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SUPPLEMENT 1 TO NUREG-0933, "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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INTRODUCTION

I. BACKGROUND

A. NRR Operating Plan

The NRC Policy and Planning Guidance, 1983 (NUREG-0885,[210] 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).[48] These issues are covered in Section 1. The priority recommendations in the present report exclude the issues that are being implemented under NUREG-0737.[98]

2. Task Action Plan items, previously-proposed issues in NUREG-0371[2] and NUREG-0471[3], 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. Issues identified by AEOD and others will be prioritized in future supplements to this report and included in Section 3.

4. Human Factors Program Plan (HFPP) items under development in NRR and outlined in NUREG-0985.[603] 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), DST, 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).

Resolution of an issue is considered complete, indicated by NOTE 3 in Table II, when resolution has resulted in the establishment of regulatory requirements or guidance (by rule, Standard Review Plan change, or equivalent) or in a documented authoritative decision that no change in requirements is warranted. 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).

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 or 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. Generally, a LOW priority indicates that it is not clear from currently available information whether the issue merits pursuit. Development of additional information bearing on the merits of the issue could clarify whether pursuit with a MEDIUM priority or a decision to DROP are warranted.

The DROP category covers proposed issues that are without merit or whose significance is clearly negligible. They are recommended for summary elimination 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 simplied 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,[1] 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:

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

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,[16] 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 midwest site (Braidwood) meteorology was used for the dose calculations. However, the calculated doses were adjusted to reflect the mean of the population density within a 50-mile radius of U.S. nuclear power plants.[64] Assumptions and parameters used for the calculations 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,[70] 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.[64]

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[64] cited above) and multiplying by the estimated number of man-hours involved.

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

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

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 estimateu 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[392]) 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,[64] Section 5) for the general case of combining uncertainties for random variables with unknown distributions (as well as some special cases) are used. (See also Paragraph 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.

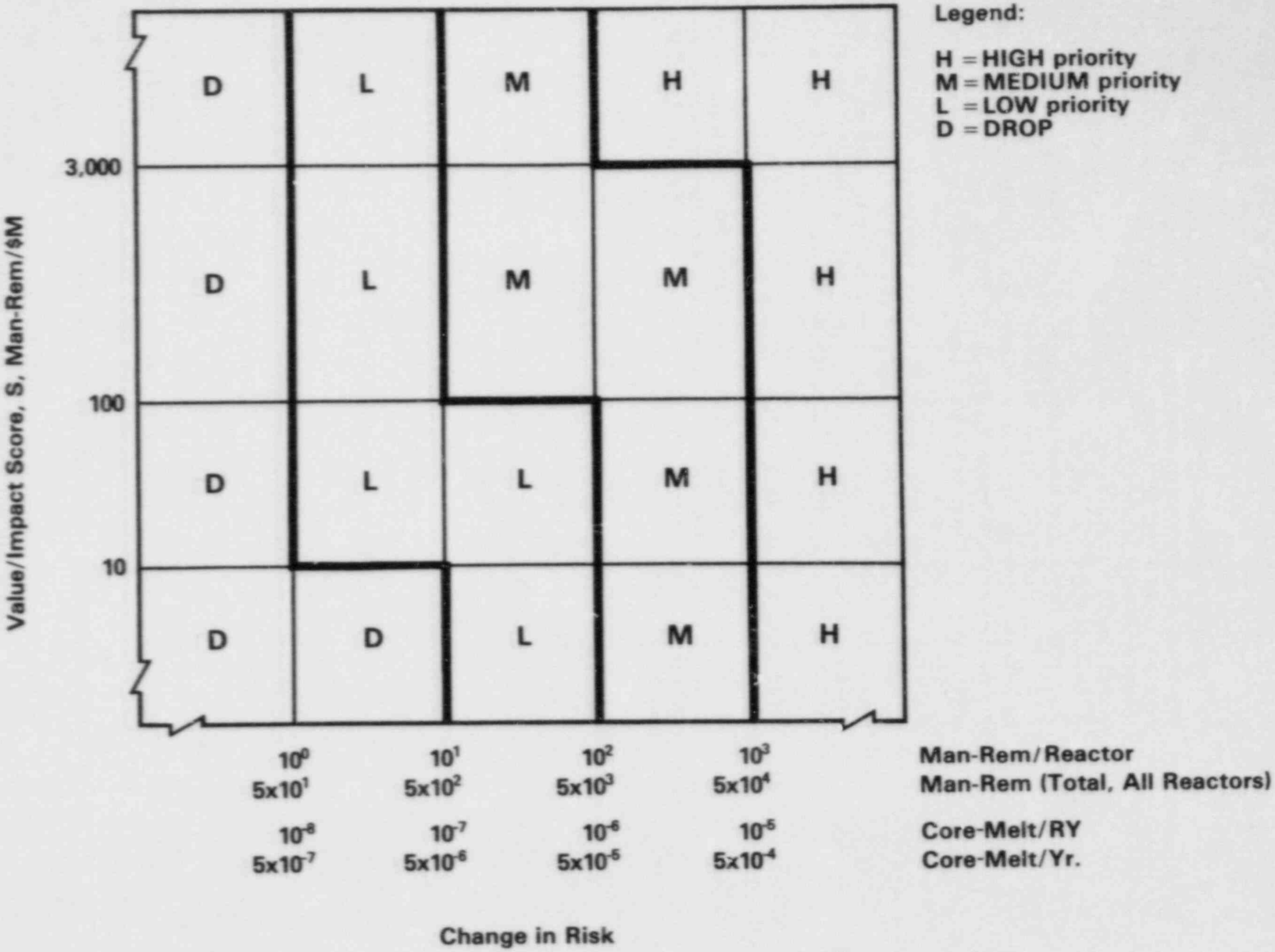

Figure 1-Priority Ranking

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) 5×10^{-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 in Table III.

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, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission.

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 PROGRAM PLAN ITEMS

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
I - TMI Action Plan Item With Implementation of Resolution Mandated by NUREG-0737[98]
LI - Licensing Issue
MPA - Multi-Plant Action (See Status in NUREG-0748)[57]
NA - Not Applicable
RI - Regulatory Impact Issue
USI - Unresolved Safety Issue (See Status in NUREG-0606)[60]

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I			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/DFO/HFBR	NOTE 3	1	6/30/84	
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-	-	-			
I.A.2.1(1)	Qualifications - Experience	-	NRR/DHFS/LQB	I			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	HIGH			
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I			
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/OLB	LI			NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	LOW			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	HIGH			
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3			
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2			NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	MEDIUM			
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3			
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP			NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	MEDIUM			
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	1	6/30/84	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3	1	6/30/84	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DFO/HFBR	HIGH	1	6/30/84	
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	MEDIUM	1	6/30/84	
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	1	6/30/84	NA
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-			
I.A.4.1(1)	Short-Term Study of Training Simulators	Thatcher	NRR/DHFS/LQB	NOTE 3			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/LQB	NOTE 3			
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	RES/DFO/HFBR	HIGH			
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3			
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3			
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DHFS/LQB	HIGH			
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)			NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI			
I.B.	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFS/LQB	MEDIUM			
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFS/LQB	MEDIUM			
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFS/LQB	MEDIUM			
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFS/LQB	MEDIUM			
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3			NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	RES/DFO/HFBR	MEDIUM			
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	RES/DFO/HFBR	MEDIUM			
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/DL/ORAB	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)			NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)			NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)			NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)			NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision 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)			NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)			NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)			NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)			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)			NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)			NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)			NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)			NA
I.C	OPERATING PROCEDURES						
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			
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	I			
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR	I			
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PTRB	MEDIUM			
I.D	CONTROL ROOM DESIGN						
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	MEDIUM			
I.D.4	Control Room Design Standard	Thatcher	RES/DFO/HFBR	MEDIUM			
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3			
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3			
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DFO/ICBR	NOTE 1			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3			
I.D.5(5)	Disturbance Analysis Systems	Thatcher	RES/DFO/HFBR	MEDIUM			
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)			NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	1	6/30/84	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	1	6/30/84	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	1	6/30/84	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	1	6/30/84	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	Pittman	OIE/DQASIP/QUAB	HIGH			
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	LOW			NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3			NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3			NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW			NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW			NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3			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			NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW			NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3			NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW			NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR	I			
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PTRB	MEDIUM			
II.A	SITING						
II.A.1	Siting Policy Reformulation	V'Molen	NRR/DE/SAB	MEDIUM			
II.A.2	Site Evaluation of Existing Facilities	V'Molen	NRR/DE/SAB	V.A.1			
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
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			
II.B.5(2)	Behavior of Core Melt	V'Molen	RES/DAE/CSRB	HIGH			
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DAE/CSRB	MEDIUM			
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DSI/RSB	HIGH			
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8			
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/ASTOP	HIGH			
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRA/RRBR	HIGH			
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	HIGH			
II.C.3	Systems Interaction	Pittman	NRR/DST/RRAB	A-17			
II.C.4	Reliability Engineering	Pittman	RES/DRA/RRBR	HIGH			
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I			F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW			
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	Revision Date	MPA No.
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I			F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I			F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3			
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3			
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	MEDIUM			
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW			
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			
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45			
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3			
II.E.3.5	Regulatory Guide	Riggs	RES/DAE/FBRB	A-45			
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			
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3			
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3			
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3			
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3			
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3			
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3			
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 2			
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	Thatcher	NRR/DE/EQB	MEDIUM			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
II.F	INSTRUMENTATION AND CONTROLS						
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			
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP			
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DFO/ICBR	MEDIUM			
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I			
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3			
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DAE/FBRB	HIGH			
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	II.H.2			NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SEBR	LI (NOTE 3)			NA
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)			NA
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)			NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)			NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)			NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)			NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)			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			NA
II.J.3.2	Issue Regulatory Guide	Pittman	RES/DFO/HFBR	I.B.1.1			NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	RES/DRA/RABR	NOTE 2			
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 All Safety-Related Valve Positions and Positioning Requirements and Positive Controls	-	NRR	I			
II.K.1(2)	Review and Modify Procedures for Removing Safety-Related Systems from Service	-	NRR	I			
II.K.1(3)	Provide a Trip for the Pressurizer Low-Level Bistable	-	NRR	I			
II.K.1(4)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip	-	NRR	I			
II.K.1(5)	Provide Automatic Safety-Grade Anticipatory Reactor Trip	-	NRR	I			
II.K.1(6)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems	-	NRR	I			
II.K.1(7)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation of Safety Systems	-	NRR	I			
II.K.2	Commission Orders on B&W Plants	-	NRR	I			
II.K.3	Final Recommendations of B&O Task Force	-	NRR	I			
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness		-	-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I			
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	I			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
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			
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 1			
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E		-	-			
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	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
III.A.3	Improving NRC Emergency Preparedness						
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			NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	Riggs	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	MEDIUM			
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3			NA
III.A.3.6	Interaction of NRC and Other Agencies	-	-	-			
III.A.3.6(1)	International	Pittman	OIE/DEPER/EPLB	NOTE 3			NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3			NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3			NA
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	Milstead	OIE/DEPER/IRDB	NOTE 3			NA
III.B.2	Implementation of NRC and FEMA Responsibilities	-	-	-			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
III.B.2(1)	The Licensing Process	Milstead	OIE/DEPER/IRDB	NOTE 3			NA
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3			NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public	-	-	-			
III.C.1(1)	Review Publicly Available Documents	Pittman	PA	LI (NOTE 3)			NA
III.C.1(2)	Recommend Publication of Additional Information	Pittman	PA	LI (NOTE 3)			NA
III.C.1(3)	Program of Seminars for News Media Personnel	Pittman	PA	LI (NOTE 3)			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)			NA
III.C.2(2)	Provide Training for Members of the Technical Staff	Pittman	PA	LI (NOTE 3)			NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-			
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I			
III.D.1.1(2)	Review Information on Provisions for Leak Detection	-	NRR/DSI/METB	NOTE 4			
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	-	NRR/DSI/METB	NOTE 4			
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP			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			NA
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP			NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP			NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3			NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP			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
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

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
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 .ehavior	Emrit	NRR/DSI/RAB	NOTE 3	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	1	6/30/84	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	1	6/30/84	NA
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3	1	6/30/84	
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	1	6/30/84	NA
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
III.D.3.1	Radiation Protection Plans	V'Molen	NRR/DSI/RAB	HIGH			
III.D.3.2	Health Physics Improvements	-	-	-			
III.D.3.2(1)	Amend 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)			
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)			NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)			
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)			
III.D.3.3	Inplant 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/DFO/ORPBR	LI			
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)			NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)			NA

TABLE II (Continued)

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

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)			NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI			
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)			NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)			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)			NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer	-	NRR/DST/GIB	USI			
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	-	NRR/DST/GIB	USI			D-10
A-3	Westinghouse Steam Generator Tube Integrity	-	NRR/DST/GIB	USI			
A-4	CE Steam Generator Tube Integrity	-	NRR/DST/GIB	USI			
A-5	B&W Steam Generator Tube Integrity	-	NRR/DST/GIB	USI			
A-6	Mark I Short Term Program	-	NRR/DST/GIB	USI			
A-7	Mark I Long Term Program	-	NRR/DST/GIB	USI			D-01
A-8	Mark II Containment Pool Dyanmic Loads - Long Term Program	-	NRR/DST/GIB	USI			
A-9	ATWS	-	NRR/DST/GIB	USI			
A-10	BWR Feedwater Nozzle Cracking	-	NRR/DST/GIB	USI			B-25
A-11	Reactor Vessel Materials Toughness	-	NRR/DST/GIB	USI			
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	-	NRR/DST/GIB	USI			
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3			
A-14	Flaw Detection	Matthews	NRR/DE/MTEB	DROP			
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	Pittman	NRR/DE/CHEB	NOTE 3			
A-16	Steam Effects on BWR Core Spray Distribution	Emrit	NRR/DSI/CPB	NOTE 3			D-12
A-17	Systems Interaction	-	NRR/DST/GIB	USI			
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP			
A-19	Digital Computer Protection System	Thatcher	NRR/DSI/ICSB	NOTE 4			
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI			
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	V'Molen	NRR/DSI/CSB	LOW			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	V'Molen	NRR/DSI/CSB	DROP			NA
A-23	Containment Leak Testing	Matthews	NRR/DSI/CSB	RI			
A-24	Qualification of Class 1E Safety Related Equipment	-	NRR/DST/GIB	USI			B-60
A-25	Non-Safety Loads on Class 1E Power Sources	Thatcher	NRR/DSI/PSB	NOTE 3			
A-26	Reactor Vessel Pressure Transient Protection	-	NRR/DST/GIB	USI			B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI			
A-28	Increase in Spent Fuel Pool Storage Capacity	Colmar	NRR/DE/SGEB	NOTE 3			
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	Colmar	NRR/DSI/ASB	MEDIUM			
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	HIGH			
A-31	RHR Shutdown Requirements	-	NRR/DST/GIB	USI			
A-32	Missile Effects	Pittman	NRR/DE/MTEB	A-37/A-38/B-68			NA
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	E(NOTE 3)			NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3			NA
A-35	Adequacy of Offsite Power Systems	Emrit	NRR/DSI/PSB	NOTE 3			
A-36	Control of Heavy Loads Near Spent Fuel	-	NRR/DSI/GIB	USI			C-10
A-37	Turbine Missiles	Pittman	NRR/DE/MTEB	DROP			NA
A-38	Tornado Missiles	Sege	NRR/DSI/ASB	LOW			NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	-	NRR/DST/GIB	USI			
A-40	Seismic Design Criteria - Short Term Program	-	NRR/DST/GIB	USI			
A-41	Long Term Seismic Program	Colmar	NRR/DE/MEB	MEDIUM			
A-42	Pipe Cracks in Boiling Water Reactors	-	NRR/DST/GIB	USI			B-05
A-43	Containment Emergency Sump Performance	-	NRR/DST/GIB	USI			
A-44	Station Blackout	-	NRR/DST/GIB	USI			
A-45	Shutdown Decay Heat Removal Requirements	-	NRR/DST/GIB	USI			
A-46	Seismic Qualification of Equipment in Operating Plants	-	NRR/DST/GIB	USI			
A-47	Safety Implications of Control Systems	-	NRR/DST/GIB	USI			
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	-	NRR/DST/GIB	USI			
A-49	Pressurized Thermal Shock	-	NRR/DST/GIB	USI			
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	E (NOTE 3)			NA
B-2	Forecasting Electricity Demand	-	NRR	E (NOTE 3)			NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (DROP)			NA
B-4	ECCS Reliability	Emrit	NRR/DSI/RSB	II.E.3.2			NA
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	Thatcher	NRR/DE/SGEB	MEDIUM			
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DE/MEB	HIGH			
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (DROP)			NA
B-8	Locking Out of ECCS Power Operated Valves	Riggs	NRR/DSI/RSB	DROP			NA
B-9	Electrical Cable Penetrations of Containment	Emrit	NRR/DSI/PSB	NOTE 3			
B-10	Behavior of BWR Mark III Containments	V'Molen	NRR/DSI/CSB	HIGH			
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3			
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI			
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	Emrit	NRR/DST/GIB	A-48			NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (DROP)			NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Emrit	NRR/DE/MEB	A-18			NA
B-17	Criteria for Safety Related Operator Actions	Milstead	NRR/DHFS/LQB	MEDIUM			
B-18	Vortex Suppression Requirements for Containment Sumps	Emrit	NRR/DST/GIB	A-43			NA
B-19	Thermal-Hydraulic Stability	Colmar	NRR/DSI/CPB	NOTE 4			
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI			
B-21	Core Physics	-	NRR/DSI/CPB	LI (DROP)			NA
B-22	LWR Fuel	V'Molen	NRR/DSI/CPB	NOTE 4			
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (DROP)			NA
B-24	Seismic Qualification of Electrical and Mechanical Components	Emrit	NRR	A-46			NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI			
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTEB	MEDIUM			
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI			
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	E (NOTE 3)			NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	NOTE 4			
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI			
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	NOTE 4			
B-32	Ice Effects on Safety Related Water Supplies	Pittman	NRR/DE/EHEB	NOTE 4			
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)			NA
B-34	Occupational Radiation Exposure Reduction	Emrit	NRR/DSI/RAB	III.D.3.1			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			
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			
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	E			
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	E (DROP)			NA
B-39	Transmission Lines	-	NRR/DE/EHEB	E (DROP)			NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	E (DROP)			NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	E (DROP)			NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	E (NOTE 3)			NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	E			
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	E (NOTE 3)			NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	E (B-2)			NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	E (DROP)			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
B-47	Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	Colmar	NRR/DE/MTEB	DROP			NA
B-48	BWR CRD Mechanical Failure (Collet Housing)	Emrit	NRR/DE/MTEB	NOTE 3			
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI			
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	NOTE 4			
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	Emrit	NRR/DE/MEB	A-40			NA
B-52	Fuel Assembly Seismic and LOCA Responses	Emrit	NRR/DST/GIB	A-2			NA
B-53	Load Break Switch	Sege	NRR/DSI/PSB	RI (NOTE 3)			
B-54	Ice Condenser Containments	Milstead	NRR/DSI/CSB	MEDIUM			
B-55	Improved Reliability of Target Rock Safety-Relief Valves	V'Molen	NRR/DE/MEB	MEDIUM			
B-56	Diesel Reliability	Milstead	NRR/DSI/PSB	HIGH			
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44			
B-58	Passive Mechanical Failures	Colmar	NRR/DE/EQB	MEDIUM			
B-59	N-1 Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	NOTE 4			
B-60	Loose Parts Monitoring System	Emrit	NRR/DSI/CPB	NOTE 2			
B-61	Allowable ECCS Equipment Outage Periods	Pittman	NRR/DST/RRAB	MEDIUM			
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (DROP)			NA
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	Emrit	NRR/DE/MEB	NOTE 3			
B-64	Decommissioning of Reactors	Colmar	NRR/DE/CHEB	NOTE 2			
B-65	Iodine Spiking	Milstead	NRR/DSI/AEB	DROP	1	6/30/84	NA
B-66	Control Room Infiltration Measurements	Matthews	NRR/DSI/AEB	NOTE 3			
B-67	Effluent and Process Monitoring Instrumentation	Colmar	NRR/DSI/METB	III.D.2.1			NA
B-68	Pump Overspeed During LOCA	Riani	NRR/DSI/ASB	DROP			NA
B-69	ECCS Leakage Ex-Containment	Riani	NRR/DSI/METB	III.D.1.1			NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	Emrit	NRR/DSI/PSB	NOTE 3			
B-71	Incident Response	Riani	NRR	III.A.3.1			NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI			
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	Thatcher	NRR/DE/MEB	C-12			NA
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Milstead	NRR/DE/EQB	NOTE 3			
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	Emrit	NRR/DSI/CSB	NOTE 3			
C-3	Insulation Usage Within Containment	Emrit	NRR/DST/GIB	A-43			NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSI/RSB	NOTE 4			
C-5	Decay Heat Update	Riggs	NRR/DSI/CPB	NOTE 4			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
C-6	LOCA Heat Sources	Riggs	NRR/DSI/CPB	NOTE 4			
C-7	PWR System Piping	Emrit	NRR/DE/MTEB	NOTE 3			NA
C-8	Main Steam Line Leakage Control Systems	Milstead	NRR/DSI/ASB	HIGH			
C-9	RHR Heat Exchanger Tube Failures	Thatcher	NRR/DSI/RSB	DROP			NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3			NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Matthews	NRR/DE/MEB	MEDIUM			
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3			NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17			NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	NOTE 4			
C-15	NUREG Report for Liquids Tank Failure Analysis	-	NRR/DE/EHEB	LI (DROP)			NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	E (DROP)			NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3			NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	LOW			NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	NRR/DSI/RSB	NOTE 4			
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3			NA

NEW GENERIC ISSUES

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP			NA
2.	Failure of Protective Devices on Essential Equipment	Colmar	NRR/DSI/ICSB	NOTE 4			
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSI/ICSB	NOTE 2			
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3			NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1			NA
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	V'Molen	NRR/DSI/CPB	NOTE 3			
7.	Failures Due to Flow-Induced Vibrations	V'Molen	NRR/DSI/RSB	DROP			NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1			NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3			NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP			NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37			NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	MEDIUM			
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP			NA
14.	PWR Pipe Cracks	Matthews	NRR/DE/MTEB	NOTE 2			
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	LOW			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
16.	BWR Main Steam Isolation Valve Leakage Control Systems	Milstead	NRR/DSI/ASB	C-8			NA
17.	Loss of Offsite Power Subsequent to LOCA	Colmar	NRR/DSI/PSB, ICSB	DROP			NA
18.	Steam Line Break with Consequential Small LOCA	Riggs	NRR/DSI/RSB	I.C.1			NA
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	Sege	NRR/DST/GIB	A-47			NA
20.	Effects of Electromagnetic Pulse on Nuclear Plant Systems	Thatcher	NRR/DSI/ICSB	NOTE 3	1	6/30/84	NA
21.	Vibration Qualification of Equipment	Thatcher	NRR/DE/EQB	NOTE 4			
22.	Inadvertent Boron Dilution Events	V'Molen	NRR/DSI/RSB	DROP			NA
23.	Reactor Coolant Pump Seal Failures	Riggs	NRR/DSI/ASB	HIGH			
24.	Automatic Emergency Core Cooling System Switch to Recirculation	V'Molen	NRR/DSI/RSB	NOTE 4			
25.	Automatic Air Header Dump on BWR Scram System	Milstead	NRR/DSI/RSB	NOTE 3			
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	Emrit	NRR/DSI/ASB	17			NA
27.	Manual vs. Automated Actions	Pittman	NRR/DSI/RSB	B-17			NA
28.	Pressurized Thermal Shock	Emrit	NRR/DST/GIB	A-49			NA
29.	Bolting Degradation or Failure in Nuclear Power Plants	V'Molen	NRR/DE/MTEB	HIGH			
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR	NOTE 4			
31.	Natural Circulation Cooldown	Riggs	NRR/DSI/RSB	I.C.1			NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	Emrit	NRR/DSI/ASB	51			NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	Pittman	NRR/DSI/ICSB	A-47			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			NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, RSB, AEB	NOTE 1	1	6/30/84	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Matthews	NRR	NOTE 4			
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	Matthews	NRR	NOTE 4			
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25			NA
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Colmar	NRR/DSI/ASB	NOTE 3	1	6/30/84	B-65
41.	BWR Scram Discharge Volume Systems	V'Molen	NRR/DSI/RSB	NOTE 3			B-58
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	18			NA
43.	Contamination of Instrument Air Lines	Milstead	NRR/DSI/ASB	DROP			NA
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43			NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3	1	6/30/84	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76			NA
47.	Loss of Off-Site Power	Thatcher	NRR/DSI/RSB, ASB	NOTE 3			
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	Sege	NRR/DL/ORB4	NOTE 2			
49.	Interlocks and LCOs for Redundant Class 1E Tie Breakers	Sege	NRR	NOTE 4			
50.	Reactor Vessel Level Instrumentation in BWRs	Thatcher	NRR/DSI/RSB, ICSB	NOTE 1			
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Emrit	NRR/DSI/ASB	MEDIUM			
52.	SSW Flow Blockage by Blue Mussels	Emrit	NRR/DSI/ASB	51			NA
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	V'Molen	NRR	NOTE 4			
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	Matthews	NRR	NOTE 4			
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Thatcher	NRR	NOTE 4			
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	Colmar	NRR/DHFS/HFEB	A-45/I.D.1			NA
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	Matthews	NRR	NOTE 4			
58.	Inadvertent Containment Flooding	Sege	NRR/DSI/ASB, CSB	DROP			
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	Thatcher	NRR	NOTE 4			
60.	Lamellar Tearing of Reactor Systems Structural Supports	Colmar	NRR/DST/GIB	A-12			NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	Milstead	NRR/DSI/CSB	MEDIUM			
62.	Reactor Systems Bolting Applications	V'Molen	NRR	NOTE 4			
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	Thatcher	NRR	NOTE 4			
64.	Identification of Protection System Instrument Sensing Lines	Thatcher	NRR/DSI/ICSB	NOTE 3			
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	V'Molen	NRR/DSI/ASB	HIGH			
66.	Steam Generator Requirements	Riggs	NRR/DL/ORAB	NOTE 2			
67.	Steam Generator Staff Actions	Riggs	NRR	NOTE 4			
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/DL/ORB4	NOTE 1			
70.	PORV and Block Valve Reliability	Riggs	NRR/DSI/RSB	MEDIUM	1	6/30/84	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Matthews	NRR	NOTE 4			
72.	Control Rod Drive Guide Tube Support Pin Failures	V'Molen	NRR	NOTE 4			
73.	Detached Thermal Sleeves	Sege	NRR	NOTE 4			
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR	NOTE 4			
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Thatcher	NRR/DSI	NOTE 1			
76.	Instrumentation and Control Power Interactions	Colmar	NRR	NOTE 4			
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	NRR/DSI/ASB	HIGH			
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Riggs	NRR	NOTE 4			
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	NRR/DE/MEB	MEDIUM			
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			NA
81.	Potential Safety Problems Associated With Locked Doors and Barriers in Nuclear Power Plants	Colmar	NRR	NOTE 4			
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen		MEDIUM			
83.	Control Room Habitability	Matthews	NRR	NOTE 4			
84.	CE PORVs	Riggs	NRR	NOTE 4			
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR	NOTE 4			
86.	NRC Pipe Cracking Review Group Study	Matthews	NRR	NOTE 4			
87.	Failure of HPCI Steam Line Without Isolation	Pittman	NRR	NOTE 4			
88.	Earthquakes and Emergency Planning	Riggs	NRR	NOTE 4			
89.	Stiff Pipe Clamps	Riggs	NRR	NOTE 4			
90.	Technical Specifications for Anticipatory Trips	V'Molen	NRR	NOTE 4			
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	NRR	NOTE 4			
92.	Fuel Crumbling During LOCA	V'Molen	NRR	NOTE 4			
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	NRR	NOTE 4			

HUMAN FACTORS PROGRAM PLAN

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
HF.01.1.0	STAFFING AND QUALIFICATIONS						
HF.01.1.1	Establish Staffing Requirements	Pittman	NRR/DHFS	NOTE 4			
HF.01.1.2	Personnel Qualification Requirements	Pittman	NRR/DHFS	NOTE 4			
HF.01.1.3	Guidance on Limits and Conditions for Shift Work	Pittman	NRR/DHFS	NOTE 4			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Revision Date	MPA No.
HF.01.1.4	Maintenance Staffing and Qualification	Pittman	NRR/DHFS	NOTE 4			
HF.01.1.5	Fitness for Duty	Pittman	NRR/DHFS	NOTE 4			
HF.01.2.0	TRAINING						
HF.01.2.1	Develop Training Regulation and Guidance	Pittman	NRR/DHFS	NOTE 4			
HF.01.2.2	Training Assessment Procedures	Pittman	NRR/DHFS	NOTE 4			
HF.01.3.0	LICENSING EXAMINATIONS						
HF.01.3.1	Examination Content	Pittman	NRR/DHFS	NOTE 4			
HF.01.3.2	The Examination Process	Pittman	NRR/DHFS	NOTE 4			
HF.01.4.0	PROCEDURES						
HF.01.4.1	Procedures, Guidance, and Evaluation Criteria	Pittman	NRR/DHFS	NOTE 4			
HF.01.4.2	Initial Test Program Training Effectiveness	Pittman	NRR/DHFS	NOTE 4			
HF.01.5.0	MAN-MACHINE INTERFACE						
HF.01.5.1	Man-Machine Interface Guidance for Existing Designs	Pittman	NRR/DHFS	NOTE 4			
HF.01.5.2	Guidance for Designs Based on Advanced Technologies	Pittman	NRR/DHFS	NOTE 4			
HF.01.6.0	MANAGEMENT AND ORGANIZATION						
HF.01.6.1	Management Organization and Guidance and Regulatory Position	Pittman	NRR/DHFS	NOTE 4			
HF.01.6.2	Assessment Procedures	Pittman	NRR/DHFS	NOTE 4			

TABLE III

SUMMARY OF THE PRIORITIZATION OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS PROGRAM PLAN (HFPP) ITEMS

ACTION ITEM/ISSUE GROUP	I	COVERED IN OTHER ISSUES	RESOLVED STAGES NOTE 1	NOTE 2	NOTE 3	USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
1. TMI ACTION PLAN ITEMS (255)													
(a) Safety													
(i) Generic Safety	63	14	6	2	50	0	17	20	13	7	2	-	194
(b) Non-Safety													
(i) Licensing	-	0	0	4	51	-	-	-	-	0	0	6	61
2. TASK ACTION PLAN ITEMS (142)													
(a) Safety													
(i) Generic Safety	-	17	0	2	20	27	5	10	3	9	1	-	94
(ii) Regulatory Impact	-	0	0	0	1	-	-	-	-	0	7	1	9
(b) Non-Safety													
(i) Licensing	-	0	0	0	1	-	-	-	-	7	5	11	24
(ii) Environmental	-	1	0	0	6	-	-	-	-	6	0	2	15
3. NEW GENERIC ISSUES (93)													
(a) Safety													
(i) Generic Safety	-	20	5	4	8	0	5	6	3	9	33	0	93
4. HFPP ITEMS (16)													
(a) Safety													
(i) Generic Safety	-	-	-	-	-	-	-	-	-	-	16	-	16
TOTAL:	63	52	11	12	137	27	27	36	19	38	64	20	506

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

TASK I.A: OPERATING PERSONNEL

TASK I.A.1: OPERATING PERSONNEL AND STAFFING

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

ITEM I.A.1.4: LONG-TERM UPGRADING

DESCRIPTION

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

SECY-81-440[250] was prepared by the NRC staff in July, 1981 and resulted in a Commission policy statement on working hour limitations which was issued in the Federal Register on February 17, 1982. Working hour limitations have been incorporated into Regulatory Guide 1.33[225] [see Item I.B.1.1(7)]. The specific issues are the following:

(1) Number of licensed operators based on number of reactors, control room configuration and operating model
(2) Should current rulemaking be expanded to include nonlicensed operators?
(3) Should current rulemaking be expanded to include "position titles" in addition to the type of NRC license?
(4) Should STAs or SEs be required on shift?
(5) Should shift supervisors (SSs) be licensed?

A proposed rule was published on August 30, 1982. After the comment period expired, the final rule was submitted to the Commissioners in SECY-83-52A[595] on March 14, 1983. In response to the TMI Action Plan, licensing has required, through technical specification, the great majority of the substantive features of the expected changes to regulation. Therefore, adoption of the rule will have the effect of codifying existing requirements and is expected to have a minimal impact on licensees.

CONCLUSION

The final rule was approved[596] by the Commission on April 28, 1983. Thus, this 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.

225. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, February 1978.

250. SECY-81-440, "Nuclear Power Plant Staff Working Hours," July 22, 1981.

595. SECY-83-52A, "Final Rulemaking Concerning Licensed Operator Staffing at Nuclear Power Units and Draft Policy Statement on Shift Crew Qualifications," March 14, 1983.

596. Memorandum for W. Dircks from S. Chilk, "Staff Requirements - Affirmation/ Discussion and Vote, 3:35 p.m., Thursday, April 21, 1983, Commissioners' Conference Room (Open to Public Attendance)," April 28, 1983.

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 EXAMINATIONS

DESCRIPTION

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

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.[593] Approval of this rule would effectively close out this item.

ITEM I.A.3.2: OPERATOR LICENSING PROGRAM CHANGES

DESCRIPTION

This TMI Action Plan item[48] 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."[88]

(2) A study of the staffing of the operator licensing program and the qualifications and training of examiners was completed in November, 1980 and documented in NUREG/CR-1750.[89]

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

CONCLUSION

This item has been RESOLVED.

ITEM I.A.3.3: REQUIREMENTS FOR OPERATOR FITNESS

DESCRIPTION

Historical Background

This safety issue as described in NUREG-0660[48] 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[289] and NUREG/CR-2076.[290] 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[64] 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 ongoing activity for individuals who have been granted an unescorted access authorization. The background check would examine an individual's past for unstable activities, a criminal record, credit problems, and previous employment problems. It 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,[64] 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 x 10^{-7}/RY for PWRs and 1.8 x 10^{-7}/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) = $9.94M.

(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) = $37.2M.

(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 ($0.3M/RY)(3,798 RY) or $1,140M.

The total industry cost for the SIR is $[1,140 + 9.94 + 37.2]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/RY 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:

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

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

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

Based on the estimated risk reduction of 16,000 man-rem and the value/impact score of 13.4 man-rem/$M, the priority ranking for this issue would be medium. However, in view of the advanced state of completion of the issue, it is concluded that the safety priority ranking should be HIGH.

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

DESCRIPTION

Historical Background

This TMI Action Plan item[48] 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 DETERMINATION

Assumptions

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.[64]

Frequency/Consequence 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 1.4×10^{-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 6.3×10^{-7}/RY.

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:

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

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

Therefore, the total risk reduction for this issue is 1.01×10^4 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:

$$S = \frac{10{,}100 \text{ man-rem}}{\$(240 + 250)\text{M}}$$

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

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[181] that licensing of managers should not be required. The other portion of the issue (i.e., licensing of other personnel--engineers, maintenance personnel, etc.) is still under study and is to be concluded in FY 1983.

CONCLUSION

We believe that the study should be completed with a MEDIUM priority. Although the value/impact score is low, the potential for risk reduction suggests pursuit of the issue to resolution.

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

DESCRIPTION

As a part of the overall evaluation of the TMI incident, it was determined[48] that a statement of understanding was needed to address the mutual intent of NRC and INPO concerning the extent to which NRC should review or rely upon training, certification, and other 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[148] 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[594] 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.

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.

TASK I.E: ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE

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

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

DESCRIPTION

Historical Background

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

Safety Significance

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

Solution

The Commission approved the establishment of AEOD in July, 1979. An interim office was established in October, 1979. AEOD is currently staffed and functioning in accordance with its purpose and scope described in Chapter 0143 of the NRC Manual.

CONCLUSION

This Licensing Issue has been resolved.

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

DESCRIPTION

Historical Background

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

Safety Significance

Systematic evaluation of operational data can identify potential significant safety problems or their precursors. Dissemination to NRC offices and industry of such evaluations, along with their resolution, can avoid occurrence of these problems at other facilities of similar design that are conducting similar operations. This item is related to improving the NRC capability to make independent assessments of safety and, therefore, is considered a licensing issue.

Solution

Each of the NRC Offices has established responsibility and procedures for evaluating operational data as follows:

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

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

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

(4) NMSS has issued a procedure[305] for achieving a more formal review and evaluation of inspection and operational data and event reports to identify and correct generic problems. The procedure includes criteria for identifying operational events that warrant detailed review and evaluation.

Evaluation reports that identify safety significant operational events are distributed within NRC.

CONCLUSION

This Licensing Issue has been resolved.

ITEM I.E.3: OPERATIONAL SAFETY DATA ANALYSIS

DESCRIPTION

Historical Background

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

Safety Significance

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

Solution

Thus far, the program has resulted in:

(1) Publication of data summaries of LERs on pumps, control rods and drive mechanisms, diesel generators, valves, primary containment penetrations, and instrumentation and control components (See References 344, 345, 346, 347, 348, 349, 350, and 351).

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

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

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

CONCLUSION

This item is an on-going effort to collect and analyze data. While no quantified safety benefits can be directly assigned to it, the benefits occur as the results of the equipment failure rate data and reliability analysis are used in assessing other specific related safety issues, including Items II.C.1, II.C.2 and II.C.4. This Licensing Issue has been resolved. The on-going activities will be conducted as described in Section 16.1 of the NRC Long Range Research Program.[133]

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

DESCRIPTION

Historical Background

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

Safety Significance

Licensee evaluations of plant operating experience, coordinated with industry and NRC evaluations using common data bases, will assure that licensee, industry, and NRC corrective action recommendations are properly coordinated and applied. Effective feedback of prioritized and analyzed event descriptions to plant operating personnel and incorporation into training programs can avoid occurrence of these problems at other plants of similar design. This item is related to improving the capability to assess safety and, therefore, is considered a licensing issue.

Solution

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

CONCLUSION

This Licensing Issue has been resolved and is being implemented.

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

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

ITEM I.E.6: REPORTING REQUIREMENTS

Items I.E.5 and I.E.6 have been combined and evaluated together.

DESCRIPTION

Historical Background

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

NPRDS is a reliability-oriented data collection and reporting system for selected components and systems related to the safety of nuclear power plants. Periodic reports containing failure statistics are issued. Licensee participation is voluntary.

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

Safety Significance

Improvements in reporting significant events of operating plants can identify potential significant safety problems or their precursors and can avoid occurrence of these problems at other plants of similar design. Improved reporting of system/component reliability data will increase the validity of operating experience assessments and PRA programs. This item is related to improving the capability to assess safety and, therefore, is considered a licensing issue.

Solution

As of January, 1982, INPO had assumed responsibility for the NPRDS. NRC is represented on the NPRDS Users Group and participates in various NPRDS work groups. AEOD submits semi-annual reports to the Commission on the effectiveness of the INPO management of NPRDS. A proposed rule on LERs was published in the Federal Register (47 FR 19543) on May 6, 1982. The final rule[597] was published in July, 1983.

CONCLUSION

Based on the actions described above, Items I.E.5 and I.E.6 are Licensing Issues that have been resolved.

ITEM I.E.7: FOREIGN SOURCES

DESCRIPTION

Historical Background

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

AEOD also sponsors a program by which the Nuclear Operations Analysis Center at ORNL screens and stores for ready access reports of foreign reactor incidents and provides monthly summaries of these events that are potentially significant and relevant to U.S. LWRs.

Foreign reactor incident and operating experience reports are now being routinely received and disseminated to NRC technical staffs. IP also routinely sends these foreign reactor incident reports to INPO for use by industry in evaluating plant operating experience under Item I.E.4.

Safety Significance

Foreign reactor incident and operating experience reports are being assessed by AEOD and affected NRC Offices as described in Items I.E.1 and I.E.2, respectively, to identify potential significant safety problems or their procedures which may be applicable to U.S. plants. Dissemination within NRC and to industry of such assessments, along with their resolutions, can avoid occurrence of these problems at other facilities of similar design. This item is related to improving the capability to assess safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM I.E.8: HUMAN ERROR RATE ANALYSIS

DESCRIPTION

Historical Background

The March 28, 1979 incident at TMI-2 increased the concern of the effect of human reliability on reactor safety. The lack of human reliability data applicable to nuclear power plants compared to hardware reliability data highlights this concern in nuclear safety assessments and regulation.

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

Safety Significance

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

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

Solution

The Human Factors Branch of RES is implementing an expanded Human Reliability research program to accomplish the purpose of this item and provide the human error information for its end use as described above. Major reports issued thus far include: (1) a human reliability data bank,[338] (2) a draft handbook for human reliability analysis,[339] (3) procedures for estimating human error probabilities,[341,342] and (4) a workbook for conducting human reliability analysis.[343]

Future work includes finalizing the handbooks, workbooks, and reliability models and maintaining the data bank. This work is described in Section 7.1 of the NRC Long Range Research Plan.[133]

CONCLUSION

This Licensing Issue has been resolved with regard to establishing and implementing the human reliability research program.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

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

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

124. NUREG-0193, "FRANTIC - A Computer Code for Time Dependent Unavailability Analysis," U.S. Nuclear Regulatory Commission, October 1977.

133. NUREG-0784, "Long Range Research Plan," U.S. Nuclear Regulatory Commission, August 1982.

138. NUREG/CR-1924, "FRANTIC II - A Computer Code for Time Dependent Unavailability Analysis," U.S. Nuclear Regulatory Commission, April 1981.

238. NRC-INPO Memorandum of Agreement, June 1, 1981.

260. SECY-81-494 "Integrated Operational Experience Reporting System," August 18, 1981.

305. Memorandum to Distribution from J. Davis, "NMSS Procedure for Review of Routine Inspection Operational Data and Licensee Event Reports," March 9, 1982.

338. NUREG/CR-2744, "Human Reliability Data Bank for Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, November 1982.

339. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, February 1983.

341. NUREG/CR-2255, "Expert Estimation of Human Error Probabilities in Nuclear Power Plant Operations: A Review of Probability Assessment and Scaling," U.S. Nuclear Regulatory Commission, May 1982.

342. NUREG/CR-2743, "Procedures for Using Expert Judgment to Estimate Human Error Probabilities in Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, February 1983.

343. NUREG/CR-2254, "Workbook for Conducting Human Reliability Analysis," U.S. Nuclear Regulatory Commission, February 1983.

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TASK III.D.2: PUBLIC RADIATION PROTECTION IMPROVEMENT

The objective of this task is to improve public radiation protection in the event of a nuclear power plant accident by improving: (1) radioactive effluent monitoring; (2) the dose analysis for accidental releases of radioiodine, tritium, and carbon-14; (3) the control of radioactivity released into the liquid pathway; (4) the measurement of offsite radiation doses; and (5) the ability to rapidly determine offsite doses from radioactivity release by meteorological and hydrological measurements so that population-protection decisions can be made appropriately.

ITEM III.D.2.1: RADIOLOGICAL MONITORING OF EFFLUENTS

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

DESCRIPTION

Historical Background

This TMI Action Plan[48] item requires development and implementation of acceptance criteria for monitors used to evaluate effluent releases under accident and postaccident conditions. Criteria would be developed for pathways to be monitored (stack, plant vent, steam dump vents) as well as for monitoring instrumentation. To meet the new criteria, licensees would have to develop, procure, and install monitoring systems which are currently beyond the state-of-the-art. This is seen to encompass the requirements in NUREG-0578,[57] Recommendation 2.1.8-b, and Appendix 2 to NUREG-0654.[224]

The envisioned monitoring system would provide automatic on-line analysis of airborne effluents including isotopic analyses of particulate, radioiodine and gas samples. To prevent saturation of detectors, an automatic sample cartridge changeout feature would be included. The system would include microprocessor control and real-time readouts, and would be located in a low postaccident background area. The sampling system would be designed to provide a representative sample under anticipated accident release conditions.

A PWR steam-dump sampling and monitoring system would be provided for PWR safety relief and vent valves. Such a system might consist of a noble gas monitor and a radioiodine sampling and monitoring system. The features of such a system would be similar to the above described airborne effluent monitor with two notable differences: (1) the system would be required to function in a very high humidity (steam-air mixture) environment, and (2) operation would only be required during actual steam venting. Because such venting is usually of a short-term or intermittent duration, the monitoring system activation could be keyed to the opening of the vents.

Liquid effluents are not envisioned as posing a major release pathway because licensees typically have installed, or are installing, adequate storage capacity to prevent discharges. Consequently, present liquid effluent monitoring systems are considered adequate.

Safety Significance

This issue has no impact on core-melt and very little impact on public risk.

Possible Solution

For the purpose of this analysis, it will be assumed that improved radiological monitoring of airborne effluent would result in a reduction of public risk.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to improved radiological monitoring of airborne effluents is not certain, but it was estimated by PNL[64] to range from zero to 1%, based on the following logic.

Present radiological monitoring requirements, as contained in NUREG-0737,[98] require real-time noble gas monitoring with sampling and laboratory analysis capabilities for radioiodines and particulates. Design basis conditions defined in NUREG-0737[98] (100 μCi/cc radioiodines and particulates, 30-minute sample time) indicate that sample collection devices would pose special handling and analysis problems due to very high radioactivity buildup. Consequently, licensees have typically provided alternate sample collection and analysis procedures. Execution of those procedures is estimated to require between 2 to 3 hours. During this time, radioiodine and particulate releases would be estimated based on computer-modeled interpretation of noble gas monitor readings, or on previous postaccident containment atmosphere analysis results, if such results were available. Public protective action recommendations would be made based on modeled estimates rather than actual effluent data. It is assumed that these recommendations would err on the conservative side (e.g., evacuate when not really required) due to the conservatism built into the modeled source terms for radioiodine and particulate releases.

Requiring licensees to have more sophisticated airborne effluent monitors would reduce the time required for obtaining actual radioiodine and particulate release data to 15 minutes, and essentially eliminate reliance on conservative theoretical release models extrapolated from noble gas monitor readings. As projected by this safety issue resolution, real-time isotopic monitoring would save nearly two hours in arriving at realistic protective action recommendations based on actual releases.

Under these circumstances, the public risk reduction would be directly attributed to the decrease in public radiation exposure which results from a more rapid assessment of the radioactive releases (about a 2-hour savings in analysis time). There may also be a public risk reduction due to non-evacuation. This could result from better knowledge of the isotopic releases eliminating the need for evacuation (presumed to exist if release knowledge is based only on noble gas monitor data). Non-evacuation results in less evacuation-related risks (e.g., traffic accidents), the avoidance of which may outweigh the radiation exposure received. However, for this analysis, it is assumed that the public risk reduction results primarily from the first effect (decrease in exposure due to more rapid assessment).

While protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario is such that actions would be recommended based on anticipated releases, prior to the actual release themselves. Under that assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification. This safety issue resolution would not impact core-melt accident frequency.

There are 134 plants affected by this issue: 71 operating (47 PWRs and 24 BWRs) and 63 planned (43 PWRs and 20 BWRs). It will be assumed that the average remaining plant life is 27.4 years for 44 BWRs and 28.8 years for 90 PWRs. The dose factors for PWR Release Categories 1 through 7 and BWR Release Categories 1 through 4 are assumed to be affected by the possible solution. From NUREG/CR-2800,[64] a 1% decrease in the dose factors results in an estimated total public risk reduction of 8,500 man-rem for all plants. Assuming a decrease in the dose factors of 0.5% for this issue, the estimated public risk reduction is 4,250 man-rem.

Cost Estimate

Industry Cost: The industry cost for equipment development, installation, support facilities, and construction is estimated at $600,000 per plant. Development of procedures, software, and calibration for the equipment is estimated to require 16 man-weeks of effort, with an additional 4 man-weeks of effort for the initial training of all licensee operators and health physics personnel. This is estimated to add $45,400 per plant to the implementation cost. Based on estimated costs of $645,000/plant for labor and equipment, the total industry cost for implementing the possible solution is (134 plants)($645,000/plant) or $86.5M.

The recurring industry operation and maintenance costs are estimated at 2 man-weeks/plant-yr for retraining, 1 man-week/plant-yr for calibration, and a reduction of 1 man-week/plant-yr (reduced laboratory analyses due to a fully automated system) for a net increase of 2 man-weeks/plant-yr at a cost of $4,540/plant-yr. As a result, industry costs for labor and material associated with operation and maintenance of the possible solution are estimated to be $17.2M.

The total industry cost associated with this issue is $(86.5 + 17.2)M or $103.7M.

NRC Cost: The NRC cost is assumed to be limited to implementation costs for development and plant installation. Since it is assumed that the new radiological monitoring systems would require no periodic inspection effort beyond that required for current systems, no additional NRC operation cost is envisioned. The NRC development costs include 1.5 man-year and $200,000 for research, criteria development, and engineering development for a total cost of $350,000. NRC administrative and technical effort associated with the review and approval of licensee submittals is estimated at 0.3 man-wk/plant for a total cost of $91,000 for all plants. Therefore, the total NRC cost associated with this issue is $441,000.

Value/Impact Assessment

Based on a total risk reduction of 4,250 man-rem, the value/impact score is given by:

$$S = \frac{4{,}250 \text{ man-rem}}{\$(103.7 + 0.441)M}$$

$$\cong 41 \text{ man-rem/\$M}$$

Other Considerations

It is anticipated that improvement of radiological monitoring of airborne effluents would have no significant impact on occupational risk. The dose required to install equipment would probably not exceed 0.5 man-rem, which is negligible compared to the typical 600 man-rem/yr required to operate a plant. Minor man-rem savings might occur under accident conditions due to better direction of field survey teams; however, such savings would be negligible compared to the 19,900 man-rem total associated with response and cleanup following an accident.

Based on an estimated occupational dose of 0.5 man-rem/plant for implementation of the possible solution in 71 operating plants, the total risk increase is 36 man-rem for all plants. Inclusion of this factor into the above calculation would reduce the value/impact score.

There is no accident avoidance cost for the resolution of this issue because improved radiological effluent monitoring systems would have no impact on accident frequency or cleanup and refurbishing costs.

CONCLUSION

This issue has a LOW priority ranking.

ITEM III.D.2.1(1): EVALUATE THE FEASIBILITY AND PERFORM A VALUE-IMPACT ANALYSIS OF MODIFYING EFFLUENT-MONITORING DESIGN CRITERIA

This item was evaluated in Item III.D.2.1 above and was determined to be LOW priority.

ITEM III.D.2.1(2): STUDY THE FEASIBILITY OF REQUIRING THE DEVELOPMENT OF EFFECTIVE MEANS FOR MONITORING AND SAMPLING NOBLE GASES AND RADIOIODINE RELEASED TO THE ATMOSPHERE

This item was evaluated in Item III.D.2.1 above and was determined to be LOW priority.

ITEM III.D.2.1(3): REVISE REGULATORY GUIDES

This item was evaluated in Item III.D.2.1 above and was determined to be LOW priority.

ITEM III.D.2.2: RADIOIODINE, CARBON-14, AND TRITIUM PATHWAY DOSE ANALYSIS

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

DESCRIPTION

Historical Background

This TMI Action Plan[48] Item addressed the issue of further research for improving the understanding of radioiodine partitioning in nuclear power reactors and of the environmental behavior of radioiodine, carbon-14, and tritium following an accident and during normal operation.

Iodine isotopes are considered to be major contributors to the occupational and public dose during a LOCA, along with noble gases and fission products. Recent study in these areas is documented in NUREG-0772.[212] Major conclusions from NUREG-0772[212] state that: (1) uncertainties in predicting atmospheric release source terms are very large (at least a factor of 10), (2) source terms for certain accident sequences may have been overestimated in past studies, e.g., WASH-1400[16], and (3) cesium iodide should be the predominant chemical form of iodine under severe accident conditions.

Safety Significance

The above conclusions indicate that methodology and assumptions currently used for evaluating radioiodine release may result in unrealistic estimates (e.g., Regulatory Guides 1.3[213] and 1.4[214] Also indicated is that more research in aerosol behavior and fission product chemistry is needed in order to improve and support calculational methodology concerned with radioiodine partitioning, fission product behavior, etc.

Possible Solution

It could be assumed that further study will improve understanding of this issue and result in more realistic assumptions and methods for evaluating source terms, releases, and environmental behavior of radioiodine, carbon-14, and tritium following an accident. This research will not affect accident frequencies at nuclear power plants. However, the results of these studies are assumed to be used to revise the Standard Review Plan[11] and Regulatory Guides.

It is then assumed that these Regulatory Guide revisions could result in reducing the size of current emergency planning zones (EPZs) from a 10-mile radius to a 2-mile radius. This assumption is based upon a reduction of source terms in a core-melt accident by a factor of 10. This results in reducing dose concentration at a particular distance from the nuclear reactor by a factor of 10, also. Assuming neutral weather conditions with a 30-meter-high plume, the offsite dose predicted at 2 miles from the accident scene, using the reduced source term assumption, would be the same as that currently predicted at 10 miles from the reactor.

CONCLUSION

Item III.D.2.2(1), related to the study of radioiodine, carbon 14, and tritium behavior at TMI-2, was completed in June 1981 and documented in NUREG-0771[455] and NUREG-0772.[212] Items III.D.2.2(2), (3), and (4) call for a series of studies and evaluations of various radionuclide pathways and models followed, if necessary, by revisions to several SRP sections and Regulatory Guides. As part of the staff's task to prepare and publish a manual (Offsite Dose Calculation Manual) to be used by the NRC and industry to estimate individual and population doses during normal and accident conditions, Items III.D.2.2(2), (3), and (4) were assessed. This Offsite Dose Calculation Manual was prepared under Item III.D.2.5 and fully describes each of the theoretical models used to predict radionuclide transport.[149] Thus, Items III.D.2.2(2), (3), and (4) are covered under Item III.D.2.5.

ITEM III.D.2.2(1): PERFORM STUDY OF RADIOIODINE, CARBON-14, AND TRITIUM BEHAVIOR

This item was evaluated in Item III.D.2.2 above and was determined to be RESOLVED.

ITEM III.D.2.2(2): EVALUATE DATA COLLECTED AT QUAD CITIES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2.(3): DETERMINE THE DISTRIBUTION OF THE CHEMICAL SPECIES OF RADIOIODINE IN AIR-WATER-STEAM MIXTURES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2.(4): REVISE SRP AND REGULATORY GUIDES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.3: LIQUID PATHWAY RADIOLOGICAL CONTROL

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

DESCRIPTION

This TMI Action Plan[48] item is concerned with improving public radiation protection in the event of a nuclear power plant accident by improving the control of radioactivity released into the liquid pathway. This control can be accomplished by the application of various interdictive measures at the source of the release and/or along the liquid pathway. Techniques have been developed and are being used to evaluate the liquid pathway effects of a class and accident for each reactor site. Those sites that might require interdictive measures related to liquid pathway releases will be determined. Interdictive measures will be assessed as to their effectiveness in improving public radiation protection.

CONCLUSION

A liquid pathway analysis for Zion was completed by DE in 1980.[391] In addition to this, a liquid pathway analysis was performed for Indian Point and both analyses were utilized in NUREG-0850.[390] A BTP on Liquid Pathway Analysis has been drafted by EHEB and requires further staff work for completion. With technical assistance from ANL (FIN B2454), reports on Slurry Wall Barriers and Groundwater Interdiction Methods have been drafted and are scheduled for publication in 1983.[384] Thus, a solution to this issue has been identified.

ITEM III.D.2.3(1): DEVELOP PROCEDURES TO DISCRIMINATE BETWEEN SITES/PLANTS

This item was evaluated in Item III.D.2.3 above and its solution has been identified.

ITEM III.D.2.3(2): DISCRIMINATE BETWEEN SITES AND PLANTS THAT REQUIRE CONSIDERATION OF LIQUID PATHWAY INTERDICTION TECHNIQUES

This item was evaluated in Item III.D.2.3 above and its solution has been identified.

ITEM III.D.2.3(3): ESTABLISH FEASIBLE METHOD OF PATHWAY INTERDICTION

This item was evaluated in Item III.D.2.3 above and its solution has been identified.

ITEM III.D.2.3(4): PREPARE A SUMMARY ASSESSMENT

This item was evaluated in Item III.D.2.3 above and its solution has been identified.

ITEM III.D.2.4: OFFSITE DOSE MEASUREMENTS

ITEM III.D.2.4(1): STUDY FEASIBILITY OF ENVIRONMENTAL MONITORS

DESCRIPTION

This TMI Action Plan[48] item called for the staff to study the feasibility of environmental monitors capable of measuring real-time rates of exposures to noble gases and radioiodines. Monitors or samplers capable of measuring respirable concentrations of radionuclides and particulates were also considered. This activity supports proposed revisions to Regulatory Guide 1.97[55] (Item II.F.3).

CONCLUSION

The establishment of Regulatory Guide 1.97[55] requirements for fixed monitors for detecting unidentified releases was postponed pending the outcome of a feasibility study. This study was completed in April 1982.[188] Using this study as a basis, the staff concluded that environmental monitors of this nature are not practical and that proposed requirements for these monitors be dropped from consideration.[189] All required action on this item has been completed[382] and the issue has been RESOLVED.

ITEM III.D.2.4(2): PLACE 50 TLDs AROUND EACH SITE

DESCRIPTION

This TMI Action Plan[48] item called for OIE to place 50 TLDs around each site in coordination with states and utilities. During normal operation, OIE quarterly reports from these dosimeters were to be provided to NRC, state, and federal organizations. In the event of an accident, the dosimeters could then be read at a frequency appropriate to the needs of the situation.

The specific objectives of this program were as follows:

(1) To establish preoperational, historical, baseline radiation dose levels, whenever possible, for each monitored facility
(2) To provide ongoing radiation dosimetry data during routine operations
(3) To provide postaccident radiation dosimetry to aid in assessment of population exposures and radiological impact
(4) To allow for independent verification of the adequacy of NRC licensees' environmental radiation monitoring programs
(5) To provide uniform treatment of dosimeters with respect to handling, shipping, calibrating, reading, and data processing for all monitored facilities in the United States
(6) To provide uniform, consistent environmental radiation monitoring data for use by the Congress, federal and state agencies, monitored facilities, and the public.

OIE completed installation of TLDs at all operating reactors in August 1980 in accordance with the TMI Action Plan schedule. A Direct Radiation Monitoring Network was established and a program for routine reporting was begun. The completion of these activities are described in an OIE memorandum.[236]

With the establishment of the NRC TLD Direct Radiation Monitoring Network, the installation of TLDs at all operating reactor sites, and the routine reporting of the TLD measurements, all work required by this item has been completed.[236,379] This item is related to improving the capability to make assessments of safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM III.D.2.5: OFFSITE DOSE CALCULATION MANUAL

DESCRIPTION

Historical Background

TMI Action Plan[48] item requires that NRR prepare a manual to be used by the NRC and plant personnel to estimate maximum individual doses and population doses during an accident.

Safety Significance

This issue does not affect core-melt frequencies or the amount of radioactivity released. Instead, it is intended to reduce the consequences of a major release by assuring that licensees have a rapid and sufficiently accurate method of estimating dose and that communication between licensees and the NRC is expedited by having a common standard calculation method for both.

Possible Solution

The proposed manual would include formulations with which to combine source term and meteorological measurements. This would determine offsite dose rates in a manner that would be standard among all parties making decisions on public protection and emergency response. Appendix 2 to NUREG-0654[224] establishes criteria for automated assessment of radiation doses in the event of an accident.

PRIORITY DETERMINATION

The assessment of this issue and its proposed resolution were performed by PNL.[64]

Frequency Estimate

The proposed solution to the issue does not affect accident frequencies. The frequencies for the various release categories given for Oconee and Grand Gulf were used unchanged in the value/impact calculation.

Consequence Estimate

The PNL experts judged that a 1% reduction in public dose (man-rem) might be expected as a result of having the offsite dose calculation manual available. We estimate the changes in consequences would be much less, 0.01% to 0.1%. Since all sequences would be affected and the risk from both PWRs and BWRs is about 210 to 250 man-rem/RY, the risk reduction is estimated to be 0.02 to 0.2 man-rem/RY.

Currently, there are 43 PWRs and 27 BWRs operating with cumulative experience of 350 RY and 260 RY, respectively. If we add to these the 36 PWRs and 21 BWRs under construction and assume a plant lifetime of 40 years, there are 2,810 PWR-years and 1,660 BWR-years in the future for a total of 4,470 RY. Therefore, the total risk reduction associated with this issue is (0.2)(4,470)man-rem or 894 man-rem.

Cost Estimate

Industry Cost: For the utilities, 4 man-weeks of training for implementation are assumed, since operators are now retrained periodically and this retraining could include dose calculation methods. This different method would not incur additional recurring costs. Thus, total industry cost is estimated to be $7,700/plant or $0.98M for 127 plants.

NRC Cost: The NRC has already completed work on development of a portable computerized system for dose calculations to be used by the NRC Regional Offices. This is part of the program for NUREG-0654.[224] This program has been developed to the point of field trials for the computerized system. Based on the current development costs, an additional $125,000 to develop this package into a manual form for use by utilities will be assumed. It is estimated that NRC site representatives could spend a minimal amount of time (~2 days) to evaluate initial utility performance with the package. This is estimated to be $600/plant. The total NRC cost is approximately $200,000.

Value/Impact Assessment

Based on a total public risk reduction of 894 man-rem, the value/impact score is given by:

$$S = \frac{894 \text{ man-rem}}{\$(0.98 + 0.2)M}$$

$$= 758 \text{ man-rem/\$M}$$

CONCLUSION

Based on the value/impact score, the issue was identified as medium priority. However, since the prioritization was completed, the Offsite Dose Calculation Manual was published as NUREG/CR-3332[599] in September, 1983. Thus, this item has been RESOLVED and no new requirements were issued.[598]

ITEM III.D.2.6: INDEPENDENT RADIOLOGICAL MEASUREMENTS

DESCRIPTION

This TMI Action Plan[48] item deals with independent radiological measurements, i.e., means of collecting data independently of the licensees' programs to do this. An OIE task force has developed a plan and requirements for upgrading the capability of Regional Offices to perform independent radiological measurements during routine inspections and emergency response operations. The objective of the upgrade is to achieve consistent capability among the regional offices, including standardization in major equipment items, such as mobile laboratory vans, gamma spectrum analysis equipment, radiation survey instrumentation, and air-sampling and monitoring devices.

Based on the recommendations of this task force, each Region was equipped with complete mobile laboratories.[235] In some cases, this represented upgrading certain equipment or purchasing new equipment. This action item required that revisions be made to the inspection program to include the upgrading of the independent radiological measurements. The program is included in the routine OIE program for review and revision of the inspection program. As new equipment needs are identified, the program will be revised and the equipment acquired.

With the upgrading of independent radiological measurements and the implementation of other recommendations made by the task force, all work required by this item has been completed.[235,379] This item is related to improving the NRC capability to make independent assessments of safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

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ITEM B-65: IODINE SPIKING

DESCRIPTION

Historical Background

This NUREG-0471[3] task is to develop and confirm a model for the iodine spiking phenomenon, in which the iodine concentration in the reactor coolant rises to many times its equilibrium concentration level (peak concentration) followed by a decay back to a level below the peak concentration. Procurement of data from operating plants and the development of a fuel release model for predicting the magnitude of the spikes will provide an understanding of this phenomenon which is not presently available. Improved knowledge of this topic would establish a better basis for accident calculations and could be used as a basis for establishing new reactor coolant activity limits.

Safety Significance

The calculated radiological consequences for some postulated design basis accidents are highly dependent upon the magnitude of the iodine spike assumed in the dose calculation model. These calculations are made with conservative assumptions, incorporating an iodine spiking factor which is based on a limited sample of plant data, and are in turn used to establish allowable coolant activity limits in the TS governing plant operations. However, the iodine spiking is a significant effect in only non-core melt accident consequences, which are not major contributors to nuclear plant risk.

PRIORITY DETERMINATION

A technical analysis of the proposed resolution of this issue was performed by PNL.[64] The resolution of this issue would apply to all operating and planned LWRs.

Frequency/Consequence Estimate

For converting thyroid exposure to equivalent whole body exposure, PNL derived a PWR expected public risk of 0.0143 man-rem/RY for a non-core melt SGTR and a coincident iodine spike using: (a) the PWR SGTR Task Force estimates for the probability of non-core melt SGTR events (1.3×10^{-3}/RY) and the amount of radioiodine (I-131) released (53,600 Ci/event); (b) the Prairie Island 1 conversion factor for translating curies of I-131 released to thyroid exposure; and (c) the conversion factor derived in the prioritization of Item III.A.1.3, "Maintain Supplies of Thyroid Blocking Agent." Using the ratioing technique described in NUREG/CR-2800[64] and a BWR small break LOCA frequency of 1.4×10^{-3}/RY, a BWR expected public risk due to a small break LOCA with a coincident iodine spike of 0.0185 man-rem/RY was derived.

Peak iodine concentration levels were estimated by AEB based on the average measured PWR and BWR coolant activity levels and an average peaking factor of 500, which was derived from the small population of data available on the iodine spiking phenomena. The peak primary coolant activity levels derived in this manner were estimated to be 60 μCi/gm and 4 μCi/gm for PWRs and BWRs, respectively, and represent the base case average peak iodine concentrations before resolution of this issue.

Dose calculations used by the STGR Task Force were performed using an assumed coolant iodine activity level increase by a factor of 20 and a maximum allowed primary coolant iodine concentration of 1.0 μCi/gm for PWRs and 0.2 μCi/gm for BWRs, or an allowable primary coolant peak iodine concentration of 20 μCi/gm and 4 μCi/gm for PWRs and BWRs, respectively. It was assumed that new coolant activity limits established after the iodine spiking phenomena was better understood and would not permit allowable peak iodine concentrations greater than those derived above. Thus, the above values are assumed to represent the adjusted case peak allowable coolant activity concentrations after resolution of this issue.

The post-implementation or adjusted public risk was determined by multiplying the pre-implementation or base case public risk by the ratio of the post-implementation reactor primary peak iodine concentration level to the pre-implementation average primary peak iodine concentration. As a result, the adjusted case public risk of 0.00477 man-rem/RY and 0.0185 man-rem/RY was calculated for PWRs and BWRs, respectively.

The change in public risk which might be realized by completion of this issue was determined by subtracting adjusted public risk from the base case public risk. The change in public risk was thus calculated to be 0.00953 man-rem/RY and 0 man-rem/RY by multiplying the above changes in public risk by the respective number of reactors and their average remaining lifetime (i.e., PWRs - 90 reactors and 28.8 years; BWRs - 44 reactors and 27.4 years) and adding the products. Total public risk reduction was estimated to be 25 man-rem for completion of this issue.

Since this iodine spiking issue does not significantly affect core-melt accident consequences, resolution of the issue would not result in a core-melt frequency change.

Cost Estimate

From the currently available data, it was judged that the 4-hour sampling interval following a transient, which is currently proposed in LCOs, would probably miss some spiking peaks. A change to a 2-hour interval was thus assumed to provide adequate information for peak activity determination. The total sampling period following a major power transient was estimated to be 33 hours. At a sampling interval of 2 hours, rather than 4 hours, it was estimated that 8 additional samples would be required following each major transient. A survey of the available iodine spiking data resulted in an estimated frequency of iodine spiking events of 0.52/RY and 0.14/RY for PWRs and BWRs, respectively.

Industry Cost: It was assumed that the costs to industry are due to the increased frequency of iodine sampling after each transient. No new equipment for sampling and analysis was assumed to be required. However, some minor modification of the sampling systems was assumed to be required at operating plants to accommodate the increased sampling frequency.

At the 71 operating plants, 4 man-weeks of labor were assumed to upgrade the sampling and analysis capability to accommodate the shorter sampling interval. At a cost of \$2,270/man-week, a total industry implementation cost of \$645,000 was calculated.

Increased industry operating costs were estimated using the 8 estimated additional samples per major transient, the above estimated iodine spike frequencies for PWRs and BWRs, the respective number of reactors and their average remaining life, and an estimated 2 man-hours to obtain and analyze a reactor coolant sample. A total industry operating cost of \$1.38M was calculated. Therefore, the total industry cost associated with this issue was estimated to be \$2M.

NRC Cost: Efforts required by the NRC to develop and confirm a model for the iodine-spiking phenomenon could be significant because little is known about the physics associated with the phenomenon. Two staff-years of NRC effort were estimated for the development of new requirements. Contractor support of the development of new requirements was estimated to be \$300,000. At a cost of \$100,000/man-year for NRC personnel, a total NRC cost of \$0.5M for resolution of the issue and development of new requirements was estimated.

It was assumed that NRC staff time would be expended in the review of increased sampling requirements and the resulting information during the lifetime of the plants. It was estimated that 0.1 man-week/RY would be required to monitor the new sampling requirements and plant results at a total NRC cost of \$860,000. Thus, the total NRC cost is estimated to be about \$1.4M.

Value/Impact Assessment

Based on a public risk reduction of 25 man-rem, the value/impact score is given by:

$$S = \frac{25 \text{ man-rem}}{\$(2 + 1.4)M}$$

$$= 7.4 \text{ man-rem/\$M}$$

Uncertainties

Uncertainty in cost was found to be small, about ± 50%. Uncertainty in the public risk reduction estimate ranged from about plus 2 orders of magnitude on the upper bound to about minus 1 order of magnitude on the lower bound.

Other Considerations

It was assumed that all the labor associated with obtaining and analyzing additional record coolant samples would, of necessity, be expended in moderate radiation fields. In addition, one-fourth of the labor estimated for modification of the sampling systems at operating plants was assumed to occur in a moderate radiation field. Assuming a field of 25 millirem/hr a total increased ORE of 370 man-rem was estimated.

CONCLUSION

The total public risk reduction calculated for this issue is insignificant. Furthermore, the value/impact ratio is poor. The estimated increase in ORE due to the assumed resolution of the iodine spiking issue is large in comparison to the estimated public risk reduction, which would also be an incentive for a drop priority assignment. Uncertainty, although high for the public risk reduction estimate, would only support a remote possibility that the issue could warrant as high as a medium-priority assignment. Therefore, based upon the above considerations, we recommend that this issue be assigned a DROP priority.

REFERENCES

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

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

ISSUE 20: EFFECTS OF ELECTROMAGNETIC PULSE ON NUCLEAR POWER PLANTS

DESCRIPTION

Historical Background

This concern was raised[183] because of the potential for a high-altitude nuclear weapon detonation causing a large electromagnetic pulse (EMP) which subsequently could induce large currents and voltages in electrical systems. The concern was that sensitive electronics at nuclear power plants could be irreparably damaged. In addition, Petitions for Rulemaking on EMP (PRM-50-32, 32A, and 32B) have been filed.

Safety Significance

This issue is unique because of its ability to affect more than one plant at the same time. Portions of a nuclear power plant's electrical, instrumentation and control systems may be disabled due to the large currents and voltages which could be induced. Loss of critical systems such as offsite power, emergency onsite power, etc., could lead to loss of core cooling with subsequent core melt.

The original concern was that sensitive electronics would be irreparably damaged, but it now appears that, if failure occurred, it would likely be only momentary (i.e., trip breakers, etc.) and the failed equipment could be restored to service to continue core heat removal.[115]

Possible Solution

If the electrical equipment necessary for safe shutdown displayed sensitivity to EMP, then a minimum safety margin for the ratio of the peak EMP voltage to its damage threshold voltage could be chosen and the equipment would then be required to meet this criterion. This could then provide assurance that the equipment would not be damaged.

CONCLUSION

Detailed programmatic information on this issue was presented in SECY-81-641[604] and subsequent program status reports were provided in SECY-82-157[605] and SECY-82-157A.[203] A study on the effects of EMP on nuclear power plants was documented in NUREG/CR-3069[115] and forwarded to the Commission in SECY-83-367.[606] This issue was RESOLVED with the Commission approval[607] of the staff's report and no new requirements were established. Continuing staff work in response to the PRMs is a separate entity and does not affect this conclusion.

REFERENCES

115. NUREG/CR-3069, "Interaction of Electromagnetic Pulse with Commercial Nuclear Power Plant Systems," U.S. Nuclear Regulatory Commission, February 1983.

183. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR to NRR Staff," December 1976.

203. SECY-82-157A, "Status Report on the NRR Investigation of the Effects of Electromagnetic Pulse (EMP) on Nuclear Power Plants," July 16, 1982.

604. SECY-81-641, "Electromagnetic Pulse (EMP) - Effects on Nuclear Power Plants," November 5, 1981.

605. SECY-82-157, "Status Report on the Evaluation of the Effects of Electromagnetic Pulse (EMP) on NUclear Power Plants," April 13, 1982.

606. SECY-83-367, "Staff Study of Electromagnetic Pulse (EMP) Effects on Nuclear Power Plants and Discussion of Related Petitions for Rulemaking (PRM-50-32, 32A, and 32B)," September 6, 1983.

607. Memorandum for W. Dircks from S. Chilk, "SECY-83-367 - Staff Study of Electromagnetic Pulse (EMP) Effects on Nuclear Power Plants and Discussion of Related Petition for Rulemaking (PRM-50-32, 32A, and 32B)," November 15, 1983.

ISSUE 34: RCS LEAK

DESCRIPTION

Historical Background

This issue was raised by AEOD[492,569] and involved isolation of the reactor coolant system charging and letdown system following a spurious safety injection transient at H. B. Robinson on January 29, 1981. Following the spurious safety injection, the plant operators initiated actions to bring the plant to hot shutdown. During automatic isolation of the CVCS letdown line due to the SI, it is believed that the outermost isolation valves closed faster than the two open orifice isolation valves. Leakage past the orifice isolation valves resulted in opening of the relief valve and rupture of the valve bellows. Also, a pressure surge due to isolation valve closure caused an upstream valve drain line cap to blow off. Other concerns related to spurious actuation of SI are discussed in Issue 8 (Inadvertent Actuation of Safety Injection in PWRs).

Safety Significance

In a detailed review of the H. B. Robinson event by AEOD,[569] it was determined that no safety concern was involved and that the resultant small LOCA inside containment was within the analyzed SBLOCA. However, AEOD concluded that improved procedures to handle the spurious safety injection actuation signal could have prevented overpressurization of the CVCS piping run downstream of the orifice isolation valves but upstream of the CVCS containment isolation valves.

AEOD recommended that NRR review the procedures for identification and recovery from spurious SI actuations and the closing sequences in the CVCS isolation system. In regard to the latter, AEOD suggested that closing the orifice isolation valves prior to the CVCS containment isolation valves, could eliminate actuation of the relief valve. However, it is noted that the closure sequence suggested by AEOD most likely would not have eliminated blowoff of the valve drain cap (valve CVC-200E), which appears to have been the major RCS leakage path. Also, even though the relief valve bellows failed, failure of the bellows did not affect the pressure relieving function of the relief valve in the low design pressure (600 psi) section of the CVCS piping run.

Solution

The RCS leakage to containment was primarily from a blown-off drain line cap in the letdown piping. A manual valve that is normally-closed upstream of the drain cap was either left partially open by maintenance error or opened by vibration. The licensee's fix to this problem consisted of locking closed the manual drain line valve in this piping segment. Other similar arrangements in the letdown piping were either locked closed or verification of closure required. The drain cap was replaced. The staff position is that the licensee's fix to preclude similar events at H. B. Robinson is acceptable.

The letdown piping configuration at H. B. Robinson is typical for W designs, but not typical for the CE or B&W designs. The CE and B&W designs principally have the containment isolation upstream of the letdown orifices and orifice isolation valves. The event that occurred at H. B. Robinson should bound any similar events for all PWR designs.

CONCLUSION

The leak rate of 5 to 7 gpm that resulted from the event was well within the makeup capability of the charging system at H. B. Robinson. The event did not result in unacceptable consequences and is more appropriately termed a small leak and not an SBLOCA. Issue 58 (Containment Flooding) also bounds the event that occurred and NRR review of procedures for identification and recovery from spurious SI actuations is addressed in TMI Action Plan Item I.C.1.

The resultant SBLOCA described above is within the analyzed SBLOCA and over-pressurization of the low pressure CVCS piping section was mitigated by the relief valve. Therefore, we are in agreement with AEOD that the event did not identify a new safety concern and recommend that this issue be DROPPED from further consideration as a generic safety issue.

REFERENCES

492. Memorandum for C. Michelson from H. Denton, "H. B. Robinson RCS Leak on January 29, 1981," June 15, 1981.

569. "Engineering Evaluation of the H. B. Robinson Reactor Coolant System Leak on January 29, 1981," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 23, 1981.

ISSUE 36: LOSS OF SERVICE WATER

DESCRIPTION

Historical Background

On June 19, 1981, AEOD issued a preliminary report[557] for review and comment on their case study of the incident at the Calvert Cliffs Unit 1 plant in which the plant lost both redundant trains of service water when the system became air bound. On August 5, 1981, NRR provided comments[494] on this report, but indicated that their review was limited because of the preliminary nature of the AEOD report which did not contain recommendations and conclusions. Subsequently, AEOD issued the final version[558] of this case study on December 17, 1981, in which three generic recommendations were presented in the cover letter and NRR was requested to comment on this version of the AEOD work. NRR provided another review[559] of the case study on September 23, 1982, which was more detailed and specifically addressed the AEOD generic recommendations. However, in an AEOD memorandum[560] to NRR on May 2, 1983, additional clarification to the case study was provided. In response to this memorandum, DL requested[461] DSI to review the latest AEOD memorandum. DSI performed this review and provided specific and detailed responses[562] to all concerns identified by AEOD. Based on the content of the DSI memorandum,[562] the AEOD concerns were addressed and solutions were identified in a memorandum[563] from NRR to AEOD on September 15, 1983. Also, in an AEOD memorandum[564] to OIE on August 18, 1983, detailed information was provided so that an appropriate Information Notice could be issued by OIE. As a result, IE Information Notice No. 83-77[565] was issued on November 14, 1983.

Safety Significance

Calvert Cliffs Unit 1 experienced a loss of both redundant trains of service water when the system became air-bound as a result of the failure of a non-safety-related instrument air compressor aftercooler. The significance of this event lies in the fact that it involved two fundamental aspects in the design of safety-related systems: (1) interaction between safety and non-safety-related systems and components, and (2) common cause failure of redundant safety systems.

Possible Solutions

A summary of the AEOD recommendations and the NRR responses are as follows:

I. AEOD Recommendations of Section 8, Part (a) of the AEOD Report[558]

   1. AEOD Recommendation (1)

      It is recommended that butterfly valves SW-4 and SW-5 have valve operators added (pneumatic or electric motor) and that these valves either close automatically, as do the valves on the turbine supply header, or as a minimum have the capability to be remote manually-operated from the control room.

NRR Responses[559,563]

Inasmuch as the service water system at Calvert Cliffs is a "dual purpose" system, as defined in SRP[11] 3.6.1, a single failure in one redundant safety-related train is not postulated concurrently with a moderate-energy line break in the other train. Therefore, it is always assumed that the check valve will function as designed to provide the isolation of both trains. Consequently, this configuration is acceptable under the current staff criteria.

However, during the review of operating reactor compliance with the moderate energy pipe break criteria, the staff considered the effect of leak rate from postulated cracks on the system operability and the time required to isolate the break before loss of system safety function. Based on these reviews, the NRR staff believes that sufficient instrumentation (service water head tank level alarms) and time (30 minutes) are available for the Calvert Cliffs operators to locally close the manual isolation valve in such an event. Further, makeup to the heat tank provides additional time for taking this action.

2. AEOD Recommendation (2)

It is recommended that the four check valves and the four solenoid operated three-way valves in the instrument air lines that provide control air for the four diesel-generator 12 service water supply and return valves be added to the IST program.

NRR Response[559]

A number of plant-specific recommendations were made for the Calvert Cliffs plant. Although NRR agrees that implementation of the AEOD plant-specific recommendations may be beneficial to the operation of the facility, NRR does not believe that ordering such changes would be accompanied by an appropriate increase to the health and safety of the public. Considering that the licensee participated in the peer review of the AEOD case study, they are aware of the AEOD plant-specific recommendations. Therefore, NRR does not believe that further regulatory actions are necessary.

NRR does, however, have the following comment on this plant-specific recommendation. The NRR/OIE working group on instrument and service air system, which was formed after growing concerns of air system degradation, was considering generic recommendations regarding the isolation and boundary between safety and non-safety-related air systems. By copy of the NRR memorandum,[559] the AEOD recommendation to include system boundary valve in the IST program was forwarded to the working group for their information. This working group was disbanded before they could review this recommendation. Therefore, a memorandum[566] was issued to DL for their review and action on this item.

3. AEOD Recommendation (3)

If there is a loss of offsite power and the service water system supplying diesel generator 12 becomes unavailable, the diesel will transfer to an inactive service water loop. Since operator action is necessary to realign the diesel generator such that it energizes the bus which powers the service water pump in the loop to which it was transferred or, alternatively, to start a third service water pump, it is recommended that the human factors of these actions be evaluated against the length of time the diesel can run without service water before it trips.

NRR Response[563]

In general, NRR believes that implementation of AEOD's plant-specific recommendations may be beneficial to the operation of the facility. Therefore, under a separate letter,[567] NRR has forwarded a copy of the AEOD final report to the licensee with a statement that NRR considers the recommendations to be valid and that implementation of the AEOD plant-specific recommendations be considered by the licensee.

4. AEOD Recommendations (4) and (6)

For operating plants and plants currently in the licensing process that have service water systems that contain both safety and non-safety-related portions, it is recommended that the system isolation provisions be reviewed to identify any procedural or hardware changes necessary to protect the safety-related portion of the service water system from a failure portion of the service water system from a failure in the non-safety-related portion during normal operation and accident conditions.

It is recommended that the guidance in the SRP be clarified to emphasize automatic isolation of the non-safety-related portion of the service water system when it degrades the operability of the safety-related portion of the system.

NRR Responses[559,563]

NRR concurs that the guidelines in SRP[11] Section 9.2.2 could more clearly stated when automatic isolation of safety and non-safety-related portions of the system is necessary, such as indication of low pressure in the non-safety-related portion as would occur in a failure of the non-seismic Category I piping due to an SSE. The staff will propose revisions to the review procedures (Paragraph III.3.a) of SRP[11] Section 9.2.2 to more clearly indicate the types of isolation signals required and when isolation is necessary.

5. AEOD Recommendation (5)

It is recommended that an IE Circular on common cause failure of service water systems be issued.

NRR Response

An IE Information Notice[565] was issued on November 14, 1983.

II. AEOD Recommendations of Section 8, Part (b) of the AEOD Report[558]

All of the recommendations (1-3) in this section concern steam generator tube rupture events. Upon approval by CRGR and the Commission, the action on these recommendations will be carried forward under Steam Generator Staff Actions (Issue 67), stated in References 559 and 563 as the generic studies being performed by the staff.

III. Additional AEOD Recommendation[560]

AEOD Recommendation

The accessibility of the steam generator dump valves (ADVs) during accident conditions be reviewed to determine the acceptability of the assumption that the affected steam generator can be isolated in 30 minutes with manual operation of the ADVs.

NRR Response[563]

This matter will also be addressed under Issue 67, "Steam Generators Staff Actions" subsequent to approval. However, it is noted that there are currently no plans for the backfitting of RSB BTP 5-1. A cost/benefit analysis concerning backfit of RSB BTP 5-1 is now implicitly a part of USI A-45. A-45 will address decay heat removal system improvements including consideration of the ADV. The staff requirements for the successful completion of this effort are outlined in a DST memorandum.[568]

CONCLUSION

Based on the contents of References 563 and 564, it appears that all but one generic concern and one plant-specific matter raised by the AEOD case study on the Calvert Cliffs loss of service water have been or will be adequately addressed as part of USI A-45 or Issue 67. The remaining plant-specific matter concerning Calvert Cliffs has been brought to the attention of the DL for appropriate action.[566] In addition, the clarification of SRP[11] Section 9.2.2 is available and would resolve the remaining generic concern.

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

494. Memorandum for C. Michelson from H. Denton, "AEOD Preliminary Report on Calvert Cliffs Unit 1 Loss of Service Water," August 5, 1981.

557. Memorandum for M. Denton and V. Stello from C. Michelson, "Calvert Cliffs Unit 1 Loss of Service Water," June 19, 1981.

558. Memorandum for H. Denton and R. DeYoung, Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980," December 17, 1981.

559. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Final Report: Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980," September 23, 1982.

560. Memorandum for M. Denton from C. Heltemes, "Response to NRR Comments on AEOD Report, 'Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980,'" May 2, 1983.

561. Memorandum for W. Houston and L. Rubenstein from F. Miraglia, "Response to NRR Comments on AEOD Report 'Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980'," June 2, 1983.

562. Memorandum for F. Miraglia from W. Houston and L. Rubenstein, "Comments to AEOD Memo dated May 2, 1983, on Calvert Cliffs, Unit 1, Loss of Service Water on May 20, 1980," July 22, 1983.

563. Memorandum for C. Heltemes from H. Denton, "Response to NRR Comments on AEOD Report, Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980," September 15, 1983.

564. Memorandum for R. Baer from K. Seyfrit, "Case Study, 'Calvert Cliffs Unit 1 Loss of Service Water on May 29, 1980,'" August 18, 1983.

565. IE Information Notice No. 83-77, "Air/Gas Entrainment Events Resulting in System Failures," U.S. Nuclear Regulatory Commission, November 14, 1983.

566. Memorandum for G. Holahan from W. Minners, "Prioritization of Issue 36: Loss of Service Water at Calvert Cliffs Unit 1," November 10, 1983.

567. Letter to A. E. Lundvall (Baltimore Gas and Electric Company) from D. Eisenhut (NRC), Docket No. 50-317, September 15, 1983.

568. Memorandum for W. Houston and L. Rubenstein from F. Schroeder, "Request for Reactor Systems Branch and Auxiliary Systems Branch Support for Plant Visits on USI A-45," November 28, 1983.

ISSUE 40: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

DESCRIPTION

Historical Background

On April 3, 1981, AEOD published draft NUREG-0785, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System."[324] As a result of the development of these safety concerns and the findings presented in the report, the NRC staff met with representatives of the BWR Regulatory Response Group and GE on April 9, 1981. A letter[325] was issued on April 10, 1981 to all BWR licensees requiring a generic evaluation of the safety concerns within 45 days of receipt and a plant-specific evaluation within 120 days of receipt.

A meeting was held with GE on April 28, 1981 to discuss the status of its generic evaluation. Subsequently, NEDO-24342[326] was submitted to the NRC by letter dated April 30, 1981.[327]

A multidisciplinary group from NRR was assembled to review the generic evaluation. A three-phase approach was developed to identify generic review objectives and describe review termination points. It was agreed that this approach would be based on establishing either: (1) a low probability for the event, (2) acceptable consequences for the event, or (3) alternate cooling systems and mitigation equipment for the event.

As the review progressed, it became evident that a sufficient data base did not exist to conservatively terminate the generic review on the basis of a quantitative risk assessment. It was equally difficult to show acceptable consequences for all scram initiators, considering the potential for an unisolable leak from the reactor coolant system into the reactor building. Thus, it was necessary to generically evaluate the mitigation capability for this scenario.

As the evaluation proceeeded, several suggestions for improving and verifying piping integrity, mitigation capability, and environmental qualifications of essential equipment were made. These suggestions are discussed in NUREG-0803[328] which begins with a review of the licensing design basis for the SDV piping system. An evaluation of the SDV piping system integrity and an assessment of the mitigation capability follow. Finally, each suggestion for improvement is evaluated in NUREG-0803[328] and the final guidance for resolution of this problem is presented. NUREG-0803[328] was transmitted to the BWR licensees, CP applicants, CP holders, and OL applicants by letters.[329,332] These letters also requested appropriate responses to the safety concerns and guidelines presented in NUREG-0803.[328] In these letters, it has been noted that an acceptable plant-specific response for this issue will conform to the final approved guidance provided in NUREG-0803.[328]

However, an additional submittal[402] was forwarded to the NRC staff by GE and the BWR Owners' Group in August, 1982 in which an analysis was presented to demonstrate the probability of a pipe break in the scram discharge volume system was negligibly small and that, therefore, this issue should not be regarded as a significant safety issue. On the basis of its review of the August, 1982 submittal, the NRC staff concluded that the results of the submittal were unacceptable. However, before the submittal was formally rejected by the staff, GE and the BWR Owners' Group provided additional material which amplified the August, 1982 submittal with supporting information[403] which was presented at a meeting with the staff on February 8, 1983.

A study[337] was completed which describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated small-break LOCA outside of the primary containment. This study is contained in the first volume of a two-volume study in which a detailed analysis of the accident sequence is presented. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment will be provided in Volume 2 of the study.

Safety Significance

If a break or leak exists or develops in the SDV piping during a reactor scram, this would result in the release of water and steam at 212°F into the reactor building at a maximum flow rate of 550 gpm and is postulated to result in 100% relative humidity in the reactor building. The principal means of isolating this break would be to close the scram exhaust valves which are located on the hydraulic control units; however, this is dependent upon the ability to reset scram, which cannot be absolutely ensured immediately following the scram. Therefore, a rupture of the SDV could result in an unisolable break outside of primary containment, which is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

Solution

NUREG-0803[328] provides guidance to ensure pipe integrity, detection capability, mitigation capability and qualification of the emergency equipment to the expected environment.

CONCLUSION

This issue was RESOLVED, requirements were established, and MPA B-65 was established by DL for implementation purposes.[602]

REFERENCES

324. NUREG-0785, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," U.S. Nuclear Regulatory Commission, April 1981.

325. Letter to All BWR Licensees from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," April 10, 1981.

326. NEDO-24342, "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," General Electric Company, April 1981.

327. Letter to D. Eisenhut (NRC) from G. Sherwood (GE), "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," April 30, 1981.

328. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," U.S. Nuclear Regulatory Commission, August 1981.

329. Letter to All GE BWR Licensees (Except Humboldt Bay) from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-34)," August 31, 1981.

332. Letter to All BWR Applicants for CPs, Holders of CPs, and Applicants for OLs from D. Eisenhut, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-35)," August 31, 1981.

337. NUREG/CR-2672, "SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis," U.S. Nuclear Regulatory Commission, November 1982.

402. Letter to D. Eisenhut from T. Dente (BWR Owners' Group), "Analysis of Scram Discharge Volume Piping Integrity, NEDO-22209 (Prepublication Form)," August 23, 1982.

403. Letter to K. Eccleston (NRC) from T. Dente (BWR Owners' Group), "Transmittal of Supporting Information on Application of Scram Time Fraction to Scram Discharge Volume (SDV) Pipe Break Probability as Used in NEDO-22209," January 28, 1983.

602. Memorandum for T. Speis from R. Mattson, "Status of Generic Issues 40 and 65 Assigned to DSI," December 27, 1983.

ISSUE 45: INOPERABILITY OF INSTRUMENTATION DUE TO EXTREME COLD WEATHER

DESCRIPTION

On September 27, 1979, OIE issued Bulletin No. 79-24[501] regarding frozen lines. Issuance of the bulletin was prompted by LERs which had revealed many events involving frozen instrument, sampling, and processing lines. All licensees and CP holders were requested to "review their plants to determine the adequate protective measures have been taken to assure that safety related process, instrument and sampling lines do not freeze during extremely cold weather." The results of those reviewed were to be reported to the Regional Director by October 31, 1979.

AEOD addressed the concerns for inoperability of instrumentation due to extreme cold weather in a memorandum to NRR and OIE on June 15, 1981.[502] Highlighted in the memorandum was the December 1980 event at Arkansas Nuclear One, Unit 2 in which all four RWST instrumentation channels were lost when the level transmitters froze. The system heat-tracing circuit was de-energized because the main line fuse was removed. This situation would have prevented the automatic change over of the ECC from injection to recirculation mode under LOCA conditions (i.e., loss-of-safety function). AEOD requested that OIE and NRR address the generic problem identified in its memorandum. It was suggested that OIE issue a supplement to Bulletin 79-24[501] and that NRR address the adequacy of protective measures for freezing of safety-related instrument lines in the review of OLs.

In an August 14, 1981 memorandum to AEOD,[503] NRR advised that a BTP on freeze protection of safety-related instrument lines was being developed and would be included in the appropriate SRP[11] Section following its review and approval. NRR further advised that OIE proposed to amend the Inspection and Enforcement Manual[247] to include a module which would set forth requirements for inspection of systems and measures for protection against cold weather. This inspection module would require that regional inspectors perform plant site visits prior to the beginning of the cold season to verify the condition of heat-tracing systems and measures taken to protect plant equipment from cold weather conditions. An amendment to the Inspection and Enforcement Manual (Procedure No. 71714)[247] was issued by OIE on January 1, 1982, thus completing the OIE portion of the resolution of this issue. Acceptance criteria for the design of protective measures against freezing in instrument lines of safety-related systems were included in Draft[504] Regulatory Guide 1.151, "Instrument Sensing Lines." With inclusion of the criteria in the Draft Regulatory Guide, further work on a BTP was terminated. The Draft Regulatory Guide was issued for comment in March, 1982. Comments were collected and dispositioned and the Regulatory Guide[505] was published in July, 1983. Notice of the issuance of Regulatory Guide 1.151[505] was published in the Federal Register[506] on August 8, 1983. Implementation of the Guide is limited to all CPs issued after September 1, 1983. However, other licensees or applicants may adopt the use of the Guide on a voluntary basis. As stated in the value/impact statement for the Guide, no backfitting of requirements for freeze protection and alarms is to be accomplished other than those changes effected by IE Bulletin 79-24[501] (and the inspection requirements added to the OIE Inspection Manual).

In February, 1984, the following SRP[11] Sections were revised to incorporate the changes associated with the resolution of this issue: (1) Section 7.1, Rev. 3; (2) Section 7.1, Appendix A, Rev. 1; (3) Section 7.5, Rev. 3; and (4) Section 7.7, Rev. 3. The issuance of these changes were addressed in References 570 and 571.

CONCLUSION

This issue has been RESOLVED and requirements were issued.

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

247. IE Bulletin No. 79-24, "Frozen Lines," U.S. Nuclear Regulatory Commission, September 27, 1979.

502. Memorandum for H. Denton and V. Stello from C. Michelson "Inoperability of Instrumentation Due to Extreme Cold Weather," June 15, 1981.

503. Memorandum for C. Michelson from H. Denton "AEOD Memorandum on the Inoperability of Instrumentation Due to Extreme Cold Weather," August 14, 1981.

504. Draft Regulatory Guide and Value/Impact Statement, Task IC 126-5, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, March 1982.

505. Regulatory Guide 1.151, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, July 1983.

506. Federal Register Notice 48 FR 36029, August 8, 1983.

570. Memorandum for V. Stello from H. Denton, "Issuance of Revised Section 7.1, Appendix A to This Section, Section 7.5 and Section 7.7 of the Standard Review Plan, NUREG-0800," March 9, 1984.

571. Memorandum for H. Denton from V. Stello, "SRP Changes Concerning Resolution of Generic Issue 45, Inoperability of Instrumentation due to Extreme Cold Weather," April 3, 1984.

ISSUE 68: POSTULATED LOSS OF AUXILIARY FEEDWATER SYSTEM RESULTING FROM TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP STEAM SUPPLY LINE RUPTURE

DESCRIPTION

Historical Background

In an evaluation of INPO/NSAC Significant Operating Experience Report 81-17,[271] the operators of the Fort Calhoun nuclear power plant determined that the configuration of their plant made it susceptible to the possibility that all AFW supply could be disabled by a break of the steam supply line inside the pump room. The steam line break would disrupt the supply of steam to the turbine driven pump and concurrently disable the electric motor driven pump, if the electric pump motor were not qualified to operate in the steam environment and were located in the same pump room. The operators of Fort Calhoun reported this deficiency in LER 82-012.

In the analysis,[271] the Fort Calhoun operators identified a sequence which begins with the loss of offsite power and postulated a break in the steam supply line when the AFW pumps are required to operate. Fort Calhoun has the electric motor-driven AFW pump housed in the same room as the steam turbine-driven AFW pump. In reviewing the LER submitted by Fort Calhoun, NRR concluded that the design met current acceptance criteria and that the scenario postulated, loss of offsite power followed by the passive failure of the steam supply line disabling all AFW pumps, was outside the scope of events postulated as part of the current licensing basis and did not represent a credible accident scenario.[272]

Safety Significance

AEOD, in performing a review of this issue, expressed a concern in their technical review report AEOD/T302[273] that a single passive failure could result in the loss of a safety system that could be required to bring the plant to a safe shutdown and suggested that the pipe break criteria presented in Section 3.6.1 of the SRP[11] be reviewed to determine if additional guidance is necessary. Similar problems have been identified at San Onofre Units 2 and 3 and at Arkansas Nuclear One, Unit 1.

Possible Solutions

Several possible solutions have been identified. One solution is the relocation of the turbine driven pump to anther room separate from the electric driven pump(s). An alternative to this first solution is locating a full capacity electric pump in a room separate from the steam turbine pump and any effects resulting from a steam line supply break. A second solution is the replacement of the electric pump motor(s) with Class 1E environmentally-qualified components. A third solution, the alternative implemented at the San Onofre facility, is the addition of a forced lube oil cooling system to the electric pump motor bearings. This assumes that everything else is already qualified. Finally, an augmented ISI of the steam lines has also been proposed as an alternative solution.

PRIORITY DETERMINATION

Frequency Estimate

Three event sequences were analyzed and assessed for each CE and the B&W type reactors to determine the frequency at which core-melt would be expected to occur as a result of placement of all AFW pumps in the same room. The sequence initiators appraised were: (1) a break of the AFW turbine-driven pump steam supply line inside the pump room, (2) a transient reactor trip other than interruption of the main feedwater, and (3) a loss of offsite power for longer than 15 minutes or interruption of the main feedwater.

For the first sequence, the turbine driven AFW pump steam supply line break occurs at a frequency of 2×10^{-3}/RY. This failure rate was estimated by assuming one AFW steam line break has occurred in 440 RY, which has not happened. Since only 10% of the steam line is in the pump room, the frequency is one tenth of this or 2×10^{-4}/RY The dominant sequence events and their probabilities which follow are: a turbine trip (0.9), offsite power is not lost (~1.0), the electric motor-driven AFW pump(s) fail due to the operating environment (0.9), the operators fail to align the steam generators back to the main feedwater system (0.1), and for only the B&W reactors, the operators are unsuccessful in achieving feed-and-bleed operation to remove decay heat with a probability of P, where $P \leq 1$. The frequency of this sequence is estimated to be less than $P(2 \times 10^{-6})$ event/RY for B&W reactors and $\sim 2 \times 10^{-6}$ event/RY for CE reactors. B&W reactors were given credit for feed-and-bleed operation while CE reactors were not given credit for this mode. Many CE reactors do not have sufficient pressure relief capability to permit injection of coolant water at the lower HPCI pressure provided by the CE reactors. The success of feed-and-bleed is difficult to establish on a generic basis and it has not been demonstrated to be achievable, in that the availability of procedural guidance and training in this operation is undetermined. Therefore, no probabilistic estimate of success is assigned.

The second sequence is initiated by transient trips other than the interruption of main feedwater or loss of offsite electrical power. These transient trips have a frequency of 7/RY. The dominant sequence events which follow and their probabilities are: retention of offsite power (~1.0), a rupture or break of the steam supply line to the turbine driven AFW pump inside the pump room (2×10^{-5} based on ten demands on the AFW per year), failure of the electric motor-driven AFW pump(s) due to the operating environment (0.9), failure by the operators to align the steam generators to the main feedwater system (0.1), and the operators are unsuccessful in achieving feed-and-bleed cooling (a value of P for B&W reactors). The frequency of the sequence is estimated to be less than $P(1.4 \times 10^{-6})$ event/RY for B&W reactors and $\sim 1.4 \times 10^{-6}$ event/RY for CE reactors.

The third sequence is initiated by the loss of offsite power for more than 15 minutes which has a frequency of 0.2/RY or an interruption of main feedwater which may be expected to occur at a frequency of 3/RY. The events following in the dominant sequence are: the steam supply line to the turbine-driven pump breaks inside the pump room (2×10^{-5}), emergency power is supplied by the diesel generators (~1.0), the electric motor-driven AFW pumps fail due to the operating environment (0.9), and for B&W reactors the operators are unable to achieve feed-and-bleed cooling (a value of P). The frequency of this sequence is estimated to be less than P(5×10^{-5}) event/RY for B&W reactors and ~6×10^{-5} event/RY for CE reactors.

The sum of the three dominant sequences for B&W reactors is ~P(5.3×10^{-5}) event/RY. For CE reactors, the sum of the three dominant sequences is ~6.3×10^{-5} event/RY. These values closely approximate the increase in core-melt frequency resulting from placing the steam turbine-driven pump and non-qualified electric motor-driven pumps within the same enclosure. It should be noted that the sequences involving loss of electric power, which were the original concern, do not dominate the results.

Consequence Estimate

The consequences of these sequences are obtained using the CRAC Code[64] for the release fractions and categories of a PWR as given in WASH-1400.[16] The calculations assume an average population density of 340 persons per square mile (which is the average for U.S. domestic sites) from an exclusion area of one-half mile about the reactor out to a 50-mile radius about the reactor. A typical midwest plain meteorology is also assumed.

The sequence described would be similar to the T_1MLU sequence described in NUREG/CR-1659[54] for Oconee resulting in a Category 3 release. For B&W reactors, the risk is P(144) man-rem/RY or P(3,850) man-rem/reactor for the remaining life of the reactor. If P were considered to be 0.9, which means that only one in ten attempts to cool the core by using feed-and-bleed is successful, then the risk would be 130 man-rem/RY or 3,460 man-rem/reactor. For CE reactors, the risk is 170 man-rem/RY or 4,600 man-rem/reactor.

Cost Estimate

It was estimated by one utility[274] that relocating the turbine pump would cost \$13.5M. The cost to install Class 1E environmentally-qualified motors was estimated to be \$5.2M. The addition of the forced-cooled lube oil system was estimated to be \$2.5M. (While the forced-cooled lube system could permit the continued operation of the electric pump motors in the steam environment at San Onofre, it is not certain that a similar solution would be possible at the other affected plants.)

It is estimated that an enhanced ISI effort would reduce the frequency of steam supply line breaks and the effects which result by a factor of 5. Enhanced inspection of the AFW steam supply line would reduce the frequency of core-melt accidents involving the failure of the steam supply line from 5.3×10^{-5} and 6.3×10^{-5} core-melt/RY to 1.1×10^{-5} and 1.3×10^{-5} core-melt/RY for B&W and CE reactors, respectively. The core-melt reduction reduces the public risk by 227 and 270 man-rem, respectively, for the 27 years remaining reactor life. The cost to perform this ISI is estimated to be 0.1 person-year/RY. The person-year costs include efforts necessary to erect scaffolding, remove insulation materials, perform and evaluate the inspections, and to restore the system to an operable configuration. In addition, a one-time cost of \$30,000 is estimated to be necessary to make some piping changes to permit the inspection of some welds. The total cost estimated to perform ISI of the AFW steam supply line inside the pump room for the remaining plant life is \$0.3M.

Value/Impact Assessment

As described, the costs for modifications to eliminate the risk associated with this issue is estimated to be between \$2.5M and \$13.5M. For B&W reactors having a risk exposure of 3,850 man-rem/reactor, the value/impact score varies between 1,540 and 285 man-rem/\$M per reactor. For CE reactors having a risk exposure of 4,600 man-rem/reactor, the value/impact score varies between 1,840 and 340 man-rem/\$M per reactor. ISI has a value/impact score of 756 and 900 man-rem/\$M per reactor for B&W and CE reactors, respectively.

Other Considerations

The accident avoidance savings, based upon a resolution which reduces the core-melt frequency by 5×10^{-6} event/RY, is \$82,000/RY or \$2.2M for the lifetime of the reactor.[64]

No increase in occupational exposure is expected to result from the proposed resolution of this issue. The results of the analysis are sensitive to changes in the assigned failure rates. An order of magnitude change in accident frequency could change the priority ranking assigned.

CONCLUSION

The value/impact rating for this issue is high for those facilities that have pumps co-located and suceptible to this common cause failure. A review and evaluation of the criteria for judging the acceptability of new proposed plants to determine if similar practices would be judged acceptable is deemed to be of HIGH priority. The requirement for redundancy and diversity to assure the availability of an important safety system is of dubious benefit if designs are permitted in which environmental or single-fault conditions may render the diverse or redundant components inoperable.

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

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 Application Program," U.S. Nuclear Regulatory Commission, 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission.

271. INPO/NSAC Significant Operating Experience Report 81-17, "Potential for Steam Line Rupture to Affect Auxiliary Feedwater System," November 11, 1981.

272. Memorandum for J. Gagliardo from D. Eisenhut, "Potential Failure of Turbine Driven Auxiliary Feedwater Pump Steam Supply Line - Fort Calhoun," October 8, 1982.

273. Memorandum for H. Denton from C. Michelson, "Technical Review Report, Postulated Loss of Auxiliary Feedwater System Resulting from Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture," February 16, 1983.

274. Letter to G. Knighton (NRC) from K. Baskin (Southern California Edison Company), "Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station Units 2 and 3," October 29, 1982.

ISSUE 70: PORV AND BLOCK VALVE RELIABILITY

DESCRIPTION

Historical Background

This issue was identified in a DL memorandum[515] and involves assessing the need for improving the reliability of PORV and block valves.

Both the PORVs and block valves were originally designed as non-safety components in the reactor pressure control system for use only when at power operation. The block valves were installed because of expected leakage from the PORVs. Neither the PORVs nor the block valves were required to safely shut down the plant or mitigate the consequences of accidents. However, RSB has recently determined that PORVs are, in fact, relied upon to mitigate a design-basis SGTR. The acceptability of relying on non-safety grade PORVs to mitigate a design-basis accident (SGTR) was raised in an RSB memorandum.[575]

In most plants, the low temperature overpressure protection (LTOP) system is designed to use the PORVs. For this mode of operation, the valves are typically set to open at 500 psig rather than the high pressure (~2300 psig) setpoint used at power. Westinghouse and some CE designed plants use redundant PORVs for LTOP concerns. These plants are brought to a water solid condition during shutdown. In contrast, B&W owners use a single PORV and the gas (steam or nitrogen) space in the pressurizer functions as the primary LTOP system. The PORV and associated actuation circuitry functions as a backup should the operator fail to terminate a low temperature overpressure challenge prior to compression of the gas space. LTOP systems, as specified in SRP[11] Section 5.2.2, are to be single failure proof, testable, designed to quality standards, and operable from emergency power. Full implementation of IEEE Standard-279 to withstand an SSE is not specified, but the OBE is. At the present time, the LTOP system requirements are being implemented as MPA B-04.[578]

NUREG-0737,[98] Item II.D.1, set forth functional requirements for both PORVs and block valves. All plants were required to demonstrate the functionability of these valves for all expected flow conditions during operating and accident conditions. It was further required that the block valves be capable of closing to ensure that a stuck open relief valve can be isolated, thereby terminating a small loss of coolant accident. In response to the Item II.D.1 requirements, PORVs were tested extensively by EPRI[583] and the results reported to the staff. Only limited block valve testing has been performed by the EPRI program. Reports describing the test program results have been submitted to the NRC staff for review as well as some plant-specific evaluations. Most plants have requested exemptions to the specified completion date for Item II.D.1 to obtain additional time for required evaluation of piping associated with safety valves, PORVs, and block valves.

When PORVs are used for high point vents in some plants under Item II.B.1 of NUREG-0737,[98] both PORVs and block valves are required to meet seismic and environmental requirements for safety-related equipment.

There was, and still is, no TS requirements that these components be operational when the plant is at power. Continued operation at power with inoperable PORVs and block valves is permitted by the TS if the block valve is closed and power to the block valve(s) is removed. Many of the plants now operate with the PORVs blocked.

It is noted that the safety evaluations[572,573,574] for Item II.K.3(2) determined that an automatic PORV isolation system is not necessary. However, the possible need to improve PORV reliability was recognized. The Item II.K.3(2) conclusion was predicated on the absence of the need to reduce the PORV-SBLOCA frequency by this single modification. Therefore, the scope of the evaluation of Item II.K.3(2) was limited in that it did not consider, or combine, the automatic PORV isolation system as a subset with other measures that could be taken to improve overall PORV/block-valve system reliability. The analysis provided herein also indicates that a singular modification, like that reviewed in Item II.K.3(2), or a reliability program to improve PORV/block-valve system reliability without considering improvements to the control element (automation vs. manual) may yield at most small benefits. The broader program reviewed herein includes an automated PORV isolation system as a possible part of an overall PORV/block-valve system reliability improvement. This analysis therefore expands on the evaluation of Item II.K.3(2) by including additional means to improve the PORV/block-valve system reliability and assessing all of the modes of risk reduction (and costs) that would likely result from these improvements.

Safety Significance

PORVs and block valves are used in various modes of plant operation. When these valves have been demanded to operate, they have stuck open on a number of occasions. Such malfunctions have led to significant plant transients and aggravated others in the past. Most notable are the events at TMI-2 in 1979 and Crystal River 3 in 1980. The failure of a PORV block valve to completely close aggravated recovery from a reactor coolant system leak at H. B. Robinson in 1981. Most recently (January 1982), a malfunction of the PORV aggravated recovery from a SGTR event at Ginna.[581] Some of the accident sequences and transients can be mitigated by using the PORVs and HPI pumps for pressure and coolant inventory control. This mode of operation is known as feed-and-bleed, or bleed-and-feed, depending on the HPI capability of the injection pumps and system design. In these situations, the PORVs could experience multiple openings and closures. Life cycle testing, which is not presently required, could provide better assurance that the PORVs can withstand this type of operation.

In other cases, when the PORVs have leakage problems and the PORVs are blocked, this could cause the safety valves to be challenged and, if they stick open, could result in an unisolable SBLOCA. Also, during startup and shutdown operations, many of the plants use the PORVs as part of the LTOP system. The LTOP systems that rely on lifting the PORVs to reduce reactor pressure, in accordance with established pressure/temperature limits, have experienced several events where the LTOP systems have been inoperable.[577,591,592] As a result of the inoperable LTOP systems, the potential for brittle fracture of the reactor pressure vessel is increased. Even though the LTOP system discussed above is a separate MPA issue, the common cause failures are: inoperable PORVs due to PORV leakage, maintenance errors disabling the LTOP systems, procedural deficiencies, and inadequate inspection or surveillance of the gas (nitrogen/air) supply that provides the PORV opening force in certain plants.

Possible Solution

Resolution of this issue could involve specification of the PORV/block-valve combination to some or all of the requirements associated with safety grade systems, better initial qualifications for the valves, and specified maintenance and testing requirements. The need for an automatic actuating circuit for the motor-operated block valve with manual override and reset capabilities was evaluated and determined unnecessary as a singular requirement.[572,573,574] Any program initiated to improve PORV/block-valve system reliability should take into consideration the Item II.K.3(2) conclusions and the basis and criteria for the conclusions. This analysis includes implementation of an automatic PORV isolation system to gage the potential benefits this modification could provide if combined with other reliability improvement measures. Broader and more specific ideas relating to possible resolutions are further discussed in the conclusions.

PRIORITY DETERMINATION

To establish the priority of this issue, the potential reductions in the SBLOCA frequency that could result from certain improvements in the PORV/Block valve system are quantified. The change in the SBLOCA frequency and resulting risk reduction is assumed to estimate the baseline benefits for this issue. The effects of transients, other systems (e.g., LTOP, MPA B-04), functions (e.g., feed-and-bleed, USI A-45), ATWS events, and safety grade PORV/block-valves on the overall PORV/block-valve reliability are additional unquantified benefits considered in arriving at the overall priority ranking.

Frequency Estimate

PORV Challenge Frequency: The PORV challenge frequencies are based on operating data[572,574] for W and CE plants for the post-TMI period from April 1, 1980 to March 31, 1983. These data included PORV challenges for 7 CE plants and 28 W plants. Examination of these data showed 18 of the plants (51%) with no PORV challenges (lifts). Since approximately 55% of the plants operate with closed block valves,[572,574,586] the 18 plants with no PORV challenges are assumed to be within the null set (55%) of plants that operate with closed block valves. PORV challenges at low power, as identified in the data, were also eliminated. Graph plots of the remaining 37 PORV lifts for the 17 plants in the active set indicated that approximately 50% of the plants account for 69% of the PORV lifts over the three-year period covered. Therefore, the PORV lift frequency for 50% of the plants in the active set is 1.0/RY. The data[573] related to B&W plants was not used because it was considerably lower than that shown for CE and W plants. Therefore, this analysis is applicable only to W and CE plants.

PORV/Block-Valve Failure Frequency: The PORV-SBLOCA (given that the block valves are open) requires actuation (challenge) of the PORV, failure of the PORV to close, and failure to close the block valves. The failure frequency of the PORV to close, given that it has opened, is 0.02/demand.[366] The failure frequency of the block valves to function is estimated at 0.005/demand.[346]

Potential reduction in the failure probability of the PORV to close, given that it has opened, is highly judgmental. Since the "weak spot" in the PORV appears to be the control systems[576] and, for valves in general, malfunctions of valve operators and control equipment have occurred about twice as often as malfunctions of the valves proper,[590] a PORV failure probability of 0.01 is assumed obtainable. The improvement is assumed to result from improved maintenance, testing, surveillance, and proper matching of the valve operators with the valve body as a valve assembly. A similar reduction to 0.003 in the block valve failure to close probability is also assumed obtainable.

Operator Error Frequency: Failure of the operator to close the block valve increases the chance for a SBLOCA through the PORV flow path, given that the PORV sticks open. Based on operating experience,[76] the pre-TMI operator error rate was approximately 0.29. The 0.29 value is in agreement with the WASH 1400[16] (Table III, 6-1) estimate of 0.2 to 0.3 for plant operators under very high stress levels where dangerous activities are occurring rapidly. Since TMI, valve position indicators and more emphasis on operator training should reduce the chance of operator error. Based on the analyses performed,[572,574] we assume an operator error rate (HEP) of 0.05. This assumes an 83% improvement in operator performance (reliability) as a result of TMI improvements and the increased emphasis on operator training.

Automatic Actuation Block Valve Failure Frequency: Because the plant operator is "in fact" a control element that effects overall reliability for the PORV/block-valve system, this analysis considers automatic actuation of the block valves to replace the manual operator action. The failure rate for an automated block valve (MOV) is taken at 0.002/demand. The failure rate is based on the assumption of one failure in 486 tests;[366] actually, no failures were observed in the 486 tests. Therefore, the 0.002/demand failure rate for automatic actuation should be conservative.

PORV-SBLOCA Frequency: For the purpose of this analysis, the PORVs and safety valves were assumed to be normally closed and the block valves were assumed to be normally open.

The base-case PORV/block-SBLOCA frequency of 1.1×10^{-3}/RY with the operator controlling closure of the block valves is the product of the PORV challenge frequency (1.0), the probability that PORV sticks open (0.02), the probability that the operator will not close the block valve or that the block valve malfunctions by failing to close (0.05 + 0.005).

Improved valve maintenance, testing, surveillance, qualification and specification, and automated block valves are considered as combined improvements. They are estimated to reduce the PORV/Block-SBLOCA frequency to 5×10^{-5}/RY. The reduced SBLOCA is the product of the PORV challenge frequency (1.0), the probability that the PORV sticks open (0.01), and the probability that the automatic actuation circuitry of the block valve fails or that the block valve malfunctions by failing to close (0.002 + 0.003).

The potential 95% reduction in the PORV/Block-SBLOCA is therefore estimated to be 1.05×10^{-3}/RY. It seems likely that improvements in PORV/block-valve reliability could also reduce the PORV actuations that have resulted from low power lifts, spurious signals, maintenance, and testing that are not specifically reduced in the above analysis.

For comparison, the above generic PORV/SBLOCA frequency of 1.1×10^{-3}/RY is approximately one-third the PORV-SBLOCA frequency (3.1×10^{-3}/RY) calculations[572,574] for the CE and W plants. Also, it is estimated [588,589] that a PORV-SBLOCA has a frequency of 5.1×10^{-3}/RY. Based on the above, the PORV-SBLOCA might range from 1×10^{-3} to 5×10^{-3}/RY. The above values are all within the uncertainties inherent in this generic analysis.

Transient Frequencies: The PORV challenges (lifts) in the data base resulted from plant operating experiences that included transients, spurious actuations, and one SGTR event. Therefore, high pressure transients that challenge the PORVs most frequently, and the potential for a SBLOCA from such events, are included in the SBLOCA section of this analysis. Less frequent high pressure transients that might occur, and that are not inclusive in the 3-year data base, are expected to provide secondary effects that are in addition to the SBLOCA base-line effects. These secondary effects should not significantly alter the results, but are factors considered in the conclusion of this analysis.

LTOP Events: LTOP events and low power actuations of the PORVs were eliminated from the calculated PORV lift frequency of 1.0/RY used to estimate the SBLOCA frequency. The four such events that were evident from the 3-year data base, could be attributed in part to inadequate PORV/block-valve system reliability programs. LTOP events that result from failure to place the block valves in the open position to allow pressure relief through the PORVs, as stated earlier, is a separate issue (MPA B-04). However, resolution of the LTOP concerns (event frequencies) would likely be a subset of the overall measures to improve PORV/block-valve reliability addressed in this issue. In recognition of possible double counting of the benefits attributed to resolution of the LTOP issue, this issue will not be credited for the potential benefits that may be more appropriately obtained within the more definitive resolution of the LTOP issue. Nevertheless, coordination of the resolutions of the various operations for which the PORV/block-valve systems are used is considered an important element that is factored into the prioritization of this issue.

Feed-and-Bleed: Feed-and-bleed is being evaluated under USI A-45. Any need to require improved PORV/block-valve reliability related to specific needs in the feed-and-bleed mode of operation should be developed as part of the USI A-45 resolution. Resolution of Issue 70 should also be coordinated with USI A-45.

ATWS Frequency: Reliability improvement benefits to the existing PORV/block-valve systems with respect to ATWS events are evaluated based on three types of pressure relieving scenarios: (1) success, (2) partial success, and (3) failure. Success is the probability that the block valve, PORV, and SVs open (function) to relieve pressure build-up in the reactor. Success, however, does not indicate that sufficient relief capacity is available through the existing valves. Partial success is that either the PORV or SVs open to relieve pressure build-up in the reactor. As in the success case, partial success does not mean that the existing valves have sufficient relief capacity to mitigate high reactor pressures that might result in exceeding a 3200 psi stress level. Failure is the probability that no relief is obtained through either the PORV/block-valve system or the SVs.

ATWS Success: The probability of ATWS success (Case 1) prior to PORV/block-valve reliability improvements is 0.441A. This probability is the product of the ATWS frequency ($A \sim 10^{-5}$/RY), the probability that the plant is operating with the block valves open (0.45), the probability that the PORVs open (0.99), and the probability that the SVs open (0.99). Assuming that PORV reliability can be improved by 50%, the probability of opening the PORVs is increased to (0.995). The resulting increase in an ATWS success (as defined above) is 2×10^{-3}(A)/RY. If we assume that the ATWS event leads to core-melt, the potential reduction in core-melt frequency is 2×10^{-8}/RY.

Partial ATWS Success: Partial ATWS success (Case 2) involves the sum of three event sequences, given that an ATWS occurs ($A \sim 10^{-5}$/RY):

(1) Case 2a

Block Valves are open (0.45)
PORVs open (0.99)
SVs fail to open (0.01)

The probability for Case 2a prior to PORV/block-valve reliability improvements is (4.45×10^{-3})(A). Assuming a 50% improvement in PORV reliability (PORV opens = 0.995), the resulting increase in partial ATWS success for Case 2a is 3×10^{-5}(A)/RY. Assuming the ATWS leads to core-melt, the potential reduction in core-melt frequency for Case 2a is 3×10^{-10}/RY.

(2) Case 2b

Block Valves are open (0.45)
PORVs fail to open (0.01)
SVs open (0.99)

The probability for Case 2b prior to PORV/block-valve reliability improvements is 4.46×10^{-3}(A). Assuming a 50% improvement in PORV reliability (PORV fails to open = 0.005), the resulting increase in partial ATWS success for Case 2b is 2.23×10^{-3}(A). Assuming the ATWS leads to core-melt, the potential reduction in core-melt frequency for Case 2b is 2.2×10^{-6}/RY.

(3) Case 2c

Block Valves are closed (0.55)
SVs open (.99)

For Case 2c, plants operating with closed block valves, it is assumed that the operator does not or cannot open the block valves in time to relieve the primary system pressure. In this case, reliability improvements to the PORV/block-valve system will not affect the partial success resulting from the opening of the SVs.

Failure From ATWS: Failure to relieve system pressure from the PORVs and SVs following an ATWS event can occur whether the plants operate with the block valves open (Case 3a) or with the block valves closed (Case 3b).

(1) Case 3a

Block valves open (0.45)
PORVs fail to open (0.01)
SVs fail to open (0.01)

The probability for Case 3a prior to PORV/block-valve reliability improvements is 4.5×10^{-4}(A). Assuming a 50% improvement in PORV reliability (PORV fails to open 0.005), the resulting decrease in failure for Case 3a is 2.3×10^{-4}(A). Assuming the ATWS leads to core-melt, the potential reduction in core-melt frequency for Case 3a is 2.3×10^{-9}/RY.

(2) Case 3b

Block valves closed (0.55)
SVs fail to open (0.01)

For Case 3b, with the plants operating with closed block valves, reliability improvements to the PORV/block-valve system have no affect.

Combined ATWS Frequency: The combined reduction in core-melt frequency for the above ATWS cases is approximately 4.4×10^{-8}/RY. Based on this estimate, improvements that increase the reliability of the PORV/block-valve systems by as much as 50% will not provide significant benefits toward reducing core-melt or consequences that would result from an ATWS event.

It is restated, however, that this estimate addresses only the effects of improved PORV/block-valve reliability to provide some relief in system pressure following an ATWS. The analysis does not address the adequacy of existing valves to provide sufficient relief capacity to mitigate an ATWS. If the ATWS resolution requires modifications to the PORV/block-valve systems, Issue 70 should be coordinated with the ATWS resolution.

Safety Grade PORV/Block-Valves: A design basis event (SGTR) occurred at the Ginna plant within the 3-year period that was used to determine the PORV lift frequency (1.0/RY). Other than the stated RSB position that the PORV/block-valve system should be safety grade,[575] no evidence has been revealed that indicates that safety grade components are more reliable than control grade components (PORV/block-valve systems). In the absence of such evidence, no benefit (risk reduction) could be quantified at the present time to this proposed method to improve PORV/block-valve system reliability. Any subsequent staff information developed in the resolution of Issue 70 should be coordinated with, and factored into, the need for safety grade PORV/block-valve systems or individual components of the PORV/block-valve systems.

Core-Melt Frequency: The WASH-1400[16] median core-melt frequency (1.5×10^{-5}/RY) for a SBLOCA ($S_2=10^{-3}$/RY) is dominated by failures in the emergency core cooling injection system (S_2D), and the emergency core cooling recirculation system (S_2H), resulting in a Category 7 type release (Basemat melt-through). A more representative (although conservatively biased) value of the probability of failure of the HPI(D) systems was considered[587] to be 3×10^{-3}/demand, as opposed to the WASH-1400[16] value of 9×10^{-3}/demand. Therefore, the WASH-1400[16] S_2D sequence frequency of 9×10^{-6}/RY is replaced by the value of 3×10^{-6}/RY. In addition, RRAB proposed that, for the WASH-1400[14] S_2H sequence frequency of 6×10^{-6}/RY also involving a Category 7 type release, the operator may not need to go to recirculation, or the time available for corrective action (close the block valve) will most likely be hours instead of minutes. Therefore, the HEP=0.05 in the PORV-SBLOCA (discussed above) should be reduced to HEP~0.005 for the S_2H (WASH-1400) sequence. The adjusted (reduced) S_2H sequence frequency is therefore 6×10^{-7}/RY. The result is a core-melt frequency of 4×10^{-6}/RY that is dominated by the Category 7 type release. Ratioing the potential reduction in PORV-SBLOCA frequency (1.05×10^{-3}/RY) to the WASH-1400[16] SBLOCA frequency ($S_2 = 10^{-3}$/RY) yields an estimated potential reduction in core-melt frequency of $(1.05)(4 \times 10^{-6}/RY) = 4.2 \times 10^{-6}/RY$.

Consequence Estimate

The consequences (risk reductions) attributed to improvements in the PORV/block-valve reliabilities are shown below in Table 3.70-1. Column 1 lists the WASH-1400[16] dominant release categories. Column 2 lists the modified WASH-1400[16] S_2 (small break) core-melt frequency for the specific release category. The modifications delete the S_2C sequences and containment failure modes from steam explosions, in accordance with RRAB recommendations. The RRAB position that an S_2 (SBLOCA) with a failure of the HPI(D) system may not lead to core-melt, if aggressive cooldown via the steam generators (secondary-side) is used, could possibly be considered on a plant-specific basis. RSB is currently evaluating aggressive cooldown as a means to mitigate very small LOCA scenarios and to improve methods for decay heat removal under USI A-45. However, until these methods and capabilities are fully established, recovery credit for aggressive cooldown (elimination of the S_2D core-melt sequence) is not considered generically representative.

Table 3.70-1

1	2*	3	4
RELEASE CATEGORY	MODIFIED WASH-1400 (S_2~10^{-3}/RY) CORE-MELT FREQUENCY (RY^{-1})	DOSE (man-rem)	RISK (2)(3) (man-rem/RY)
2	2×10^{-9}	[illegible] $\times 10^6$	~
3	2×10^{-7}	5.4×10^6	1.1
4	~	2.7×10^6	~
5	1×10^{-8}	1.0×10^6	~
6	1×10^{-8}	1.4×10^5	~
7	4×10^{-6}	2.3×10^3	~

*Not including the 10% from adjacent catetories used to smooth the data.

Ratioing a reduction in the PORV-SBLOCA of 1.05 x 10^{-3}/RY to the WASH-1400[16] SBLOCA ($S_2=10^{-3}$) and remaining plant life of 27 years, the potential public risk reduction resulting from improving the reliability of the PORV/block-valve systems is (1.05)(1.1)(27) man-rem/RY = 31 man-rem/RY.

The analysis provided herein provides the baseline potential risk reductions. Consideration of LTOP, feed-and-bleed operations, ATWS events, and safety grade PORV/block-valve systems are not expected to significantly alter the potential risk reduction of this issue, but these unquantified secondary effects are additional qualitative factors considered in the prioritization (see "Other Considerations" below).

Cost Estimate

Plant Implementation Cost (Operating PWRS): PNL estimated[64] that valve backfit labor costs are $27,200/plant based on 12 man-wk/plant and $2,270/man-wk. This includes management review, QA control, licensing review, and engineering for the backfit. Material requirements are two safety grade PORVs and two instrumented (for automatic actuation) block valves, each costing $25,000. Incremental material costs such as piping, supports, hardware, etc., beyond those associated with initial installation of the safety grade PORVs and instrumented block valves at a plant are estimated at $50,000. The cost for the safety analysis is estimated at $50,000/plant. A Class III License Amendment for the valve upgrade is placed at $4,000. The implementation cost is therefore estimated to be $237,200/plant.

Plant Maintenance Cost (All PWRs): Additional annual maintenance and testing is estimated at (0.5 man-wk/RY) ($2,270/man-wk) = $1,140/RY. The present worth of this cost in constant dollars with a 4% discount rate[585] over 27 years is $17,860/plant.

Plant Implementation Cost (Planned PWRs): For new plants, an incremental effort above the analysis required for relief valves is estimated at $5,000/plant. Assuming that new valves at $25,000/each will be required, the (forward fit) implementation costs are $105,000/plant.

NRC Cost (Operating PWRs): NRC costs will most likely involve plant-specific reviews, generic studies necessary to establish reliability and performance goals for the PORV and block valves, and preparation of a Regulatory Guide. The generic studies and preparation of Regulatory Guide is estimated to require 3 man-years of effort ($300,000). The cost is assumed distributed over 47 operating plants and 48 planned plants for an NRC cost of $3,100/plant. The plant-specific reviews (which include review, SER preparation, and technical specification changes) are estimated to require 1.5 man-months ($12,500). Thus, the total NRC cost for plants affected by a backfit is estimated to be $15,600/plant.

NRC Cost (Planned PWRs): The NRC reviews are assumed to be part of the normal licensing process. However, as stated above, the NRC costs associated with the generic studies and development of a Regulatory Guide ($3,100) are distributed over both operating and planned reactors.

Value/Impact Assessment

Based on a public risk reduction of 31 man-rem/plant and a cost of \$0.27M/plant for operating plants only, the value/impact score is given by:

$$S = \frac{31 \text{ man-rem/plant}}{\$0.27\text{M/plant}}$$

$$\cong 115 \text{ man-rem/\$M}$$

Other Considerations

(1) Occupational Risk Change

(a) Implementation ORE: PNL estimated that replacement of the existing PORV/block-valve system with a safety grade (or equivalent) system would require 96 man-hours/plant. The radiation field in the region of the pressurizer is estimated at 0.2 R/hr (EPRI-NP-1139, page 3-26).[584] The implementation dose is therefore estimated to be 19.2 man-rem/plant.

(b) Maintenance ORE: The maintenance ORE for additional annual testing was estimated by PNL at 4 man-hrs/plant in the 0.2 R/hr field. Over a 27 year period, this results in an ORE of 21.6 man-rem/plant.

Based on information in EPRI-NP-1138,[431] the PORV/block-valve maintenance in two PWRs over six years required approximately 50 man-hrs/RY. In a 0.2 R/hr radiation field, this amounts to approximately 10 man-rem/RY. The EPRI report (page 4-103)[431] also concluded that plants with the least maintenance were those that contracted outside specialty vendors and/or manufacturers to perform maintenance and adjustments. This indicates a general need for additional training in maintenance procedures for those plants that perform their own maintenance. The extent of improvement (maintenance reduction) that can be attributed to improved maintenance procedures is difficult to judge. However, if we assume only a 10% improvement, the potential ORE reduction is 1 man-rem. Over a remaining plant life of 27 years, this amounts to 27 man-rem/RY. Therefore, it is estimated that the ORE resulting from the additional annual testing described above can be offset by the ORE reduction resulting from improved maintenance methods.

(c) Outage Avoidance ORE: An estimate of the potential ORE reduction that can be attributed to improved reliability of the PORV/block-valve system and from potential outage avoidance cannot be quantified due to incompleteness in the information reviewed. However, an example of such an event involved a rupture of the rupture disc in a pressurizer relief tank that resulted from improper seating of a PORV (EPRI-1139).[584] The tank repair time for this event required approximately 16 hours, of which 3 hours were classified as critical path time (lost power production). The radiation field and ORE were not given. Such events are not believed to be frequent, but they have occurred at other plants. Improved PORV/block-valve reliability may reduce similar occurrences and thereby reduce the ORE resulting from such repairs.

(d) Accident Avoidance ORE: The reduction in core-melt frequency of 4.2×10^{-6}/RY results in avoidance of ORE associated with core-melt cleanup operations (20,000 man-rem/core-melt).[64] The accident avoidance dose over a remaining plant life of 27 years is $[(27)(4.2 \times 10^{-6})(2 \times 10^{4})]$ = 2 man-rem/plant.

The combined implementation and maintenance ORE expected to result from upgrading the PORV/block-valve system is 20 man-rem/plant. However, this ORE may be offset by less frequent outages and repairs that benefit from improved PORV/block-valve system reliability. For purposes of comparison, the median annual collective dose (ORE) for PWRs appears to have leveled off at about 400 man-rem/RY.[582]

(2) Outage Avoidance Cost: Most of the repairs made to PWR RCS relief valves do not require reactor shutdown. However, as reported in EPRI-NP-2092,[114] an average of 115 Effective Full Power Hours (EFPH) per outage with an event frequency of 0.11 outage-event/RY is mainly attributed to the PORVs and block valves. We assume that improved reliability of the PORV/block-valve system can also reduce the outage frequency. If we further assume that the outage frequency reduction is proportion to only 30% of the potential reduction in the PORV/block-valve SBLOCA frequency [(0.3)(1.05/1.1) = 0.29], the reduction in outage frequency is:

$$(0.11 \text{ Event/RY}) \left[\frac{115 \text{ EFPH}}{\text{Event}}\right] \left[\frac{\text{EFPD}}{24 \text{ EFPH}}\right] (0.29) = 0.15 \text{ EFPD/RY}$$

where:

EFPD = Effective Full Power Day

Based on a replacement power cost of \$0.3M/day,[64] the potential reduction in outage frequency results in a replacement power cost savings of \$45,000/RY. Assuming a 4% discount rate[585] over 27 years yields a present worth cost savings of \$0.76M/reactor. If higher replacement power cost is used, the cost savings would be proportionately greater.

(3) Accident Avoidance Cost (On-site)

The present worth cost of a core-melt accident is estimated[64] at \$1.65 billion considering cleanup and replacement power cost over a ten-year period. The present worth of accident avoidance at each plant is $[(4.2 \times 10^{-6}/\text{RY})(\$1{,}650\text{M})(27 \text{ RY})]$ = \$0.2M.

(4) As a result of some of the TMI Action Plan items, various requirements and options have been installed in operating reactors. The options (changes) were directed primarily at reducing the PORV/block-valve challenges. Which plants have what mitigating features is beyond the scope of this evaluation. However, to adequately assess which method, or methods, would provide optimum improvements to the reliability of the PORV/block-valve systems, the current plant-specific status of related TMI Action Plan items should be determined. As previously discussed in this report, there are a number

of related issues with which the resolution of this issue must be coordinated and whose open status affect the safety mission(s) for the PORV/block-valve assembly and thus affect the appropriate reliability/qualification objectives.

Item II.D.1 is still ongoing with the outcome uncertain. A feed-and-bleed mission with or without seismic qualification has not been determined (USI A-45). The ATWS rule is not yet complete. The severe accident research program may conclude that it is desirable to depressurize the reactor vessel during pressurized core coolant boil-off (station blackout, etc.) in order to enable the accumulators to dump and thus buy time to restore HPI, or to avoid the effects of pressurized vessel melt-through such as direct heating, missiles, pressure spike, subsequent accumulator dump, etc.

Also, the need for safety grade PORV/block-valves beyond the safety features that some of the plants already have should be determined. If replacement valves that meet all the requirements of a safety grade PORV/block-valve system are not needed, the value/impact (benefit/cost) ratio for this issue could be significantly improved with a lesser pedigree of safety grade PORV/block-valve systems.

CONCLUSION

The PORV/block-valve systems were in general believed to be primarily for operational flexibility in pressure control and not required to safely shut down the reactor. However, these valves are sometimes used to mitigate certain design-basis accidents (e.g., SGTR), transients, LTOP events, reduce safety valve challenges, and potentially to help mitigate the affects of an ATWS. SBLOCAs through this system and resulting challenges to safety systems appear to be of sufficient frequency that, based on the evaluation provided herein, improved reliability of the PORV/block-valve system might yield a potential public risk reduction of 31 man-rem/reactor at a cost of \$0.27M/reactor. The resultant value/impact score of 115 man-rem/\$M and the potential reduction in core-melt frequency of approximately 4×10^{-6}/RY indicates that a medium priority ranking is appropriate for this issue. Replacement of the existing PORV/block-valves that meet all the requirements of a safety grade PORV/block-valve system may not be needed and reliability could be improved with less costly modifications. Therefore, subject to the above considerations, the value/impact score for this issue could be higher, but could not exceed the medium ranking for prioritization purposes.

The outage avoidance cost savings of \$0.76M/reactor and the accident avoidance cost savings of \$0.2M/plant are not included in the value/impact assessment. Potential cost savings through outage and accident avoidance is estimated at approximately \$1M/reactor. These potential cost savings are nearly four times greater than the estimated implementation costs and could provide an additional industry incentive for resolution of this issue. Likewise, potential increases in the ORE resulting from expanded and improved maintenance testing and surveillance procedures could be offset by ORE reductions brought about by improved PORV/block-valve system reliability.

Conversely, development of aggressive cooldown capabilities through USI A-45 could reduce the potential for core-melt from a PORV-SBLOCA. This capability could reduce the value/impact ratio of this issue.

For plants that have not yet commenced operation, the value/impact ratio could be considerably greater. This is because forward-fit costs would be less than the backfit cost. The magnitude of the value/impact ratio will be dependent on the current licensing/construction status and the extent to which upgraded PORV/block-valve systems already exist or are already planned in the plant designs. The CE plants that do not have PORVs are outside the scope of this issue and are addressed separately in Issue 84 (CE PORVs).

Because of the large uncertainty inherent in a limited assessment such as this one, issues would be assigned a medium priority ranking predicated on a base-line risk reduction and other qualitative considerations even though the estimated base-line risk reduction is not significant. Further and more careful analysis may show greater potential risk reduction, although this would not be expected to occur for all, or even most, medium priority issues. In this case, analysis previously performed under TMI Action Plan Item II.K.3(2) concluded that automating the closing of the block valve would not reduce SBLOCA frequency significantly.[572,573,574] Thus, the II.K.3(2) analysis should be considered as completed since the results of this analysis are comparable in that both estimate similar values of SBLOCA frequency. However, this issue identifies the possible need for both a broader and more specific resolution.

Resolution might entail imposition of some or all of the attributes of safety grade qualification such as: (a) redundancy for selected design basis challenges, (b) N-stamp, (c) seismic Category I qualification, (d) environmental qualification, (e) technical specifications on operability and/or the normal alignment of the block valves, and (f) QA pedigree. Resolution might also entail deterministic or probabilistic reliability qualification for one or a variety of missions for the PORV and block valve, beyond that comtemplated in NUREG-0737,[98] Item II.D.1, e.g., feed-and-bleed. Resolution might entail particular component reliability monitoring, surveillance, and follow-up in service with corrective action for instances of below-average reliability performance. Resolutions might entail systems analysis to identify common causation of PORV or block valve failures under circumstances in which their operability is important, perhaps leading to altered power supplies for valve actuations. Resolution might entail special qualification or analysis for water hammer in PORV or code safety valve discharge lines, enlarged flow capacity, or the replacement of relief valves with fast acting control valves.

Thus, this issue is proposed as a MEDIUM priority issue to coordinate efforts to look at the details of the PORV/block-valve situation (e.g., outcome of Item II.D.1, look at data, coordinate new information on, or incentives for, a broader safety mission for PORVs from each of several on-going programs), and assess, on a schedule tied to related programs, the adequacy of our existing PORV/block-valve requirements.

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14. ABSTRACT (200 words or less)

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

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