage (assumed to be analogous to that at TMI-2 in the Precursor Study) is presumably less severe than core-melt, the level of core damage normally considered in nuclear power plant risk studies. Based on these considerations, it is assumed that the frequency of core damage as assessed using the Precursor Study should be divided by INPO's factor of 30 to result in the frequency of core-melt.

Thus, the base case and adjusted case core-melt frequencies become  $1.2 \times 10^{-7}/\text{RY}$  and  $7 \times 10^{-8}/\text{RY}$ , respectively and the core-melt frequency reduction is  $5 \times 10^{-8}/\text{RY}$ . An average LWR dose factor of  $3.3 \times 10^6$  man-rem was calculated from NUREG/CR-2800,<sup>64</sup> (Appendices A-D). Based on this factor, the potential risk reduction would be  $(5 \times 10^{-8}/\text{RY})(3.3 \times 10^6 \text{ man-rem})$  or 0.17 man rem/Ry.

The result shows a slight decrease in risk. However, the calculation is heavily dependent on the assumed value for the probability of loss of feedwater on shutdown vs. the probability of loss of feedwater over a 1 day ECCS outage. This dependency can be seen by doing the same type of calculation but assuming a 1.5 day outage time. Then, for the adjusted case,  $P = \lambda t = 0.0024$  and evaluating events 12, 13, 14, and 15 yields an adjusted case frequency of about  $3.2 \times 10^{-6}/\text{RY}$ . This would then show an even smaller risk reduction when compared to the base case result of  $3.5 \times 10^{-6}/\text{RY}$ . (Again, these would be reduced by a factor of 30). Similarly, if 2 days were assumed,  $P = \lambda t = 0.32$  and the core-melt frequency would be about  $4.3 \times 10^{-6}/\text{RY}$  which would show a slight increase in risk for staying at power. These calculations show the sensitivity of the results to the assumptions and the data.

#### Cost Estimate

Industry Cost: The direct cost would be \$4,000/plant for a Class III amendment to an operating license. Other costs for implementation could be significant for analysis of various plant situations to identify the preferred mode and, therefore, justify the change.

Based on 71 operating plants, the industry cost was assumed to be (71 plants) (\$4,000/plant) or \$0.28M and 1 man-year/plant for supporting analysis or (\$100,000/plant)(71 plants) = \$7.1M. Since most changes would involve a justification for continued power operation, a potential large cost saving could be involved. For calculation purposes, it could be assumed that over the life of a typical plant, at least 1 day of shutdown may be saved. At \$300,000/day, the industry cost saving for 134 plants is \$40M.

NRC Cost: NRC cost for issue development was based on the assumption that considerable analysis (and review of licensee submittals) would be needed to quantify safety benefits associated with TS modifications. This was assumed to be about 3.5 man-years or \$3.5M.

### Other Considerations

1. Since this issue was originally raised, the NRC has published a rule which allows relief from TS requirements in an emergency situation. This rule leaves the decision to the licensees of determining: (1) what constitutes an emergency, and (2) what is the most prudent action to take. During the comment period on the rule, it was requested that comments be provided regarding whether or not the rule should have more specific guidance. It was concluded, based on comments received, that it was not feasible to provide detailed guidance as to when deviations are permissible. It was felt that this would defeat the purpose of the rule which is to provide flexibility in situations that cannot always be anticipated.
2. More recently, the general issue of whether TS are properly focused or are unduly burdensome has been raised. In response to this problem, a Technical Specification Improvement Project has been established.<sup>768</sup> This project will consider the safety relevance and burden of the TS as a whole and of specific sections. This issue is one example of a possible modification to improve the TS.
3. The above analysis was done based on assuming a situation in which a plant is at power and the question is whether to require the plant to proceed to shutdown. It was pointed out that a clearer case could be made for situations of the plant being in hot shutdown and requiring proceeding to cold shutdown. Regardless, both situations could lead to potentially large cost savings for the industry and it may be to a licensee's advantage to try to anticipate the possibility of these situations and submit modified TS to avoid crisis-type decisions (i.e., emergency TS relief) when the emergency arises or to avoid second-guessing after the emergency passes if, for example, the rule is used.
4. The McGuire event (which is part of the basis for raising this issue) could have been a case for application of the rule. The question would have been if, as postulated, the situation would have continued (i.e., no charging/SI pumps), would it have been preferable for the licensee to deviate from the TS and keep the plant on line and, if so, how long should power operation be continued?
5. The situations of concern are typically beyond the design bases of the plant and, therefore, should occur rather infrequently.
6. For cases like McGuire where shutdown would require the inoperable equipment, it appears that a TS change may not solve the problem because, no matter what length of time is chosen for continued operation, there is some probability that the equipment would not be restored in the allowed time and shutdown would be necessary anyway (either due to the TS or a transient). For such cases, it is probably best to let the licensee use the rule based on a consideration of the specific plant circumstances at that time. It should be pointed out that the AFW system TS 3.7.1.2 (which was suggested as a solution<sup>553</sup>) was written prior to the Rule<sup>767</sup> and would probably not be included in the TS if the rule had been in effect at that time.

CONCLUSION

Although we did not calculate a specific value/impact score for this issue, the calculation of potential man-rem reduction for the assumed scenario gives an indication of the uncertain nature of this type of analysis. After consideration of the new rule,<sup>767</sup> we concluded that to a large extent the safety implications of this issue have been addressed. The rule gives the licensees the flexibility to consider their individual plant circumstances and make a decision to deviate from the TS if they decide it is necessary. However, as has been pointed out, there may be specific cases where changes should be considered.<sup>767</sup> Because the risk was so hard to quantify, we originally assumed a small change in public risk, acknowledged the potential cost saving, and concluded it should be a Regulatory Impact issue to be addressed by the Technical Specification Improvement Project.<sup>768</sup>

REFERENCES

- 64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission.
- 76. NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report," U. S. Nuclear Regulatory Commission, June 1982.
- 307. EPRI NP-2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.
- 553. Memorandum for D. Eisenhower from J. Olshinski on "Loss of High Head Injection Capability at McGuire Unit 1 and Reconsideration of Technical Specification 3.0.3 and 3.5.2," April 12, 1982.
- 767. Memorandum for W. Dircks from H. Denton, "Final Rule - Applicability of License Conditions and Technical Specifications in an Emergency," February 17, 1983.
- 768. Memorandum for T. Speis from H. Denton, "Formation of a Technical Specification Improvement Project Group," December 31, 1984.

ISSUE 66: STEAM GENERATOR REQUIREMENTSDESCRIPTIONHistorical Background

Following the steam generator tube rupture (SGTR) event at Ginna in January 1982,<sup>554</sup> the staff proceeded to develop generic steam generator requirements which would help mitigate or reduce steam generator tube degradations and ruptures. To assist the staff, a supporting value/impact analysis was conducted by SAI.<sup>512</sup> The SAI analysis was based on single SGTR events for a typical W design. The W design experience was believed to bound the experience of other NSSS vendors. Based on results of the SAI study and additional staff analyses, the probability of an SGTR event involved in a core-melt accident was determined to be  $3.5 \times 10^{-6}/RY$ .<sup>511</sup> This low probability was the controlling factor in establishing the staff position that a SGTR is not of high risk to the public.

Given that the risk to the public from a SGTR is not high, the proposed staff requirements were assessed for potential reductions in occupational radiological exposure, potential reductions in SGTR frequency and tube degradation, and potential reduction frequency in forced plant outages.<sup>511</sup> The proposed steam generator requirements are identified in Table 3.66-1.

Safety Significance

Steam generator tubes are considered part of the RCS pressure boundary. Steam generator tube leaks, or tube ruptures, provide a direct path for the loss of primary system coolant through the steam generator (secondary system) and to the environment outside the primary containment structure. The two major safety implications are: (1) direct release of radioactive fission products, and (2) loss of RCS cooling water and ECCS water without the capability to recirculate the water as would be the case for LOCAs inside containment.

Solution

The solution to this issue is the implementation by the licensees of those requirements identified as providing adequate mitigation and reduction to steam generator tube degradations and ruptures.

CONCLUSION

Based on the SAI analysis,<sup>512</sup> four of the proposed requirements were recommended for implementation and assigned a high priority ranking. The requirements ranked as high priority were: Item 66.5 combined with Item 66.6, Item 66.1.1, and Item 66.12.

Four of the proposed requirements - Item 66.2.3, Item 66.2.4, Item 66.3, and Item 66.10 - were redirected as Staff Action Items (See Issue 67). These items, if given further staff development, could result in more definitive requirements at a later date.



TABLE 3.66-1

<u>Sub-Item No.</u>	<u>Requirement Title</u>	<u>Status*</u>
66.1	Prevention and Detection of Loose Parts or Foreign Objects	-
66.1.1	Secondary Inservice Inspection and Quality Assurance	NOTE 2
66.1.2	Loose Parts Monitoring System	DROP
66.2	Inservice Inspection Requirements	-
66.2.1	Full length (cold leg) Inspection	NOTE 2
66.2.2	72-Month Inspection Interval	NOTE 2
66.2.3	Supplemental Sampling	Covered in Issue 67
66.2.4	Denting Inspection	Covered in Issue 67
66.3	Improved Eddy Current Techniques	Covered in Issue 67
66.4	Upper Inspection Ports	DROP
66.5	Secondary Water Chemistry Program	NOTE 2
66.6	Condenser Inservice Inspection	NOTE 2
66.7	Standard Technical Specification Limit for Coolant Iodine Activity	NOTE 2
66.8	Primary to Secondary Leakage Limits	NOTE 2
66.9	Stabilization and Monitoring of Degraded Tubes	DROP
66.10	Reactor Coolant System Pressure Control	Covered in Issue 67
66.11	Containment Isolation and Reset	DROP
66.12	Safety Injection Signal Reset	NOTE 2

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\* For a description of the terms used for status, see Table II in the Introduction.



The 72-month inspection interval (Item 66.2.2) and the cold leg inspections (Item 66.2.1), which are sub-requirements of the ISI requirement (Item 66.2), are licensing issues as are the STS requirements on Iodine (Item 66.7) and RCS leakage (Item 66.8). These licensing issues will provide additional assurance that existing regulatory requirements on radiological releases will be maintained and further reduce the potential for the types of SGTR events which are not considered as part of the design basis. The remaining proposed requirements were dropped since they afforded little or no reduction in public risk and would likely result in additional ORE.

In general, the secondary ISI plus QA requirement (Item 66.1.1) is expected to eliminate up to 90 percent of SGTRs that can occur from loose parts in the secondary side. Operating experience has indicated that loose parts have accounted for approximately 50 percent of the SGTRs. The secondary water chemistry program (Item 66.5), in tandem with the condenser ISI (Item 66.6), is expected to reduce the remaining 50 percent of SGTRs and tube degradation by up to 70 percent in severely degraded units. The above 3 requirements (Items), in conjunction with improved eddy current techniques which will be developed further as a staff action item, offer the most effective value/impact ratios and best overall potential reductions (approximately 72%) in SGTRs. The SI reset requirement (Item 66.12) provides added assurance against a loss of defense in depth of the ECCS systems. The SI reset requirement will affect approximately 10 plants and the backfit costs are believed minimal.

Based on the above discussion, the resolution of the issue is available in SECY 84-13<sup>753</sup> and in SECY 85-62.<sup>765,766</sup> The status of each sub-item listed in Table 3.66-1 reflects the current NRR disposition of the proposed requirements.

#### REFERENCES

- 511. "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events," Science Applications, Inc., February 2, 1983.
- 512. Memorandum for D. Eisenhut from T. Speis, "DST Prioritization of Steam Generator Requirements," May 4, 1983.
- 554. Memorandum for D. Eisenhut, et al., from H. Denton, "Development of Generic Recommendations Based on the Review of the January 25, 1982 Steam Generator Tube Rupture at Ginna, May 3, 1982.
- 753. SECY-84-13, "NRC Integrated Program for the Resolution of Steam Generator USI's," January 11, 1984.
- 765. SECY-85-62, "NRC Integrated Program for the Resolution of Steam Generator USI's - Response to Commissioner Comments (Memo from Chilk to Dirks Dated January 23, 1985)," February 22, 1985.
- 766. Memorandum for W. Dircks from S. Chilk, "SECY-85-62 - NRC Integrated Program for the Resolution of Steam Generator USIs -- Response to Commissioners Comments (Memo from Chilk and Dircks Dated January 23, 1985)," March 15, 1985.

ISSUE 67: STEAM GENERATOR STAFF ACTIONSDESCRIPTION

Following the SGTR event at Ginna on January 25, 1982, increased staff effort was placed on developing means to mitigate and reduce steam generator tube degradations and ruptures. To meet these objectives, a dual approach was taken. The first approach was to develop staff requirements to be implemented by the licensees. The proposed staff requirements are evaluated in Issue 66. In addition to these proposed requirements, the staff identified and recommended certain staff actions. The status of the staff actions as determined in this evaluation are listed in Table 3.67-1. For reference purposes, the sub-item number is consistent with the staff action number provided in a DL memorandum.<sup>752</sup> These items are also included in the CRGR review package,<sup>753</sup> and EDO recommendations to the Commission.<sup>753,757,758</sup> The following is a summary of the evaluation of the 16 parts of this issue.

Three of the proposed staff actions should be considered as Licensing Improvement issues:

- 5.1 Reassessment of Radiological Consequences
- 5.2 Reevaluation of SGTR Design Basis
- 10.0 Supplemental Tube Inspections

Two of the proposed staff actions are Regulatory Impact issues that could provide cost-benefits to the NRC and industry:

- 2.1 Integrity of Steam Generator Tube Sleeves
- 8.0 Denting Criteria

The improved Eddy Current Tests (Item 67.7.0) recommendation is ranked as a MEDIUM priority issue principally because of potential reductions in ORE.

Nine of the proposed staff actions are considered part of on-going staff activities and no new staff efforts need be initiated:

- 3.1 Steam Generator Overfill
- 3.2 Pressurized Thermal Shock
- 3.3 Improved Accident Monitoring
- 3.4 Reactor Vessel Inventory Measurement
- 4.1 RCP Trip
- 4.2 Control Room Design Review
- 4.3 Emergency Operating Procedures
- 6.0 Organizational Responses
- 9.0 Reactor Coolant System Pressure Control

The remaining proposed staff action (Item 67.5.3) is in the DROP category and is not recommended for further consideration.

The basis for each of the recommended staff actions is provided in separate evaluations below.

TABLE 3.67-1

<u>Sub-Item</u>	<u>Staff Action</u>	<u>Priority*</u>	<u>Lead Office/ Division/Branch</u>
67.2.1	Integrity of Steam Generator Tube Sleeves	RI	NRR/DE/MEB
67.3.1	Steam Generator Overfill	USI A-47, 1.C.1	NRR/DST/GIB NRR/DSI/RSB
67.3.2	Pressurized Thermal Shock	USI A-49	NRR/DST/GIB
67.3.3	Improved Accident Monitoring	NOTE 3 (MPA A-17)	NRR/DSI/ICSB
67.3.4	Reactor Vessel Inventory Measurement	II.F.2 (MPA F-26)	NRR/DSI/CPB
67.4.1	RCP Trip	II.K.3(5) (MPA G-1)	NRR/DSI/RSB
67.4.2	Control Room Design Review	1.D.1 (MPA F-08)	NRR/DHFS/HFEB
67.4.3	Emergency Operating Procedures	1.C.1 (MPA F-05)	NRR/DHFS/PSRB
67.5.1	Reassessment of Radiological Consequences	LI	NRR/DSI/AEB
67.5.2	Reevaluation of SGTR Design Basis	LI	NRR/DSI/RSB
67.5.3	Secondary System Isolation	DROP	NRR/DSI/RSB
67.6.0	Organizational Responses	III.A.3	OIE/DEPER/IRDB
67.7.0	Improved Eddy Current Tests	MEDIUM	NRR/DE/MTEB
67.8.0	Denting Criteria	RI	NRR/DE/MTEB
67.9.0	Reactor Coolant System Pressure Control	USI A-45, 1.C.1	NRR/DST/GIB NRR/DSI/RSB
67.10.0	Supplemental Tube Inspections	LI	NRR/DL/ORAB

\*For a description of the terms used for priority, see Table II in the Introduction.

ITEM 67.2.1: INTEGRITY OF STEAM GENERATOR TUBE SLEEVESDESCRIPTIONHistorical Background

This item is Recommendation 2.1 of the DL memorandum.<sup>752</sup> The recommendation is for the staff to develop a SRP to clarify staff positions on the materials design, fabrication, installation, examination, and inspection of steam generator tube sleeves.

Safety Significance

At the present time there is no specific SRP to direct the staff/industry reviews related to the design, installation, and inspection of tube sleeves. The SRP would provide an acceptable means to meet GDC 14 and GDC 32 of 10 CFR Part 50, Appendix A.

PRIORITY DETERMINATIONConsequence Estimate

The public risk reduction that can be attributed to this recommendation is not quantifiable. Some small improvement in the effectiveness of the sleeves to perform their intended function (i.e., assure retention of structural integrity of degraded tubes) would result from improved guidance.

Cost Estimate

The major reason for improved guidance is reduced cost. The estimated cost to develop the SRP is 3 man-months of NRC staff time (\$25,000). We estimate that 25% of the operating and planned PWRs (22 plants) will require tube sleeve modifications. The SRP may reduce plant-specific reviews from 2 man-months to 1 man-month. The SRP is expected to also reduce industry man-power requirements by approximately the same amount. The SRP would, therefore, result in a NRC cost saving of \$158,000 and an industry cost saving of \$183,000. The combined NRC and industry cost saving is estimated to be \$341,000.

CONCLUSION

A small public risk reduction is perceived from development of an SRP on steam generator tube sleeves. However, the SRP would be cost-effective in that it would reduce NRC review cost and industry costs associated with the design, installation, and inspection requirements for tube sleeves. The earlier the SRP is developed, the greater the cost saving. Therefore, this issue is classified as a Regulatory Impact issue.

ITEM 67.3.1: STEAM GENERATOR OVERFILLDESCRIPTIONHistorical Background

This item is Recommendation 3.1 of the DL memorandum.<sup>752</sup> The recommendation is that the NRC should select a small number of PWRs representing the PWR spectrum of designs and determine the potential for and consequences of steam generator overfill as a result of a SGTR. This recommendation is closely related to Items 67.5.1, 67.5.2, and 67.9. Further NRC or licensee actions should be determined based on the results of these studies. The recommendation as addressed herein does not consider potential steam generator overfill resulting from control system failures. Steam generator overfill via control systems failures are being evaluated simultaneously under USI A-47. Issue 37 (Steam Generator Overfill and Combined Primary and Secondary Blowdown) and Issue 56 (Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event) are also related issues.

Safety Significance

Following an SGTR, the affected steam generator could fill up to the steamline safety valve due to primary-to-secondary leakage from continued operation of the safety injection pumps. The safety valve may lift at successively lower pressures and fail to fully reseal. The failure to completely reseal could contribute to steam generator overfill by lowering the damaged steam generator pressure, thus raising the differential pressure across the broken tube and sustaining the leakage despite reduced primary system pressure. Failure of the valve to reseal would also provide a direct pathway for release of radioactive primary water to the environment. This sequence of events is beyond the design basis for SGTR events in SRP<sup>11</sup> Section 15.6.3 to establish that the radiological consequences meet 10 CFR Part 100.

For the B&W OTSG design in particular, it may not be possible to stop the primary-to-secondary leakage in an SGTR while maintaining the RCS in a subcooled state. The increased tendency for the OTSG leakage to continue throughout the event is a result of the tubes being directly exposed to the OTSG steam space. Generally, the emergency procedures instruct the operator to discharge steam to the atmosphere or, if available, to the condenser to control level in the damaged SG as necessary. In at least one B&W plant, however, if the water supply for safety injection pumps is approaching a minimum level or if the offsite radiological consequences are becoming excessive, the OTSG is allowed to completely fill, thus terminating the leakage. The number of B&W plants that permit filling of the OTSG is not known at present. We do not believe the potential for prolonged leakage and the associated offsite radiological consequences has been factored into OR or NTOL FSAR SGTR accident analyses. (See Item 67.5.2).

Possible Solutions

Solutions could involve improved RCS pressure control to reduce the differential pressure and leakage across the broken SG tube (primary to secondary), and/or improved EOPs to preclude overfill. The above measures are discussed in response to staff recommendations concerning RCS pressure control and EOPs. (See Items 67.9.1 and 67.4.3). With regard to the concern that the steam lines



cannot support the dead-weight load if the lines are filled with water, additional supports or stronger steam lines could resolve this aspect of the concern.

### PRIORITY DETERMINATION

#### Cost Estimate

The NRC cost would be dependent on the number of PWRs selected for this study and the design variations within this selected group.

#### Other Considerations

Following the Ginna event, concerns were raised relative to the potential for failing the steam lines under the additional dead-weight load, if the steam lines are filled with water as a result of SG overfill. (The Point Beach SGTR, which was a relatively low leak rate, resulted in a near overfill condition.)<sup>755</sup> Should the steam lines fail, the SGTR could become a LOCA outside containment. However, analyses<sup>753</sup> conducted for 4 plants indicate that the steam lines are unlikely to fail under the additional dead-weight load.

Accordingly, the staff's risk analyses<sup>753</sup> assume a conditional probability of steam line break, given an SG overfill, of  $10^{-3}$  which is believed to be reasonably conservative. If the steam lines were redesigned to withstand an overfill condition, the analysis<sup>753</sup> would indicate a reduction in core-melt frequency of  $1.2 \times 10^{-7}/\text{RY}$ .

The consequences resulting from failure of the steam lines by overfilling the steam generators is assumed to involve releases typical of a PWR Category 4 release. Exposure is calculated assuming a typical mid-West meteorology and a population density of 340 persons/square-mile within a 50-mile radius of the plant. The potential public risk reduction is therefore  $[(1.2 \times 10^{-7})(2.7 \times 10^6)]$  man-rem/RY or  $3.2 \times 10^{-1}$  man-rem/RY. Considering an average remaining plant life of 24 years, the public risk reduction is 8 man-rem/reactor.

### CONCLUSION

This item encompasses several considerations related to steam generator overfills and is closely related to staff studies identified in Items 67.5.1, 67.5.2, and 67.9. The primary concern (mitigation of a steam generator overfill) is part of the following on-going staff programs: (1) USI A-47 and (2) NUREG-0737, Item 1.C.1, Emergency Operating Procedures. (See Item 67.4.3). Therefore, the SG overfill issue is covered by the above-stated on-going staff programs.

Rupture of steam lines as a result of a steam generator overfill is a secondary concern predicated on the condition that an overfill occurs. The public risk associated with rupture of the steam lines is low and strengthening of the steam lines is considered a LOW priority.



ITEM 67.3.2: PRESSURIZED THERMAL SHOCKDESCRIPTIONHistorical Background

This item is Recommendation 3.2 of the DL memorandum.<sup>752</sup> The recommendation is that the effects of RCS flow stagnation associated with isolation of a steam generator should be addressed by the on-going Pressurized Thermal Shock program (USI A-49).

Safety Significance

During the Ginna SGTR event, the affected steam generator was isolated and the RCPs were tripped. As a result, the flow in the 'B' Reactor Coolant Loop was reduced to a few hundred gallons per minute while cold high pressure injection water was being injected into the loop. The cold leg piping apparently experienced a cooldown of approximately 260°F in 30 minutes. The reactor vessel apparently did not experience this rapid cooldown since the flow in the cold leg was in the reverse direction, that is, from the reactor vessel towards the steam generator. Other events, as discussed in NUREG-0916,<sup>754</sup> resulting in a steam generator isolation and continued safety injection could result in adding cold water to the reactor vessel.

CONCLUSION

The probability, consequences, and resolution of such events are being addressed in USI A-49.

ITEM 67.3.3: IMPROVED ACCIDENT MONITORINGDESCRIPTIONHistorical Background

This item is Recommendation 3.3 of the DL memorandum.<sup>752</sup> The recommendation is to address the accident monitoring weaknesses of the type observed at Ginna by implementation of Regulatory Guide 1.97<sup>55</sup> and the Safety Parameter Display System.

Safety Significance

During the event at Ginna, several weaknesses in accident monitoring were apparent. These include: (1) non-redundant monitoring of RCS pressure, (2) failure of the position indication for the steam generator relief and safety valves, and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. These conditions make it more difficult for correct operator action in response to such events.

Possible Solution

Had Regulatory Guide 1.97<sup>55</sup> been implemented at Ginna before the January 1982 event, the monitoring of the event would have been substantially improved and :

there would have been more assurance of correct operator actions. Improved accident monitoring would also have improved the NRC's ability to assess the plant status and the appropriateness of the licensee's actions and recommendations.

### CONCLUSION

The recommendation was resolved by MPA A-17 and the resolution was issued in Supplement 1 to NUREG 0737 (Generic letter No. 82-33).<sup>376</sup>

### ITEM 67.3.4: REACTOR VESSEL INVENTORY MEASUREMENT

#### DESCRIPTION

##### Historical Background

This item is Recommendation 3.4 of the DL memorandum.<sup>752</sup> The staff recommends implementation of TMI Action Plan Item II.F.2 because it would have substantially improved the Ginna situation by ensuring that steam bubble formation in the reactor vessel upper head could be more accurately monitored.

##### Safety Significance

During the Ginna SGTR event, the formation of a steam bubble in the reactor vessel upper head significantly complicated the course of the event. The uncertainty about the size was a significant factor in the operator's decisions to continue safety injection beyond the point when termination is called for in the emergency procedures.

##### Possible Solution

Implementation of NUREG-0737,<sup>98</sup> Item II.F.2.

### CONCLUSION

Following Commission approval for implementation of Item II.F.2, letters to individual licensees and orders to B&W licensees and ANO-2 were issued on December 10, 1982.<sup>491</sup> This issue is part of Item II.F.2 which is being implemented as MPA F-26.

### ITEM 67.4.1: REACTOR COOLANT PUMP TRIP

#### DESCRIPTION

##### Historical Background

This item is Recommendation 4.1 of the DL memorandum.<sup>752</sup> The recommendation is that the NRC staff should develop requirements for licensees to provide RCP trip criteria that will ensure continued forced RCS flow during steam generator tube breaks up to and including the design basis tube rupture.

Safety Significance

Analyses indicate that continued operation of the RCPs following a range of small LOCAs could lead to excessive inventory loss for which the high pressure injection system would be unable to compensate. Generally, the range of break size of concern is from 0.02 to 0.2 ft<sup>2</sup> (2 to 5 inches equivalent diameter). The interim position (documented in NUREG-0623)<sup>97</sup> requires manual monitoring of the RCPs on the symptoms of a small LOCA (i.e., a safety injection signal and low RCS pressure).

CONCLUSION

This recommendation is being developed under NUREG-0737, Item II.K.3(5)<sup>98</sup> and is being implemented as MPA G-1.

ITEM 67.4.2: CONTROL ROOM DESIGN REVIEWDESCRIPTION

This item is Recommendation 4.2 of the DL memorandum.<sup>752</sup> As a result of a review of the Ginna control room following the tube rupture, several items related to the event were identified that are contrary to good human factors engineering principles. These items should be reviewed by HFEB as part of the detailed control room design review required by NUREG-0737.<sup>98</sup> This information should be used in the basis for a study to determine what changes can be made to improve control room designs.

CONCLUSION

It has been determined that items identified at Ginna have been covered in the work to be done for the TMI Action Plan Item I.D.1 control room reviews, thus assuring that these items will be factored into all Item I.D.1 control room design reviews. This recommendation will be resolved as part of NUREG-0737,<sup>98</sup> Item I.D.1 and is being implemented as MPA F-08.

ITEM 67.4.3: EMERGENCY OPERATING PROCEDURESDESCRIPTIONHistorical Background

This item is Recommendation 4.3 of the DL memorandum.<sup>752</sup> The purpose is to ensure that newly-developed EOPs consider the experiences from the Ginna SGTR event. PSRB should review the items listed below prior to emergency procedure implementation for inclusion in its review plan. This staff effort should be considered in conjunction with ongoing work under NUREG-0737,<sup>98</sup> Item I.C.1.

- ° RCP Restart
- ° Availability of Faulted SG Safety and Relief Valve

- Multiple and Second Order Failures
- Bubble Formation
- Cooling Faulted SG
- Cooling Intact SG
- Safety Injection Pump Termination and Restart Criteria
- Procedure Format and Clutter
- Criteria for Natural Circulation Determination
- Accommodation of Plant Differences from Reference Plant in Emergency Procedure Development
- Rapid Determination and Isolation of Faulted SG and Timely Depressurization of RCS to Minimize RCS Inventory Loss and Releases
- MSIV Closure During Plant Cooldown
- Use of Charging and Letdown Systems
- Operation of the RCP in the Damaged Loop
- Operation of Loop Isolation Valves
- Use of Pressurizer PORV
- Potential Complicating Events
- Site-Specific Operator Training
- SG Level Control for CE Plants

#### Safety Significance

The above list includes transients and plant conditions that form the basis of many of the emergency procedures, reliability analyses, human factors engineering, crisis management, and operator training. Plant conditions may exist in addition to those pertinent to design bases which could prevent proper operator actions during such events/conditions and possibly pose a serious threat to reactor safety.

#### Possible Solution

The solution to this recommendation is to consider the Ginna event in the development of EOPs.

#### PRIORITY DETERMINATION

Guidance for the evaluation and development of procedures for transients and accidents is covered by Item I.C.1 of NUREG-0737.<sup>98</sup> Some of the items in the above list are explicitly included in the review requirements of Item I.C.1. Other items in the list are believed to be implicitly within the intent of Item I.C.1 in that the availability of systems under expected conditions (like Ginna) should be used in developing diagnostic guidance for operator and procedural development.

#### CONCLUSION

This recommendation is covered in Item I.C.1 of NUREG-0737<sup>98</sup> and is being implemented as MPA F-05.

ITEM 67.5.1: REASSESSMENT OF RADIOLOGICAL CONSEQUENCESDESCRIPTIONHistorical Background

This item is Recommendation 5.1 of the DL memorandum.<sup>752</sup> The recommendation is that SGTR accidents should be reassessed by NRC staff to determine the effects of releases made for periods substantially longer and via other release points than those previously analyzed. These analyses should specifically address the applicability of the assumptions in SRP<sup>11</sup> Section 15.6.3 and address the costs and benefits of requiring revised analyses by licensees. This issue is closely related to Items 67.5.2 and 67.3.1.

Safety Significance

Public risk from an SGTR, even considering steam generator overfill, is considered low for typical PWRs. This low risk is expected to remain valid even if new source term results are applied. However, the safety significance of this issue is derived from concern over the number of SGTR events and potential for exceeding the bounds of the analyses that are currently required in SRP<sup>11</sup> Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR Part 100.

PRIORITY DETERMINATION

SRP<sup>11</sup> Section 15.6.3 does not address a steam generator overfill in the SGTR scenario. In addition, termination of the leak from an SGTR within 30 minutes, as assumed in typical PWR FSARs, may be non-conservative and not consistent with operating experiences. Therefore, implementation of this recommendation will allow the staff to upgrade SRP Section 15.6.3 and provide a better understanding and means to assess future SGTR events in operating plants relative to the consequence limits in 10 CFR Part 100.

Information generated from implementation of this recommendation will also assist the licensees in their understanding of similar events and help determine the course of action needed to mitigate the consequences of SGTRs and overfilling of the steam generators.

CONCLUSION

Implementation of this recommendation is not expected to result in significant overall risk reduction for the public. Therefore, with regard to potential risk reduction to the public, this recommendation is considered low priority. However, AEB considers this recommendation a Licensing Improvement issue and recommends the reassessment. DST agrees that a "best estimate" analysis modeled after plant experiences, like Ginna, could be beneficial in more realistically determining the risk and conservatisms inherent in the current SRP requirements. If this limited scope comparison of the SRP model with a best estimate analysis is followed, this issue could be considered as an improvement to current licensing positions (a licensing issue.)

ITEM 67.5.2: REEVALUATION OF SGTR DESIGN BASISDESCRIPTIONHistorical Background

This item is Recommendation 5.2 of the DL memorandum.<sup>752</sup> The recommendation is that NRC should reevaluate and consider reclassifying or redefining the design basis SGTR event. This issue is closely related to issues being addressed under Items 67.3.1 and 67.5.1.

A SGTR accident is one of the events for which the NRC requires a safety analysis to show that the reactor will respond in an acceptable manner and that the health and safety of the public is adequately protected. The SGTR accident is the loss of integrity (development of a leak) in a steam generator tube (or tubes) so that reactor coolant water from the primary system flows into the secondary water in the steam generator. This provides a potential path for the release of radioactivity to the environment.

As analyzed in SARs, the event is a break of a single steam generator tube with flow out of the full flow area of both ends of the steam generator tube at the break. The reactor is assumed to be at full power at the time of the accident.

The SGTR accident serves as the design basis for allowable reactor coolant activity since the amount of radioactivity released to the environment is directly proportional to the amount of activity in the coolant. The analysis of this event in SARs is intended to bound the potential release of radioactivity, should a SGTR occur. The behavior of reactor systems during this event has not traditionally received much emphasis, either in the analyses reported by the licensees or during review by the NRC.

Safety Significance

The safety significance of this recommendation is derived from concern over the number of SGTR events and the potential for exceeding the bounds of the analyses that are currently required in SRP<sup>11</sup> Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR Part 100.

PRIORITY DETERMINATION

The analysis of an SGTR is performed to bound potential offsite doses using many conservative assumptions (i.e., accident terminated within 30 minutes) to maximize the predicted doses (SRP Section 15.6.3).<sup>11</sup>

The probability of the simultaneous occurrence of the SRP conditions is extremely low. SGTR events have occurred at a frequency of approximately  $2 \times 10^{-2}$ /RY. This event might therefore be classified as an incident which may occur during the lifetime of a particular plant.

SGTR events which have actually occurred were not as severe as the SRP design basis event. Had the frequencies of the conservative assumptions been included in a calculation of a design basis frequency, a much lower frequency would result. A change in classification would necessarily require changes to the



conservative analysis assumptions (listed in the SRP). Changes to the design basis assumptions may include more conservative limits on the reactor coolant activity for those plants that do not have STS limits on coolant iodine concentrations, inclusion of SGTR overfill conditions, multiple ruptures of the steam generator tubes, and other conditional failure scenarios.

### CONCLUSION

The general basis for Item 67.5.2 is derived from the number of SGTR events that have occurred and the potential existing for SGTR doses exceeding 10 CFR Part 100 guidelines. However, these doses would occur only if there were an unlikely (but not impossible) set of circumstances as discussed in detail in Section 8.1 of NUREG-0916.<sup>754</sup>

For the 4 SGTRs that have occurred in domestic operating reactors, no significant consequences (doses) to the public have occurred and the existing design basis SGTR has proven to be adequate.

At the present time, and in regard to the safety significance of this issue, we believe it is premature to establish a priority for reclassification of the design basis SGTR event, prior to obtaining the results from other Staff Actions (See Item 67.5.1). Until the results from Item 67.5.1 are obtained, this issue should be considered a Licensing Issue.

### ITEM 67.5.3: SECONDARY SYSTEM ISOLATION

#### DESCRIPTION

##### Historical Background

This item is Recommendation 5.3 of the DL memorandum.<sup>752</sup> The recommendation is that the NRC should reevaluate the provisions for isolating the steam generators in conjunction with Items 67.3.1 and 67.5.1. The evaluation should consider whether the current provisions for isolating the main steam and feedwater lines are adequate with particular emphasis on isolation of the steam generator with RCS loop isolation valves, utilizing closed bonnet secondary safety valves or containing the discharge from the steam generator safety and relief (atmospheric dump) valves.

##### Safety Significance

The primary safety significance of SGTR events is the potential for a direct path for a loss of radioactive coolant from the RCS through the steam generator to outside the containment. This event could also increase the probability of a core-melt because the reactor coolant leaking from a steam generator tube cannot be recirculated. Other systems that penetrate the containment and communicate either with the RCS or the containment have two containment isolation valves that close automatically or are locked closed. The steam generator safety and atmospheric valves open automatically and, as required by the ASME Code, cannot be isolated.

### Possible Solution

Some of the older PWRs have block valves in the reactor coolant loops that could be used to isolate the steam generators and prevent the loss of coolant and radioactivity from the RCS. Alternatively, the discharge from the steam generator safety and relief valves could be routed to return to the containment or a quench tank. GDC 57 currently requires each line that penetrates containment and is neither part of the RCS nor connected to the containment atmosphere to have at least one isolation valve that is locked closed, automatic or capable of remote operation. GDC 57 is not currently interpreted to apply to the valves on the steam generator. However, some improved means of isolating the steam generator, possibly either by requiring loop isolation valves in the RCS or containment of the safety valve discharge, could be considered.

### PRIORITY DETERMINATION

Recommendation 8 of NUREG-0651<sup>755</sup> states: "For those plants provided with loop isolation valves, the use of these valves following an SGT rupture should be investigated. Isolating the affected loop would provide an almost immediate abatement of SGT leakage, but would prohibit cooldown of the damaged SG. Licensees should, therefore, examine the advantages and disadvantages in their plant of loop isolation."

As pointed out in NUREG-0651,<sup>755</sup> the determination and isolation of the damaged SG appears to be taking longer than the assumed 30 minutes in the FSAR analysis. In this regard, Item 67.5.1 could address this aspect of SG isolation.

The EOPs involved with isolation of the secondary system following an SGTR have already been identified in Item 67.4.3 as selected events for staff review. In isolating the SG, the operator's worst error could be isolating the wrong steam generator. If this were to occur, overfill of the broken steam generator could still result. In addition, the intact steam generator which is isolated could boil dry. Saturated conditions in this hot leg could result. When the operator recognizes the error, isolates the faulted steam generator, and opens the intact steam generator, he might have no steam generator cooling since natural circulation might have become inhibited through the intact steam generator due to void formation. The faulted steam generator is now isolated, resulting in minimal transfer of heat. He could unisolate the faulted steam generator and steam either to the condenser (if available) or to the atmosphere, but this would result in increased offsite doses.

The W SGTR guidelines contain a note which advises the operator not to use these loop isolation valves in the event of an SGTR. It goes on to state that "any use of LSIV's (Loop Stop Isolation Valves) must be justified on a plant specific basis." W reasons for not using these valves are: (1) their use has not been included in any accident analyses, (2) they are not meant to be safety components, (3) their use has not been recommended since steam generator isolation has not been shown necessary to limit releases to an acceptable value, (4) the valves are very slow acting, taking on the order of minutes to close, and (5) their subsequent reopening required a rather careful procedure.

### CONCLUSION

Many PWRs do not have these valves for use in an SGTR accident. For those plants that have LSIVs, modifications would likely be required.

However, based on the above discussions, the valves do not appear to be necessary. In each of the SGTR events which have occurred, the operator took correct action and in none of the events did incorrect action result in any significant adverse effect to the public. In each event, the SGTR was isolated to the faulted steam generator. This issue is considered a DROP issue.

#### ITEM 67.6.0: ORGANIZATIONAL RESPONSES

##### DESCRIPTION

###### Historical Background

This item is Recommendation 6.0 of the DL memorandum.<sup>752</sup> The recommendation is to establish as soon as possible improved NRC emergency preparedness to handle nuclear accidents at licensed reactor facilities.

###### Safety Significance

In the event of a nuclear accident, improved NRC emergency preparedness procedures will enable NRC to monitor and evaluate the situation and potential hazards, advise the licensee's operating staff as needed, and, in an extreme case, issue orders governing such operations.

###### Possible Solution

Resolution of this item centers around implementation of the TMI Action Plan Item III.A.3.

##### CONCLUSION

This item is part of the TMI Action Plan Item III.A.3.

#### ITEM 67.7.0: IMPROVED EDDY CURRENT TESTS

##### DESCRIPTION

###### Historical Background

Improved Eddy Current Tests (ECT) were originally proposed by the staff as requirements to be implemented by the licensees. Improved ECT could enhance earlier detection of degradations and thereby minimize, or mitigate, steam generator tube degradations and ruptures. The evaluation of improved ECT as a requirement (Item 66.3) concluded that use of current state-of-the-art improvements provided only small reductions in public risk. Likewise, since ECT is an evolving technology, it was determined to be premature to impose a requirement at this time. However, it was also recognized that significant potential reductions in ORE could result from use of improved ECT. Therefore, this item was believed to warrant a medium priority ranking. This ranking infers that staff efforts should be planned, but should not interfere with pursuit of high priority issues. The Item 66.3 conclusion is consistent with the position that

improved ECT should be handled as a Staff Action item and developed in accordance with the possible solution described below.

#### Safety Significance

The SG tube that ruptured at Ginna exhibited no ECT indication during earlier testing. Improved ECT techniques would most likely have given ECT indications and avoided the SGTR event at Ginna.

#### Possible Solution

This effort, conducted in parallel with ongoing ASME Code Committee activities, would incorporate updated eddy current inspection procedures in the ASME Boiler and Pressure Vessel Code, Sections V and XI for NDE and ISI, respectively. The improved test procedures would be considered part of the in-service eddy current testing of PWR steam generator tubing.

#### CONCLUSION

In a previous evaluation<sup>756</sup> by the staff, it was determined that improved ECT techniques would provide small reductions in public risk and was therefore ranked as low priority relative to public risk reduction. It was also concluded that significant reductions in ORE could result from use of improved ECT techniques. The priority ranking based on the ORE reduction potential was medium. Improved ECT would also enhance the certainty that defective or degraded tubes would be identified and removed from service to assure meeting 10 CFR 100 release limits. The latter condition could be argued to classify improved ECT as a licensing improvement issue. In either classification, an economic incentive for use of improved ECT of up to \$5M/plant based on avoided cost of forced outages could be obtainable. Based on a combination of the above potential benefits, development of improved ECT procedures is recommended as a MEDIUM priority principally because of the potential reductions in ORE.

#### ITEM 67.8.0: DENTING

##### DESCRIPTION

##### Historical Background

This item concerns a staff recommendation to develop generic inspection criteria and methods to quantify steam generator tube denting. Operating experience has shown that surveillance of steam generator tubes is necessary to identify denting and to take corrective action to mitigate the stress corrosion cracking induced by denting.

##### Safety Significance

Denting can enhance stress corrosion cracking leading to through-wall cracks and leaks in steam generator tubes. Denting, combined with flow slot hourglassing, caused the U-bend stress corrosion cracking that led to the SGTR at Surry Unit 2 in September 1976.

Possible Solution

Development of a generic inspection requirement and criteria for steam generator tube denting will provide assurance that minimum standards for denting are applied uniformly.

PRIORITY DETERMINATIONFrequency Estimate

Only one SGTR event has been attributed to the denting phenomena in approximately 300 years of reactor operation. This corresponds to a SGTR frequency of  $3 \times 10^{-3}/\text{RY}$ . The SGTR contribution to a core-melt frequency of  $4.7 \times 10^{-6}/\text{RY}$  therefore contains a contribution of approximately 15 percent ( $7 \times 10^{-7}/\text{RY}$ ) due to denting.

Consequence Estimate

The PWR Category 4 release of  $2.7 \times 10^6$  man-rem is used to estimate the consequences of a core-melt associated with an SGTR. Using the above frequencies, the public risk, annualized over a remaining plant life of 24 years, yields a public risk of  $[(7 \times 10^{-7})(2.7 \times 10^6)(24)] = 45$  man-rem/plant. If we assume that approximately 40 of the operational and planned PWRs (~90 plants) have or will experience denting problems, the total public risk is approximately 1,800 man-rem. Assuming a 30% reduction due to improved denting surveillance criteria results in a total public risk reduction of 13.5 man-rem/plant and 540 man-rem for 40 plants.

Cost Estimate

Industry Cost: It is estimated that, as a minimum, use of generic denting criteria from the STS, the industry cost benefit will parallel the NRC cost benefit.

NRC Cost: The estimated NRC cost to develop the denting criteria is based on 3 man-months of effort at \$100,000/man-year. This cost is \$25,000. The implementation mechanism is assumed to be a revision to the STS. It is assumed that the denting criteria in the STS will apply to NTOL and CP plants and those operating plants that experience denting problems. Using the same ratio (40/90) as used in the above risk determination, 40 of the total of 90 plants will require implementation of the STS denting criteria. It is also estimated that development of a generic denting criteria will reduce NRC plant-specific review time by 2 man-weeks/plant. The result is a cost savings of  $(40)(2)(\$1,920)$  or \$153,600. The net cost benefit to the NRC is therefore approximately \$128,600.

Based on the above assumptions, the total cost derived from development of a generic denting criteria is a total net cost benefit of approximately \$250,000.

Value/Impact Assessment

The public risk reduction associated with implementation of generic denting criteria is not significant. The major value in development of the generic denting criteria is that it may provide a net cost benefit to the NRC and industry. No negative impacts (adverse changes to existing plant-specific criteria) are assumed in this evaluation.



CONCLUSION

In consideration of low potential public risk reduction, development of generic denting criteria is considered low priority. However, the generic denting criteria provide a small public risk reduction potential and should result in a net cost reduction for the NRC and industry. Therefore, subject to the above implementation assumptions, development of the generic denting criteria is a Regulatory Impact issue.

ITEM 67.9.0: REACTOR COOLANT SYSTEM PRESSURE CONTROLDESCRIPTIONHistorical Background

This item addresses Recommendation 9 of the DL memorandum<sup>752</sup> and calls for a study to determine the need for controlling and reducing RCS pressure during and following an SGTR with emphasis on existing plant systems and equipment. The spectrum of possible initial conditions, RCS thermal-hydraulic conditions, and break sizes should be considered. The use of the pressurizer auxiliary system should be explicitly examined since its use may eliminate the necessity to use the pressurizer PORV in cases where forced RCS flow has been lost. The study should address the following objectives: (1) minimizing the primary to secondary leakage through the broken steam generator tube, (2) maximizing control over system pressure, and (3) minimizing the chances of producing voids in the RCS and other complicating effects.

Safety Significance

RCS depressurization following an SGTR is more difficult because of the loss of normal pressurizer spray. RCS fluid contraction, caused by the cooldown from the dumping of secondary-side steam to either the main condenser or to the atmosphere, will result in some reduction in RCS pressure, but other measures must be taken to expeditiously reduce the RCS pressure to the point where primary coolant flow into the damaged steam generator stops. The pressurizer PORV was used during the Ginna and Prairie Island SGTR events to reduce RCS pressure. However, control of RCS pressure is difficult with the PORV since its use creates an additional loss of coolant. The decrease in RCS pressure can be so rapid that steam voids may be formed in the reactor vessel upper head and at the top of the steam generator U-tubes and may further complicate the RCS depressurization. Void formation can lead to concerns regarding core cooling. The Ginna operators were sufficiently concerned that they left the safety injection pumps operating, thereby overfilling the steam generator via primary-to-secondary leakage through the ruptured tube. The resulting secondary-side pressure transient caused the main steam safety valves to lift, releasing radioactive material directly to the atmosphere. It is not apparent that the auxiliary spray from the charging system could have successfully lowered RCS pressure to the point where primary coolant flow into the steam generators could have been stopped. It may have been that, by spraying cold charging fluid into the pressurizer, the decrease in pressure would have resulted in void formation thus expanding the RCS fluid volume, filling the pressurizer, and rendering further spray flow ineffective. This phenomenon should be examined as well as the thermal stresses on the spray nozzle.



Possible Solution

With optimized RCS pressure control, risk associated with an SGTR may be reduced by reducing the potential radiological consequences.

PRIORITY DETERMINATIONFrequency Estimate

Independent analyses by the staff considered three categories of SGTR events: (1) SGTR and loss of DHR, (2) SGTR resulting from LOCA, and (3) SGTR with loss of secondary system integrity. For Categories 1 and 2 above, the core-melt probability was not dominated by SGTRs. The core-melt probabilities calculated for categories (1) and (2) were  $5.5 \times 10^{-7}/\text{RY}$  and  $3.0 \times 10^{-8}/\text{RY}$ , respectively.

Category 3 included single and multiple tube ruptures followed by stuck open SG safety valves, MSLB, failure of the MSIVs, SG overfill, and failure to depressurize the RCS before the RWST was exhausted. The latter was considered since recirculation water from the sump might not be available following a SGTR event should a loss of secondary system integrity (e.g., stuck open safety valve, MSLB) occur outside containment.

We assumed that RCS pressure control would enhance depressurization of the RCS by a factor of 10 for the Category 3 sequences involving less than 10 SGTRs. For greater than 10 SGTRs, the depressurization is assumed to be too rapid for the RCS pressure control to be effective. The result is a reduction in core-melt frequency of  $1.8 \times 10^{-6}/\text{RY}$  for enhanced RCS pressure control.

Consequence Estimate

The consequences (doses) resulting from an SGTR would involve releases typical of a PWR Category 4 release as used in WASH-1400<sup>16</sup> and modified to a typical meteorology with a population density of 340 persons/square-mile within a 50-mile radius. The public risk reduction is  $(1.8 \times 10^{-6})(2.7 \times 10^6)$  man-rem/RY or 4.9 man rem/RY. Considering an average remaining plant life of 24 years, the annualized public risk reduction is 117 man-rem/reactor.

Cost Estimate

NRC Cost: The cost of the recommended separate staff study depends on the present capability for RCS pressure control following an SGTR and the incremental improvement required. As a minimum, the study may require a review and documentation of how existing systems and procedures already provide the requisite capability. In some plants, the study may require thermal-hydraulic modeling of the primary and secondary coolant systems as well as detailed stress analysis of selected components such as the pressurizer auxiliary spray nozzle. A study of this depth and the development of an optimized approach for RCS pressure control could cost on the order of one man-year (\$100,000) or more.

TMI Action Plan Item I.C.1, clarified in NUREG-0737,<sup>98</sup> has within its scope the development of EOPs for accidents and transients including multiple SGTRs. Likewise, the USI A-45 study is also developing the adequacy of current and alternate means of satisfying LWR shutdown decay heat removal requirements. The USI A-45 study will also be looking into shutdown requirements imposed by

SGTRs in PWRs. Therefore, ongoing NRC studies, if properly coordinated, would negate the need for a separate study on RCS pressure control.

Industry Cost: The major cost of the study, as recommended, would be borne by the NRC and its contractors. However, input and consultation with specific plants, plant types, or perhaps separate PWR owners' groups would be involved. In the latter case, NSSS owners' groups are currently evaluating means of controlling reactor coolant pressure during an SGTR. The depth and scope of the steam generator owners' group (SGOG) study can be expected to at least parallel the above NRC (study) cost.

The cost of implementing an optimized approach for RCS pressure control is likely to be highly variable, depending on the adequacy of the present RCS pressure control capability and the differences between the present and the optimized approach. The cost associated with implementing an optimized approach for RCS pressure control is not presently quantifiable, but may include some or all of the following items of cost: (1) developing, validating, and implementing new emergency procedures; (2) training plant operators; or (3) replacing equipment or upgrading equipment qualification if existing equipment must be operated outside of the conditions for which it was originally designed and qualified. In the present scope of the recommended study, the implementation cost is moot. However, in an overall value/impact, the implementation cost could be significant.

#### Value/Impact Assessment

The value of the recommended NRC staff study on Reactor Coolant System Pressure Control is that it may uncover, or result in development of, optimized means (procedures, equipment, instrumentation) to control reactor coolant pressure to minimize primary to secondary leakage following an SGTR. Thus, the potential for overfilling a steam generator and the quantity of radioactive material released directly to the atmosphere following an SGTR should be reduced.

Based on the above frequency and consequence estimates, the value is a potential public risk reduction of 117 man-rem/reactor over an average remaining plant life of 24 years. The major initial impact is the cost of performing the study. Subsequent impacts will depend on the results of the study and cannot be quantified at the present time.

#### CONCLUSION

Based on the above, the potential public risk reduction of 117 man-rem/reactor that may be derived by a separate (new) NRC study on RCS pressure control is not highly significant. The potential value which would result from such a study would most likely be improved RCS pressure control for both accidents and transients. In this regard, current staff actions being developed under TMI Action Plan Item I.C.1 and USI A-45 would also resolve the objective of this issue. In addition, the on-going work by the SGOG on RCS pressure control could be factored into the on-going TMI Action Plan Item I.C.1 and USI A-45 reviews.

In summary, in view of the above findings, RCS pressure control is considered part of ongoing studies of Item I.C.1 of NUREG-0737<sup>98</sup> (being implemented under MPA F-05) and USI A-45.

ITEM 67.10.0: SUPPLEMENTAL TUBE INSPECTIONSDESCRIPTION

Supplemental Tube Inspection (STI) was originally proposed by the staff as a recommended licensee action.<sup>752</sup> The value/impact analysis<sup>756</sup> ranked the proposed staff recommendation as a licensing issue. This ranking inferred that the staff-proposed STI would provide only small potential public risk reductions and a low value/impact ratio. However, as a minimum, the statistical sample size of the proposed STI would ensure that no more than the limiting number of defective tubes would go undetected. The limiting number of sample tubes to be inspected would be based on meeting 10 CFR Part 100 release limits from and concurrent with a MSLB. Thus, STI would provide additional assurance that existing regulatory requirements on radiological releases would be maintained and further reduce SGTRs. Subsequent information<sup>753</sup> from industry indicated that the staff-proposed STI would result in higher costs and greater ORE than that previously estimated by the staff. The staff reevaluated<sup>753</sup> their proposed STI and agreed in part with the industry assessment. However, it is also the current staff position that some form of STI can be formulated that would provide added assurance of tube integrity with less ORE and an improved value/impact relationship.

In view of the above, the STI was dropped as an issue for licensee implementation and categorized as a licensee issue for further staff action and reevaluation.

CONCLUSION

Based on the above discussion, the STI is recommended as a Licensing Issue staff action to investigate more practical alternatives for STI.

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- 758. SECY-84-13B, "NRC Integrated Program for the Resolution of Steam Generator USI's - Response to Commission Comments (Memo from Chilk to Dircks dated September 13, 1984)," November 5, 1984.

ISSUE 84: CE PORVsDESCRIPTIONHistorical Background

This issue was raised by an ACRS memorandum<sup>551</sup> to the Commission in October 1983. Following the TMI-2 accident, the purpose and use of PORVs has been the subject of considerable analyses and discussions. The original purpose for which PORVs were installed was to prevent challenges to the spring-operated safety valves. However, plants designed by W and B&W sometimes rely on the PORV to depressurize the plant in certain design basis events such as a steam generator tube rupture. Another use of the PORVs for some plants is to provide low temperature overpressure protection. A more in-depth discussion on the use of the PORVs in various modes of plant operations is provided in Issue 70.

All W and B&W PWRs have at least one PORV included in their designs. Older CE plants also have PORVs, but the current designs by CE do not include PORVs. The two groups of CE-designed PWRs without PORVs are the 3410 Mwt plants (San Onofre Units 2 and 3 and Waterford Unit 3) and the 3800 Mwt plants (Palo Verde Units 1, 2, and 3 and other CE System 80 plants). Although Arkansas Nuclear One (ANO) Unit 2 also does not have a PORV, it is not part of this issue since a large, manually-actuated vent valve is presently installed on the ANO-2 pressurizer that could provide rapid depressurization capability.

Safety Significance

The staff's review<sup>738</sup> of this issue indicated that the current designed CE plants without the PORVs met regulatory requirements, but other considerations, primarily accident management for beyond design basis events and potential core-melt risk reduction, may require further study. The events for which PORVs could prove to be of benefit are of low probability and the staff is not aware of any immediate safety concerns associated with the absence of PORVs in CE plants. Therefore, a decision on CE PORVs was deferred pending resolution of USI A-45.<sup>739</sup>

CONCLUSION

This issue is deferred pending resolution of USI A-45. Therefore, a resolution has been identified.

REFERENCES

551. Letter to N. Palladino from J. Ray, "Need for Rapid Depressurization Capability in Newer Combustion Engineering, Inc., Plants," October 18, 1983.

738. NUREG-1044, "Evaluation of the Need for a Rapid Depressurization Capability for CE Plant," U.S. Nuclear Regulatory Commission, December 1984.



739. SECY-84-134, "Power Operated Relief Valves for Combustion Engineering Plants," March 23, 1984.

## ISSUE 97: PWR REACTOR CAVITY UNCONTROLLED EXPOSURES

### DESCRIPTION

#### Historical Background

Over the past several years, the NRC staff has noted an unacceptably large number of overexposures and uncontrolled exposures associated with pressurized water reactor cavity entries while incore detectors were withdrawn. In spite of industry efforts and past regulatory efforts, including OIE Circulars and Information Notices, Regional inspections, and civil penalties, these events continue to occur.

#### Safety Significance

The incore detectors referred to above are usually miniature fission chambers used for neutron flux mapping within the core. These detectors contain small amounts of highly enriched uranium. Such detectors become highly radioactive when inserted into the reactor core. When retracted out of the core, they can produce intense radiation fields in the cavity beneath the reactor vessel. These fields vary from 100 to 2000 R/hour or more in the events tabulated.<sup>716</sup> A worker entering such a field will reach his annual exposure limit in less than three minutes. Moreover, any mishap or delay could easily result in serious radiation injury or death.

#### Possible Solutions

In a draft<sup>716</sup> generic letter, it was proposed that access to the reactor cavity be controlled by a single key lock, which can be opened only with the direct concurrence of two relatively high management officials. It should be noted that, in the events tabulated,<sup>716</sup> the overexposed individual (e.g., the shift supervisor) was often the individual administratively in charge of controlling reactor cavity entries. This possible solution is included only for the purpose of obtaining a preliminary cost estimate. Alternative solutions<sup>718,719</sup> have been proposed.

### PRIORITY DETERMINATION

#### Frequency/Consequence Estimate

In the draft<sup>716</sup> generic letter, it was estimated that overexposure events occur about once a year and the incidents tabulated resulted in average exposure of 5 rem to an individual. Such an exposure rate (5 man-rem/year industry-wide) would, by itself, not justify placing this item in any category above low priority.

However, the issue is wider than this. There is a finite probability of an exposure being great enough to result in injury or death. The probability of a fatality cannot be estimated with available information; it will be necessary to use some judgement. We will assume that there is a 0.1% chance of a fatality per event. This is, of course, simply an educated guess that, in 1000 cavity

entries, a mishap will occur which will delay an individual in a high radiation field long enough to accumulate 500 rem. (For a 2000 R/hour field, this is only 15 minutes.) This is intended only as an order of magnitude estimate, i.e., we expect that 100 events without mishap should be possible, but 10,000 events without mishaps are unlikely.

The consequences of such an event do not relate directly to the usual consequence measure of total whole-body man-rem to the public. If we assume that an occupational fatality is no less undesirable than a fatality to the general public, it is possible to find a public hazard equivalent to the 0.001 fatality per year estimated for this issue.

CRAC2<sup>64</sup> calculations show a public exposure of  $5.59 \times 10^6$  man-rem and 215 early fatalities for a PWR-2 event. This implies that one fatality is equivalent to roughly 26,000 man-rem for a large-scale accident. The 0.001 fatality/year we estimate for this issue is equivalent in fatality risks to 26 man-rem/year. For a 30-year reactor lifetime, this is roughly 800 equivalent man-rem (total, all reactors).

Currently, there are 53 PWRs in operation, with a total operating history of 457.01 PWR-years. There are also about 30 PWRs not yet in commercial operation. Assuming a 30-year plant life, the estimated risk is equivalent to 12 man-rem/reactor or 1,000 man-rem for all reactors.

#### Cost Estimate

Industry Cost: The estimated<sup>716</sup> industry costs per PWR are \$500 for a lock and associated labor and \$2,000 for related paperwork and procedural changes.

NRC Cost: NRC costs are estimated to be about equal to licensee paperwork or \$2,000.

Total costs are then \$4,500/plant.

#### Value/Impact Assessment

Based on a public risk reduction of 12 man-rem/reactor and a cost of \$4,500/plant, the value/impact score is given by:

$$S = \frac{12 \text{ man-rem/reactor}}{\$0.0045\text{M/reactor}} \\ \cong 2,700 \text{ man-rem/\$M}$$

#### Other Considerations

The RAB comments<sup>718</sup> suggest that a subcategory of up to 25% of PWRs may be significantly more at risk for a fatality due to procedural and hardware variations. If a subcategory could be identified and resolution efforts targeted at these specific plants, this issue's priority would be greater. It is suggested that any task action plan written to resolve this issue first attempt to identify such a subcategory.

## CONCLUSION

These priority figures would normally indicate a medium priority category for this item. Moreover, if a specific subset of PWRs could be identified, a high priority assignment would be possible. However, DSI has elected<sup>719</sup> to pursue this matter as part of its effort on TMI Action Plan III.D.3.1. Therefore, this issue is covered in Item III.D.3.1.

## REFERENCES

- 716. Memorandum for D. Eisenhut from D. Muller, "PWR Reactor Cavity Uncontrolled Exposures, Generic Letter Implementing a Generic Technical Specification," July 12, 1984.
- 717. Memorandum for A. Thadani from W. Minners, "CRAC2 Computer Runs in Support of USI A-43," February 1, 1983.
- 718. Memorandum for W. Minners from F. Congel, "Prioritization of Generic Issue 97: PWR Reactor Cavity Uncontrolled Exposures," February 8, 1985.
- 719. Memorandum for H. Denton from R. Bernero, "PWR Reactor Cavity Uncontrolled Exposures," November 28, 1984.

## ISSUE 98: CRD ACCUMULATOR CHECK VALVE LEAKAGE

### DESCRIPTION

#### Historical Background

During the review of LaSalle, ASB identified a potential problem which could be generic to all BWRs.<sup>381,705</sup> The problem relates to ability of the CRD accumulators to retain pressure for a sufficient period of time after the failure of a CRD hydraulic pump.

The CRDs are safety-related as are the accumulators and their associated check valves. For rapid reactor shutdown, the stored hydraulic pressure in the accumulator, in conjunction with the reactor system pressure, rapidly inserts all the control rods. At reactor pressures below 500 psig the accumulators provide all the motive force to insert the control rods. Each control rod is provided with its own accumulator. With the reactor pressure above 500 psig the accumulators provide the initial acceleration force for the control rods with the majority of the work provided by the reactor pressure.

The technical specifications<sup>706</sup> for BWRs have a CRD accumulator check valve leakage surveillance statement which is ambiguous and does not have an action statement for failure to pass the surveillance requirement.

#### Safety Significance

The concern of this issue is the potential for the loss of CRD hydraulic system pump at a low reactor vessel pressure with leakage of multiple check valves followed by an accident situation that would require a reactor shutdown. During such an event, it is possible that there would be a failure to scram the reactor and the SLCS would be required to achieve cold shutdown.

#### Possible Solutions

Two possible solutions to this issue have been identified. First, the CRD pumps, associated valves, and instrumentation could be made safety-related with the redundant pump automatically starting upon failure of the running pump. The second possible solution would require both CRD pumps to be running with the reactor pressure less than 500 psig and with more than one control rod withdrawn. For those plants requiring manual action to open a stop check valve for the redundant pump to perform its function, an operator must be stationed by the valve, monitor the header pressure, and operate the valve when the header pressure drops to a predetermined value.

### PRIORITY DETERMINATION

#### Assumptions

For this issue, it is assumed that operation below 500 psig will occur only during ascent to power and controlled descent from power operation. Further,



it will be assumed that, to achieve 500 psig operation during controlled descent from power operation, the CRDs will be inserted by the time reactor pressure will have been reduced to 500 psig. During ascent to power, the time interval between going critical and reactor pressure reaching 500 psig is estimated to be usually one hour. It will also be assumed that a power ascent will occur monthly, on an average, for purposes of this calculation. The assumption of monthly power ascents will result in conservative calculations since the average number of plant trips is about eight per year, not all of which result in reactor pressure falling below 500 psig.

It will be assumed that check valve leakage will reduce the accumulator pressure below the pressure required to insert the control rod in ten minutes.

It is assumed that the accident requiring a scram is one that results in the loss of primary system pressure. With system pressure at or below 500 psig the negative reactivity feedback will, with decreasing temperature as a result of decreasing pressure, increase reactivity without control rod insertion. Thus, only those accident situations in which system pressure is lost and primary coolant temperature decreases requires the insertion of the control rods to limit the core reaction, which would be the LOCA events.

Major PRA studies have assumed that a minimum of three adjacent control rods in a BWR must remain withdrawn for the reactor to remain critical. For this analysis, the same assumption will be considered valid.

#### Frequency/Consequence Estimate

The undesired event (U), that of being unable to shutdown the reactor with the reactor protection system in an accident situation due to the loss of CRD accumulator pressure can be defined as the product of the following probabilities:

A, the probability that an accident event requiring reactor trip occurs during any one year ( $1.4 \times 10^{-3}$ ). This quantity is based upon the total LOCA initiating event frequency as given in WASH-1400.<sup>16</sup>

B, the probability that the reactor vessel pressure is less than 500 psig with the reactor critical ( $1.7 \times 10^{-3}$ ). This probability is based upon the assumption that 12 ascents to power occur annually; that one hour elapses from attaining criticality until the reactor vessel pressure is greater than 500 psig; and that the average operating time per year is 7,000 hours.

C, the probability the operators fail to scram the reactor within 10 minutes following the failure of the CRD hydraulic pump, (0.1). This value is based upon the NUREG/CR-1278<sup>339</sup> nominal model for operator error.

D, the frequency that the on-line CRD hydraulic pump fails during a one year interval (0.7). The WASH-1400<sup>16</sup> failure rate for pumps was between  $3 \times 10^{-6}/\text{hr}$  and  $3 \times 10^{-4}/\text{hr}$  with a median of  $3 \times 10^{-5}/\text{hr}$ . Since the CRD hydraulic pump is not a safety-related classified component but is believed to have a quality level above standard off-the-shelf hardware, a failure rate of  $10^{-4}/\text{hr}$  was assigned. As previously stated, an annual operating time of 7,000 hours was assumed.

E, the probability that the operators will fail to start the standby CRD hydraulic pump within 10 minutes after the failure of the on-line pump (0.1). This value is also based on the NUREG/CR-1278<sup>339</sup> nominal model for operator error. Pump failure to start is negligibly small in comparison.

F, the probability that three adjacent accumulator check valves leak, (0.1). This probability value was chosen with the belief that it conservatively covers common failure causes as well as the multitude of 3 adjacent control rod combinations involving independent failures. Even with an ambiguous action statement, it is unlikely that a large number of check valves will leak.

G, the probability that the operator failed to follow procedures by pulling a control rod adjacent to two other rods which are already pulled, (0.1).

H, the probability that the reactor protection system failed to detect the pulling of the out-of-sequence rod and then failed to initiate a scram signal, (0.01).

Z, the probability that the loss of CRD hydraulic pressure occurs before the accident event (0.5).

Hence, 
$$U = A \cdot B \cdot C \cdot D \cdot E \cdot F \cdot G \cdot H \cdot Z$$
$$= (1.4 \times 10^{-3})(1.7 \times 10^{-3})(0.1)(0.7)(0.1)(0.1)(0.1)(0.01)(0.5)/RY$$
$$= 8.4 \times 10^{-13}/RY$$

Subcriticality following a LOCA cannot usually be maintained by the SLCS, but may be maintained for a time in some LOCAs. The ECCS could control some LOCAs even if some of the control rods are not inserted. As a conservative assumption, no credit will be taken for the SLCS and it will be assumed that the accident-initiating event and the failure of the reactor protection system will result in a core-melt accident.

As defined in NUREG/CR-1659,<sup>54</sup> accident sequences involving LOCAs and the reactor protection system were dominated by the Category 2 releases. The whole body man-rem dose obtained by using the CRAC Code<sup>64</sup> assuming an average population density of 340 persons per square mile (which is the mean for U.S. domestic sites) from an exclusion area of a one-half mile radius about the reactor out to a 50-mile radius about the reactor. A typical midwest meteorology is also assumed. Based upon these assumptions the public dose resulting from a BWR Category 2 release is  $7.1 \times 10^6$  man-rem. Based on an average life of 25 years for each BWR, the public risk is  $1.5 \times 10^{-4}$  man-rem/reactor. For 44 BWRs, the risk is  $6.6 \times 10^{-3}$  man-rem.

#### Cost Estimate

The least expensive resolution to this issue involves turning on the standby CRD hydraulic pump and assigning a dedicated operator at the stop check valve control to monitor pressure and to transfer to the standby system if the hydraulic pressure drops. While it is not exactly known the number of plants having this configuration, for purposes of the calculation it will be assumed that 25% are so configured. For each reactor requiring the dedicated operator, assuming 12 power ascents and descents per year at one hour per change, will utilize 24

operator-hours per year. Based upon 1984 dollars and assuming a cost of \$52 per operator-hour for 11 reactors, the lifetime cost for all BWRs will be \$0.3M.

The cost of upgrading the CRD hydraulic system to a safety related quality level system will be much more expensive. If 0.5 person-years of technical experience were required for evaluation of the existing system and no hardware changes were required, the cost would be \$50,000/reactor or \$2.2M for the 44 reactors involved.

#### Value/Impact Assessment

Based on a risk reduction of  $6.6 \times 10^{-3}$  man-rem and a cost of \$0.3M, the value/impact score is given by:

$$S = \frac{6.6 \times 10^{-3} \text{ man-rem}}{\$0.3\text{M}}$$
$$\leq 2.2 \times 10^{-2} \text{ man-rem}/\$M$$

#### CONCLUSION

In general, accident frequencies on the order of  $10^{-13}$ /yr, even for a very specific sequence, must be used with caution. Errors of incompleteness, and overlooked dependencies, as well as other modeling errors, will generally be very large compared to such frequency estimates. In this case, a conscientious effort has been made to identify other sequences and dependencies. Even with a large error, this issue poses a very small risk. Therefore, the issue should be placed in the DROP category.

#### REFERENCES

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.
54. NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program," U.S. Nuclear Regulatory Commission, 1981.
64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.
339. NUREG/CR-1278, "Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, February 1983.
381. Memorandum for W. Minners from O. Parr, "Prioritization of Proposed Generic Issue on CRD Accumulator Check Valve Leakage," August 13, 1984.
705. Memorandum for C. Thomas from O. Parr, "CRD Accumulators - Proposed Improved Technical Specification," August 13, 1984.
706. NUREG-0123, Revision 3, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)," U.S. Nuclear Regulatory Commission, December 1980.

## ISSUE 101: BREAK PLUS SINGLE FAILURE IN BWR WATER LEVEL INSTRUMENTATION

### DESCRIPTION

#### Historical Background

Issue 50 addressed several areas of concern with BWR water level instrumentation and its resolution involved voluntary implementation of water level measurement improvements for all of the staff concerns, except the one related to a break in an instrument line in conjunction with the worst single failure.<sup>720</sup>

This concern was first identified in an AEOD draft report<sup>721</sup> which was later issued as AEOD/C201<sup>322</sup> in January 1982. In the interest of the expeditious resolution of Issue 50, it was decided<sup>697</sup> to address the AEOD concern as Issue 101.

#### Safety Significance

Water level is measured in BWRs by means of differential pressure sensors connected between the reactor vessel (at a point low enough in elevation to be below the expected water level) and reference columns (which are completely full of water and connected at the top to the steam dome). The differential pressure sensed by the dp cell corresponds to the difference in elevation between the "collapsed" water level in the reactor and the water level in the reference column. If the reference column is broken, the water in it will flash to steam and the water level indication in all channels connected to the broken column will give a false "high" reading.

Typically, a BWR will have two reference columns. (There is a variety of design, however.) A break in one column will cause all instrumentation associated with that column to indicate full scale high level. This can simultaneously cause a transient and interfere with safety systems. A single failure associated with the other reference column can completely defeat mitigation systems. The following points were stated in an RRAB memorandum:<sup>722</sup>

"Consequences of such an event depend upon (1) the location of the postulated reference leg break, whether it is a single reference leg or a common line; (2) the physical location of an additional postulated single failure, and (3) the various combinations thereof.

"Further, effects of such an event depend upon plant specific design. In some older plants, a postulated reference leg break itself without any additional single failure will cause failure of ECC system initiation due to a reactor water level condition.

"The greatest vulnerability occurs when the same sensor is used to initiate more than one system. In one plant where core spray initiation and MSIV initiation share the same set of sensors, a single failure in either system in addition to a pipeline break in the instrument reference leg may cause a core uncover. In another plant, the consequences of the additional single failure becomes of concern only when the coolant injection system initiation transmitter fails. In such an event, operator action is required to prevent

core uncover in about 45 minutes. Further, several indications are available in the control room to give the operator information relative to the accident progression and status of the plant."

### Proposed Solution

The references cited above do not recommend specific modifications since individual plant designs are apparently too varied to permit generic solutions.<sup>723</sup> However, it appears to be possible to fix the problem by modification to the logics which use reactor level as an input.<sup>722</sup>

### PRIORITY DETERMINATION

#### Frequency/Consequence Estimate

The RRAB memorandum<sup>722</sup> contains a probabilistic assessment of the concern. This assessment estimated a core-melt frequency of  $1 \times 10^{-5}/\text{RY}$  and a public risk of 50 man-rem/RY. The affected plants were estimated to have roughly 20 effective full-power years of remaining life for a total risk of 1,000 man-rem/reactor.

#### Cost Estimate

The RRAB assessment<sup>722</sup> contained a cost-benefit ratio of \$1,000/man-rem for the concern in this issue. This translates into \$1M/reactor.

#### Value/Impact Assessment

Based on an estimated risk reduction of 1,000 man-rem/reactor and a cost of \$1M/reactor, the value/impact score is given by:

$$\begin{aligned} S &= \frac{1,000 \text{ man-rem/reactor}}{\$1\text{M/reactor}} \\ &= 1,000 \text{ man-rem}/\$M \end{aligned}$$

### Other Considerations

It must be emphasized (as virtually every reference points out) that both the affected accident sequences and the modifications to resolve the issue will vary from plant to plant. The resolution of this issue will be more case-specific than most, and some plants may not require modification.

The RRAB calculations<sup>722</sup> assume an operator error probability of 0.1. This figure is based on judgment balancing the relatively high likelihood of initial operator confusion, due to conflicting level indicators, against a relatively long time (45 minutes) available for problem diagnosis before core uncover in the primary sequence. Specific plant designs and other more rapid sequences may well indicate a higher figure for operator error probability, which would increase the priority figures above.

In some cases, ORE associated with the modifications may be a significant factor. This area should be addressed in specific plant reviews.



## CONCLUSION

The priority parameters are on the borderline between medium and high priority. However, some specific plants will undoubtedly fall well into the high area, others well into medium or below. At present, without further study on this issue, the specific plants for which this issue is particularly important cannot be identified. Thus, it is recommended that this issue be placed into the HIGH priority category. As work progresses, it should be possible to target the issue more specifically.

## REFERENCES

- 322. AEOD/C201, "Report on the Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, January 1982.
- 697. Memorandum for D. Eisenhut from R. Bernero, "Resolution of Generic Issue 50, Reactor Vessel Level Instrumentation in BWR," September 6, 1984.
- 720. Memorandum for T. Speis from R. Bernero, "Request for Prioritization of Generic Safety Issue - Break Plus Single Failure in BWR Water Level Instrumentation," October 10, 1984.
- 721. Memorandum for H. Denton and V. Stello from C. Michelson, "Case Study Report - Safety Concern Associated with Reactor Vessel Instrumentation in Boiling Water Reactors," September 2, 1981.
- 722. Memorandum for B. Sheron from A. Thadani, "Reactor Vessel Level Instrumentation in BWR's (Generic Issue 50)," August 2, 1984.
- 723. Memorandum for H. Denton from T. Speis, "Reactor Vessel Level Instrumentation in BWRs (Generic Issue 50)," August 2, 1984.



## ISSUE 102: HUMAN ERROR IN EVENTS INVOLVING WRONG UNIT OR WRONG TRAIN

### DESCRIPTION

#### Historical Background

In January 1984, AEOD issued a special study report (AEOD/S401)<sup>640</sup> describing the number of events that resulted from human error in identification of the correct unit or train. This study focused on LERs issued during 1981, 1982, and part of 1983.

Although the scope of its study was narrow, AEOD found that 19 out of 27 events identified resulted from human error during maintenance and surveillance testing; 16 of these occurred while the plants were at power. Although most of the events had limited safety significance because of the short duration of the condition and/or because redundant systems were operable and available, AEOD considered them to be examples of events that could have high safety significance under other circumstances. As a result, AEOD concluded that the above statistic was evidence that human errors in maintenance and testing operations are major contributors to loss of safety system events.

Following the issuance of the AEOD report,<sup>640</sup> OIE issued Information Notice Nos. 84-51<sup>641</sup> and 84-58<sup>642</sup> to alert licensees to the potential problem.

#### Safety Significance

Safety functions can be inadvertently defeated by human errors involving the wrong unit, wrong train, or wrong system.

#### Possible Solutions

The possible solutions to this issue are considered to be the two AEOD recommendations and one suggestion that were transmitted to NRR.<sup>640,643</sup>

##### AEOD Recommendations

- (1) Consider the need for further clarification and/or guidance on what constitutes an acceptable independent verification program.
- (2) Review the wrong unit/train events and develop appropriate guidance to minimize such events.

##### AEOD Suggestion

- (1) As part of the Maintenance and Surveillance Program Plan (MSPP), consider the high proportion of events that were due to human error in maintenance and testing operations at power.

### CONCLUSION

All of the AEOD concerns outlined above are to be addressed<sup>645,646</sup> in the MSPP, a draft<sup>644</sup> of which was issued on July 7, 1984. In this plan, six Technical Issues were identified:

- (1) Human Error in the Performance of Maintenance
- (2) Indicators of Maintenance Effectiveness
- (3) Counteracting Aging Effects and the Role of Preventive Maintenance
- (4) Management and Organization Impacts on Maintenance Effectiveness
- (5) Maintenance Program Criteria and Standards
- (6) The Maintenance and Operations Interface

AEOD Recommendations (1) and (2) outlined above are to be addressed in MSPP Technical Issues (6) and (5), respectively.<sup>645</sup> The AEOD suggestion is covered in MSPP Technical Issue (1).<sup>645</sup> In April 1985, the MSPP (Issue HF02) was presented to the Commission in SECY-85-129.<sup>764</sup> Issue 102 was specifically addressed as Item 3.2.9 of the MSPP. Thus, Issue 102 is covered in the MSPP which was evaluated in its entirety as Issue HF02.

#### REFERENCES

640. Memorandum for H. Denton from C. Heltemes, "Special Study Report-Human Error in Events Involving Wrong Unit or Wrong Train," January 13, 1984.
641. IE Information Notice No. 84-51, "Independent Verification," U. S. Nuclear Regulatory Commission, June 26, 1984.
642. IE Information Notice No. 84-58, "Inadvertent Defeat of Safety Function Caused by Human Error Involving Wrong Unit, Wrong Train, or Wrong System," U. S. Nuclear Regulatory Commission, July 25, 1984.
643. Memorandum for H. Denton from C. Heltemes, "Human Error in Events Involving Wrong Unit or Wrong Train," August 8, 1984.
644. Memorandum for D. Eisenhut, et al., from H. Thompson, "Maintenance and Surveillance Program Implementation Plan," July 7, 1984.
645. Memorandum for C. Heltemes from H. Denton, "Special Study Report - Human Error in Events Involving Wrong Unit or Wrong Train," May 2, 1984.
646. Memorandum for C. Heltemes from H. Denton, "Human Error in Events Involving Wrong Unit or Wrong Train," September 17, 1984.
764. SECY-85-129, "Maintenance and Surveillance Program Plan," April 12, 1985.

## ISSUE 105: INTERFACING SYSTEMS LOCA AT BWRs

### 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 and 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. These actuators are used to test the operability of these valves. This operability test was considered sufficient to assure the integrity of the pressure isolation function and leak-testing of pressure isolation valves in BWRs was not required. However, beginning in 1980, the BWR STS Section 3.4.6.2 required the leak-testing of all RCS pressure isolation valves 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 inservice testing program for review.

Recent BWR operating experience indicates 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 have been three reported failures of the boundary between the RCS and low pressure injection systems in approximately 200 BWR-years of operation.<sup>762</sup> 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 of these cases, there was a degradation of the pressure isolation valves due to personnel errors. None of these plants was required to leak test pressure isolation valves.

This issue, which is limited to pressure isolation valves in BWRs, is related to Issue 96 which considers the failure of the pressure isolation valves 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 pressure isolation valves 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.<sup>761</sup> 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 is 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 are not anticipated.

### PRIORITY DETERMINATION

The prioritization of this issue is based on analysis performed by PNL.<sup>64</sup>

### Frequency/Consequence Estimate

Since this generic issue applies only to BWRs, the Browns Ferry, Unit 1, IREP<sup>760</sup> PRA was used in the estimation of public risk reduction. The general approach was 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 is 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  $1.5 \times 10^{-2}/\text{RY}$  is estimated. Analysis of the low pressure piping reveals 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 estimates the conditional probability of an inter-system LOCA, via the pipe cracking scenario, of  $10^{-1}/\text{event}$  given an overpressurization of the low pressure piping. This gives a new estimate of Ls of  $1.5 \times 10^{-3}/\text{RY}$ , as opposed to the value of Ls derived in the Browns Ferry PRA ( $3 \times 10^{-8}/\text{RY}$ ). In NUREG-0677,<sup>763</sup> a probability of BWR intersystem LOCA of  $6.2 \times 10^{-4}/\text{RY}$  was calculated. No contribution from maintenance and operator errors was included in deriving the above frequency of BWR intersystem LOCA. The BWR intersystem LOCA frequency derived for this prioritization analysis ( $1.5 \times 10^{-3}/\text{RY}$ ) which is based on recent LERs is dominated by operator and maintenance errors and appears to be an expected value when compared to the value derived in NUREG-0677.<sup>763</sup> When this new value of Ls ( $1.5 \times 10^{-3}/\text{RY}$ ) is input to the affected core-melt minimal cutsets in the Browns Ferry PRA, a base case core-melt frequency due to

isolation boundary failures is calculated to be  $6.31 \times 10^{-6}/\text{RY}$ . The effect of accidents resulting in direct core-melt releases outside containment is assumed to be best estimated by the BWR Release Category 2. When the dose conversion factor for BWR Category 2 events ( $7.1 \times 10^6$  man-rem/event) is multiplied by the base case core-melt frequency, a public risk of 44.7 man-rem/Ry is calculated.

Implementation of the assumed resolution for this issue is 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.,  $1.22 \times 10^{-10}$  events/Ry and  $8.66 \times 10^{-4}$  man-rem/Ry, respectively). Therefore, implementation of the assumed resolution of this issue is estimated to result in a reduction in core-melt frequency of  $6.3 \times 10^{-6}/\text{RY}$  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 is calculated to be  $2.3 \times 10^4$  man-rem.

#### Cost Estimate

Resolution of the issue is 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."

Industry Cost: Implementation of the assumed resolution of this issue is 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 is estimated. Increased surveillance testing, reduction of allowable concurrent testing and improved post-maintenance inspection procedures are 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 is calculated to be about \$650,000. Total industry cost for resolution (and implementation) of this issue is therefore estimated to be about \$875,000.

NRC Cost: It is 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 (NRR and OIE) review of licensee implementation of the assumed resolution of the issue would require 5 staff-weeks/plant for a cost of about \$230,000. Resident inspector surveillance of site actions emanating from the resolution of this issue are estimated to require 0.5 staff-weeks/Ry for a present worth of about \$325,000 over the remaining lifetime of the 20 affected BWRs. The total present worth of NRC cost for this issue is thus estimated to be about \$650,000.

Total NRC and industry costs for resolution and implementation of this issue are thus estimated to be approximately \$1.5M.



### Value/Impact Assessment

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

$$S = \frac{2.3 \times 10^4 \text{ man-rem}}{\$1.5\text{M}} \\ = 15,000 \text{ man-rem}/\$M$$

### Other Considerations

It should be noted that the probability of intersystem LOCA 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 pressure isolation valves contributes further to the risk. Although the risk from other interfaces has not been calculated, the evaluation of Issue 96 shows that the risk from failures of the valves isolating the RHR system in a PWR is at least an order of magnitude less than the risk calculated for this issue. The failure of the pressure isolation valves 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, LOCA 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 could not be included within the scope of the limited efforts performed for a prioritization analyses. However, were they to be included, we would expect the estimate of frequency for intersystem LOCA and resultant core-melt to be greater. For that reason, we believe that the priority conclusion reached on the basis of the simplified analysis performed for this generic issue is conservative.

A relatively small total increase in ORE (530 man-rem) is calculated due to assumed increases in surveillance and post-maintenance inspections. The calculation assumes 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 is calculated to result in a total averted ORE of 215 man-rem: 65 man-rem due to cleanup of core-melt events and 150 man-rem due to cleanup of non-core-melt intersystem LOCAs. Both the increased ORE and the averted operator exposure are insignificant in comparison to the calculated public risk reduction of  $2.3 \times 10^4$  man-rem and would not alter the recommendations indicated by the value/impact assessment.

At an estimated industry cleanup and replacement power costs 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 intersystem LOCA estimated for resolution of this issue would result in an averted accident cost savings with a present worth of about \$2.7M. This exceeds the total expected NRC and industry cost for resolution of the issue and would therefore lend support to a decision to pursue resolution of the issue.



## CONCLUSION

Significant reductions in public risk and the frequency of core-melt accidents are calculated for the resolution of this issue. The analysis indicates that these reductions may be achieved at a relatively small cost to the NRC and the industry, resulting in a very favorable value/impact ratio (15,000 man-rem/\$M). If the averted cost of the cleanup of intersystem LOCAs is included, the net impact is a cost saving. Therefore, we recommend that resolution of the issue be pursued with a HIGH priority.

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761. AEOD/E414, "Stuck Open Check Valve on the Residual Heat Removal System at Hatch Unit 2," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, May 31, 1984.
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## ISSUE 108: BWR SUPPRESSION POOL TEMPERATURE LIMITS

### DESCRIPTION

#### Historical Background

BWRs are equipped with SRVs to control primary system pressurization. Upon SRV actuation and following the clearing of air from the discharge lines, essentially pure steam is injected into the pool. Experiments indicate that the steam jet/water interface at the discharge line exist during this phase is relatively stationary when the local pool temperature is low.<sup>732</sup> Thus, the condensation proceeds in a stable manner and no significant hydrodynamic loads are experienced. Continued steam blowdown into the pool will increase the local pool temperature. The condensation rates at the turbulent steam/water interface are eventually reduced to levels below those needed to readily condense the discharged steam. At this threshold level, the condensation process may become unstable; for example, steam bubbles may be formed and shed from the pipe exit, oscillate, collapse, and give rise to severe pressure oscillations which are then imposed on the pool boundaries.

Current practice for dealing with this phenomenon in BWRs is to restrict the allowable pool operating temperature so that the threshold temperature is not reached. This restriction is referred to as "the pool temperature limit." USI A-39 resolved, among other things, the concern about steam condensation behavior in the suppression pool of MARK I, II, and III containments. Resolution of this issue has been reported in NUREG-0661<sup>702</sup> and NUREG-0487.<sup>733</sup> Criteria for the pool temperature limit were established and included in those reports. However, the staff also indicated that the evaluation of this issue would continue in an attempt to improve the criteria and that future progress would be reported.

NUREG-0783<sup>734</sup> presents the results of the staff evaluation of the safety issue concerning suppression pool temperature limits. Acceptance criteria for the pool temperature limits, the events and associated assumptions used to analyze pool temperature response, and the suppression pool temperature monitoring systems are included. The resolution applies to MARK I, II and III containments using the SRV quencher devices specified in NUREG-0783.<sup>734</sup>

In a subsequent letter<sup>735</sup> to the NRC staff from the BWR Owners' Group, additional test data were summarized which form the basis for a BWR Owners' Group proposal that the current local pool temperature limits for SRV discharge be eliminated. The BWR Owners' Group proposal will be contained and supported in a forthcoming report to the staff which is expected in January 1985. It is the staff effort required for the review of BWR Owners' Group report that is to be prioritized here.<sup>736</sup>

#### Safety Significance

It is assumed that the report will provide adequate technical justification that there is no risk increase and hence no safety significance associated with

the elimination of the suppression pool temperature limits from the TS. This issue represents a reduction of NRC requirements and is a regulatory impact issue.

#### Possible Solution

A resolution of this issue would be the presentation of test data and analysis to demonstrate that suppression pool loads associated with SRV discharge are well within the containment design capability with the deletion of the suppression pool temperature limits.

#### PRIORITY DETERMINATION

##### Frequency/Consequence Estimate

A premise of this analysis is that there would be no increase in core-melt frequency as a result of the implementation of the resolution of this issue.

##### Cost Estimate

Industry Cost: The testing and analysis that support the BWR Owners' Group proposal are virtually completed and it is expected that the final report supporting the proposal will be submitted to the NRC staff in January 1985. The additional cost that can be anticipated for the industry is that associated with clarifications of the report and responding to NRC staff questions. This is estimated to cost \$50,000. In addition, there will be the licensee cost for the preparation of amendments to eliminate the suppression pool technical specifications on temperature limits. This cost is assumed to be approximately \$15,000 per plant. Assuming that requests for this change is made for 37 plants,<sup>737</sup> the total industry cost is estimated to be \$555,000.

It is noted that the present plant technical specifications require a plant shutdown in the event that the suppression pool temperature exceeds 110°F. A review of the LERs from 1974 to date reveals that in most cases the temperature excess in the suppression pool was caused by hot weather, testing of the HPCI system operability and, occasional operator errors or equipment failure for a total of approximately 22 reported events. Of these events, only two shutdowns were initiated because of violations of the suppression pool TS: Cooper Nuclear Plant (July 24, 1984) and Quad Cities Nuclear Plant No. 2 (August 16, 1983).

If shutdowns can be avoided at the rate indicated in the past, say, once every 9 years or so, the following estimate of cost savings can be made. It is assumed that the minimum downtime following a shutdown is 24 hours. For a shutdown that lasts for 24 hours and a cost of replacement power of \$300,000 per day, the present worth (PW) of these savings at a real discount rate of 5%, over an average reactor plant lifetime of 28 years and assuming 1 shutdown every nine years, is given by:

$$\begin{aligned} PW &= (\$300,000)(1 + 0.05)^{-9} + (\$300,000)(1 + 0.05)^{-18} \\ &\quad + (\$300,000)(1 + 0.05)^{-27} \\ &= (\$300,000)(0.645 + 0.416 + 0.268) \\ &= (\$300,000)(1.33) \\ &\approx \$400,000 \end{aligned}$$

Added to the cost saving in eliminating plant shutdowns would be the savings implied in the additional 20 cases of operating experience by the elimination of the need for suppression pool cooling by use of the RHR and/or core spray systems (LER 112, 8/2/83, Susquehanna 1) and which also reduces the challenges to this safety-related equipment. The cost of operating the necessary equipment to cool the suppression pool will be assumed to be \$10,000 per event. Based on the past experience of about 15 events required over the 10-year span of LERs reviewed above, the present value ( $PW_1$ ) of this cost with a real discount rate of 5% is given by:

$$PW_1 = \$10,000 (15/10)(0.05)^{-1} (1 - \exp(-0.05t))$$

For  $t = 28$  years,  $PW_1 = \$262,000$ . The present worth of the total cost savings to the industry is, therefore, estimated to be  $\$(262,000 + 400,000) = \$662,000$ . The present total cost to the industry for implementation of this change in technical specifications is estimated to be  $\$(50,000 + 555,000) = \$605,000$ .

In addition, it is noted in the DSI memorandum<sup>736</sup> that it is expected that the implementation of the proposal, if the NRC staff accepts it, will result in plant improvements by: (1) the elimination of monitoring requirements of the pool temperatures by plant operators during SRV discharge; (2) increased plant operator flexibility insofar as any necessary operator action could be delayed during SRV discharge, and (3) improved potential for additional strategies for plant control during ATWS and station blackout scenarios in which the use of the suppression pool without restrictions may provide enhanced plant control. These items represent benefits accruing from the elimination of the temperature limits, but the value of these benefits are somewhat offset by the requirement of alternative suppression pool limits that will have to be established by the licensee for proper plant operation. These alternative limits, however, will be based on performance requirements such as NPSH, containment design factors, environmental qualification, and so forth, but not as a result of technical specification requirements so that the trade-off still represents a net benefit to the licensee.

NRC Cost: Based on a CSB memorandum,<sup>737</sup> it is expected that the NRC will require a technical assistance program effort in the review of the BWR Owners' Group report for a total estimated NRC cost of \$90,000. The administrative NRC costs to process amendments for TS changes are in the order of 1 to 2 man-weeks/plant. For 37 BWRs, this will require 55 man-weeks at a cost of  $(55 \times \$2,270) = \$125,000$ . Therefore, the total cost to the NRC is estimated to be  $\$(90,000 + 125,000) = \$215,000$ .

### CONCLUSION

Based on the assumption that the elimination of suppression pool temperature limits can be adequately justified by the BWR Owners' Group data and analysis, it is concluded that a savings can accrue to the industry with an estimated present worth of \$661,000 at a cost of approximately  $\$(215,000 + \$350,000)$  or \$820,000 for both the NRC and the industry, with no increased risk to the public. In addition, there is some additional potential for the overall improvement of plant safety through some small but intangible increase in plant operation flexibility as well as safety equipment reliability by the elimination of the

suppression pool TS on temperature limits. Therefore, it is concluded that the potential savings from the proposal initiated by the BWR Owners' Group are in the same order as the costs, so that NRC effort on this Regulatory Impact issue should have a low priority.

#### REFERENCES

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## ITEM HF02: MAINTENANCE AND SURVEILLANCE PROGRAM

### DESCRIPTION

#### Historical Background

The purpose of the Maintenance and Surveillance Program (MSP) effort is to provide direction for the NRC's efforts to assure effective nuclear power plant maintenance. The program will be based on the current NRC regulatory approach to maintenance and an evaluation of the effectiveness of current industry efforts in the maintenance area.

The NRC's current regulatory approach to nuclear power plant maintenance is concentrated on: (1) QA during design, construction, and operation for structures, systems and components important to safety (10 CFR 50, Appendix B), and (2) surveillance requirements to assure that the necessary availability and quality of such systems and components is maintained (10 CFR 50.36). Despite extensive surveillance testing requirements, the NRC's rules and regulations provide no clear programmatic treatment of maintenance. NRC additionally requires reporting of significant events (10 CFR 50.72), including personnel errors and procedural inadequacies which could prevent fulfillment of safety functions and exceeding of TS Limits.

The NRC does not stipulate maintenance requirements for systems which are not safety-related. Many challenges to safety systems originate from systems/components which are classified as not safety-related. The relationship between non-safety grade control systems and safety systems is being addressed in USI A-47.

The MSP is intended to integrate the NRC's efforts to assure effective nuclear power plant maintenance and to do so in a manner that is consistent with and responsive to the Commission's 1984 Policy and Planning Guidance.<sup>745</sup> The program addresses the problems and issues which exist and proposes development of alternative NRC approaches to regulating nuclear utility maintenance activities consistent with the Policy and Planning Guidance. The scope of the program includes all aspects of maintenance required to carry out a systematic maintenance and surveillance program. It includes conventional maintenance and repair plus such things as surveillance and test activities, equipment isolation, post-maintenance testing, independent verification, maintenance management, administrative control, personnel selection and training, procedures, and technical documentation.

#### Safety Significance

Since the TMI-2 accident in 1979, it has been evident that faulty maintenance practice is a principal contributing factor to operating abnormalities. Preliminary estimates indicate that, aside from design deficiencies, more than 35 percent of the abnormal nuclear power plant occurrences reported to Congress since 1975 may be directly attributed to maintenance error, with the trend towards a worsening maintenance situation as plants age.<sup>740</sup> Reviews of operating experience show a high frequency of degraded system performance due to

both the lack of maintenance (especially preventive maintenance) and improperly performed maintenance, including human error during repair and surveillance testing.<sup>740</sup>

#### Possible Solutions

The proposed solution to this issue is to implement a systematic maintenance program as addressed in the NRC's preliminary MSP with the following five objectives:

- (1) To assure that needed maintenance is being accomplished, especially in counteracting system and equipment aging effects, by taking appropriate preventive and corrective action to minimize equipment failures.
- (2) To reduce failures from improper maintenance to an acceptable level and to assure safety through effective maintenance management, personnel selection and training, procedures, administrative control, and design for maintainability.
- (3) To assure proper integration of maintenance operations and other organizational interfaces for maintenance activities which can affect plant safety.
- (4) To improve the effectiveness of nuclear power plant maintenance programs in reducing the number of challenges to safety systems (e.g., reactor scrams).
- (5) To optimize surveillance requirements to assure equipment availability when required without excessive equipment out-of-service intervals for testing and to eliminate the unnecessary exposure for transient trips due to excessive test frequencies of logic and initiation systems.

#### PRIORITY DETERMINATION

##### Frequency Estimate

This issue affects all 134 BWRs and PWRs operating or under construction. For this analysis, Oconee 3 was selected as the representative PWR and Grand Gulf 1 was selected as the representative BWR.

##### Assumptions

The following paragraphs describe the background and approach to quantifying the base and adjusted cases for this issue. The background description relates to the subjects of aging and maintenance in an overall sense. The approach makes assumptions based on the background information. The subjects of aging and of overall effect of maintenance are considered below.

Aging: The effects of system and equipment aging is considered as part of the MSP because adequate maintenance and surveillance can counteract aging effects. NUREG/CR-2497<sup>76</sup> (p. 5-4) discusses the variation of significant precursors with plant age. This can be assumed to reflect general equipment deterioration and the subsequent impact on plant safety in general. Trends for a number of

initiating events or demand failures were presented for data up through 1979. For PWRs, failure rate trends for long-term core cooling were given as constant and perhaps increasing. For BWRs, only the ADS demand failure showed an increasing failure trend, based on a small number of observed events. The emergency power system failure trend was given as constant, perhaps increasing. The general conclusion was that no clear variation in the number of significant events with plant age has been demonstrated.

It is further suggested that the operating time on the majority of the plant safety systems is very small. In many cases, the operating time is only that experienced during testing intervals. While aging effects cannot be ruled out at this time, the likelihood of their showing any significant role in safety systems is small. As a result, it is proposed that aging effects, if any, would best be modeled by failure rates increasing in the balance-of-plant. This would manifest itself as an increase in plant transients requiring shutdown.

Maintenance: A central aspect of the maintenance and surveillance program is the increased efficiency of maintenance operations and the assumed resultant reduction in errors committed during maintenance. It is believed that, if an integrated maintenance program were implemented, increases in preventive maintenance would reduce the need for corrective maintenance during plant operation. The MSP would provide a decrease in improper maintenance due to better training, procedures, human factors engineering, etc. The maintenance program is also seen as improving maintenance such that fewer transients will occur because of better maintained equipment.

In order to evaluate this issue, it is necessary to estimate and bound the likely magnitude of these effects and the degree to which current maintenance and surveillance approaches can deal with the program. Existing information is reviewed below.

Utility Maintenance Experience: The idea of this program is to increase the role of preventive maintenance and thus decrease corrective maintenance required for failures during operation. An examination of current experience indicates that corrective actions now represent the smaller fraction of recorded maintenance actions.

To characterize existing utility maintenance programs, NUREG/CR-3543<sup>741</sup> (p. 20) indicates that between 64% and 80% of the age-related LER failures examined were detected by routine testing and surveillance performed in accordance with the plant TS or maintenance program. Detection after failure during plant operation could then be assumed to occur in 20% to 36% of the failures.

This indicates that the present unsystematic preventive maintenance programs at the majority of nuclear power plants is still detecting a substantial number of the events related to equipment degradation before actual failure in operation occurs. The 36% figure could be assumed to bound the category of failures during operation.

This result could also be expected to follow for a large part of the safety-related systems since the operating time on these systems is only during periodic test. Some examples of exceptions are the AFW systems in some designs

and instrumentation channels for systems such as the reactor protection system and the electrical power systems.

An examination of transients in an EPRI study<sup>307</sup> indicates that almost every transient category can also be considered as involving equipment failures, although these would be in the BOP. Although one could argue that preventive maintenance may not be as strict in this portion of the plant as opposed to safety systems, the utilities obviously have an economic incentive to maintain this portion of the plant as well. As a result, it is assumed that BOP failures and hence transients are subject to the same detection percentages mentioned above.

Base Case: The base case is the same as for the original Oconee and Grand Gulf risk assessments with the following exception. To model the effects of plant aging, it is proposed that the BOP transient frequencies be increased by 10%. This would reflect increased failures due to plant aging. This 10% value is felt to be an appropriate "trip level" beyond which the present surveillance programs would detect the increased failures. Thus  $T_2$ ,  $T_3$ , and  $T_{23}$  frequencies of 3/RY, 4/RY, and 7/RY, respectively, are increased by 10% for the base case. The base case parameters are  $T_2$  and  $T_3$  for Oconee and  $T_{23}$  for Grand Gulf.

The resolution of USI A-47 (which is expected to be resolved prior to this issue) may reduce the base and adjusted case transient frequencies and also result in a decrease in the predicted reduction in transient frequency for parameters  $T_2$  and  $T_3$ . However, it is anticipated that the resulting changes will not significantly impact the results and conclusion contained herein.

Adjusted Case: The adjusted case involves transients affected in the base case by reducing them because of the implementation of a systematic maintenance program. A comparison of U.S. and Japanese data on automatic scrams for 1981 and 1982 provide the basis for an adjusted case reduction in frequency of transients. U.S. data for 1981 and 1982 automatic scrams indicate a frequency of 5.3/RY whereas the comparable Japanese data indicate a frequency of 0.4/RY. After discussions with PNL researchers in the human factors maintenance area, it is assumed for this analysis that, if an integrated maintenance program were implemented in the U.S., the U.S. automatic scram frequency could be reduced to 2/RY. This factor of 2.65 reduction from the 1981 and 1982 U.S. average of 5.3/RY is assumed to be applied to the base case transient frequencies  $T_2$  and  $T_3$  for Oconee and  $T_{23}$  for Grand Gulf. Thus, applying the factor of 2.65 related to improved maintenance results in the adjusted case transient frequencies.

Also, for the adjusted case, it is proposed here that integrated maintenance and avoidance of errors can impact unscheduled maintenance outages during power operation, reducing the duration ( $t$ ) and the outage frequency ( $f$ ). The model that is used in this analysis to represent maintenance outages is expressed as the following equation for unavailability  $Q(TM)$  of systems due to test and maintenance where  $H1$  and  $H2$  are contributions from human performance,  $D1$  and  $D2$  are contributions for design,  $t$  is expressed in hours/act, and  $f$  is expressed in acts/month.

$$Q(TM) = [(H1 + D1)t] [(H2 + D2)f/720]$$

Factors  $H1$  and  $D1$  initially add to one as do factors  $H2$  and  $D2$ . The model initially assumes  $H1$  and  $D1$  as 50% each and  $H2$  and  $D2$  as 25%/75% split,

respectively. It is assumed for this analysis that improved maintenance from an implemented maintenance program results in a 10% improvement in human performance related to outage duration (t) and a 25% improvement in human performance related to outage frequency (f). Thus, a test and maintenance term of 0.0021 in the base case become 0.0019 in the adjusted case.

#### Frequency/Consequence Estimate

The improvement affects all categories of PWR and BWR releases as defined in WASH-1400.<sup>15</sup> The total whole body man-rem dose is obtained by using the CRAC Code<sup>64</sup> assuming an average population density of 340 persons per square mile (which is the mean for U.S. domestic sites) from an exclusion area of a half-mile radius about the reactor out to a 50-mile radius about the reactor. A typical midwest plain meteorology is also assumed. Based upon these assumptions and the proceeding discussions, the base case core-melt frequencies are  $4.95 \times 10^{-5}/\text{RY}$  and  $3.81 \times 10^{-5}/\text{RY}$  for PWRs and BWRs, respectively. With the maintenance improvements as described, the adjusted case core-melt frequencies become  $3.08 \times 10^{-5}/\text{RY}$  and  $2.08 \times 10^{-5}/\text{RY}$  for PWRs and BWRs, respectively.

The base case and adjusted case core-melt frequencies are distributed over the following release categories:

<u>Base Case (RY)<sup>-1</sup></u>	<u>Adjusted Case (RY)<sup>-1</sup></u>
PWR-1 = $2.7 \times 10^{-8}$	PWR-1 = $2.6 \times 10^{-8}$
PWR-2 = $3.1 \times 10^{-6}$	PWR-2 = $1.2 \times 10^{-6}$
PWR-3 = $2.3 \times 10^{-5}$	PWR-3 = $1.3 \times 10^{-5}$
PWR-4 = $5.0 \times 10^{-8}$	PWR-4 = $1.9 \times 10^{-8}$
PWR-5 = $3.3 \times 10^{-7}$	PWR-5 = $2.2 \times 10^{-7}$
PWR-6 = $3.4 \times 10^{-6}$	PWR-6 = $3.1 \times 10^{-6}$
PWR-7 = $2.3 \times 10^{-5}$	PWR-7 = $1.5 \times 10^{-5}$
BWR-1 = $1.1 \times 10^{-7}$	BWR-1 = $8.4 \times 10^{-8}$
BWR-2 = $3.5 \times 10^{-6}$	BWR-2 = $1.8 \times 10^{-6}$
BWR-3 = $1.4 \times 10^{-6}$	BWR-3 = $1.0 \times 10^{-6}$
BWR-4 = $1.6 \times 10^{-6}$	BWR-4 = $1.2 \times 10^{-6}$

The reduction in core-melt frequency per release category results in a per-plant reduction in public risk of 64 man-rem/Ry and 123 man-rem/Ry for PWRs and BWRs, respectively. Based upon an average remaining life of 28.8 years for the 90 PWRs and 27.4 years for 44 BWRs, the total best estimate public risk is reduced by  $3.1 \times 10^5$  man-rem.

#### Cost Estimate

Industry Cost: The implementation of all tasks, identified in the Draft Maintenance Program Plan,<sup>740</sup> involve principally costs associated with labor. Some of the labor intensive costs for each plant are:

Task 2.2.4	Evaluation of Regulatory Alternative (Principally the establishment of the preventive maintenance program by the licensee)	= 10 man-years
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Task 2.3	Assess role of Safety System Monitoring (Principally to code and label piping, valves, etc.)	= 1 man-year
Task 2.5.2	Plant Maintainability (Job performance aids and task analysis)	= 2 man-years
Task 2.5.3.d	Upgrade Maintenance Procedures (3.5 man-years for rewriting maintenance procedures and 1.5 man-years for improv- ing document control)	= 5 man-years
Task 2.5.4	Maintenance Personnel Qualifications and Training (Qualification and Training Program)	= 1.25 man-years

Equipment and other labor costs are expected to cost \$0.2M/plant. Thus, the total industry cost for implementation is given by:

$$(134 \text{ plants})[(19.25 \text{ man-years/plant})(\$100,000/\text{man-year})] + \$0.2\text{M} = \$280\text{M}$$

Labor and engineering costs are estimated to be 3.7 man-years/RY. In addition, 16 days will be added to the annual plant outage time to permit additional maintenance. At an estimated cost of \$300,000/day for replacement power costs, the additional maintenance costs will be \$4.8M/RY. However, the added maintenance is expected to reduce the automatic scram frequency by 3.3 events/RY. Based on 1981 data, each scram results on the average in a two-day outage time. Hence, about 7 days replacement power costs are saved each plant-year, or  $(7 \times \$300,000) = \$2.1\text{M}$  saved per plant-year. Thus, the annual cost for operation and maintenance is  $\$(4.8 - 2.1)\text{M/RY} + (3.7 \text{ man-yr/RY})(\$0.1\text{M/man-yr}) = \$3.1\text{M/RY}$ .

Total industry cost for maintenance and operation  
 $= [(90 \text{ PWRs})(28.8 \text{ years}) + (44 \text{ BWRs})(27.4 \text{ years})](\$3.1\text{M/RY})$   
 $= \$12,000\text{M}$

Total industry cost for the implementation and maintenance and operation is  
 $\$(12,000 + 280)\text{M} = \$12,280\text{M}$ .

NRC Cost: Estimated NRC costs for implementation are \$1.2M and \$10,000/RY for operation and maintenance review. Thus, total NRC costs are estimated to be  $\$1.2\text{M} + (\$10,000)[(90 \times 28.8) + (44 \times 27.4)] = \$38\text{M}$ .

#### Value/Impact Assessment

Based on a risk reduction of  $3.1 \times 10^5$  man-rem, the value/impact score is given by:

$$S = \frac{3.1 \times 10^5 \text{ man-rem}}{\$(12,280 + 38)\text{M}}$$

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

#### Other Considerations

The total industry cost benefit resulting from accident avoidance costs is calculated to be \$100M. Occupational dose calculations predict a total industry implementation dose of  $2.7 \times 10^4$  man-rem. This dose results principally



from a 1% increase in occupational dose to provide better identification (labels, etc.) for piping, valves, and other control devices. The operation and maintenance dose is believed to be nil. It is felt that the increase in preventive maintenance requirements will be more than offset by improved maintenance training, maintainability, and less frequent unplanned maintenance. Some have estimated as much as a 50% dose reduction to maintenance personnel.

### CONCLUSION

The total potential reduction justifies a HIGH priority ranking. However, the value/impact score is low, which indicates that care must be taken to select the best cost-beneficial alternatives. Hence, it is recommended that a value/impact analysis be accomplished before developing individual requirements to assure that the optimum benefit/cost ratios are implemented.

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NUREG-0933

## APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING  
AND FUTURE PLANTS

This appendix contains a listing of those safety issues that are applicable to operating plants as well as future plants. The priority designations for all issues are consistent with those listed in Table II of the Introduction. This listing includes: issues that have been resolved with new requirements [NOTE 3(a)]; USI, HIGH and MEDIUM priority issues that are under development; nearly-resolved issues (NOTES 1 and 2) whose impact is not yet known; issues that are scheduled for prioritization (NOTE 4); and issues that are covered in other issues that fall in any of the above categories.

### 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

- |        |   |
|--------|---|
| HIGH   | - High Safety Priority  |
| MEDIUM | - Medium Safety Priority  |
| HFPP   | - Human Factors Program Plan  |
| I      | - TMI Action Plan Item With Implementation of Resolution Mandated by NUREG-0737 <sup>98</sup> |
| NA     | - Not Applicable  |
| TBD    | - To Be Determined  |
| USI    | - Unresolved Safety Issue (See Status in NUREG-0606) <sup>69</sup>                            |

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
<u>TMI ACTION PLAN ITEMS</u>						
<u>I.A.</u>	<u>OPERATING PERSONNEL</u>					
<u>I.A.1</u>	<u>Operating Personnel and Staffing</u>					
I.A.1.1	Shift Technical Advisor	I	All	All	9/13/79	9/27/79
I.A.1.2	Shift Supervisor Administrative Duties	I	All	All	9/13/79	9/27/79
I.A.1.3	Shift Manning	I	All	All	7/31/80	6/26/80
I.A.1.4	Long-Term Upgrading	NOTE 3(a)	All	All	4/28/83	4/28/83
<u>I.A.2</u>	<u>Training and Qualifications of Operating Personnel</u>					
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-				
I.A.2.1(1)	Qualifications - Experience	I	All	All	3/28/80	3/28/80
I.A.2.1(2)	Training	I	All	All	3/28/80	3/28/80
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	I	All	All	3/28/80	3/28/80
I.A.2.3	Administration of Training Programs	I	All	All	3/28/80	3/28/80
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-				
I.A.2.6(1)	Revise Regulatory Guide 1.8	HFPP	-	-	-	-
I.A.2.6(4)	Operator Workshops	MEDIUM	All	All	TBD	TBD
<u>I.A.3</u>	<u>Licensing and Requalification of Operating Personnel</u>					
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	I	All	All	3/28/80	3/28/80
I.A.3.3	Requirements for Operator Fitness	HFPP	-	-	-	-
<u>I.A.4</u>	<u>Simulator Use and Development</u>					
I.A.4.1	Initial Simulator Improvement	-				
I.A.4.1(2)	Interim Changes in Training Simulators	NOTE 3(a)	All	All	4/-/81	3/28/81
I.A.4.2	Long-Term Training Simulator Upgrade	-				
I.A.4.2(1)	Research on Training Simulators	HFPP	-	-	-	-
I.A.4.2(2)	Upgrade Training Simulator Standards	NOTE 3(a)	All	All	4/-/81	4/-/81
I.A.4.2(3)	Regulatory Guide on Training Simulators	NOTE 3(a)	All	All	4/-/81	4/-/81
I.A.4.2(4)	Review Simulators for Conformance to Criteria	HFPP	-	-	-	-
<u>I.B.</u>	<u>SUPPORT PERSONNEL</u>					
<u>I.B.1</u>	<u>Management for Operations</u>					
I.B.1.1	Organization and Management Long-Term Improvements	-				

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
I.B.1.1(1)	Prepare Draft Criteria	HFPP	-	-	-	-
I.B.1.1(2)	Prepare Commission Paper	HFPP	-	-	-	-
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	HFPP	-	-	-	-
I.B.1.1(4)	Review Responses to Determine Acceptability	HFPP	-	-	-	-
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	75, HFPP	-	-	-	-
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	75, HFPP	-	-	-	-
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants	-				
I.B.1.2(1)	Prepare Draft Criteria	I	A11	A11	NA	6/26/80
I.B.1.2(2)	Review Near-Term Operating License Facilities	I	A11	A11	NA	6/26/80
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	I	A11	A11	NA	6/26/80
<u>I.C OPERATING PROCEDURES</u>						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-				
I.C.1(1)	Small Break LOCAs	I	A11	A11		
I.C.1(2)	Inadequate Core Cooling	I	A11	A11	9/13/79	9/13/79
I.C.1(3)	Transients and Accidents	I	A11	A11	9/13/79	9/27/79
I.C.2	Shift and Relief Turnover Procedures	I	A11	A11	9/13/79	9/27/79
I.C.3	Shift Supervisor Responsibilities	I	A11	A11	9/13/79	9/27/79
I.C.4	Control Room Access	I	A11	A11	9/13/79	9/27/79
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	I	A11	A11	5/7/80	6/26/80
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	I	A11	A11	10/31/80	10/31/80
I.C.7	NSSS Vendor Review of Procedures	I	A11	A11	NA	6/26/80
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	I	A11	A11	NA	6/26/80
I.C.9	Long-Term Program Plan for Upgrading of Procedures	HFPP	-	-	-	-
<u>I.D CONTROL ROOM DESIGN</u>						
I.D.1	Control Room Design Reviews	I	A11	A11	6/26/80	6/26/80
I.D.2	Plant Safety Parameter Display Console	I	A11	A11	6/26/80	6/26/80
I.D.3	Safety System Status Monitoring	HFPP	-	-	-	-
I.D.4	Control Room Design Standard	HFPP	-	-	-	-
I.D.5	Improved Control Room Instrumentation Research	-				
I.D.5(2)	Plant Status and Post-Accident Monitoring	NOTE 3(a)	A11	A11	NA	12/-/80
I.D.5(3)	On-Line Reactor Surveillance System	NOTE 1	A11	A11		
I.D.5(5)	Disturbance Analysis Systems	HFPP	-	-	-	-

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
<u>I.F.</u>	<u>QUALITY ASSURANCE</u>					
I.F.1	Expand QA List	HIGH	A11	A11	NA	TBD
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)	A11	A11	NA	7/-/81
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	NOTE 3(a)	A11	A11	NA	7/-/81
I.F.2(6)	Increase the Size of Licensees' QA Staff	NOTE 3(a)	A11	A11	NA	7/-/81
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	NOTE 3(a)	A11	A11	NA	7/-/81
<u>I.G.</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>					
I.G.1	Training Requirements	I	A11	A11	NA	6/26/81
I.G.2	Scope of Test Program	NOTE 3(a)	A11	A11	NA	7/-/81
<u>II.B.</u>	<u>CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW</u>					
II.B.1	Reactor Coolant System Vents	I	A11	A11	9/13/79	9/27/79
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	I	A11	A11	9/13/79	9/27/79
II.B.3	Post-Accident Sampling	I	A11	A11	9/13/79	9/27/79
II.B.4	Training for Mitigating Core Damage	I	A11	A11	3/28/80	3/28/80
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-				
II.B.5(1)	Behavior of Severely Damaged Fuel	HIGH	A11	A11	TBD	TBD
II.B.5(2)	Behavior of Core Melt	HIGH	A11	A11	TBD	TBD
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	MEDIUM	A11	A11	TBD	TBD
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	HIGH	A11	A11	TBD	NA
II.B.7	Analysis of Hydrogen Control	II.B.8	-	-	-	-
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	HIGH	A11	A11	TBD	TBD
<u>II.C.</u>	<u>RELIABILITY ENGINEERING AND RISK ASSESSMENT</u>					
II.C.1	Interim Reliability Evaluation Program	HIGH	A11	A11	TBD	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	HIGH	A11	A11	TBD	NA

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.C.3	Systems Interaction	A-17	-	-	-	-
II.C.4	Reliability Engineering	HIGH	A11	A11	TBD	TBD
<u>II.D</u>	<u>REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES</u>					
II.D.1	Testing Requirements	I	A11	A11	9/13/79	9/27/79
II.D.3	Relief and Safety Valve Position Indication	I	A11	A11	7/21/79	9/27/79
<u>II.E</u>	<u>SYSTEM DESIGN</u>					
<u>II.E.1</u>	<u>Auxiliary Feedwater System</u>					
II.E.1.1	Auxiliary Feedwater System Evaluation	I	NA	A11	3/10/80	3/10/80
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	I	NA	A11	9/13/79	9/27/79
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	NOTE 3(a)	A11	A11	NA	7/-/81
<u>II.E.2</u>	<u>Emergency Core Cooling System</u>					
II.E.2.1	Reliance on ECCS	II.K.3(17)	-	-	-	-
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	MEDIUM	A11	A11	TBD	TBD
<u>II.E.3</u>	<u>Decay Heat Removal</u>					
II.E.3.1	Reliability of Power Supplies for Natural Circulation	I	NA	A11	9/13/79	9/27/79
II.E.3.2	Systems Reliability	A-45	-	-	-	-
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	A-45	-	-	-	-
II.E.3.5	Regulatory Guide	A-45	-	-	-	-
<u>II.E.4</u>	<u>Containment Design</u>					
II.E.4.1	Dedicated Penetrations	I	A11	A11	9/13/79	9/27/79
II.E.4.2	Isolation Dependability	I	A11	A11	9/13/79	9/27/79
II.E.4.3	Integrity Check	HIGH	A11	A11	TBD	TBD
II.E.4.4	Purging	-	-	-	-	-
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	NOTE 3(a)	A11	A11	11/28/78	NA
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	NOTE 3(a)	A11	A11	10/22/79	NA
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	NOTE 3(a)	A11	A11	9/27/79	NA
<u>II.E.5</u>	<u>Design Sensitivity of B&amp;W Reactors</u>					
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		



## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.E.6	In Situ Testing of Valves					
II.E.6.1	Test Adequacy Study	MEDIUM	A11	A11	TBD	TBD
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>					
II.F.1	Additional Accident Monitoring Instrumentation	I	A11	A11	9/13/79	9/27/79
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	I	A11	A11	7/2/79	9/27/79
II.F.3	Instruments for Monitoring Accident Conditions	NOTE 3(a)	A11	A11	NA	12/-/80
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	MEDIUM	A11	A11	TBD	TBD
<u>II.G</u>	<u>ELECTRICAL POWER</u>					
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	A11	9/13/79	9/27/79
<u>II.H</u>	<u>TMI-2 CLEANUP AND EXAMINATION</u>					
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	NA	B&W	5/-/80	NA
II.H.3	Evaluate and Feed Back Information Obtained from TMI	II.H.2	-	-	-	-
II.J.3	Management for Design and Construction					
II.J.3.1	Organization and Staffing to Oversee Design and Construction	I.B.1.1	-	-	-	-
II.J.3.2	Issue Regulatory Guide	I.B.1.1	-	-	-	-
II.J.4	Revise Deficiency Reporting Requirements					
II.J.4.1	Revise Deficiency Reporting Requirements	NOTE 2	A11	A11	TBD	TBD
<u>II.K</u>	<u>MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS</u>					
II.K.1	IE Bulletins	-				
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	NOTE 3(a)	A11	A11	3/31/80	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	3/31/80	NA

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	A11	3/31/80	NA
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	A11	A11	3/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	A11	A11	3/31/80	3/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	A11	A11	3/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W	3/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W	3/31/80	NA
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	NOTE 3(a)	A11	A11	3/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	A11	A11	3/31/80	3/31/80
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)	A11	A11	3/31/80	NA
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	NOTE 3(a)	A11	A11		NA
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	NOTE 3(a)	A11	A11	1/1/81	1/1/81
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, W	3/31/80	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		NA
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	3/31/80	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	3/31/80	3/31/80

## Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	NOTE 3(a)	NA	B&W	3/31/80	3/31/80
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	3/31/80	3/31/80
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA	3/31/80	3/31/80
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	1/1/81	1/1/82
II.K.2	Commission Orders on B&W Plants	-				
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	NOTE 3(a)	NA	B&W		NA
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	NOTE 3(a)	NA	B&W		NA
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	NOTE 3(a)	NA	B&W		NA
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	NOTE 3(a)	NA	B&W		NA
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	NOTE 3(a)	NA	B&W		NA
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	NOTE 3(a)	NA	B&W		NA
II.K.2(7)	Reevaluate Transient of September 24, 1977	NOTE 3(a)	NA	B&W		NA
II.K.2(8)	Continued Upgrading of AFW System	II.E.1.1, II.E.1.2	-	-	-	-
II.K.2(9)	Analysis and Upgrading of Integrated Control System	I	NA	B&W	1/1/81	1/1/81
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	I	NA	B&W	1/1/81	1/1/81
II.K.2(11)	Operator Training and Drilling	I	NA	B&W	1/1/81	1/1/81
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	I.C.1	-	-	-	-
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	1/1/81	1/1/81
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	I	NA	B&W	1/1/81	1/1/81
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	I	NA	B&W	6/1/80	6/1/80
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	I	NA	B&W	6/1/80	6/1/80
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	I	NA	B&W		NA
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	I.C.1	-	-	-	-

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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	I	NA	B&W	1/1/81	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	1/1/81	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	7/1/81	7/1/81
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	I	NA	All	1/1/81	1/1/81
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	I	All	All	4/1/80	4/1/80
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	II.C.1, II.C.2, II.C.3	-	-	-	-
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	I	NA	All	1/1/81	1/1/81
II.K.3(6)	Instrumentation to Verify Natural Circulation	I.C.1, II.F.2, II.F.3	-	-	-	-
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	I	NA	B&W	1/1/81	1/1/81
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	II.C.1, II.E.3.3	-	-	-	-
II.K.3(9)	Proportional Integral Derivative Controller Modification	I	NA	W	7/1/80	7/1/80
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	I	NA	W		
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	7/1/80	7/1/80
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	I	GE	NA	10/1/80	10/1/80
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	I	GE	NA	1/1/81	NA
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	I	GE	NA	1/1/81	1/1/81
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	I	GE	NA	1/1/81	1/1/81
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	I	GE	NA	1/1/81	1/1/81
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	I	GE	NA	1/1/81	1/1/81
II.K.3(19)	Interlock on Recirculation Pump Loops	I	GE	NA	1/1/81	NA

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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.K.3(20)	Loss of Service Water for Big Rock Point	I	GE	NA	1/1/81	NA
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	1/1/81	1/1/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	1/1/81	1/1/81
II.K.3(23)	Central Water Level Recording	I.D.2, III.A.1.2, III.A.3.4	-	-	-	-
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	1/1/82	1/1/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	1/1/82	1/1/82
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	II.E.2.1	-	-	-	-
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	10/1/80	10/1/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	1/1/82	1/1/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	4/1/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	1/1/83	1/1/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	1/1/83	1/1/83
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	II.E.2.2	-	-	-	-
II.K.3(33)	Evaluate Elimination of PORV Function	II.C.1	-	-	-	-
II.K.3(34)	Relap-4 Model Development	II.E.2.2	-	-	-	-
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	I.C.1	-	-	-	-
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	I.C.1	-	-	-	-
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	I.C.1	-	-	-	-
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	I.C.1	-	-	-	-
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	I.C.1	-	-	-	-
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	II.K.2(16)	-	-	-	-
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	I.C.1	-	-	-	-
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensable Gases	I.C.1	-	-	-	-
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	II.K.2(15)	-	-	-	-



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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	1/1/81	1/1/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	1/1/81	1/1/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	7/1/80	7/1/80
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	I.C.1, II.E.2.2	-	-	-	-
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	II.C.1, II.C.2	-	-	-	-
II.K.3(49)	Review of Procedures (NRC)	I.C.8, I.C.9	-	-	-	-
II.K.3(50)	Review of Procedures (NSSS Vendors)	I.C.7, I.C.9	-	-	-	-
II.K.3(51)	Symptom-Based Emergency Procedures	I.C.9	-	-	-	-
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	I.B.1.1, I.C.2, I.C.5	-	-	-	-
II.K.3(53)	Two Operators in Control Room	I.A.1.3	-	-	-	-
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	I.A.4.1	-	-	-	-
II.K.3(55)	Operator Monitoring of Control Board	I.C.1, I.D.2, I.D.3	-	-	-	-
II.K.3(56)	Simulator Training Requirements	I.A.2.6, I.A.3.1	-	-	-	-
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	10/1/80	NA
<u>III.A</u>	<u>EMERGENCY PREPAREDNESS AND RADIATION EFFECTS</u>					
<u>III.A.1</u>	<u>Improve Licensee Emergency Preparedness - Short Term</u>					
III.A.1.1	Upgrade Emergency Preparedness	-				
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	I	A11	A11	10/10/79	8/19/80
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	I	A11	A11	10/10/79	8/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-				
III.A.1.2(1)	Technical Support Center	I	A11	A11	9/13/79	9/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	A11	A11	9/13/79	9/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	A11	A11	9/13/79	9/27/79
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-				
III.A.1.3(2)	Public	NOTE 1	A11	A11	5/-/80	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	I	A11	A11		

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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
III.A.2.1(2)	Conduct Public Regional Meetings	I	A11	A11		
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	I	A11	A11		
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	A11	A11		
III.A.2.2	Development of Guidance and Criteria	I	A11	A11		
III.A.3	Improving NRC Emergency Preparedness					
III.A.3.3	Communications	-				
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	NOTE 3(a)	A11	A11		
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	NOTE 3(a)	A11	A11		
<u>III.D</u>	<u>RADIATION PROTECTION</u>					
III.D.1	Radiation Source Control					
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-				
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	I	A11	A11	7/2/79	9/27/79
III.D.1.1(2)	Review Information on Provisions for Leak Detection	NOTE 4	TBD	TBD	TBD	TBD
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	NOTE 4	TBD	TBD	TBD	TBD
III.D.2	Public Radiation Protection Improvement					
III.D.2.3	Liquid Pathway Radiological Control	-				
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	NOTE 1	A11	A11	TBD	TBD
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	NOTE 1	A11	A11	TBD	TBD
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	NOTE 1	A11	A11	TBD	TBD
III.D.2.3(4)	Prepare a Summary Assessment	NOTE 1	A11	A11	TBD	TBD
III.D.3	Worker Radiation Protection Improvement					
III.D.3.1	Radiation Protection Plans	HIGH	A11	A11	TBD	TBD
III.D.3.3	Implant Radiation Monitoring	-				
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	A11	A11	9/13/79	9/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	I	A11	A11	9/13/79	9/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	I	A11	A11	9/13/79	9/27/79
III.D.3.3(4)	Issue a Regulatory Guide	I	A11	A11	9/13/79	9/27/79
III.D.3.4	Control Room Habitability	I	A11	A11	5/7/80	6/26/80

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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
<u>IV.E</u>	<u>SAFETY DECISION-MAKING</u>					
IV.E.5	Assess Currently Operating Reactors	HIGH	A11	A11	TBD	NA
<u>TASK ACTION PLAN ITEMS</u>						
A-1	Water Hammer	USI [NOTE 3(a)]	A11	A11	NA	3/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	USI [NOTE 3(a)]	NA	A11	1/-/81	1/-/81
A-3	Westinghouse Steam Generator Tube Integrity	USI	NA	W	TBD	TBD
A-4	CE Steam Generator Tube Integrity	USI	NA	CE	TBD	TBD
A-5	B&W Steam Generator Tube Integrity	USI	NA	B&W	TBD	TBD
A-6	Mark I Short-Term Program	USI [NOTE 3(a)]	GE	NA	12/-/77	NA
A-7	Mark I Long-Term Program	USI [NOTE 3(a)]	GE	NA	8/-/82	8/-/82
A-8	Mark II Containment Pool Dynamic Loads - Long Term Program	USI [NOTE 3(a)]	GE	NA	8/-/81	8/-/81
A-9	ATWS	USI [NOTE 3(a)]	A11	A11	6/26/84	6/26/84
A-10	BWR Feedwater Nozzle Cracking	USI [NOTE 3(a)]	A11	NA	11/-/80	11/-/80
A-11	Reactor Vessel Materials Toughness	USI [NOTE 3(a)]	A11	A11	10/-/82	NA
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	USI [NOTE 2]	NA	A11	NA	TBD
A-13	Snubber Operability Assurance	NOTE 3(a)	A11	A11	1980	1981
A-16	Steam Effects on BWR Core Spray Distribution	NOTE 3(a)	GE	NA		NA
A-17	Systems Interaction	USI	A11	A11	TBD	TBD
A-19	Digital Computer Protection System	NOTE 4	A11	A11	TBD	TBD
A-24	Qualification of Class 1E Safety Related Equipment	USI [NOTE 3(a)]	A11	A11	8/-/81	8/-/81
A-25	Non-Safety Loads on Class 1E Power Sources	NOTE 3(a)	A11	A11		9/-/78
A-26	Reactor Vessel Pressure Transient Protection	USI [NOTE 3(a)]	NA	A11	9/-/78	9/-/78
A-28	Increase in Spent Fuel Pool Storage Capacity	NOTE 3(a)	A11	A11	4/17/78	NA
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	MEDIUM	A11	A11	TBD	TBD
A-30	Adequacy of Safety-Related DC Power Supplies	HIGH	A11	A11	TBD	TBD
A-31	RHR Shutdown Requirements	USI [NOTE 3(a)]	A11	A11	5/-/78	1/1/79
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	II.F.3	-	-	-	-
A-35	Adequacy of Offsite Power Systems	NOTE 3(a)	A11	A11	6/2/77	1981
A-36	Control of Heavy Loads Near Spent Fuel	USI [NOTE 3(a)]	A11	A11	7/-/80	7/-/80
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	USI [NOTE 3(a)]	GE	NA	2/29/80	9/30/82
A-40	Seismic Design Criteria - Short Term Program	USI	A11	A11	TBD	TBD
A-42	Pipe Cracks in Boiling Water Reactors	USI [NOTE 3(a)]	A11	NA	2/-/81	2/-/81
A-43	Containment Emergency Sump Performance	USI	NA	A11	TBD	TBD
A-44	Station Blackout	USI	A11	A11	TBD	TBD

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Action Plan Item/ Issue No.	Title	Safety Priority Ranking	Affected NSSS Vendor		Operating Plants- Effective Date	Future Plants- Effective Date
			BWR	PWR		
A-45	Shutdown Decay Heat Removal Requirements	USI	A11	A11	TBD	TBD
A-46	Seismic Qualification of Equipment in Operating Plants	USI	A11	A11	TBD	NA
A-47	Safety Implications of Control Systems	USI	A11	A11	TBD	TBD
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	USI	A11	W		
A-49	Pressurized Thermal Shock	USI	NA	A11	TBD	TBD
B-4	ECCS Reliability	II.E.3.2	-	-	-	-
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	MEDIUM	A11	A11	TBD	TBD
B-6	Loads, Load Combinations, Stress Limits	HIGH	A11	A11	TBD	TBD
B-10	Behavior of BWR Mark III Containments	NOTE 3(a)	GE	NA		9/-/84
B-12	Containment Cooling Requirements (Non-LOCA)	NOTE 3(a)	A11	A11	NA	
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	A-48	-	-	-	-
B-17	Criteria for Safety Related Operator Actions	MEDIUM	A11	A11	TBD	TBD
B-18	Vortex Suppression Requirements for Containment Sumps	A-43	-	-	-	-
B-22	LWR Fuel	NOTE 4	A11	A11	TBD	TBD
B-24	Seismic Qualification of Electrical and Mechanical Components	A-46	-	-	-	-
B-29	Effectiveness of Ultimate Heat Sinks	NOTE 4	A11	A11	TBD	TBD
B-31	Dam Failure Model	NOTE 4	A11	A11	TBD	TBD
B-32	Ice Effects on Safety Related Water Supplies	NOTE 4	A11	A11	TBD	TBD
B-34	Occupational Radiation Exposure Reduction	III.D.3.1	-	-	-	-
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)	A11	A11		3/-/78
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	A-40	-	-	-	-
B-52	Fuel Assembly Seismic and LOCA Responses	A-2	-	-	-	-
B-55	Improved Reliability of Target Rock Safety Relief Valves	MEDIUM	A11	NA	TBD	TBD
B-56	Diesel Reliability	HIGH	A11	A11	TBD	TBD
B-57	Station Blackout	A-44	-	-	-	-
B-58	Passive Mechanical Failures	MEDIUM	A11	A11	TBD	TBD
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	A11	A11	TBD	TBD
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	NOTE 3(a)	A11	A11	4/20/81	
B-64	Decommissioning of Reactors	NOTE 2	A11	A11	TBD	NA
B-66	Control Room Infiltration Measurements	NOTE 3(a)	A11	A11	NA	7/-/81
B-69	ECCS Leakage Ex-Containment	III.D.1.1	-	-	-	-
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	NOTE 3(a)	A11	A11	NA	7/-/81

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			BWR	PWR		
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	NOTE 3(a)	A11	A11	5/27/80	5/27/80
C-3	Insulation Usage Within Containment	A-43	-	-	-	-
C-4	Statistical Methods for ECCS Analysis	NOTE 4	A11	A11	TBD	TBD
C-5	Decay Heat Update	NOTE 4	A11	A11	TBD	TBD
C-6	LOCA Heat Sources	NOTE 4	A11	A11	TBD	TBD
C-8	Main Steam Line Leakage Control Systems	HIGH	A11	NA	TBD	TBD
C-10	Effective Operation of Containment Sprays in a LOCA	NOTE 3(a)	A11	A11	NA	-
C-11	Assessment of Failure and Reliability of Pumps and Valves	MEDIUM	A11	A11	TBD	TBD
C-13	Non-Random Failures	A-17	-	-	-	-
C-14	Storm Surge Model for Coastal Sites	NOTE 4	A11	A11	TBD	TBD
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	NOTE 3(a)	A11	A11	12/27/82	12/27/82
D-2	Emergency Core Cooling System Capability for Future Plants	NOTE 4	A11	A11	NA	TBD

NEW GENERIC ISSUES

2.	Failure of Protective Devices on Essential Equipment	NOTE 4	A11	A11	TBD	TBD
3.	Set Point Drift in Instrumentation	NOTE 2	A11	A11	TBD	TBD
5.	Design Check and Audit of Balance-of-Plant Equipment	I.F.1	-	-	-	-
8.	Inadvertent Actuation of Safety Injection in PWRs	I.C.1	-	-	-	-
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	II.K.3(5)	-	-	-	-
14.	PWR Pipe Cracks	NOTE 2	NA	A11	TBD	TBD
16.	BWR Main Steam Isolation Valve Leakage Control Systems	C-8	-	-	-	-
18.	Steam Line Break with Consequential Small LOCA	I.C.1	-	-	-	-
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	A-47	-	-	-	-
21.	Vibration Qualification of Equipment	NOTE 4	A11	A11	TBD	TBD
23.	Reactor Coolant Pump Seal Failures	HIGH	A11	A11	TBD	TBD
24.	Automatic Emergency Core Cooling System Switch to Recirculation	NOTE 4	A11	A11	TBD	TBD
25.	Automatic Air Header Dump on BWR Scram System	NOTE 3(a)	A11	NA	1/9/81	1/9/81
27.	Manual vs. Automated Actions	B-17	-	-	-	-
28.	Pressurized Thermal Shock	A-49	-	-	-	-
29.	Bolting Degradation or Failure in Nuclear Power Plants	HIGH	A11	A11	TBD	TBD
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	NOTE 4	A11	A11	TBD	TBD
31.	Natural Circulation Cooledown	I.C.1	-	-	-	-
32.	Flow Blockage in Essential Equipment Caused by Corbicula	51	-	-	-	-
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	A-47	-	-	-	-
36.	Loss of Service Water	NOTE 1	A11	A11	TBD	TBD

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			BWR	PWR		
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	A-47, I.C.1	-	-	-	-
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	NOTE 4	A11	A11	TBD	TBD
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	25	-	-	-	-
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	NOTE 3(a)	A11	NA	8/31/81	8/31/81
41.	BWR Scram Discharge Volume Systems	NOTE 3(a)	A11	NA	12/9/80	NA
42.	Combination Primary/Secondary System LOCA	I.C.1	-	-	-	-
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	NOTE 3(a)	A11	A11	NA	9/1/83
46.	Loss of 125 Volt DC Bus	76	-	-	-	-
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	NOTE 2	A11	A11	TBD	TBD
49.	Interlocks and LCOs for Redundant Class 1E Tie Breakers	MEDIUM	A11	A11	TBD	NA
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	MEDIUM	A11	A11	TBD	TBD
52.	SSW Flow Blockage by Blue Mussels	51	-	-	-	-
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	II.E.6.1	-	-	-	-
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	NOTE 4	A11	A11	TBD	TBD
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	A-47, I.D.1	-	-	-	-
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	NOTE 4	A11	A11	TBD	TBD
60.	Lamellar Tearing of Reactor Systems Structural Supports	A-12	-	-	-	-
61.	SRV Line Break Inside the BWR Wetwell Airspace of of Mark and II Containments	MEDIUM	GE	NA	TBD	TBD
62.	Reactor Systems Bolting Applications	NOTE 4	A11	A11	TBD	TBD
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	NOTE 4	A11	NA	TBD	TBD
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	HIGH	A11	A11	TBD	TBD
66.	Steam Generator Requirements	NOTE 2	NA	A11	TBD	TBD
67.	Steam Generator Staff Actions	-	-	-	-	-
67.3.1	Steam Generator Overfill	A-47, I.C.1	-	-	-	-
67.3.2	Pressurized Thermal Shock	A-49	-	-	-	-
67.3.3	Improved Accident Monitoring	I	A11	A11	12/17/82	12/17/82
67.3.4	Reactor Vessel Inventory Measurement	II.F.2	-	-	-	-
67.4.1	RCP Trip	II.K.3(5)	-	-	-	-
67.4.2	Control Room Design Review	I.D.1	-	-	-	-
67.4.3	Emergency Operating Procedures	I.C.1	-	-	-	-



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			BWR	PWR		
67.6.0	Organizational Responses	III.A.3	-	-	-	-
67.7.0	Improved Eddy Current Tests	MEDIUM	NA	A11	TBD	TBD
67.9.0	Reactor Coolant System Pressure Control	A-45, I.C.1	-	-	-	-
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	HIGH	NA	B&W, CE	TBD	TBD
70.	PORV and Block Valve Reliability	MEDIUM	NA	A11	TBD	TBD
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	NOTE 4	A11	A11	TBD	TBD
72.	Control Rod Drive Guide Tube Support Pin Failures	NOTE 4	NA	W	TBD	TBD
73.	Detached Thermal Sleeves	NOTE 4	A11	A11	TBD	TBD
74.	Reactor Coolant Activity Limits for Operating Reactors	NOTE 4	A11	A11	TBD	NA
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	NOTE 1	A11	A11	TBD	TBD
76.	Instrumentation and Control Power Interactions	NOTE 4	A11	A11	TBD	TBD
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	HIGH	A11	A11	TBD	TBD
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	NOTE 4	A11	A11	TBD	TBD
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	NA	B&W	TBD	TBD
82.	Beyond Design Basis Accidents in Spent Fuel Pools	MEDIUM	A11	A11	TBD	TBD
83.	Control Room Habitability	NOTE 4	A11	A11	TBD	TBD
84.	CE PORVs	NOTE 1	NA	CE	TBD	TBD
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	NOTE 4	A11	NA	TBD	TBD
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	NOTE 2	A11	NA	TBD	TBD
87.	Failure of HPCI Steam Line Without Isolation	NOTE 4	A11	A11	TBD	TBD
88.	Earthquakes and Emergency Planning	NOTE 4	A11	A11	TBD	TBD
89.	Stiff Pipe Clamps	NOTE 4	A11	A11	TBD	TBD
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	NOTE 4	A11	A11	TBD	TBD
93.	Steam Binding of Auxiliary Feedwater Pumps	HIGH	NA	A11	TBD	TBD
94.	Additional Low Temperature Overpressure Protection	NOTE 4	NA	A11	TBD	TBD
95.	Issues for Light Water Reactors	NOTE 4	A11	A11	TBD	TBD
96.	Loss of Effective Volume for Containment Recirculation Spray	NOTE 4	A11	A11	TBD	TBD
97.	RHR Suction Valve Testing	NOTE 4	A11	A11	TBD	TBD
99.	PWR Reactor Cavity Uncontrolled Exposures	III.D.3.1	-	-	-	-
100.	RCS/RHR Suction Line Valve Interlock on PWRs	NOTE 4	NA	A11	TBD	TBD
101.	OTSG Level	NOTE 4	NA	B&W	TBD	TBD
101.	Break Plus Single Failure in BWR Water Level Instrumentation	NOTE 4	A11	NA	TBD	TBD

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102.	Human Error in Events Involving Wrong Unit or Wrong Train	HF02	-	-	-	-
103.	Design for Probable Maximum Precipitation	NOTE 4	A11	A11	TBD	TBD
104.	Reduction of Boron Dilution Requirements	NOTE 4	NA	A11	TBD	TBD
105.	Interfacing Systems LOCA at BWRs	HIGH	A11	NA	TBD	TBD
106.	Piping and Use of Highly Combustible Gases in Vital Areas	NOTE 4	A11	A11	TBD	TBD
107.	Generic Implications of Main Transformer Failures	NOTE 4	A11	A11	TBD	TBD
109.	Reactor Vessel Closure Failure	NOTE 4	A11	A11	TBD	TBD
110.	Equipment Protective Devices on Engineered Safety Features	NOTE 4	A11	A11	TBD	TBD
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	NOTE 4	A11	A11	TBD	TBD
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	NOTE 4	NA	W	TBD	TBD
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	NOTE 4	A11	A11	TBD	TBD
114.	Seismic-Induced Relay Chatter	NOTE 4	A11	A11	TBD	TBD
115.	Reliability of Westinghouse Solid State Protection System	NOTE 4	NA	W	TBD	TBD
116.	Accident Management	NOTE 4	A11	A11	TBD	TBD

HUMAN FACTORS ISSUESHF01 HUMAN FACTORS PROGRAM PLAN (HFPP)

HF01.1.0	Staffing and Qualifications	-				
HF01.1.1	NPP Staffing Requirements	HIGH	A11	A11	TBD	TBD
HF01.1.2	NPP Personnel Qualifications Requirements	HIGH	A11	A11	TBD	TBD
HF01.1.3	Guidance on Limits and Conditions of Shift Work	HIGH	A11	A11	TBD	TBD
HF01.1.4	Fitness for Duty	HIGH	A11	A11	TBD	TBD
HF01.2.0	Training	-				
HF01.2.1	Development of Training Regulation and Guidance	HIGH	A11	A11	TBD	TBD
HF01.2.2	NRC Training Evaluation Program	HIGH	A11	A11	TBD	TBD
HF01.3.0	Licensing Examination	-				
HF01.3.1	The Examination Content	HIGH	A11	A11	TBD	TBD
HF01.3.2	The Examination Process	HIGH	A11	A11	TBD	TBD
HF01.4.0	Procedures	-				
HF01.4.1	Procedures Guidance and Criteria	HIGH	A11	A11	TBD	TBD

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			BWR	PWR		
HF01.5.0	Man-Machine Interface (MMI)	-				
HF01.5.1	MMI Guidance for Existing Designs	HIGH	A11	A11	TBD	TBD
HF01.5.2	MMI Guidance for Designs Based on Advanced Technologies	HIGH	A11	A11	TBD	TBD
HF01.6.0	Management and Organization	-				
HF01.6.1	Regulatory Position on Management and Organization at Operating Reactors	HIGH	A11	A11	TBD	TBD
HF01.6.2	NRC Management and Organization Guidelines and Assessment Procedures for Operating License Reviews	HIGH	A11	A11	TBD	TBD
HF02	Maintenance and Surveillance Program	HIGH	A11	A11	TBD	TBD