Volume One

Regulatory Review Group

Summary and Overview

U.S. Nuclear Regulatory Commission
Office of the Executive Director For Operations

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1.1 SUMMARY

On January 4, 1993, the Executive Director for Operations established a Regulatory Review Group. The Review Group has conducted a review of power reactor regulations and related processes, programs, and practices with special attention placed on the feasibility of substituting performance-based requirements and guidance for prescriptive requirements and guidance. This examination of processes included industry's role and their potential roles. The findings and recommendations of the Review Group focused on identifying specific problems, their causes, and achievable solutions. A draft report containing the recommendations of the Review Group and a summary of the basis for them was made available for a 60-day comment period on May 28, 1993. Public, staff, and ACRS comments have been considered in this final report.

A review was conducted of the regulations affecting operating reactors, four selected licenses, supporting guidance in selected areas, public comments from the Marginal-to-Safety Program and the 1992 CRGR review, and recent industry requests. Based on this review, specific issues were identified and pursued both as to regulatory substance and process. Several public meetings were held to discuss and receive comments on material placed in the public document room. The Review Group focused on the analysis of information and the development of recommendations rather than the collection of material that would duplicate information previously accumulated.

In their comments, the staff and industry fundamentally agreed with the problems identified. The Office of Nuclear Regulatory Research, ACRS, and utilities agreed with the major rulemaking recommendation--to publish for comment a revision of 10 CFR 50.54. This rule applies to the control of changes to quality assurance, security, emergency response, guard training, security contingency, fire protection plans, and commitments. The Review Group raised additional issues that were narrower in scope and provided recommendations requiring rulemaking and changes in the processing of license amendments that are focused on relief of burden. Of primary importance in each of these areas is the question of responsibility for the development of technical justifications as it pertains to the role of the NRC and the nuclear industry.

The review of risk assessment methods resulted in the development of a graded approach to the incorporation of risk analysis into the regulatory process. The full implementation of the approach recommended in the report needs focused attention on the part of the staff and industry. NUMARC, representing the industry, has advised the NRC that it has established a working group for this purpose.

A review of the NRR Inspection Program Assessment and how it treated the question of the relationship of overall inspection effort to industry safety performance was conducted. It appears that inspection program changes could result in NRC resource savings and a reduction in industry burden.

1.2 REPORT ORGANIZATION

The report is organized into five parts. Volume One contains a summary, findings and recommendations, an analysis of comments, inspection program review results, and a summary of public comments. The review of the inspection process was based on the NRR assessment report submitted to the Commission in August 1993. The resources applied by NRR to the Inspection Program Assessment in parallel with the Review Group effort were not duplicated by the Review Group. Due to timing, the inspection program review was not published for public comment.

Volume Two of the report contains the details of our analysis of regulations, guidance documents, comments, and the rulemaking process. The review was limited to 10 CFR 21, 10 CFR 26, 10 CFR 50, 10 CFR 73, and Division 1 regulatory guides as they relate to operating reactors. Specific subjects were picked for development based on both our review and past comments received by the Commission. Each subject was treated separately with a summary of the analysis, discussion of comments, and recommendations. The main purpose of each analysis was problem definition. This made the comments on these issues most important as they related to the scope and depth of the problems. Detailed data sheets are included to ensure that the bases for our recommendations are traceable.

Volume Three of the report covers the details of our analysis of four representative licenses. This section provides recommendations where immediate action could yield tangible results. The analysis involved an item-by-item evaluation of each license selected, with a specific sample of items being selected as representative of a class of items that can be dealt with in a similar manner. A major factor in the consideration of which items appeared to be candidates for reduction in regulatory burden or enhanced flexibility was the proposed disposition of the item in the Technical Specification Improvement Program.

Volume Four of the report deals with the integration of risk analysis techniques into the regulatory process. The emphasis in the report is on trying to define specific applications and then to generalize these applications to categories with similar characteristics. The approach to quality assurance and Technical Specifications was developed in some detail. The intent is to provide a structure for using risk techniques as an integral part of safety regulation. The importance of timely evaluation of this volume's recommendations is the coincident start of a major EPRI effort and the establishment of a NUMARC working group to address the possible expanded use of risk analysis in support of regulatory requirements.

Volume Five contains the public comments on the report. Significant comments were received that could have a bearing on the implementation of the recommendations. The comments in several cases exceeded the level of detail contained the draft report. All of this material has been included.

1.3 FINDINGS AND RECOMMENDATIONS

The findings of the Review Group are discussed in order of importance based on the ability of the Commission to implement the changes in a timely manner and the potential impact on the efficient operation of power reactors. The Review Group also realized that the view of what is a performance-based requirement varies with the perspective of an individual or organization. This difference in perspective has resulted in general agreement between the staff, industry, and the Review Group on the problems to be addressed. Disagreement exists among the staff and the Review Group on the need to amend 10 CFR 50.54. A basic consensus exists between the staff, commenters, and the Review Group on the specific recommendations and potential improvements of Section 1.4 with the obstacle to implementation being identified by the staff and industry as a lack of NRC resources. Comments received from utilities question the will of the NRC to institute change.

The Review Group found that the technical substance of the body of regulations was either acceptable as a performance objective or was being addressed in an ongoing program. However, a deficiency was found to exist in the regulations governing the processes that control changes to programs adopted by licensees to implement the regulations. Anecdotal information received from the industry over the last five years repeatedly addresses the effects of licensees' commitments to the NRC in the areas of security, quality assurance, and fire protection. The Review Group found that the word "commitment" lacked both a definition and a defined change mechanism and that the plans listed have no fixed standard above which a licensee can make changes on its own volition. The lack of definition of commitment and the lack of a fixed standard for plans leave uncertain the degree of autonomy that licensees can exercise in carrying out their safety function. Additionally, the opportunity for informal backfits to occur as part of the review and inspection processes is enhanced by the lack of definition and standard. The contribution to regulatory burden from past commitments beyond what is required by the regulations is potentially large, making this the major rulemaking recommendation.

The Review Group believes that fundamental change can be achieved and maintained only through rulemaking which unambiguously sets the standards to be pursued by both the staff and the industry. If the performance objectives of the regulations are deficient they should be changed and be subject to the Backfit Rule. This recommendation is so dominant in effecting change that the Review Group has included as Appendix A a Commission paper and proposed rule. (See Sections 2.3.18, 2.3.9, and 2.3.5 of Volume Two; A.2.2.2 and B.2.2.2. of Volume Three.)

The next four findings deal with broad problems created by the way in which the Commission staff and industry interact within the existing regulatory framework.

1. The need for the staff to be responsive to licensee submittals that are safety neutral but have a primary aim of economic relief.

2. The need for each licensee to clearly identify the regulatory vehicle that is the cause of unnecessary expenditures and to then aggressively pursue corrective action fully utilizing the flexibility already available.

3. The need for industry to take a more proactive approach to the interaction with the staff on issues that require rule changes, and the need for the staff to establish clear ground rules for this interaction.

4. The need to establish a clear set of performance standards for the use of risk analysis within the regulatory structure.

The first two issues listed are closely linked. Coincident with the Review Group effort, several licensees have initiated similar tasks. These licensees include Virginia Power, Florida Power Corporation, and Entergy. In response to these initiatives, the Office of Nuclear Reactor Regulation (NRR) has established a temporary organization to develop an efficient management approach to address the anticipated increase in workload. The interaction between these licensees and NRR is continuing at this time. The importance of this initiative becomes evident in examining how a backlog of such actions developed. The effect of the NRR priority system promulgated in 1988 was a tempering of the number and type of licensing action requests and commitment changes that licensees submitted. As fewer actions were requested, resources were diverted to support the review of advanced reactors and plant life extension. The reversal of this process will have to address both receptiveness and resources. NRR has recently updated its priority instruction to start to address responsiveness. This area will need to receive senior management attention and support, especially during the formative stages.

Resource allocations continue to be guided by a set of priorities, which place the safety of operating reactors first, advanced reactors and plant life extension second, and the elimination of unnecessary burden at operating reactors last. Advanced reactors, plant life extension, and burden relief are all driven by economic considerations. Some parity, based on public benefit, should be considered in the allocation of resources between these economic topics. Specific resources are allocated for advanced reactors and plant life extension to support a preplanned level of effort. Items that will result in economic benefits for operating reactors generally have to compete for marginal resources. The past effect of the NRR priority system on the rate of submission of requests by licensees needs to be modified if unnecessary burden is to be reduced.

The industry initiatives mentioned earlier illustrate the need to identify the exact regulatory cause for actions resulting in undue burden. Only a minority of the items identified involved changes related to requirements contained in a license or regulation. Most actions involved "commitments" which require no formal regulatory action to change but for which licensees currently request NRC review prior to making the change. The lack of need for formal regulatory action should allow significant reductions in burden in the near term if the licensees who have initiated reviews are representative of the industry. The results of these licensee reviews also identify the need to define commitment and a change mechanism so that staff involvement is clearly spelled out for the industry and staff. For amendments to technical specifications, line-item improvements consistent with the Improved Standard Technical Specifications should be made available to individual licensees on a plant-specific basis. (See Volume Three Section B.2.2.4.)

The last two broad findings are longer term relative to yielding relief of burden, but the recommendations are reasonably easy to implement. The industry should be encouraged to take advantage of the opportunity provided by the petition-for-rulemaking process, 10 CFR 2.802. These petitions should be as complete as normally found in a staff-prepared notice of proposed rulemaking. NRC resources would focus, as a first priority, on petitions whose supporting analysis is found complete and compelling. These petitions could then be issued as a proposed rulemaking with a staff-prepared Federal Register notice stating the staff position and requesting public comments. The initial staff obligation in this approach would be to issue guidance addressing the level of detail expected in a petition that reduces burden and has no safety impact. Commission responsiveness to any resulting industry initiative should be included within the Marginal-to-Safety Program. This approach would leverage the staff resources in those areas thought by industry to be important enough to be worth the investment of preparing a petition. (See Section 2.3.17 of Volume Two.)

Regulations cannot be considered in isolation. The real impact of moving toward a higher degree of performance orientation in regulations will be realized only when the alternative practices and implementing programs are provided. The industry is in the most knowledgeable position to provide these alternatives. Quality assurance, as an area with significant potential for relief of burden with no safety impact, was found to fall into this category. By letter dated July 20, 1993, NUMARC informed the NRC that it had established a working group with this goal. (See Sections 2.3.1, and 2.3.13 of Volume Two.)

Volume Four of the report, Risk Technology, provides a starting point for near-term applications and industry-staff discussions. Since the establishment of the Review Group, NUMARC has established a working group to interface with the staff on the application of risk technology to the regulatory process. Because the interface deals with implementation, the interface and proponent for broader applications should be in the Office of Nuclear Reactor Regulation. NRR has assigned the Chief of the Probabilistic Safety Assessment Branch as the focal point for this interface with industry.

1.4 SPECIFIC RECOMMENDATIONS AND PROGRAM IMPROVEMENTS

1.4.1 Regulations

In its assessment of the regulations and implementing guidance, the Review Group identified a number of items that were inconsistent, needed improvement, or would be amenable to a more performance-based approach. The items are listed with their report sections identified. The items are classified as recommendations and potential improvements. Recommendations could have significant impact on the regulatory process, while potential improvements are narrower in scope. The items listed should result in a safety-neutral reduction of regulatory burden or an improvement in safety based on the elimination of unimportant issues that now divert NRC and utility attention.

1.4.1.1 Recommendations

- A definition of the term "current licensing basis" should be consistently incorporated into both 10 CFR Part 50 and 10 CFR Part 54. (See Section 2.3.10.)

- 10 CFR Part 21 should be revised to recognize the existing procurement practices and conditions and allow the level and type of dedication to be graded based on the safety significance of each part. (See Section 2.3.1.)

- Appendix B to 10 CFR Part 50 does not need near term revision. The implementing documents and guidance should be revised to facilitate implementation of Appendix B in a performance based and graded manner. This should facilitate licensees revising their quality assurance (QA) plans. Future revision may be necessary, if the literal reading of Appendix B is in conflict with a proposed alternative. (See Sections 2.3.13, 2.3.14, and 4.4.)

- Significant staff resources should not be devoted to a wholesale revision or updating of the regulatory guides. Revisions should be made only on a case-by-case basis. Eight guides dealing with quality assurance and ASME Code cases were identified as requiring periodic updates. (See Section 2.3.15.)

- Change staff practice to allow multiple actions to be proposed by a licensee in a license amendment request if the overall level of safety remains the same. (See Section 2.3.17.)

- Revise existing guidance to provide an approach to compensatory measures in security similar to that used for safety systems, such as allowed outage times. (See Sections 2.3.18 and A.2.2.4.)

- The staff should ensure that the system for Improved Technical Specifications will maintain appropriate control of changes to material that is removed from Technical Specifications and placed in licensee-controlled documents. (See Section 2.3.10.)

- The reporting requirements in 10 CFR 26.71(d) should be modified to allow submittal of fitness-for-duty performance data on an annual basis. (See Sections 2.3.5 and 2.3.16.)

- The fitness-for-duty program audit period should be allowed to be increased based on sustained satisfactory performance. (See Section 2.3.5.)

- The industry and staff should continue to build a consensus view in the ASME Code committees to revise the Code based on risk techniques. (See Section 2.3.7.)

- The Review Group does not recommend amending 10 CFR 50.54f. The procedures that require public comments on the technical and cost-benefit aspects of the generic correspondence adequately addressed previous concerns identified with generic communications. (See Section 2.3.6.)

- License amendments to delete reporting requirements of the type identified as "Not Required" in the Review Group's review of the Seabrook and Surry licenses should be acted upon. Reporting requirements relocated to licensee-controlled programs should also be reviewed. (See Section 2.3.16.)

- Generic actions should be taken to eliminate the seven periodic reports identified in Section 2.3.16, revise Regulatory Guide 1.16 dated 1975 as it relates to the content of required reports, and evaluate the remaining 29 reports listed. For example, eliminate the requirement for submittal of quarterly security logs. (See Sections 2.3.16 and 2.3.18.)

- Eliminate the policy statements that appear to have been interpreted as a requirement and impose a regulatory burden on licensees. The policy statements should be reviewed and evaluated to determine whether revisions are necessary or rulemaking should be undertaken. (See Section 2.3.11.)

- Revise NRC guidance documents addressing fire protection to emphasize that the use of alternative methods of compliance with the regulations is acceptable. (See Section 2.3.4.)

- Continue the effort to address industry concerns with the current fire protection regulations. (See Section 2.3.4.)

1.4.1.2 Potential Improvements

- Revise and reissue Regulatory Guide 1.86 to reflect the current NRC organization (not AEC) and address the areas of inconsistency with 10 CFR 50.82. (See Section 2.3.3.)

- Replace the guidance of Generic Letter 88-16, which states that the revised Technical Specifications reference the specific-approved topical report number and date used in the reload analysis, with a more generic flexible statement such that a license amendment would not be needed to reference the most current NRC-approved topical report used in performing the analysis. The approved topical report revision would still be required to be referenced in the core operating limits report. (See Section 2.3.8.)

- The staff should continue work on the Inservice Testing (IST) Program guidelines. These guidelines would provide a type of framework in which licensees may take advantage of using the most recent addenda and editions referenced in the regulations rather than what was originally committed to in the IST programs. The Review Group recommends issuing a generic letter informing licensees of the NUREG (which will offer generic approval of subsequent addenda and editions of the code per 50.55a(f)(4)(iv)) and the flexibility allowed if licensees take advantage of it. The Review Group also recommends that Inspection Procedure 73756 be reviewed for continued applicability in light of the issuance of the NUREG. (See Section 2.3.7.)

- Develop a generic communication system that differentiates between generic communications that request action and those that provide information. (See Section 2.3.6.)

- Add the following definition to 10 CFR 50.2, "Material alteration means any modification to a facility which changes the design bases of the facility." Modify the following four areas of the regulations (10 CFR 50.23, 50.45, 50.56, and 50.92(a)) to be consistent with the definition. (See Section 2.3.10.)

- 10 CFR 50.7 should be revised to the extent necessary to reflect statutory changes regarding the time frame for an employee who believes he or she has been discriminated against to file a complaint to the Department of Justice. (See Section 2.3.10.)

- The following policy statements have been superseded by rulemakings or are no longer applicable. These policy statements should be deleted. (See Section 2.3.11.)

  Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents
  Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel
  Commission Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel
  Nuclear Power Plant Access Authorization Program; Policy Statement
  Maintenance of Nuclear Power Plants; Revised Policy Statement
  Policy Statement on Information Flow

- The following policy statements address conduct of operations at nuclear power plants. The policy statements should be reviewed and evaluated to determine whether they should be combined or go into rulemaking or whether the portions of the policy statements superseded by rulemakings should be deleted. (See Section 2.3.11.)

  Nuclear Power Plant Staff Working Hours
  Commission Policy Statement on Engineering Expertise on Shift
  Policy Statement on the Conduct of Nuclear Power Plant Operations
  Education for Senior Reactor Operators and Shift Supervisors at Nuclear Power Plants; Policy Statement
  Power Plants; Policy Statement

- Provide a discussion of the Office of Nuclear Regulatory Research prioritization system in the preface of the Regulatory Agenda. In the Regulatory Agenda, identify under which category each rulemaking belongs. (See Section 2.3.17.)

- Schedules should be established for all rulemaking activities in the Regulatory Agenda. The schedules should include the date the action was originally approved by the EDO to improve tracking of the action. (See Section 2.3.17.)

- The abstract information for each rulemaking in the Regulatory Agenda should be kept current. (See Section 2.3.17.)

- Rulemakings and petitions for rulemaking whose schedules are significantly beyond the 2-year and 1-year resolution guidelines should be brought to resolution promptly. In those cases where the priority of the rulemaking is so low that staff resources are not anticipated to be available for several years, the rulemaking should be dropped from the agenda. (See Section 2.3.17.)

- Consider the issuance of an Information Notice to reenforce the importance of the behavioral observation element of FFD programs. (See Section 2.3.5.)

- Review the existing security requirements (particularly Appendix B to 10 CFR Part 73) to determine how they can be expressed in a more performance-based manner. (See Section 2.3.18.)

- While licensees provide sufficient space for residents, the rule (10 CFR 50.70(b)(2)) should be updated to address the current policy of two full-time inspectors at single unit sites. (See Section 2.3.10.)

1.4.2 Operating Licenses

The Review Group identified the following specific recommendations in its review of four operating licenses in addition to those discussed in Section 1.4.1.

1.4.2.1 Recommendations

- Eliminate the past practice of treating Commission policy statements, regulatory guides, and other non-requirements as legal requirements by generically including them in licenses. (See Section A.2.2.1 of Volume Three.)

- Develop NRC criteria on providing credit to licensees in the development of their technical specifications for design features that provide redundancy in excess of existing applications. (See Section A.2.2.1 of Volume Three.)

- Information/data requirements without a clear nexus to safety and duplicate reporting requirements should be eliminated. (See Section A.2.2.2 of Volume Three.)

- Adopt a graded approach to limiting conditions for operation and surveillance requirements wherever practicable, and to the implementation of specific review committee functions, e.g., station onsite review committee procedure and design change reviews. The appropriate application of risk assessment methodology could be valuable in establishing both the bounds and direction of such an approach. (See Section A.2.2.4 of Volume Three.)

- Expand the use of performance-based requirements to replace prescriptive criteria in license conditions and technical specifications. The functional requirement should be distinguishable from the technical details needed to implement that requirement. As evidenced in the Technical Specification Improvement Program, licensee-controlled programs that govern such implementation details can provide both flexibility and the requisite assurance of system functionality. (See Section C.2.2.4 of Volume Three.)

- Each licensee should conduct a comprehensive and thorough assessment of its own license to identify any items that have the potential for reducing regulatory burden or enhancing flexibility without decreasing the current level of safety. Each licensee should inform NRR of any license changes that it would likely pursue and the schedules on which it would pursue them. (See Section 3.3.4.)

1.4.3 Risk Technology

The current state of the art in probabilistic risk analysis (PRA) technology was examined to determine under what circumstances information, either qualitative or quantitative, gleaned from PRA methods could be used in the regulatory process. PRA methods provide an integral tool that can be used to help ensure coherence and consistency in the regulatory process, and provide a means of converting diverse deterministic requirements to performance-based requirements, providing equivalent protection to public health and safety while offering increased flexibility to licensees if the PRA-based criteria are met. To this end, the current state of the art in PRA methods was assessed considering how the strengths of these methods could be exploited, while minimizing the significance of weaknesses that still remain in the application of PRA-based methods in regulation.

1.4.3.1 Recommendations

Based on our review, we recommend that the use of PRA-based techniques in the regulatory process be characterized into three general classes, each having similar requirements in terms of the boundary conditions and assumptions used in the analysis, as well as similar requirements in terms of the depth and breath of the review that would be required by the NRC staff.

1. Reliance on Quantitative Results from Multiple Plant-Specific PRAs
   Examples of this category would include performance-based responses to the Maintenance Rule and PRA-based approaches to graded quality assurance. (See Sections 4.3 and 4.4.)

2. Reliance on Single Plant-Specific PRA Quantitative Results in Selected Areas
   Examples of this category would include optimization of selected technical specifications, evaluations of "unreviewed safety question" under 10 CFR 50.59, and use of precalculated configuration management analyses to support extension of allowed outage times under certain circumstances. (See Sections 4.3 and 4.5.)

3. Reliance on Numerical Results from Single Plant-Specific PRA
   An example of this type of usage would be the development of PRA-based technical specifications requiring on-line updating of PRA models. (See Sections 4.3 and 4.6.)

Beyond the technical recommendations, more specific recommendations regarding the nature of the regulatory environment needed to introduce the use of PRA-based analyses in a broad fashion are offered.

- The current state of development and use of probabilistic techniques in the industry can support use of PRA-based regulatory approaches at the present time. Several utilities have ongoing programs using PRA methods and "living" probabilistic analyses to improve operations and maintain plant safety and efficiency that could be extended to the regulatory environment and provide increased licensee flexibility while maintaining or improving the safety envelope. It is recommended that the Commission elicit licensee proposals in this regard to support such an effort. (See Sections 4.2 and 4.8.)

- The development by NRC of methods for optimizing technical specifications using PRA-based techniques is nearing completion and, with publication of a handbook early in CY 1994, will provide a technical basis for judging the acceptability of PRA-based approaches proposed by licensees. This handbook can provide guidelines for methods or similar techniques that could be used in a pilot program in the near future if there is industry interest in such an application. In addition, this handbook could serve as the point of departure for discussions between the NRC staff and the industry leading to industry-proposed guidance, suitably endorsed by NRC. It is recommended that this handbook be published as a regulatory guide. (See Section 4.9.3.)

- NRC programs and interests on the development and implementation of PRA-based methods in regulation currently span multiple offices and organizations. An integral agency plan covering the research, development, implementation, and use of PRA-based techniques in regulation is needed in maintaining a consistency of approach throughout the agency and in allocating resources. This plan would also assist in the efficient use of the limited number of NRC staff with expertise in quantitative risk assessment. (See Section 4.9.)

- A gradual approach is recommended for the transition to the more PRA-based approaches, testing benefits gained versus costs of implementing pilot programs before proceeding to complete implementation industry wide. As indicated above, certain PRA-based approaches can be implemented now, while others will be suitable for trial investigation in the near future. (See Section 4.9.)

- The NRC generally uses PRA insights to add requirements to the industry. This use of PRA needs to be extended to allow PRA-based insights to reduce regulatory burden when it is shown that such a reduction does not reduce the safety envelope of the plant. Thresholds (e.g., NRC guidelines) on content of submittals, acceptable PRA methods, and decision criteria must, therefore, be established by the NRC for each PRA usage class (as described above) in concert with any industry-proposed pilot applications of these potential uses. (See Section 4.3.)

1.5 INSPECTION PROGRAM ASSESSMENT REVIEW

In August 1993, the Office of Nuclear Reactor Regulation reported on its assessment of the reactor inspection program. The NRR assessment and the individual regional reports were reviewed by the Review Group. These reports and the Review Group members' experience served as the basis for these comments on the inspection program. The significant effort expended by NRR was generally expected to yield sufficient information concerning the program so that the most efficient application of resources for the Review Group was to review the results. The Review Group limited its comments on the assessment to two questions. Does it address how well safety oversight and resource utilization objectives are being achieved? Does it address the relationship between Commission inspection resources and industry performance?

The assessment fully addresses the first question. The feedback from the review and the communications between NRR and the regions should result in an improved program. While not contained in the overall NRR assessment report, the case studies conducted relative to early detection of falling licensee performance provided insights not previously available. The cumulative impact of the corrections to be made will be significant in allowing the inspection program to be more effective. The recommendations to set direct inspection goals based on plant performance should not be acted upon until a decision is made on the desired relationship between overall resources and licensee performance. If the current planning level of 2700 to 2800 direct inspection hours per unit is maintained, independent of industry performance, the recommendations of the assessment should be fully implemented.

The relationship between inspection resources and industry performance may provide insight into the root cause of resource and planning problems described in the Assessment Report and, therefore, the corrective steps needed. However subjective, a perspective should be developed on this relationship prior to the implementation of the recommendation to assign direct inspection hour goals based on plant performance. When combined with a historic perspective and available data, some quantitative insights can be developed. Facts from the report, Table 1, and approximations based on past events give some insights to the answer. The quantitative input to the process could be refined with the use of actual agency manpower data.

Table 1

Average Number of Direct Inspection Hours Per Operating Reactor

FISCAL YEAR	TOTAL HOURS OF DIRECT INSPECTION	PLANTS	DIRECT INSPECTION PER PLANT
1987	305,200	109	2,800
1988	314,400	109	2,884
1989	340,200	112	3,037
1990	328,200	113	2,904
1991	335,000	112	2,991
1992	300,860	108	2,786
1993	299,600	107	2,800
1994	302,400	108	2,800
1995	291,600	108	2,700

Table 1 shows that the level of direct inspection has been relatively stable. The rise in resources in 1989 was driven by the simultaneous effort expended at TVA facilities, Peach Bottom, Seabrook, Pilgrim, Comanche Peak One, South Texas, and others which was a combination of problem plant and issuances of operating licenses. The conclusion from those data are:

> THE AVERAGE LEVEL OF INSPECTION EFFORT HAS REMAINED RELATIVELY CONSTANT.

The workload that formed the initial basis for the established levels has changed with time. The resource intensive inspection effort associated with the issuance of licenses has been replaced by the ongoing oversight of operations. The broad measures of industry performance as maintained by the Office for Analysis and Evaluation of Operational Data (AEOD) and the industry organizations such as the Institute of Nuclear Power Operations (INPO) indicate improved industry performance. The NRC's own indirect indicators such as the number of generic communications, orders, confirmatory action letters, escalated enforcement cases, and identified problem plants have all decreased. Two conclusions can be drawn from this body of indicators:

THE MIX OF EFFORT HAS SHIFTED TO CHIEFLY INSPECTING OPERATIONAL SAFETY.

INDUSTRY PERFORMANCE HAS IMPROVED AS RELATED TO OPERATIONS SINCE 1987.

The assessment completed by NRR added significant new information which allows better quantification of the potential resource impacts of changes in both industry and NRC performance. The conclusions drawn from the assessment are:

POOR PERFORMING PLANTS ARE RECEIVING SUFFICIENT OVERSIGHT.

THE CURRENT PROGRAM AS IMPLEMENTED IS PROVIDING SUFFICIENT INFORMATION FOR THE IDENTIFICATION OF PROBLEM PLANTS BY THE SENIOR MANAGEMENT REVIEW PROCESS.

INSPECTION RESOURCES PER PLANT, GENERALLY, DO NOT REFLECT PLANT PERFORMANCE AS REPRESENTED BY SALP SCORES.

These last five conclusions summarize the major reasons a mismatch in workload and resources has developed. Detailed manpower data could be used to partially quantify this difference. Based on general observations and discussions, we would estimate this difference to be 10%. While not mentioned in the assessment, the regional administrators have had conflicting constraints placed on them that bear significantly on the application of this excess inspection resource.

THE LEVEL OF INSPECTION SHOULD BEAR A RELATIONSHIP TO THE PERFORMANCE OF EACH FACILITY AND SHOULD BE SUFFICIENT TO MAKE THE NECESSARY SAFETY JUDGMENTS.

THE DIRECT INSPECTION GOALS ARE BASED ON THE EFFICIENT USE OF EACH INDIVIDUAL INSPECTOR'S TIME.

One additional fact, which is not covered in the assessment but which could be quantified based on manpower records:

> THE LEVEL OF EFFORT REQUIRED TO BE EXPENDED ON INDIVIDUAL PROBLEM PLANTS HAS GONE DOWN DUE TO THE SUCCESSFUL EARLY IDENTIFICATION BY THE SENIOR MANAGEMENT PROCESS OF POTENTIAL PROBLEMS.

By examining the resources related to these statements some estimate can be achieved. The Review Group did not do a detailed analysis of manpower data but estimated values based on general program knowledge. The Review Group estimates that 50% of direct inspection is mandatory and 25% is reactive to information developed from the mandatory elements of the program or events. Approximately 25% of the total direct inspection effort is discretionary including the perceived 10% excess. Some discretionary resources are necessary to allow for on-the-job training, peaks in allegation followup, definition of potential generic safety questions, and mismatches in inspector capability versus current needs.

The excess effort can be categorized into two parts: (1) the portion due to the excess specialist inspectors in specific areas and (2) the portion due to the combined improved performance of the industry and the NRC. Each of these categories requires different approaches to mitigating the excess burden on licensees. Before making any significant changes in the management process involved in the inspection program, the resource mismatch should be addressed. Existing processes were promulgated on the assumption that a relationship should exist between the total level of direct inspection effort and industry performance. If excess resources exist and the direct inspection goals for individual inspectors are followed, then excess inspection must result. The program has placed the regional administrators in a position of having conflicting management objectives. The assessment finding that resources do not correlate to plant performance is not surprising.

1.5.1 Recommendations

The recommendations address direct inspection effort that impacts licensees through both the payment of fees and onsite support during inspections. The overall budget levels are not addressed directly but would be affected by changes in direct effort.

Reduce the total level of direct inspection effort 10% over two years to deal with the excess due to improved industry and NRC performance. This is a small enough increment that, compensating for problem plants returning to their historic high resource demands, could be achieved through scheduling shifts with no safety impact. Current direct inspection effort projections show a reduction in FY 1995. This reduction should be advanced to FY 1994. Permanent correction of the imbalance can be achieved through attrition. Intermediate relief could be accomplished by shifting reactor inspectors into programs with existing or potential backlogs such as licensing actions. This decentralized shift could take advantage of the current computer network for the transfer of documents between project engineers in the regions supporting project managers in headquarters. Regional staff, in small numbers, could be detailed to headquarters organizations without geographic displacement. If a reasonable gradation of inspection effort results, the amount of corresponding total direct inspection effort should be appropriate.

Regional administrators should identify specialist inspectors whose skills are not transferable within the reactor program. For these specialists, shift direct inspection goals to other programs, such as materials licensing and agreement state programs that may be conducive to employing decentralized resources. Regional administrators should not be penalized for missing direct inspection goals within the limitations of the reactor inspection program for these identified specialists.

1.6 ANALYSIS OF STAFF COMMENTS

Program offices, regional offices, and ACRS commented on the report. In general, the comments on the draft report were favorable and reflected agreement with the findings and recommendations or presented new or updated information that was incorporated into the report. The remaining differences are summarized in Table 2. As requested, the Office of Nuclear Reactor Regulation provided a general estimate of the level of effort associated with implementation of the recommendations as 30 to 60 FTE. Additional comments, which could not be classed as agreement or disagreement, were cautions on what needs to be considered if the Commission decides to move forward in particular areas.

These comments along with potential burden relief, impact on safety, and individual NRC office responsibilities were considered in the ordering of the Findings and Recommendations of this report. In considering these factors, the recommendations dealing with rulemaking were viewed as not competing for the resources necessary to carry out the recommendations concerning licensing activity. Based on existing organizational assignments, these efforts would be split between the offices of NRR and RES and therefore achievable in parallel. As a result, long-term rulemaking actions and short-term licensing actions are assigned equal priority by the Review Group as it concerns implementation. If the resources are considered as competing on an absolute scale, the immediate relief provided by licensing actions would become the highest priority recommendation. The level of effort required for responding in a timely manner to licensee requests will largely be dependent on the proactive posture taken by the utilities. A key indicator of this posture will be the internal backlog of licensing action and commitment reviews tracked by the Office of Nuclear Reactor Regulation.

Table 2

Staff Comments Which Disagree With Recommendations

Revise 50.54	NRR, AEOD, OGC, CRGR - Partial disagreem nt, should improve sta view and inspection guidance and management oversight, burden on inspectors v d increase. Definition of the term commitment should be pursued in a deliberate ner. NMSS - NRC should regulate against an inclusive but d or acceptance criteria and establish strong management control. OE - disagrees without a better understanding of the impacts, the proposed standard is not well defined and will be difficult to enforce, proposes a veto-type approach for plan and commitment changes, or include commitments in the FSAR. Region I - changing the reference to safeguards effectiveness may not be appropriate, changes should be evaluated to determine effect on entire plan.
PRA Development	NRR - disagrees, resource needs are incompatible with the present staffing workload levels and priorities.
AOT Approach for Security	NRR - agrees in principle - however, systems are less redundant and therefore need more immediate compensatory measures. NMSS - would create vulnerabilities that do not previously exist
Definition of CLB Clarification of Design Bases Evaluate Design Basis Policy Statement	NRR - disagrees with need NRR - scope and depth of design bases should be considered in ongoing study of design basis reconstitution. NRR - disagrees, does not burden licensees and design basis information is required
Delete Quarterly Submittal of Security Logs	Region V - logs provide meaningful indicator of weaknesses and assist inspectors, imposes a small impact on licensees and makes NRC inspections more efficient. NMSS - logs provide needed information for threat assessments
Audit Frequency for FAD	NRR - disagrees, would reduce viability of programs and could lead to increasing complaints and allegations. Region I - changes may be premature, audits are identifying problems that licensees are finding hard to correct, may be appropriate in the future.
Revise Guidance for Fire Protection	NRR - disagrees - would not reduce regulatory burden or impact on licensees
Definition of alteration	NRR - disagrees, not necessary, regulations rarely used.

1.7 ANALYSIS OF PUBLIC COMMENTS

The Review Group received 27 letters of public comment. The letters were supportive of the Review Group's effort and the recommendations. Some of the comments were incorporated into the report. Comments addressing areas that were not under the original scope of the Review Group's charter were not reviewed in detail, and disagreements are identified in the individual summaries of Appendix B to this volume. The letters themselves can be found in Volume Five of the report.

Many of the comments address concerns and details that need to be addressed as part of any follow-on effort to the Review Group recommendations. While the report was not changed to account for these comments, they supply important insights for recommendations that move forward. The comments which remain as disagreements are summarized in the Table 3.

Several items, while listed in the table, are not major conflicts. These include correcting the regulations governing space for resident inspectors, defining alteration, and updating 10 CFR 50.54 (f) on generic communications. These comments reveal a need to update our regulations to reflect current practices as resources permit.

A significant difference exists on the need to define commitment. The Review Group believes the spectrum of comments demonstrate the need for clarity. The safety importance of the need for a clear understanding of the applicable regulations will increase as the impact of commitments on utility expenditures is addressed. Based on both the staff and public comments, the need for an additional set of comments on a definitive proposal exists.

One utility disagreed with the Potential Improvement to issue an information notice based on a generalization of the findings from a security incident described in Section 2.3.5. This deals with behavioral observation programs being narrowly focused on substance abuse. The Review Group continues to feel this could be a precursor of problems at other utilities.

While two commenters disagreed with the recommendation not to upgrade regulatory guides, each commenter offered different solutions. The Review Group had considered each of these options in its work and rejected them and continues to feel that rejection was valid. First, withdrawing out-of-date guides would affect future use only since existing license references would remain. If outdated guides are being used inappropriately by the staff, this should be corrected. The Review Group did not develop the information necessary to confirm or deny this assertion. Second, updating all existing regulatory guides, would be beneficial. However, the industry could receive the benefit through the use of NRC-approved topical reports. In addition, the safety impact would be small, and the impact on NRC resources so large as to make this option unachievable.

The discussion on the use of design based generic reactor data in PRA classification schemes has been expanded. The actual and best approach to the problem will result from the interaction of the Commission staff and the NUMARC Threshold Working Group.

Table 3

Public Comments That Disagree With Review Group Recommendations

No Changes to 10 CFR 50.54(f)	NUMARC - disagrees, the regulation needs to be changed because it is still subject to misuse, advocate NRC management discipline. NUBARG - disagrees, threshold for issuance of an information request should be raised to cross-reference 50.109 and incorporate into staff procedures allowance for alternative actions and schedules, and a process to seek relief.
Update Limited Number of Regulatory Guides	NUMARC - disagrees, regulatory guides are used as informal requirements and should be kept up to date. GPU Nuclear - disagree, outdated regulatory guides should be voided rather than left in place.
Definition and Change Process for Commitments	TU Electric - disagrees, not needed, change process already codified. GPU nuclear - disagrees, unnecessary and adds burden, licensing basis commitments should be evaluated under 10 CFR 50.59.
Issue Generic Communication Regarding Training in Behavioral Observation for Aberrant Behavior	Entergy - disagrees, no basis for recommendation.
Inclusion of All SSCs Important in PRAs As A Way of Dealing with Variations in Analyses and Modeling	Commonwealth Edison - disagrees. Entergy - disagrees, not justified EPRI - difficult to implement, use as part of verification and validation. NUMARC - significant plant differences exist and there other means are available for adequately addressing this issue.
Revise 10 CFR 50.2, define Alteration	Yankee Atomic - disagree, proposal should be dropped .
Revise 10 CFR 50.7	Entergy - disagrees, do not include unless real situations exist.

APPENDIX A

PROPOSED AMENDMENTS TO 10 CFR 50.2 AND 50.54

Appendix A is a commission paper reflecting the Review Group recommendation to publish proposed amendments to 10 CFR 50.2 and 10 CFR 50.54 for public comment. While the changes are limited to the areas of quality assurance, security, emergency planning, fire protection, and plant specific commitments, the principles behind the changes have far broader application.

1. The regulations represent the safety standard to which licensees should be held accountable.

2. The licensee retains primary responsibility for compliance with the safety standard established by the regulations.

3. The amount of regulatory oversight should bear a relationship to the safety significance of the regulation.

Based on the comments received from all the program offices this represents a major change in commission practice by shifting from a process of pre-approval to one of post-implementation review. The proposed amendments to 10 CFR 50.54 would allow licensees to make changes to their required plans without NRC prior approval if the changes do not reduce the program content below the standards set in the regulations. Commitments could be changed without prior notification provided no degradation of safety or public protection occurs. The amendments represent a major advance toward performance based regulation and are analogous to the establishment of tier two material for advance reactors and the change process currently established for FSAR material. In the long run, this should lead to licensees pursuing the most economic alternative for implementing required programs in the most timely manner.

In their comments, the program offices fundamentally agreed with the problem being addressed, but stated that the necessary change would be better achieved through improved management oversight and development of staff guidance.

We continue to believe fundamental change can only be achieved and maintained through rulemaking, which then unambiguously sets the standards to be pursued by both the staff and the industry. Two significant staff comments have been accommodated in the amendments: (1) an extended implementation period of one year has been incorporated into the rule to allow for an orderly transition and (2) the reporting frequencies that currently exist have been retained to allow for timely staff action if necessary.

Utility comments received on the Review Group report clearly support the recommended action on the proposal with the exception of the need to define the term "commitment." The utility comments on this part of the proposal demonstrated that a lack of clarity exists on exactly what a commitment is and what the governing regulations are. Another round of dialogue, such as would be provided by the publication as a proposed rule, is warranted. Only one comment was received from a public interest group. They recognized the recommendations as a step toward increased self-regulation and objected to the Commission considering any such shift. Publishing the proposed rule would provide the opportunity for more public comments to be solicited and considered.

An additional benefit of the change should be increased attention to the safety impact of program changes on the part of the licensees and the staff. Licensees would now be subject to enforcement action for inappropriate changes to required plans and commitments rather than a staff comment. Similarly, the staff would carry the burden of proving noncompliance with the regulations if they disagree with implemented changes. The approach would allow the staff/licensee interaction to more sharply focus on real safety problems.

The Review Group considered two options for soliciting public comments: (1) publication as a proposed rule and (2) publication as an advance notice of proposed rulemaking. We believe the Review Group report provided an equivalent avenue for comments as would be provided by an advance notice, therefore, a proposed rule is warranted.

RULEMAKING ISSUE
(Affirmation)

________, 1993 SECY-93-____

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: INCREASED OPPORTUNITY FOR PUBLIC COMMENTS ON REGULATORY REVIEW GROUP RULEMAKING RECOMMENDATION ADDRESSING THE AMENDING OF 10 CFR 50.54 FOR SECURITY, QUALITY ASSURANCE, EMERGENCY PLANNING, FIRE PROTECTION, AND OTHER COMMITMENTS

PURPOSE

Obtain Commission approval of a notice of proposed rulemaking.

SUMMARY

The Regulatory Review Group was established to conduct a comprehensive and disciplined review of power reactor regulations and related processes, programs, and practices for their implementation. The Regulatory Review Group emphasized the feasibility of substitution of performance-based requirements and guidance for unnecessarily prescriptive requirements and guidance. Of the recommendations in the draft report issued May 28, 1993, the one with the greatest potential effect addresses the processes by which licensees can make and change commitments to the NRC. This recommendation was to revise 10 CFR 50.54, "Conditions of Licenses," as it applies to the control of changes to

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security plans, quality assurance plans, and emergency preparedness plans. The Regulatory Review Group also recommended that fire protection plans be included under the provisions of 10 CFR 50.54, that the term "commitment" be defined in 10 CFR 50.2, and that a change process for commitments be included in 10 CFR 50.54.

Recognizing the importance of this recommendation to moving to performance-based regulation, the staff has enclosed a proposed rule package to allow immediate action to be taken to implement the recommendation. The following discussion and enclosures address each area in more detail than the Regulatory Review Group report to provide a basis for soliciting comments on implementation. Action can be taken on an individual area or collectively.

BACKGROUND

In reviewing anecdotal information received from the industry over the last five years, the Review Group repeatedly found references to the effects of licensees' commitments to the NRC, in the areas of security, quality assurance, and fire protection. The Review Group found "commitment" lacked both an exact definition and a defined change mechanism and the plans listed have no fixed standard for changes to be judged against. The Review Group also determined that the lack of definition of commitment and the lack of a fixed standard for plans left uncertain the degree of autonomy the licensees can exercise in carrying out their safety function.

Each level in the NRC's hierarchical regulatory structure should include a change mechanism to allow the NRC staff to review the licensees' actions at a level consistent with the safety significance of the action. To a degree, the Commission has taken analogous approaches for both operating reactors and advanced reactors although using different nomenclature, i.e., 10 CFR 50.59 and tier two. The Review Group made recommendations for the definition and control of commitments to give to the licensees of operating reactors the stability intended by the tier-two designation in the standard plant process. The lack of clarity in allowing "commitment" to remain undefined and in allowing the change process to be based on floating criteria creates a situation in which requirements can be backfit informally. The staff is making this recommendation to the Commission to eliminate instability in the regulatory process for operating reactors using an approach analogous to that taken for advanced reactors.

The staff recommends defining "commitment" and the associated change process in a manner similar to the relationship between the Final Safety Analysis Report (FSAR) and 10 CFR 50.59. Thus, the NRC would capture "commitments" upon which the Commission has based safety findings not found in the FSAR or in submitted and approved plans. For specific plans required by the regulations and currently addressed in 10 CFR 50.54, the staff proposes establishing the regulations as the standard changes must meet rather than an undefined level of effectiveness. Promulgation of the amendments in the enclosure would bring greater discipline and coherence to the regulatory process, which should improve safety and reduce cost. The recommended rule changes would also enable the NRC to eliminate duplication between licenses and the regulations.

In the enclosure, the staff discusses security, quality assurance, emergency planning, fire protection, and commitments. Based on internal discussions and comments received by the Regulatory Review Group at public meetings, several topics on implementation are significant enough to be highlighted.

1. The regulations would establish the criteria that required plans must meet. Licensees could change the content of plans that exceed the requirements of the regulations without prior NRC approval. Plant-specific requirements resulting from a unique aspect of an individual licensee would have to be imposed as license conditions or specifically identified as a "commitment." This should promote a consistent application of regulatory requirements across the industry. This change would also increase the flexibility of licensees to optimize staffing in areas such as security as detailed in the Regulatory Review Group report.

2. The NRC staff would be notified "for information only" at the same reporting frequency that is currently required (e.g., security - 60 days, quality assurance - refueling cycle, emergency preparedness - 30 days) and require fire protection plan changes be submitted within 60 days of the change. This provides the staff the most current version of the plan and allows the staff to act in a timely manner if the staff determines a change reduces the plan below that needed to meet the regulations. Changes to commitments would be submitted "for information only" annually or on a refueling outage basis, consistent with 10 CFR 50.71(e)(4).

3. The NRC would not need to consider backfits when implementing the proposed revisions to 10 CFR 50.54 because implementation would be prospective in nature. However, problems could arise because "commitment" was not defined in the past. Licensee submittals and the associated staff safety evaluation reports may not have clearly identified the commitments.

4. The definition of "commitment" as proposed is derived from the Commission's Enforcement Policy. In accordance with this policy, responses to enforcement actions would be excluded because compliance with the regulations is required. How compliance is achieved would not be a commitment since continued operation of a facility is based on compliance with the requirement and not the method of achieving compliance. This same rationale would apply to generic communications promulgated to impose a compliance backfit. In other generic communications, the staff should clearly state whether the response will be considered a commitment, such that future changes to that commitment would be governed by the proposed change process in 10 CFR 50.54. This definition of commitment is consistent with the definition of commitment as an administrative mechanism used to supplement the NRC's enforcement program. This is discussed in Section H of Appendix C to 10 CFR Part 2.

5. The ability to measure performance against an established standard is a key element in a performance-based system. The staff has demonstrated the ability to evaluate performance against current

requirements. For example, the regional staff that conducts inspections are also responsible for conducting licensing evaluations of changes to the programs for quality assurance and security. The impact on the staff would be limited to the established standard being the regulations rather than a level of effectiveness which is plant specific and undefined.

6. The NRC could continue to revise the technical substance of the regulations referenced by 10 CFR 50.54 consistent with the Backfit Rule, 10 CFR 50.109, independent of the process changes recommended herein.

7. Implementation of these changes would be handled in the same manner as the recent maintenance rule and the Part 20 rule. Implementation would be effective one year from the date of publication of the final rule to allow for the development of guidance, where necessary, to support the transition.

RESOURCE CONSIDERATIONS

By implementing the changes discussed herein, the NRC should reduce the burden on licensees. The burden should be reduced by (1) eliminating delays for changes which now require NRC prior approval based on the use of the generally undefined term "effectiveness," (2) eliminating license amendments for changes to security and fire protection plans, and (3) consistently applying throughout the industry the principle that the regulations establish the performance criteria. The estimated reduction in burden will vary according to the practices at each plant.

The changes recommended should help reduce personnel cost because they apply to the process for meeting regulatory requirements. These can be from a reduction in overall processing effort or the direct reduction of staff as described for physical security in Volume Two of the Regulatory Review Group report.

For perspective on potential savings, the elimination of one full time equivalent engineer per site for processing at a cost of $100,000 annually integrated across 70 sites with a 20-year average plant life represents a saving of $140M. The reduction of three technicians per shift in a five shift rotation at a cost of $60,000 annually per person over the same period and number of plants represents a savings of $1.2B.

The NRC would expect modest savings. The NRC effort would shift from pre-approval reviews to post-implementation reviews with no response required. The recommendations do not address review responsibility. Currently, the regions process the change requests required for security and quality assurance, while the NRR staff reviews the report of changes to the FSAR as required by 10 CFR 50.71. The recommendations do not change the reporting frequencies for submittal of changes to plans addressed in 10 CFR 50.54; however, the changes would be submitted for information only and not for review and approval.

COORDINATION

The Office of the General Counsel (OGC) has reviewed this paper and has found no legal objection to the staff's proposal. A copy of the proposal has been supplied to the Advisory Committee on Reactor Safeguards review. In view of the policy nature of the proposed amendments, the staff believes that appropriate decisions on the disposition of this specific recommendation of the Regulatory Review Group can best be achieved by early review by the Commission and the public. The staff will resubmit the amendments to ACRS and CRGR for review and comment before making recommendations to the Commission regarding the final rule.

RECOMMENDATION

That the Commission Approve the publication of this paper and the enclosure as a proposed rule to allow publication in the Federal Register by , 1993, for a 60 day comment period.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Federal Register Notice for
   proposed amendments to 10 CFR 50.2 and 50.54
2. Regulatory Analysis Statement
3. Public Announcement
4. Draft Congressional Letter

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

RIN 3150-AE70, AE71, AE72, AE73, and AE74

Changes to Quality Assurance Programs, Security Plans, Emergency Plans, and Fire Protection Plans, and Definition of and Changes to Commitments

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to amend its power reactor safety regulations to allow licensees to make changes to their quality assurance program, security plan, emergency plan, and fire protection plan that do not reduce the program or plan's content below that necessary to implement the requirements prescribed in the regulations without prior NRC approval. The proposed amendments would continue to require licensees to submit the changes at the current reporting frequency. However, the changes would be submitted to the NRC for information purposes only. The NRC staff would not review and approve the change prior to the licensee implementing the change. The NRC is also proposing to amend its power reactor safety regulations to define the term commitment and outline the change mechanism for a licensee who wishes to modify a commitment. The proposed amendment would also include a requirement that licensees notify the NRC if a previous change to a program or plan inadvertently decreases the program or plan's content

below that necessary to implement the regulations or if a change to a commitment inadvertently reduces the degree of protection provided to public health and safety. Implementation of the proposed amendments would be one year after publication of the final rules. The proposed amendments are intended to reduce the regulatory burden on the NRC staff and the licensees by having each licensee held to the same standard of acceptability as prescribed in the regulations.

DATES: Comment period expires (60 days after publication). Comments received after this date will be considered if it is practical to do so, but the Commission is able to assure consideration only for comments received on or before this date.

ADDRESSES: Mail written comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch.

Deliver comments to: 11555 Rockville Pike, Rockville, Maryland, between 7:45 am and 4:15 pm Federal workdays.

Copies of the regulatory analysis, the supporting statement submitted to OMB, and comments received may be examined at: the NRC Public Document Room at 2120 L Street NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Claudia M. Craig, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 504-1281.

SUPPLEMENTARY INFORMATION:

Background

In February 1993, the Commission approved the establishment of a regulatory review group (RRG) to conduct a comprehensive and disciplined review of power reactor regulations and related NRC processes, programs, and practices for their implementation. Within the framework of this review, the RRG identified two areas that may cause undue regulatory burden on both the NRC staff and licensees: commitments contained in programs and plans described in 10 CFR 50.54 and the change mechanisms, and commitments not contained in programs or plans under 10 CFR 50.54 and the associated mechanisms for changes.

Three areas were identified in 10 CFR 50.54 -- quality assurance, security, and emergency planning. Although not currently included in 10 CFR 50.54, the RRG also identified the fire protection plan as being under the licensee's control and applicable to treatment similar to that of the three areas listed above. Often during the licensing process, licensees made commitments to perform actions in each of these areas in excess of what is required by the regulations. In the cases of security and fire protection, a provision regarding each plan was added as a condition of the license and a license amendment was required to make changes to either the security or fire protection plan. Therefore, as 10 CFR 50.54 is currently written, each quality assurance program, security plan, emergency plan, and fire protection plans is held to a different standard of acceptability at each facility. Additionally, the change mechanism and record retention requirements described in the regulations are different for quality assurance, security, and emergency planning at each facility. The RRG determined that the regulatory burden could be reduced if each licensee is held to the requirements contained in the regulation and if the change mechanism and record retention requirements within the regulation are the same for each area. This would eliminate the need for plant-specific license conditions and license amendments for changes to the security or fire protection plans.

In 10 CFR 50.54(a), a licensee is free to change commitments in the quality assurance program as long as the commitments are not reduced. If the commitment is reduced, the licensee needs NRC approval prior to implementation of the change. Although used in the regulations and in other regulatory documents, the term commitment is not defined either in 10 CFR 50.54 or in 10 CFR 50.2. The regulations do, however, make a clear distinction between a commitment and the plans or programs that are implemented to satisfy the commitment. In the past, there has not been a clear distinction between commitment and its implementing mechanism. Many times they have been treated as one and the same. This has led to ambiguity and has raised questions in determining whether NRC approval is needed for a change to a commitment.

It is important to define the term commitment and how potential changes to a commitment should be handled so that the NRC staff and licensees know the issues that should be submitted for review and approval and the issues that licensees are free to change without prior NRC approval. A common understanding of the term and of how commitments are to be treated should lead to reduce the regulatory burden for both the NRC staff and the licensees by allowing the staff and licensees to focus on changes to commitments that are significant contributors to the safety of the plant.

Actions to which licensees agreed should be treated in two separate fashions. First, if an action is determined to be of such safety significance that it needs to be upgraded from an intent to carry out an action, the action should be included in the license as a condition of the license. Licensees are not allowed to make changes to an action that becomes part of the license except through the license amendment process under §50.90. Second, if an action is not significant enough to be elevated to the regulatory status of a condition of the license, but is considered a commitment, the licensee's administrative controls should be permitted to govern the change process.

Description

The proposed amendments address the quality assurance, security, and emergency preparedness portions of 10 CFR 50.54, add a section to 10 CFR 50.54 to address the fire protection plan, add a definition of the term commitment to 10 CFR 50.2, and add a change mechanism for commitments to 10 CFR 50.54. The proposed amendments would allow licensees to change their quality assurance program, security plan, emergency plan or fire protection plan if those changes do not reduce the program or plan's content below that necessary to implement the requirements contained in Appendix B to Part 50, 10 CFR 73.55a or Appendices B and C to Part 73, 10 CFR 50.47(b), or 10 CFR 50.48(a), respectively. The licensee would be required to continue to submit those changes at the current reporting frequency, with the exception of fire protection. Licensees will be required to submit changes to the fire protection plan 60 days after the change is implemented. However, the changes would be submitted to the NRC for information purposes only. This ensures the

NRC staff has the current version of the program or plan and allows the staff to act if it determines a change reduces the program or plan's implementation below that necessary to implement the requirements in the regulations. Changes to the security plan and fire protection plan would no longer need a license amendment as long as the plan continued to implement the requirements of the regulations.

The proposed amendments regarding commitments not contained in a program or plan described in 10 CFR 50.54 would add a definition of commitment to §50.2 and add a paragraph to §50.54 that would address the change mechanism for a commitment. The definition of the term commitment would include any condition or action agreed to or volunteered by a licensee which has been submitted in writing on the docket and the source of the commitment is based on administrative mechanisms such as bulletins, generic letters, or confirmatory action letters, as discussed in Section H to Appendix C to 10 CFR Part 50, "General Statement of Policy and Procedures for Enforcement Actions." Commitments in specific programs or plans addressed under 10 CFR 50.54, in the FSAR addressed by §50.59, and in response to compliance issues would be excluded from this definition. Licensees would be free to change a commitment provided an analysis has been performed by the licensee which demonstrates the proposed change does not reduce the degree of protection provided to public health and safety. For those changes to commitments that do not reduce the degree of protection, the licensee would be required to submit a report containing the changes and the supporting analysis to the NRC in accordance with the filing requirements of 10 CFR 50.71(e) for submittal of Final Safety Analysis Report revisions. If a change does reduce the degree of protection provided public health and safety, the licensee would be required to receive

NRC approval before implementing the change. This would allow licensees to make changes to commitments by evaluating the changes and determining whether the change results in a reduction in the degree of protection provided to the public health and safety. The evaluations should be documented and should demonstrate that a reduction in the degree of protection does not exist and that the changes represent an equivalent level of safety.

The proposed definition of commitment would include only certain licensee correspondence. For example, responses to enforcement actions would be excluded because compliance with the regulations is required. How compliance is achieved would not be considered a commitment since continued operation of a facility is based on compliance with the requirements in the regulations and not the method of achieving compliance. The same rationale would apply to generic communications promulgated to impose a compliance backfit. In other generic communications, however, the staff would have to clearly state whether the response will be considered a commitment such that future changes to that commitment would be governed by the proposed mechanism to change a commitment. This approach is consistent with the implied definition of commitment in the enforcement policy.

If at any time a licensee identifies that a previous change to a plan or program has inadvertently decreased the plan or program below that necessary to implement the regulations, or if a change to a commitment has inadvertently reduced the level of protection to the public health and safety, the licensee shall notify the NRC of the situation and any corrective action taken to ensure the compliance with the regulations. This is consistent with the NRC's policy and procedures for enforcement actions contained in Appendix C to 10 CFR Part 2. In that policy, the Commission states that it attaches great importance to licensee programs for detection, correction, and reporting of problems that may constitute, or lead to, violation of regulatory requirements. Adjustments to civil penalties often occur when a licensee identifies the violation, promptly reports it to the NRC, and takes prompt action to correct the problem upon discovery.

The licensee would also be required to maintain the changes and the bases for the changes as facility records for three years. Three years was determined to be adequate to allow for NRC review at the site while minimizing the potential for dual record retention because the changes would be submitted on the docket.

Implementation of the final amendments will be similar to that of the recent maintenance rule and 10 CFR Part 20. The final amendments will be effective one year after publication in the Federal Register to allow a transition period for the development of staff and industry guidance, if necessary. One year was determined to be adequate because there are no technological advances which need to be addressed and there is existing guidance and industry practices available.

Environmental Impact: Categorical Exclusion

The NRC has determined that these proposed regulations are the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for these proposed regulations.

Paperwork Reduction Act Statement

The proposed rules amend information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq). These rules have been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this collection of information is estimated to average a reduction of 40 hours per site per each response for security plans, fire protection plans, quality assurance plans and commitments, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The Commission has prepared a draft regulatory analysis on these proposed regulations. The analysis examines the costs and benefits of the alternatives considered by the Commission. The draft analysis is available for inspection in the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from Claudia M. Craig, Washington, DC 20555, telephone (301) 504-1281.

The Commission requests public comment on the draft regulatory analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that these rules will not, if promulgated, have a significant economic impact on a substantial number of small entities. These proposed rules affect only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act, or the Small Business Size Standards set out in the regulations issued by the Small Business Administration at 13 CFR Part 121.

Backfit Analysis

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to the quality assurance, security, emergency preparedness, and fire protection portions of the proposed amendments, and, therefore, that a backfit analysis is not required for those portions of the proposed rules because the amendments do not impose more stringent safety requirements on 10 CFR Part 50 licensees. However, as required by 10 CFR 50.109, the Commission has completed a backfit analysis for the proposed definition of and change mechanism for the term commitment. The Commission has determined based on this analysis, that the backfit, although imposing a new regulatory staff position, is justified in that it will reduce costs through the deletion of submittals for NRC approval of changes to voluntary licensee obligations in excess of the regulations. The backfit analysis on which this determination is based reads as follows:

Objectives. By defining the term commitment and providing a mechanism to change a commitment, regulatory burden on the licensees and NRC staff should be reduced because submittal of changes for NRC approval to voluntary licensee obligations in excess of the regulations will no longer occur.

Description. The backfit will define the term commitment and will recognize that licensees can modify commitments without NRC prior approval if the change does not reduce the level of protection to the public health and safety. Licensees will be required to submit a report of the changes to commitments that do not reduce the level of protection to the public health and safety in accordance with the filing requirements of 10 CFR 50.71(e) for submittal of Final Safety Analysis Report revisions. Licensees will be required to submit for NRC review and approval changes to commitments which do reduce the level of protection to the public health and safety prior to implementation.

Change in Risk. There is no potential change in the risk to the public from the accidental offsite release of radioactive material as a result of this backfit. This backfit only affects how licensees define commitments and how commitments can be changed by licensees.

Impact on Radiological Exposure. There is no potential impact on the radiological exposure of facility employees as a result of this backfit. This backfit only affects how licensees define commitments and how commitments can be changed by licensees.

Costs. There are no installation costs associated with this backfit, and it is anticipated that the long-term cost should be decreased because of the deletion of unnecessary submittals of changes to voluntary obligations in excess of the regulations for NRC approval. There will be some minor costs associated with the preparation and submittal of the periodic report, but this is estimated at far less than the costs associated with the current process of individual submittal of changes to commitments for NRC approval.

Safety Impact. There is no potential safety impact of changes in plant operational complexity because of this backfit.

NRC Resource Burden. The burden on the NRC staff should also be reduced in the long term because unnecessary submittals of changes to commitments will be alleviated. The NRC staff could perform one review of changes to commitments that do not reduce the level of protection to the public health and safety annually or on a refueling cycle basis for each plant instead of performing numerous reviews of individual changes to commitments. The NRC staff will also continue to perform reviews of unreviewed safety questions.

Impact on Different Facilities. There is no potential impact on different facility types, designs, or ages as a result of this backfit because all plants will be implementing prospectively the same definition of commitment and the same change mechanism for commitment.

Status of Backfit. This backfit is considered final, although the backfit will be implemented only in a prospective manner.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

Part 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 946, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L.91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and

50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. In §50.2, the definition of commitment is added to read as follows:

§ 50.2 Definitions.

* * * * *

Commitment means any condition or action agreed to or volunteered by a license holder that has been submitted to the Commission in writing when the source of the commitment is based on such administrative mechanisms as bulletins, generic letters, and confirmatory action letters, excluding responses from legally binding requirements.

* * * * *

3. In §50.54, paragraph (a) is revised to read as follows:

§ 50.54 Conditions of licenses.

* * * * *

(a)(1) Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance requirements in Appendix B to this part shall maintain and implement the quality assurance program described or referenced in the Final Safety Analysis Report, as kept up to date pursuant to 10 CFR 50.71(e)(4).

(2) Each licensee described in paragraph (a)(1) of this section may make a change to a previously accepted quality assurance program if the change does

not reduce the program's content below that necessary to implement the quality assurance requirements in Appendix B to this part. Changes to the quality assurance program description that do not reduce the program's content below that necessary to implement the quality assurance requirements in Appendix B to this part must be submitted to the NRC as follows:

(i) Changes made to a previously NRC-accepted quality assurance program must be submitted as specified in §50.4, in accordance with the requirements of §50.71(e).

(ii) The submittal of a change to the quality assurance program must include all pages affected by that change and must be accompanied by a forwarding letter that identifies the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the requirements of Appendix B to this part. However, the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items.

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(3) A licensee desiring to make a change that reduces the program's content below that necessary to implement the quality assurance requirements in Appendix B to this part shall request an exemption pursuant to 10 CFR 50.12.

(4) The licensee shall notify the NRC if it identifies a previous change to the quality assurance plan inadvertently decreases the plan's content below that necessary to implement the requirements of Appendix B to this part. The licensee shall also provide to the NRC the action taken to correct the deviation.

(5) The requirements of this section shall be implemented by each licensee no later than 1994.

* * * * *

4. In §50.54, paragraph (p) is revised to read as follows:

§ 50.54 Conditions of licenses.

* * * * *

(p)(1) The licensee shall prepare, implement, and maintain safeguards contingency plan procedures in accordance with Appendix C to Part 73 of this chapter for effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan.

(2) The licensee may make changes to the plan referenced in paragraph (p)(1) without prior Commission approval if the changes do not decrease the plan's content below that necessary to implement Appendix C to Part 73. Changes to the plan that do not reduce the plan's content below that necessary to implement the requirements of Appendix C to Part 73 must be submitted to the NRC as follows:

(i) Changes made to a previously NRC-accepted safeguards contingency plan must be submitted as specified in §50.4 within 60 days after the change is made.

(ii) The submittal of a change to the safeguards contingency plan must include all pages affected by that change and must be accompanied by a forwarding letter that identifies the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the requirements of Appendix C to Part 73. However, the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items.

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(3) Prior to the safeguards contingency plan being put into effect the licensee shall have --

(i) All safeguards capabilities specified in the safeguards contingency plan available and functional;

(ii) Detailed procedures developed according to Appendix C to Part 73 available at the licensee's site; and

(iii) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed Procedures.

(4) A licensee desiring to make a change which would decrease the plan's content below that necessary to implement the criteria of Appendix C to Part 73 or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, Responsibility Matrix) contained in a licensee safeguards contingency plan prepared pursuant to §50.34(d) shall request an exemption pursuant to 10 CFR 73.5.

(5) The licensee shall implement and maintain a security plan and guard training and qualification plan prepared to comply with the requirements of 10 CFR 73.55a and Appendix B to Part 73. The licensee may make changes to the security plan and guard training and qualification plan without prior Commission approval if the changes do not decrease the plans' content below that necessary to implement the applicable requirements of 10 CFR 73.55a and Appendix B to Part 73. Changes to the plans that do not reduce the plans' content below that necessary to implement the requirements of 10 CFR 73.55a and Appendix B to Part 73 must be submitted to the NRC as follows:

(i) Changes made to a previously NRC-accepted security plan and guard training and qualification plan must be submitted as specified in §50.4 within 60 days after the change is made.

(ii) The submittal of a change to these plans must include all pages affected by that change and must be accompanied by a forwarding letter that identifies the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the applicable requirements of 73.55a and Appendix B to Part 73. However, the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items.

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(6) A licensee desiring to make a change which would decrease the security plan or the guard training plan and qualification plan contents below that necessary to implement the requirements of 10 CFR 73.55a or Appendix B to Part 73 shall request an exemption pursuant to 10 CFR 73.5.

(7) The licensee shall provide for the development, revision, implementation, and maintenance of its safeguards contingency plan. To this end, the licensee shall provide for a review of the safeguards contingency plan at least every 12 months. This review must be conducted by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program. The review must include a review and audit of safeguards contingency procedures and practices, an audit of the security system testing and maintenance program, and a test of the safeguards systems along with commitments established for response by local law enforcement authorities. The results of the review and audit, along

with recommendations for improvements, must be documented, reported to the licensee's corporate and plant management, and kept available at the plant for inspection for a period of two years.

(8) The licensee shall notify the NRC if it identifies a previous change to the plans inadvertently decreases the plans' contents below that necessary to implement the requirements of 10 CFR 73.55a, Appendix B to Part 73 or Appendix C to Part 73. The licensee shall also provide to the NRC the action taken to correct the deviation.

(9) The requirements of this section shall be implemented by each licensee no later than 1994.

* * * * *

5. In §50.54, paragraph (q) is revised to read as follows:

§ 50.54 Conditions of licenses.

* * * * *

(q)(1) A licensee authorized to possess and operate a nuclear power reactor shall implement and maintain in effect emergency plans which implement the requirements in 10 CFR 50.47(b) and Appendix E to this part. A licensee authorized to possess and/or operate a research reactor or a fuel facility shall follow and maintain in effect emergency plans which implement the requirements in Appendix E to this part.

(2) The nuclear power reactor licensee may make changes to these plans without Commission approval if the changes do not decrease the plans' content below that necessary to implement the requirements of §50.47(b) and Appendix E to this part. The research reactor and/or the fuel facility licensee may make changes to these plans without Commission approval if these changes do not

decrease the plans' content below that necessary to implement the requirements of Appendix E to this part. Changes to the emergency plans that do not reduce the plans' content below that necessary to implement the applicable emergency planning requirements in 10 CFR 50.47(b) and Appendix E to this part must be submitted to the NRC as follows:

(i) Changes made to previously NRC-accepted emergency plans must be submitted as specified in §50.4 within 30 days after the change is made.

(ii) The submittal of a change to the emergency plans must include all pages affected by that change and must be accompanied by a forwarding letter that identifies the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the applicable requirements of 10 CFR 50.47(b) and Appendix E to this part. However, the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items.

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(3) The licensee shall notify the NRC if it identifies a previous change to the emergency plans inadvertently decreases the plans' contents below that necessary to implement the requirements of 10 CFR 50.47(b) and Appendix E to this part. The licensee shall also provide to the NRC the action taken to correct the deviation.

(4) The requirements of this section shall be implemented by each licensee no later than 1994.

* * * * *

6. In §50 54, paragraph (ff) is added to read as follows:

§ 50.54 Conditions of licenses.

*   *   *   *   *

(ff)(1) A licensee authorized to operate a nuclear power reactor shall implement and maintain in effect a fire protection plan that satisfies Criterion 3 of Appendix A to this part as implemented by the requirements of §50.48(a).

(2) Each licensee may make a change to a previously accepted fire protection plan provided the change does not reduce the plan's content below that necessary to implement the applicable requirements in §50.48(a). Changes to the fire protection plan that do not reduce the plan's content below that necessary to implement the fire protection requirements must be submitted to the NRC as follows:

(i) Changes made to a previously NRC-accepted fire protection plan must be submitted within 60 days after the change is made.

(ii) The submittal of a change to the fire protection plan must include all pages affected by that change and must be accompanied by a forwarding letter that identifies the change, the reason for the change, and the basis for concl ding that the revised program incorporating the change continues to satisfy the requirements of 10 CFR 50.48(a). However, the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items.

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(3) A licensee desiring to make a change that does reduce the plan's content below that necessary to implement the fire protection requirements of §50.48(a) shall request an exemption pursuant to 10 CFR 50.12.

(4) The licensee shall notify the NRC if it identifies a previous change to the fire protection plan inadvertently decreases the plan's content below that necessary to implement the requirements of 10 CFR 50.48(a). The licensee shall also provide to the NRC the action taken to correct the deviation.

(5) The requirements of this section shall be implemented by each licensee no later than 1994.

* * * * *

7. In §50.54, paragraph (gg) is added to read as follows:

§50.54 <u>Conditions of Licenses</u>.

(gg)(1) The licensee may change a commitment without NRC approval provided an analysis has been performed which demonstrates the change does not reduce the degree of protection provided to the public health and safety. The licensee shall submit a report containing the changes and the analysis in accordance with the filing requirements of 10 CFR 50.71(e) for submittal of FSAR revisions. The licensee shall maintain the reports as facility records for a period of three years.

(2) A change to a commitment that reduces the degree of protection provided to the public health and safety must be submitted to the NRC and receive approval prior to implementation.

(3) The licensee shall notify the NRC if it identifies a previous change to a commitment inadvertently reduced the level of protection to the public health and safety. The licensee shall also provide to the NRC the action taken to correct the deviation.

(4) The requirements of this section shall be implemented by each licensee no later than 1994.

Dated at Rockville, Maryland, this day of 1993.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,
Secretary of the Commission.

APPENDIX B

SUMMARY OF PUBLIC COMMENTS

APPENDIX B

SUMMARY OF PUBLIC COMMENTS

Accident Prevention Group

The Accident Prevention Group believes the report represents a modest step in the right direction, but falls short of providing what is badly needed - focus on nuclear safety issues. The Accident Prevention Group also believes the document probably represents the most significant step towards risk-based regulation, but too much concern is given to uncertainties and the report ignores the existence of developed human reliability technologies. The document is also believed to be narrow in perspective with regard to the regulations and provides neither historical nor current uses of probabilistic risk assessment (PRA) by the NRC. The Accident Prevention Group recommends that the Commission form an external group to supplement the NRC staff's knowledge with regard to PRA techniques. The Accident Prevention Group also provided specific comments on Volume 4.

BWR Owners' Group

The BWR Owners' Group (BWROG) fully supports the RRG effort and believes the RRG has done an excellent job. The BWROG also believes it is highly important that the NRC, with assistance from the industry, move quickly to establish appropriate priorities for pursuing resolution of the various items identified. The BWROG plans to participate and contribute to that process under the auspices of NUMARC.

Carolina Power & Light

Carolina Power & Light (CP&L) supports the efforts of the RRG and urges close cooperation between NRC and NUMARC. CP&L believes it may be more cost-effective to pursue generic improvements rather than plant-specific and the Cost Beneficial Licensing Action task force should be continued and expanded or integrated into the regulatory review process. CP&L encourages the NRC to promulgate appropriate guidance throughout the regions such that regional staff are aware of the changes resulting from the recommendations. CP&L supports the increased use of PRA and encourages efforts to define criteria for utility use of PRA in the regulatory arena.

Centerior Energy

Centerior Energy strongly supports the NRC's and industry's efforts to identify and eliminate burdensome regulatory requirements and commitments that do not contribute to safety. Centerior Energy provided detailed comments on Volume 4, including identifying a need for pilot studies to evaluate the existing review criteria and define the guidelines for the use of PSA application.

Commonwealth Edison

Commonwealth Edison (CECo) is generally in agreement with most of the report's recommendations and provided detailed comments. CECo, however, does not agree with the report's positions on: design basis inservice testing, the reevaluation of the policy statement on design basis documentation, the criteria for petitions for rulemaking which reduce regulatory burden, clarification of the scope and depth of the term design bases, and inclusion of structures, systems, and components (SSCs) that have been found to be relatively important in plants of similar design. CECo suggested the NRC staff consider using the Department of Energy's Order 5700.6c for quality assurance (QA) as a starting point in developing an approach to performance-based quality assurance. Additional suggestions were: allow all core limits that can changed as a function of fuel type or cycle-specific parameters to be relocated to the Core Operating Limit Report, two tiers of commitments - enforceable and voluntary, generic communications identifying alternative courses of action, licensees being encouraged to present additional alternatives, and providing licensees with a process by which to seek review of and relief from staff rejection of alternatives, and a possible extensive rewriting of 10 CFR Part 50 using PRA as a basis.

Entergy Operations, Inc.

Entergy believes the initiative will ultimately result in a significant benefit to safety. Entergy urges NRC to pursue these and additional efforts and that the changes recommended be expeditiously implemented by the NRC. In order for the RRG effort to continue, Entergy believes a continuing charter and champion within the NRC will be required. Entergy also provided detailed comments on individual sections of the report: it may be premature to state that Appendix B should not be modified, training in risk technology for the staff should be undertaken, a policy statement to address the use of risk technology should be developed, new definitions should be adopted for the different types of generic communications, a clear definition of performance-based is needed, and a reevaluation of the human reliability component is needed.

Electric Power Research Institute

In general, the Electric Power Research Institute (EPRI) believes the report is a significant step forward toward the utilization of risk concepts in nuclear plant regulation. EPRI provided detailed comments on Volume Four of the RRG report. EPRI feels that the graded implementation of Appendix B described in the report will be difficult, if not impossible, to implement if SSCs found important in any PRA are included. A comparison of PRAs in the same class is important; however, it should be done as part of the validation and verification process of a plant's SSC prioritization process. EPRI believes that a more detailed discussion of optimizing AOTs should be included.

GPU Nuclear Corporation

GPU Nuclear provided detailed comments on sections of the report. GPU believes that adding a definition to 10 CFR 50.2 and 50.54 is unnecessary and adds burden. GPU strongly supports the NUMARC petition for rulemaking on Part 21 and the NUMARC position on security requirements. GPU believes that the enforcement policy, SALP, and ISI should be included in the regulatory review. GPU also recommends that outdated regulatory guides be voided, that the term "performance-based" be defined, that NRC consider revising its method of issuing generic letters and cancelling generic communications when they are superseded by regulations or other actions, that Appendix B may need to be changed to change the culture, that IPE results should be endorsed as appropriate sources of justification to extend allowed outage times (AOTs) and surveillance test intervals (STIs), and that PRA should be used to arrive at QA safety classification of systems.

Illinois Department of Nuclear Safety

The Illinois Department of Nuclear Safety (IDNS) is in favor of simplifying the regulatory process and shifting to risk-and performance-based regulations where it makes sense to do so. IDNS believes that two basic sources of information are necessary for adequate operational and regulatory decision-making: complete design basis information and current and comprehensive component level analysis of the risks. IDNS believes that the information in the policy statement on design basis information should be stipulated in a rule. IDNS is not in favor of relaxing the reporting requirements for emergency plan changes. IDNS fears that a series of reduction in safety that are individually marginal may end up being collectively significant. IDNS outlined a number of applications of PRA technology and made the following recommendation: through rulemaking require all licensees to submit at least a grade 2 level PRA, and work toward an eventual grade 3. IDNS does not agree with a pilot program approach for the use of PRA technology and believes the NRC should establish firm criteria for IPEs and require the use of living PRAs.

Illinois Power

Illinois Power agrees with the proposed changes identified and believes implementation of the report recommendations will result in a reduction of regulatory burden. However, with regard to the proposed changes to 10 CFR 50.54, Illinois Power believes that licensees may be less likely to make changes to plans if a notice of violation is served on licensees when the NRC disagrees with the licensee's change to its program. Illinois Power believes that licensees will instead opt to make changes with prior NRC approval and thus increase regulatory burden, which is contrary to the RRG's efforts.

Marvin Lewis

Mr. Lewis indicated that he was unable to comment on the report due the unavailability of a free copy of the report.

Nuclear Management and Resources Council

NUMARC endorses the central theme in the report that most of the apparent inflexibility does not reside in the regulations, but rather in implementing practices and guidance documents. A clear distinction between formal regulatory requirements and informal regulatory guidance must be established. NUMARC endorses the recommendation that each licensee should review its license to identify instances of overly prescriptive or unnecessary requirements. NUMARC urges the NRC to establish a formal steering committee to direct the implementation of the recommendations. Detailed comments on sections of the report were provided and areas of disagreement include: 10 CFR 50.54(f) needs to be changed, no definition of CLB is needed, no clarification of design bases is needed, do not delete the policy statement on severe accidents, it is premature to say Appendix B does not need to be changed, regulatory guides are used as informal requirements and should be kept up to date, an entire new rulemaking process should be implemented, revisions to 10 CFR 50.34a, 50.36a and Appendix I are needed, discontinue work on Regulatory Guide 1.28 in light of the graded QA approach work, the report's criterion for risk application are not endorsed, and the results of one plant's PRA should not be applied to another plant. Additionally, NUMARC recommended additional actions that were beyond the scope of the RRG effort, but should be evaluated by the NRC staff: reviewing 10 CFR Part 20 two years after implementation, revise the format and content guide for applications for licenses, review 10 CFR 50.63 for performance-based applicability.

Ohio Citizens for Responsible Energy, Inc.

The Ohio Citizens for Responsible Energy, Inc. (OCRE) stated it is essential to ensure that efforts to streamline regulations and enhance efficiency do not have negative impacts. OCRE feels that cost reduction is not a proper matter of concern for the NRC and should be the concern of the state rate regulatory agencies. The only basis for eliminating or relaxing regulatory requirements is competing risk. Licensees should develop well-justified alternatives and aggressively pursue their interests. OCRE is concerned that changes to regulatory requirements and programs have the potential to eliminate opportunities for public participation and access to information, and therefore, recommends establishing an electronic docketing system and PDR. OCRE questions the purpose of the review of the four operating licenses. OCRE is concerned with erosion of safety margins. Using risk-based allowed outage and repair times may yield a net safety benefit. OCRE believes that careful NRC oversight is needed with a risk-based approach.

Philadelphia Electric Company

Philadelphia Electric Company (PECo) endorses the concept of using Probabilistic Safety Assessment (PSA) in the regulatory process and firmly agrees that risk-based technology is a necessary step toward optimizing nuclear facilities while maintaining the public health and safety. Testing of methodologies and pilot programs must be performed expeditiously. PECo fully supports NUMARC's position and comments. PECo also provided detailed comments on Volume 4.

PLG

Overall, PLG believes the document represents an excellent step toward practical risk-based regulatory philosophy and restricted its comments to Volume 4 of the report. Most of the comments were technical in nature and would reduce the potential for confusion or misinterpretation, but some provided alternative approaches that may be more appropriately addressed in the implementation phase of the recommendations.

Reedy Associates, Inc.

Reedy Associates believes the RRG's conclusion that Appendix B is performance-based is without foundation and since no one understands that a performance-based QA approach is allowed by Appendix B, that is reason enough to rewrite Appendix B. Reedy Associates believes Appendix B as implemented does not assure product quality and that for the sake of safety, quality, and economics, Appendix B must be changed.

Science Applications International Corporation

Science Applications International Corporation (SAIC) provided comments specifically on Volumes 1 and 4. SAIC believes that Volume 1 should have provided vision and direction for risk-based regulations, specific goals, programs and directions, and should have clarified the differences between risk-based, performance-based, and prescriptive regulations. SAIC believes, that contrary to what the RRG recommended, risk-based regulation should be an NRC initiative and a joint NRC-industry working group should resolve associated criteria for risk-based regulation. SAIC also provided detailed notes on Volume 4.

Southern Nuclear Operating Company

The Southern Nuclear Operating Company is in agreement with the NUMARC comments. However, the criteria described in the report for the use of PRA are not endorsed and need to be addressed in more detail. Both the NRC and licensees must put into place a more disciplined process for evaluating the benefits of new regulations and commitments. Performance-based initiatives should be considered and old regulations and commitments which have little or no overall benefit need to be eliminated. The RRG initiative provides an important framework for the initiation of such an effort.

Sunil Weerakkody

Mr. Weerakkody believes that standards on the process of deriving PRA-based conclusions is at least as important as deriving accurate risk-based conclusions, as was done in the report. Detailed comments and additional information were provided.

Tennessee Valley Authority

The Tennessee Valley Authority (TVA) strongly supports the basic conclusion of the report that the means by that existing regulatory requirements have been implemented has contributed to burdens on nuclear utilities which do not result in a commensurate safety benefit. TVA intends to review activities at each of its nuclear plants to identify unnecessary regulatory-induced burdens. TVA encourages NRC to carry forward with the efforts recommended by the RRG, but this is only a first step. NRC management should take steps to implement the changes and ensure that resources are made available to approve those burden reduction actions which require prior NRC approval. TVA endorses the specific comments provided by NUMARC.

TU Electric

TU Electric does not believe the revisions to 10 CFR 50.2 and 10 CFR 50.54 are needed. Commitments that have a safety significance are adequately addressed by existing regulations. The remaining commitments are required to be addressed by 10 CFR 50.59 safety evaluations. Notification of changes to these commitments is achieved by the annual 10 CFR 50.59 summary submittal. TU Electric believes that promises made to the NRC are part of the current licensing basis and, therefore, should be evaluated under 50.59.

Union Electric Company

Union Electric Company strongly supports the overall intent of the report and the majority of the recommendations. Several of the recommendations may have significant immediate or potential value to the licensee. These include: revision of 10 CFR Part 21, revision of 10 CFR 50.54, delineation of NRC expectations in security, elimination of the requirement to submit security logs, revision of implementing documents for Appendix B.

Virginia Power

Virginia Power encourages timely implementation of the report's recommendations and urges the NRC to continue its effort to eliminate or simplify regulatory requirements marginal to safety. Virginia Power supports a regulatory environment that includes performance and risk-based considerations. Virginia Power recommends that NRC reviews for regulatory reduction take place on a periodic basis.

Westinghouse Electric Corporation

Westinghouse applauded the NRC for taking the initiative to perform the necessary research and analysis to support the development of the report. The report will be one of the fundamental steps in the overall nuclear industry initiative to reduce the regulatory burden while maintaining safety. Westinghouse strongly supports this initiative, but noted that utilities are not quick to implement the line item improvements associated with certain programs because of the lack of a near-term return on investment. Westinghouse also provided comments on each section of the report. These comments included: allowing changes without NRC approval to plans in 10 CFR 50.54 would give licensees flexibility and reduce burden, the use of risk technology should neither be barred nor required as part of unreviewed safety question determinations but should be allowed as a tool for the licensee, optimization beyond the new improved technical specification program is possible, parameters like core damage frequency and system unavailabilities should not be made regulatory compliance criteria, graded QA will not work for systems whose primary function is to mitigate releases from containment, setting all human error probabilities to a pre-determined value would provide erroneous results, more detailed information should be provided concerning the information needed to submit technical specification changes with risk considerations, and the report needs to address the level of risk degradation acceptable for changes in AOTs and STIs.

Winston & Strawn

Winston and Strawn submitted comments on behalf of the Nuclear Utility Backfitting and Reform Group (NUBARG). NUBARG strongly supports the RRG initiative and provided comments on the continued need for change to 10 CFR 50.54(f). NUBARG disagreed with the RRG's conclusion that 10 CFR 50.54(f) does not need revision. NUBARG recommended the NRC raise the threshold for issuance of an information request by cross-referencing the backfit rule for cases in which the request involves: a new program or an extensive analysis for which the backfitting rule should be invoked, and for compliance issues to require identification of the specific existing regulatory requirement for which verification of compliance is sought. NUBARG also recommended changes to staff procedures to include recognition that alternative actions and schedules must be considered by NRC for responses to 10 CFR 50.54(f) requests and to provide an informal process for a licensee to seek relief from the requested actions if the actions would impose a substantial burden on the licensee without a comparable safety benefit.

Wisconsin Public Service Corporation

Wisconsin Public Service Corporation (WPSC) believes the overall report serves as an excellent voice for change and firmly supports the conclusions. WPSC provided comments on Volume 4 to the report. WPSC believes there are different degrees of safety-related which must be taken into account when issues are regulated, and encourages an aggressive effort by the NRC and utilities in recognition and implementation of the conclusions. Specific comments include: the Group 3 application of PRA (on-line configuration control) is not worth directing any NRC resources into at the current time, PRA should be used as one of the factors in decision-making, the recommended screening values for human reliability should be modified, and the extent of updating PRAs should be clarified.

Yankee Atomic Electric Company

Yankee Atomic Electric Company believes the report heralds the onset of a fundamentally new approach to regulation that would set the stage for a process in which real advantage can be taken of the insights from risk analysis. Yankee provided a list of recommendations that they strongly support and believe should be implemented as soon as reasonably practicable. Yankee provided detailed comments on several areas and provided the following recommendations: the definition of the term commitment be adopted and CLB be modified to be consistent with the definition, changes to the term design basis should await completion of design basis verification programs, adopt the NUMARC changes to 10 CFR Part 21, use PRA as an option in 10 CFR 50.59 evaluations, complete revisions to NUREGs to assist licensees in developing rulemaking packages, clarify the concept of integrated risk approach, eliminate the dual standard for petitions for rulemaking that reduce burden and those that are for public health and safety, integrate the definition of commitment into the current licensing basis definition, revise and update the regulations and regulatory guides for possession-only licenses, approach design basis testing such that test conditions are not imposed into IST programs, drop the recommendation to define alteration, delete the policy statement on design basis information, issue an information notice on behavioral observation for aberrant behavior, review the prioritization method for rulemakings, avoid unilateral dropping of rulemaking activities without public input. Yankee also provided a discussion on issues which warrant criticism and include: fire protection, Appendix B QA program, acceptability should not be based on public perception, and defense in depth should be an element in graded QA for determining relative importance. Yankee Atomic also provided detailed discussion on several issues which went beyond the scope of the RRG. Due to time restraints, the RRG was unable to evaluate these fully; however, the areas listed appear, on the surface, to merit an evaluation by the appropriate NRC staff in the future: inspection program improvements, seismic design requirements, generic relief process, decommissioning regulations, fitness for duty recommendations.

Volume Two

Regulatory Review Group

Regulations

U.S. Nuclear Regulatory Commission
Office of the Executive Director For Operations

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2.1 INTRODUCTION AN SUMMARY

Volume Two of the Regulatory Review Group Report addresses the reviews of and findings regarding nuclear power reactor regulations and their implementation. Volume II:

- describes the methodology and criteria used to select and to review each regulation,
- provides position papers on certain regulations which were developed as a result of the review (the recommendations developed in these position papers are summarized in Volume One), and
- provides the completed data sheets used during the review for reference.

The regulations affecting power reactors in Parts 21, 26, 50, and 73 were reviewed to determine if they were prescriptive or performance based, if they went beyond what was necessary for safe operation, or if they were in need of clarification. The results of these reviews were documented on forms for the individual rules reviewed, and position papers were developed on certain regulations and regulatory processes to highlight significant findings and provide more detail. The statements of considerations for the rules reviewed were also assessed to ascertain background information for each rule. The rules were categorized according to the basis of the rule, type of rule (prescriptive or performance based), and safety group so that they could be aggregated into classifications as desired.

Although the review of the rules revealed a number of minor problems, it led to the conclusion that the rules themselves were not a major source of unnecessary burden on operating reactors.

Implementing guidance such as regulatory guides, the standard review plan, generic letters and bulletins, and inspection procedures were also reviewed in varying degrees to determine if the regulations and regulatory practices were consistent.

Recommendations were made based on the position papers developed from the review of regulations. These recommendations identify regulations and regulatory processes that should be eliminated, revised, or further evaluated by the staff.

2.2 METHODOLOGY

This section describes the method used to review the regulations affecting reactors under the umbrella of the Regulatory Review Task Force. The regulations screened include 10 CFR Part 21 (for commercial procurement), 10 CFR Part 26, 10 CFR Part 50 (with some exceptions), and 10 CFR Part 73 (for power plants).

The initial step was to screen the regulations. To perform the screening, standard questions were developed and collated on data forms for each regulation reviewed. These questions were developed to address the issues delineated in the Review Group's charter and provided a method to capture a minimum data set for each regulation. The questions were also used to classify or to categorize the regulations so that they could be classified in sets or groups according to attributes (e.g., prescriptive regulations, regulations that protect a barrier such as the fuel cladding). Most of the questions required the reviewer to make subjective judgments, but short explanations of the judgments were documented on the forms.

The data sheets have been included in Appendix A to this volume to provide readers the results of the subjective review performed on each regulation that was selected for review. The data sheets, in addition to providing information on each rule, identify the rules which were chosen for further evaluation and provide insights as to why those rules were chosen for further review at this time. Considerations that were used to determine which rules were chosen for further review included the fact that possible recommendations could be implemented in the near term, that there were existing data and no new information would have to be developed or researched to implement the recommendation, and that the industry had an interest in the subject. Additionally, some rules were chosen for further evaluation because of their high visibility and level of interest in many forums. Some rules were identified for potential improvements on the data sheets but did not rise to the level of those that met the criteria and were therefore not reviewed by the Review Group. These rules also should be evaluated in the future.

After the initial screening, a followup review was performed on those regulations whose data forms identified rules that had areas of potential improvement. The followup process included review of various supporting and implementing documents. This review included regulatory guides, the standard review plan (and referenced branch technical positions), generic letters, bulletins, temporary instructions, and inspection procedures.

When the followup review was completed, positions papers were developed for each item. These position papers provide the background for the issue or current problem, a discussion of the issue or problems and potential resolutions, and a recommendation. In some cases, the recommendations are very specific, but in others, the recommendations are directed toward changes in processes. It should be noted the recommendations are provided as starting points for discussion and are not seen as final resolutions. The existing processes for rulemaking, revisions to regulatory guides, amendments to licenses, etc., would still have to be executed.

The goal was to identify regulations that appeared to be unnecessarily restrictive or otherwise imposed a burden on operating nuclear power reactors. Regulations such as those affecting initial licensing that did not affect operating reactors were generally not pursued. Other regulations affecting power plants were eliminated from the review if they had been the subject of recent Commission action or were being reviewed under another program. Examples of these are 10 CFR Part 20 and Appendix J to 10 CFR Part 50.

2.3 POSITION PAPERS

2.3.1 COMMERCIAL GRADE PROCUREMENT AND 10 CFR PART 21

I. INTRODUCTION

This discusses the review of the Regulatory Review Group (referred to hereinafter as Review Group) of commercial grade procurement and the regulations (10 CFR Part 21). Part 21 provides definitions of "commercial grade" and "dedication." These definitions are important to the assurance of safety when commercial replacement parts are bought for safety-related equipment.

II. BACKGROUND

Part 21 of Chapter 10 of the Code of Federal Regulations is based on the requirements delineated in Section 206 of the Energy Reorganization Act, which reads as follows:

> "Sec. 206. (a) Any individual director, or responsible officer of a firm constructing, owning, operating, or supplying the components of any facility or activity which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954 as amended, or pursuant to this Act, who obtains information reasonably indicating that such facility or activity, or basic components supplied to such facility or activity -(1) fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission relating to substantial safety hazards, or (2) contains a defect which could create a substantial safety hazard, as defined by the regulations which the Commission shall promulgate, shall immediately notify the Commission of such failure to comply, or of such defect, unless such person has actual knowledge that the Commission has been adequately informed of such defect or failure to comply.
> (b) Any person who knowingly and consciously fails to provide the notice required by subsection (a) of this section shall be subject to a civil penalty equal to the amount provided by section 234 of the Atomic Energy Act of 1954, as amended.
> (c) The requirements of this section shall be prominently posted on the premises of any facility licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended.
> (d) The Commission is authorized to conduct such reasonable inspections and other enforcement activities as needed to insure compliance with the provisions of this section."

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Part 21 of the regulations (in 10 CFR 21.3) provides the following definitions that are applicable:

> "**Basic Component** - When applied to nuclear power reactors, means a plant structure, system, component or part thereof necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in possible offsite exposures comparable to those referred to in Part 100.11 of this chapter.
>
> Basic component includes safety related design analysis, inspection, testing, replacement parts, or consulting services that are associated with the component hardware whether these services are performed by the component supplier or others.
>
> **Commercial Grade** - is an item [that] is not a part of a basic component until after dedication.
>
> A commercial grade item means an item that is (i) not subject to design or specification requirements that are unique to facilities or activities licensed pursuant to Parts 30, 40, 50, 60, 61, 70, 71, or 72 of this chapter **and** (2) used in applications other than facilities or activities licensed pursuant to Parts 30, 40, 50, 60, 61, 70, 71, or 72 of this chapter **and** (3) to be ordered from the manufacturer/supplier on the basis of specifications set forth in the manufacturer's published product description (for example a catalog).
>
> **Dedication** - of a commercial grade item occurs after receipt when that item is designated for use as a basic component."

III. DISCUSSION

Section 206 of the Energy Reorganization Act appears to be directed at making manufacturers and suppliers as well as licensees responsible for providing information on defects in material, designs, etc. Part 21 clearly envisions a situation in which procurement is conducted under the provisions of the standard quality assurance approach and appears to treat commercial grade procurement as a scenario that might happen on occasion, but not as the principal means of procurement. Under the inherent assumptions of these rules, Part 21 requirements become a simple addition to the procurement documents. This does not, however, represent nuclear procurement today. Currently, procurement is directed at replacement parts and material. Most procurement by power plant licensees is of commercial grade equipment that is then dedicated to "safety-related" use.

By the definitions, Part 21 recognizes that all basic components cannot be purchased as under the standard quality assurance program (i.e., Appendix B to Part 50). It also recognizes that replacement parts are a subset of basic components. It would also (we believe) allow one to differentiate between replacement parts that are vital to the basic component's ability to perform its safety function and replacement parts that do not affect the ability of the basic component to perform its safety function.

We also believe that the definition of commercial grade is so narrow (with the three **"and"** conditions specified) that it leaves a regulatory gap between replacement parts purchased under the Appendix B program and those that do not meet all three parts of the definition. For example, it does not describe the condition in which licensees find themselves when manufacturers and suppliers refuse to accept Part 21, but will supply replacement parts nominally manufactured to the same specifications as was the original basic component. The position in which licensees should be is that they must dedicate any replacement part that cannot be procured under an Appendix B (or acceptable equivalent) quality assurance program.

The definition of dedication could also be expanded to articulate that the dedication process needs to go far enough to provide a reasonable confidence that the replacement part will allow the basic component to perform its intended safety function. For example, if the dedication and engineering review indicate that the replacement part, even though an element of a basic component, could not, through its own failure or other malfunction, affect the ability of the component to carry out its safety function, then the dedication process need only document this determination briefly. On the other hand, dedication of a replacement part that is of significant importance to safety (e.g., fasteners for a steam generator primary side manway) should be subject to stringent dedication measures.

IV. RECOMMENDATIONS

It is recommended that Part 21 be reviewed and revisions proposed that recognize the existing procurement practices and conditions and allow the level and type of dedication to be graded based on the safety significance of each part. In addition, the Review Group recommends: (1) replacing the word "and" with "or" in the definition of commercial grade; and (2) adding a section to the definition of dedication to include action necessary to provide reasonable assurance that it will perform in service. The sections of 10 CFR 21.3 would read as follows:

> "**Commercial Grade** - is an item [that] is not a part of a basic component until after dedication.

A commercial grade item means an item that is (i) not subject to design or specification requirements that are unique to facilities or activities licensed pursuant to Parts 30, 40, 50, 60, 61, 70, 71, or 72 of this chapter **or** (ii) used in applications other than facilities or activities licensed pursuant to Parts 30, 40, 50, 60, 61, 70, 71, or 72 of this chapter **or** (iii) to be ordered from the manufacturer/supplier on the basis of specifications set forth in the manufacturer's published product description (for example a catalog).

Dedication - of a commercial grade item occurs after receipt when that item is designated for use as a basic component and consists of those actions necessary to provide reasonable assurance that it will perform in service."

V. ANALYSIS OF PUBLIC COMMENTS

Several public comments noted the NUMARC petition for rulemaking on 10 CFR Part 21. The staff should evaluate that petition along with the above Review Group recommendations to determine the appropriate changes to Part 21 and any associated definitions.

2.3.2 COMMITMENTS

I. INTRODUCTION

The term "commitment" has been used freely in the regulations and other correspondence, but the term is not defined in the regulations and there is no guidance on how a commitment is treated in the regulatory process or how a licensee should change a commitment. This paper discusses the term commitment and how it is used in the regulatory process.

II. DISCUSSION

Although the term "commitment" is not defined in the regulations, the regulations do make a clear distinction between a commitment and the plans that are implemented to satisfy the commitment. In past practice, however, there has not been a clear distinction between the two and many times they have been treated as one and the same. This practice has placed an undue burden on both the staff and the licensee. When the commitment and the method of meeting the commitment (the plan) are treated as the same, the details of and subsequent changes to the plan implementing the commitment need to be submitted to the NRC. This places a burden on the licensee staff. Additionally, those same details and subsequent changes to the plan need to be reviewed and approved by the NRC. This places a burden on the NRC staff. However, the staff should recognize that whether contained in the final safety analysis report or in other documents on the docket, including the staff's safety evaluation reports, commitments are not directly enforceable through notices of violation unless they are converted to Technical Specifications or other conditions on the operating license. Obviously, a commitment that is converted to a license condition may be changed only by amendment of the license. At present, any other commitment may be changed by a licensee without either notice to the NRC or NRC approval unless NRC regulations, such as 10 CFR 50.54(a), (p), or (q), which are deemed to be conditions on the license, require such notice or approval.

It is important to define the term "commitment" and how it should be handled so that the staff and licensees know what issues should be submitted to the staff for review and approval and what the licensees are free to change without prior NRC approval. It is believed that a common understanding of the term and of how commitments are to be treated will lead to reduced regulatory burden for both the staff and the licensees. It will allow the staff and licensees to focus on changes to commitments that may be significant contributors to the safety of the plant.

The first step in the process of correctly applying the term "commitment" is to define it. The key elements of a commitment are: (1) the licensee agreed to perform some action to comply with an NRC requirements or request, (2) on the basis of licensee's agreement, the Commission made a safety decision that affects the interest of the licensee, and (3) both the commitment and the safety decision are in writing. Based on these key elements, the Review Group recommends that the following definition be added to 10 CFR 50.2:

> **Commitment** - any condition or action agreed to or volunteered by a license holder that has been submitted to the Commission as a basis for a safety decision and both the condition or action and the decision that made use of the information are contained in the docket file.

Commitments should be treated in two separate fashions. First, if a commitment is of such safety significance that it needs to be upgraded from an intent to carry out an action, the commitment should be placed in the license. Licensees are not allowed to make changes to a commitment that becomes part of the license except through the license amendment process. In this case, the change to the commitment must be proven by the licensee to be equivalent to the original commitment.

Second, where a commitment is not significant enough to be elevated to the regulatory status of a condition on the license, the licensee's administrative controls should govern the change process. This would allow licensees to make changes to a commitment by evaluating the change to the commitment and providing the change to the NRC for information at a specified time interval. The evaluation of the change to the commitment should be documented and should demonstrate that an unreviewed safety question is not involved and that there is equivalence in safety. An unreviewed safety question would be deemed to exist if, as a result of the change, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or if a possibility of an accident of malfunction of a different type from any evaluated previously in the safety analysis report may be created; or if the margin of safety as defined in the basis for any Technical Specification is reduced. Similar to the 10 CFR 50.59 process, the burden would be on the licensee to review the change to the commitment and determine whether the change would involve an unreviewed safety question. By having licensees evaluate the change to the commitment in this manner, there should be no effect on the safety of the plant.

To clearly delineate the licensee's and NRC's responsibility and authority relative to commitments and the change mechanism for commitments, the Review Group recommends the following section be added to 10 CFR 50.54 to formalize the process by which changes may be made to commitments:

Proposed changes in commitments that do not involve an unreviewed safety question shall be submitted to the NRC in accordance with the reporting requirements of 10 CFR 50.71(e). Changes to commitments that involve an unreviewed safety question must be submitted to the NRC and receive approval prior to implementation. A proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or if a possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or if the margin of safety as defined in the basis for any technical specification is reduced.

The definition and codification of the change process would impose new requirements and a backfit analysis would have to be performed in order to amend the rule; however, the new rule would be implemented only prospectively.

III. RECOMMENDATION

Because of the importance of the recommendation in this section, the Review Group developed a proposed rulemaking to define the term commitment and to provide a change mechanism for a licensee who wishes to modify a commitment. The Review Group received comments from the NRC offices on the above definition and change mechanism and revised the proposed rulemaking. The detailed proposed rulemaking is located in Appendix A to Volume One. Although a definition and change process have been proposed, they are not considered the final approach to commitments, rather they have been proposed to begin a dialogue and solicit comments. The Review Group envisions more licensees in the future will begin to change commitments, and it is important that a common understanding of the issue and change mechanism be established for both NRC and licensee staff.

IV. ANALYSIS OF PUBLIC COMMENTS

As a result of public comments and recent staff interactions with specific utilities on this issue, it appears there is a need for more information to be developed regarding what constitutes a commitment and a need for a set of ground rules for changing commitments. The above definition has been modified to acknowledge the use of the term "commitment" in Appendix C, General Statement of Policy and Procedure for NRC Enforcement Actions, to 10 CFR Part 2.

2.3.3 DECOMMISSIONING

I. INTRODUCTION

This paper discusses two issues related to decommissioning. The first issue regards the decommissioning regulation, 10 CFR 50.82, and Regulatory Guide 1.86. The second issue concerns the length of time for which a license may be issued. Some of the issues discussed below were also addressed in SECY-92-382, "Decommissioning - Lessons Learned," and the Commission's SRM dated June 30, 1993, on that SECY.

II. DISCUSSION OF 10 CFR 50.82 AND OF REGULATORY GUIDE 1.86

The regulation, 10 CFR 50.82, speaks in terms of applying "to surrender a license voluntarily" and decommissioning a facility. Regulatory Guide 1.86 was promulgated in 1974 to provide guidance to licensees and to describe methods and procedures considered acceptable by the staff for termination of an operating license. The regulation that this regulatory guide addresses was promulgated in 1988 and last amended in 1992. The regulatory guide was written before the rule reached its present form.

Section 50.82(b)(iii)(5) requires a decommissioning plan to describe Technical Specifications, quality assurance provisions, and security plan provisions that will be in place during decommissioning. Section 50.36 deals with Technical Specifications only in connection with operation of a production or utilization facility. The quality assurance requirements of Appendix B to 10 CFR Part 50 apply only to design, construction, and operation of a facility. The physical protection requirements of 10 CFR Part 73 apply as long as special nuclear material is present at the site. Thus, the regulatory basis for judging the adequacy of the Technical Specifications and the quality assurance and physical security plan provisions related to decommissioning is not specified in NRC regulations.

Regulatory Position C.3 of Regulatory Guide 1.86 addresses surveillance and security for the retirement alternative, the final status of which requires a possession-only license. Subpart b. of this guide states that physical barriers to unauthorized entrance into the facility should be inspected at least quarterly; subpart c. states that a facility radiation survey should be performed at least quarterly; and subpart d. states that an environmental radiation survey should be performed at least semiannually. It appears that there is no regulatory basis for these frequencies, and the regulatory guide has been structured as a surrogate for the rule.

Subpart g(1) of Regulatory Position C.3 in Regulatory Guide 1.86 states that the licensee should submit an annual report. There appears to be no regulatory basis for this reporting requirement. Subpart g(2) states that abnormal occurrence reports should be submitted by telephone within 24 hours and reported in the annual report. It is unclear whether this reporting requirement supersedes the reporting requirements of 10 CFR 50.72, which apply to all Section 103 and 104 licensees. It was noted as part of the staff comments that the reporting requirements should be evaluated to determine whether rule modifications are needed to notify the NRC of events unique to decommissioned plants.

Subpart h. of Regulatory Position C.3 states that records and logs should be kept and retained until the license is terminated, after which they may be stored. These records include environmental surveys, facility radiation surveys, inspections of physical barriers, and abnormal occurrences. Section (g) of 10 CFR 50.75 requires retention of records important to safe and effective decommissioning, including unusual occurrences involving the spread of contamination (the regulatory guide appears to go beyond this definition), as-built drawings of areas where radioactive material are used or inaccessible, and cost estimates. It appears the records listed in the regulatory guide go beyond those listed in the rule.

Subpart e. of Regulatory Position C.4, "Decontamination for Release for Unrestricted Use," states that a survey report should be filed with the AEC at least 30 days prior to the planned date of abandonment. This does not appear to have a regulatory basis.

Regulatory Position C.5, "Reactor Retirement Procedures," states that any planned activities involving an unreviewed safety question or a change in the Technical Specifications should be reviewed and approved in accordance with 10 CFR 50.59. However, 10 CFR 50.59 applies only to holders of a license authorizing operation of a production or utilization facility. A possession-only license does not authorize operation. It should be noted that in an SRM dated June 30, 1993, the Commission directed the staff to amend 10 CFR 50.59 to make it expressly applicable to holders of licenses not authorizing operation.

Regulatory Position C.5.d. discusses a "dismantling order." The regulation, 10 CFR 50.82, does not discuss a dismantling order but does discuss a decommissioning order. It is unclear if these are one and the same.

III. DISCUSSION OF LENGTH OF LICENSE

The licenses issued under Section 103 of the Atomic Energy Act, whether to possess and use (operating license) or to possess only, are to be issued for a specified period not to exceed 40 years. Section 50.51 imposes a 40-year limit on all licenses issued. Therefore, any authority to possess or to use that exists under a Section 104 license also expires after 40 years unless it is renewed or a timely application for its renewal is pending. It appears inappropriate to speak of converting an operating license to a possession-only license for decommissioning near the end of the term of the operating license when decommissioning will extend well past the end of the term of the operating license. In such cases, either a new possession-only license should be issued or the authority to possess under the original license must be renewed.

For older plants, which may decommission near the end of the term of their license, it is not clear that a possession-only license should be issued by amendment of the operating license, as has apparently been done for some of the plants that were prematurely decommissioned. The appropriate course in this case would seem to be to issue either a new possession-only license or to renew the authority to possess under the original license. Both of these choices imply that there could be a hearing, although for the possession-only case, if a hearing were held, it could be conducted after the license is issued.

IV. RECOMMENDATIONS

Obtain OGC opinion to determine if a revision to 10 CFR 50.82 is needed to discuss the type of license needed to undergo decommissioning after the original 40-year license has expired. This should explore the option of a new possession-only license or renewal of the original license for possession-only purposes. The guidance discussed should not be treated as a surrogate for the regulation.

V. POTENTIAL IMPROVEMENTS

Revise and reissue Regulatory Guide 1.86 to reflect the current NRC organization (not AEC) and address the areas of inconsistency with the regulation as discussed above.

It should be noted that Regulatory Guide 1.86 is currently being revised and is scheduled to be completed in 1993.

VI. ANALYSIS OF PUBLIC COMMENTS

The comments identified a number of areas that should be evaluated when revising Regulatory Guide 1.86. The staff should take these into consideration.

2.3.4 FIRE PROTECTION

I. INTRODUCTION

The Review Group conducted a comprehensive review of plant fire protection programs by examining the regulatory bases and safety relevance of several requirements and assessing the impact of these requirements upon overall plant performance, with emphasis upon the affected processes. In this review, the pertinent fire protection regulations were examined to evaluate the degree of flexibility provided in current implementation practices and assess their importance to safety. Included in the evaluation was proposed action to replace Appendix R to 10 CFR 50 with a non-prescriptive, performance-based regulation. Such action was published as a proposal for comment in a Federal Register notice dated February 4, 1992, and is being considered in the NRC Marginal-to-Safety Program. This issue also merited the attention of the Committee to Review Generic Requirements (CRGR) in its Special Review of Existing NRC Regulations. The CRGR concluded that since the proposed action did not meet the special review criteria, this initiative should continue to be pursued in the Marginal-to-Safety Program. The potential for a reduction in regulatory burden and other aspects of plant fire protection provisions were evaluated by the Review Group to determine whether performance-based criteria could be effectively used to enhance the flexibility of programmatic requirements.

II. BACKGROUND

As discussed in the NRC Standard Review Plan (NUREG-0800), the purpose of a fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment in accordance with 10 CFR 50, Appendix A General Design Criteria (GDC) 3 and 5. This defense-in-depth principle is aimed at achieving an adequate balance in preventing fires from starting; detecting, suppressing, and extinguishing fires quickly to limit their damage; and designing plant safety systems so that even if a fire burns for a considerable time, no essential plant safety functions will be prevented from being performed. The fire protection requirements intended to ensure that the defense-in-depth principle is achieved in practice are codified in the fire protection rule (i.e., 10 CFR 50.48, which references 10 CFR 50, Appendix A, and Branch Technical Position Auxiliary Power Conversion System Branch (BTP APCSB 9.5-1), as well as Appendix A to BTP APCSB 9.5-1). These requirements are also currently discussed in Section 9.5.1 of the Standard Review Plan (SRP). Since, with few exceptions, the SRP and the various editions of the Branch Technical Position generally all contain the same provisions, the information provided therein can be considered to be the guidelines of a generic document, hereafter referred to as BTP 9.5-1.

Appendix R to 10 CFR 50 establishes the fire protection features required to satisfy Criterion 3 of the GDC with respect to certain generic issues for nuclear power plants licensed to operate prior to January 1, 1979. For those plants not operating before 1979 whose applications for construction permits were docketed after July 1, 1976, BTP 9.5-1 provides guidelines for basic fire protection programs, while Appendix A to BTP 9.5-1 does the same for those plants docketed prior to July 1, 1976. Other documents relating to the fire protection requirements delineated in 10 CFR 50, Appendix R, which with 10 CFR 50.48 was promulgated as a fire protection rule on November 19, 1980, include: Generic Letters 81-12, 83-33, 85-01, 86-10, and 88-12 and Information Notices 83-41, 83-69, and 84-09. Regulatory Guide 1.120 also provides fire protection guidelines for nuclear power plants and not only incorporates BTP 9.5-1 guidance, but also references several National Fire Protection Association (NFPA) and other relevant industry standards.

Of the above referenced documents, Information Notice 84-09, which discusses the lessons learned from NRC inspections of fire protection safe shutdown systems, and Generic Letters 86-10 and 88-12, which provide guidance on the removal of fire protection requirements from Technical Specifications, are of particular interest. Shortly after the issuance of Information Notice 84-09 in February 1984, the NRC conducted a series of regional workshops with the industry on the implementation of fire protection requirements. NRC guidance and responses to questions posed by the industry were provided to licensees relative to not only 10 CFR 50, Appendix R, and BTP 9.5-1, but also licensing and inspection policies related to plant fire protection programs. Subsequently, the NRC guidance on fire protection was appended to Generic Letter 86-10 and new Inspection and Enforcement Manual inspection procedures (64704, 64100, and 64150) were issued. These procedures govern inspection activities which assess the adequacy and implementation of a licensee's approved fire protection program and evaluate the existing licensee controls to achieve postfire safe shutdown at individual reactor facilities. Generic Letter 88-12 provides guidance on the preparation of a license amendment request to implement Generic Letter 86-10, effecting the removal of unnecessary fire protection Technical Specifications.

Other recent NRC generic communications (Bulletin 92-01 with supplement and Generic Letter 92-08) discuss problems with Thermo-Lag fire barrier material. The information and requirements in these documents relate directly to Appendix R provisions and exemplify the regulatory impact which can result from the application of prescriptive fire protection criteria, albeit soundly based in the current regulations.

III. DISCUSSION

The regulatory bases for reactor facility fire protection plans and programs are provided in 10 CFR 50.48; 10 CFR 50, Appendix R; and 10 CFR 50, Appendix A, GDC 3 and, to the extent the sharing of systems may apply, GDC 5. Depending upon the age of the plant license and other site-specific considerations, fire protection programs also contain commitments to the guidance provided in the applicable editions of BTP 9.5-1 and to the guidelines of various codes and standards (e.g., American Society for Testing and Materials (ASTM), American Nuclear Insurers (ANI), Uniform Building Code (UBC), Underwriters Laboratories (UL), Institute of Electrical and Electronics Engineers (IEEE), and particularly the NFPA). Plant operating licenses generally contain a license condition addressing fire protection. Such license conditions vary from plant to plant. For plants licensed prior to January 1, 1979, the license condition typically mandates the implementation of the plant-specific modifications required to meet the BTP 9.5-1 guidance. For plants licensed after January 1, 1979, implementation of the approved fire protection program as described in the Final Safety Analysis Report (FSAR), the Safety Evaluation Report (SER) as supplemented, and other commitments are generally imposed as a condition of the license. While some older plants have an amended fire protection license condition similar to that in more recent licenses, the reference to specific plant SER details indicates that the fire protection program is generally constrained by past licensee commitments and the programmatic provisions previously accepted by the NRC.

As discussed in Generic Letter 86-10 and confirmed by the Review Group's assessment of specific license requirements, these various license conditions can lead to difficulties not only in identifying the operative and enforceable fire protection requirements at each facility, but also in creating an unnecessary regulatory burden if the operating license has to be amended when the fire protection program is revised. Plants with more recent or amended licenses typically allow a licensee to make changes to the approved fire protection program without the prior approval of the NRC if the changes would not adversely affect the ability to achieve and maintain shutdown in the event of a fire. A typical fire protection license condition, representative of the standard condition discussed in Generic Letter 86-10, is included as an attachment. Since this license condition indicates that the fire protection program details are described in the FSAR, licensees are allowed to make revisions to their programs in accordance with 10 CFR 50.59 requirements.

However, even those fire protection license conditions which clearly allow certain program changes to be implemented without prior NRC approval can be subject to interpretation problems. For example, the phrase in the attachment that allows changes that "would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire" has a different meaning depending on whether "changes" are viewed in an absolute or relative sense. As indicated in the attachment, the baseline for the approved fire protection program is the Safety Evaluation Report (SER) as supplemented, which is typically a document that is several years old. In a relative sense, a licensee could reduce the level of certain fire protection commitments discussed in the SER without NRC approval, as long as the net effect of the changes would not adversely impact a safe plant shutdown, assuming a fire under present-day conditions. Since several improvements in fire protection technology and the fire response capability of the surrounding community are likely to have occurred since the date of the last applicable SER supplement, the relative credit given for such enhancements could counterbalance what might be viewed as a degradation in the individual plant fire protection program, when compared to prior commitments. Therefore, even though credit may be taken for changed conditions and technological advances, it is understandable why licensees may be reluctant to reduce prior programmatic commitments if a negative connotation is attached to the reductions, even if they would result in justifiable cost savings. This reluctance is further exacerbated by a general perception among licensees that they would be penalized in the inspection/assessment arena for any such program reductions.

As a consequence of both the hesitancy of licensees to implement program changes that are subject to NRC review and the low priority assigned by NRR to those issues with no safety significance, the submittal of operating license amendment requests, even where NRC generic guidance is available, has become a delayed process. This is illustrated by the fact that many licensees have not yet taken advantage of the NRC guidance on incorporation of the fire protection program into the plant FSAR and removal of the fire protection requirements from the Technical Specifications, as discussed in Generic Letters 86-10 and 88-12. Additionally, licensees do not appear to be taking full advantage of some inherent flexibility allowed by the regulations. For example, while scores of different NFPA standards may be used to implement a fire protection program at a plant, these standards and other similar codes delineate only guidelines which, while acceptable to the NRC staff, are not mandatory and thus may be viewed as similar to NRC regulatory guides. While a licensee that commits to specific NFPA codes is expected to meet the stated standards and is subject to inspection against the criteria delineated therein, an exemption from any NFPA Code is not required since the provisions are guidelines and not regulatory requirements. Thus, a licensee could document in either the FSAR or the plant Fire Hazards Analysis specific exceptions from the codes and standards to which general commitment has been made. Such exceptions differ from deviations from the regulations in that the latter requires specific exemption requests to be submitted to the NRC for approval. However, as in the case of regulatory guide usage, licensees appear to adopt NFPA and other fire protection codes and standards, for the most part, without exception, rather than propose alternative means of meeting the intent of the accepted guidance.

Appendix R to 10 CFR 50 establishes some requirements that are prescriptive, but also provides options in certain cases. For example, as delineated in Section III.G.2 of Appendix R, a licensee may select from several options the means of ensuring that one of the redundant safe shutdown trains will remain protected during a fire. Even with the selection of a specific option, like where a fire barrier with a 3-hour rating is stipulated for redundant train separation, the NRC may grant exemptions to such requirements if the licensee provides sufficient justification that the alternative barriers establish adequate protection against the fire hazards to which they are exposed. A typical plant design may provide several thousand penetrations in the fire barriers located throughout the plant. If existing documentation does not support verification of a 3-hour rating for some penetration seals, the licensee must either implement corrective measures to comply with the Appendix R requirements or submit justification as to how the existing barrier seals meet the intent of the regulations and why literal compliance would not significantly enhance the fire protection capability at the plant. Although the acceptance of such justification and approval of any exemptions lies within the purview and regulatory authority of the NRC, licensee engineering evaluations can provide the basis for the adequacy of existing fire barriers, even if the penetration seal qualification records and other quality documentation are deficient or uncertain. In any case, while it is the licensee's prerogative to seek exemptions from the regulations, until such problems are corrected or the identified discrepant conditions are analyzed to be adequate, the affected fire barrier would be considered to be in a degraded state.

In the above example, once a regulatory option is selected, the particular requirement (e.g., the 3-hour fire barrier penetrations) not only establishes the intended technical criteria for acceptable component design and installation, but may also inadvertently create a situation for licensees where total compliance may be a less costly and more certain option in specific cases than the pursuit of an equally sound and technically defensible exemption. In general, a relatively small number of exemptions from NRC regulations have been issued in current plant operating licenses. However, several plant-specific exemptions (approximately 1600 industry-wide) to Appendix R requirements have been approved and are reflected in plant fire protection programs, rather than directly in the licenses. This fact, along with the improvements in fire protection material and component performance and the years of fire protection experience and data gained since the issuance of the fire protection rule in 1980, appear to indicate that additional flexibility in the applicable regulations could be allowed without adverse safety impact. However, the net benefit of any added flexibility for a particular licensee is dependent upon several plant-specific factors, such as the age of the license, the design of the plant fire protection systems, the fire protection program commitments, and the cost of implementing major changes.

The impact of these plant-specific factors upon the enhanced flexibility potential of the fire protection area, in general, has to be considered in assessing the relative effect that specific regulations have upon program costs and licensee management decisions relative thereto. For example, in older licensed plants, compensatory measures for degraded barriers appear to be a routine activity. Older plants are also likely to have more fire protection exemptions, which not only is consistent with the increased compensatory actions, but also is indicative of licensee management decisions to bear the cost of continuing compensation in lieu of plant modifications. On the other hand, newer plants typically are designed with an expanded fire detection capability. While the increased use of detectors may reduce firewatch costs, the expense relative to surveilling and maintaining detectors in accordance with the full NFPA test requirements often becomes the single most significant fire protection cost factor at a plant with enhanced fire detection capabilities. A more balanced approach to both fire detector usage/surveillance costs and firewatch/patrol compensation costs for degraded barriers is achievable if performance-based criteria could be introduced into fire protection programs. While certain Appendix R requirements, like Section III.J, specifying battery power provisions for emergency lighting, are prescriptive and may lead licensees to opt to replace batteries rather than conduct testing, it is generally the licensee's own commitment documents and implementing procedures that prescribe the most restrictive requirements. As discussed above, licensee fire protection programs routinely ascribe to the prescriptive provisions of industry standards (e.g., NFPA) which go beyond the detail specified in the rules and regulations.

Thus, while the modification of 10 CFR 50, Appendix R, to a more performance-based regulation would certainly provide additional flexibility to licensees for the implementation of their fire protection programs, it appears that a significant potential for enhanced flexibility already exists. A reduction in regulatory burden would ensue if licensees re-evaluated current commitments to certain fire protection codes and standards and eliminated the unnecessary restrictions to their programmatic options in this area. A recommendation (discussed further in Volume Three of the Regulatory Review Group Report) to expand the use of performance-based requirements to supplant prescriptive criteria could be applied to the fire brigade and training provisions of 10 CFR 50, Appendix R. The assessment of selected plant licenses also confirmed that licensee-controlled programs are well suited to govern the implementation details of several technical programs and that prescriptive license conditions are not required to provide an assurance of quality. Similarly, performance-based data could be utilized in fire detector surveillance testing and frequency applications to develop alternative approaches to NFPA testing requirements if licensees opted to pursue such flexibility in revisions to their own fire protection programs.

However, given that the existing plant fire protection programs have already been developed and in place for several years and are currently being implemented to meet the requirements of 10 CFR 50, Appendix R, the savings that would actually be realized by licensees if the fire protection regulations were amended would probably vary considerably from plant to plant. While licensees would certainly benefit from reduced regulatory requirements affecting specific program activities, such as fire barrier penetration seal qualification and surveillance, fire hose acceptability testing, and emergency lighting qualification, the effectiveness of any enhanced flexibility in providing real savings is dependent upon several other factors. Both the NRC and the industry would have to address the need for revision of the applicable NRC fire protection guidance documents, not just the regulations. Also, it would have to be understood that a shift to the implementation of more performance-based requirements implies more than merely a reduction in regulatory burden; a soundly based, auditable and demonstrably effective program must be ready to replace the existing controls. The development and initiation of such major programmatic changes are neither cost-free in terms of financial and efficiency impact, nor value-independent from the standpoint of performance. As is discussed above, the commitments to various industry standards which provide the scope, criteria, and implementing details for the bulk of fire protection activities at a site appear to underlie the most significant, resource-intensive costs incurred by licensees in the fire protection area. If licensee management is reluctant to amend existing plant commitments, it is not clear that the resolve exists for major programmatic changes in the fire protection area. It is also not evident that licensees are likely to make substantial capital investments in the fire protection program changes that would be necessary to realize future potential savings.

It is therefore concluded that, while the review of requirements that are considered to be marginal to safety is an initiative that merits the continued attention of both the NRC and industry in the fire protection area, licensees should also be evaluating both their own licenses and detailed fire protection programs to identify, for elimination or revision, any unnecessary and prescriptive requirements. As an example, one licensee, which previously had taken advantage of the relocation of the technical fire protection requirements from its Technical Specifications to the FSAR, found that a further reduction in regulatory burden is achievable in the administrative controls section of the plant Technical Specifications where the conduct of specific fire protection program audits at an annual, biennial, and triennial frequency is specified. The Improved Standard Technical Specifications delineate more flexible audit requirements, while still ensuring that adequate and appropriate fire protection controls and implementing provisions are maintained. The Review Group believes that licensees, as a collective group, have not taken full advantage of the available options for enhanced flexibility (e.g., Generic Letter 86-10 and 88-12 recommendations) and have not pursued all the exceptions to the fire protection codes and standards which may be unreasonable or unnecessarily restrictive at their plants.

Each licensee's perspective of the inefficiencies and inconsistencies associated with the fire protection program requirements for its plant is unique. The wide range of views as to which fire protection implementation details represent the most onerous and costly requirements is not only indicative of plant-specific differences, but also illustrative of the point that the problem with prescriptiveness may be more related to unique plant commitments, rather than the fire protection regulations in general. The use of performance-based criteria and the related information which is available from existing fire protection inspection/testing results could form the technical foundation for programmatic revisions that are cost effective and provide for a reduction in regulatory burden. Provided that the NRC is receptive to performance-based programs which comply with all regulatory requirements and licensees feel that they will not be penalized for adopting such beneficial internal program changes, enhancements to plant fire protection program flexibility can be achieved without the need for a major revision to the existing rules and regulations.

IV. SUMMARY

Since 10 CFR 50, Appendix R, contains certain prescriptive requirements that could be replaced by performance-based criteria, the NRC should continue to address industry concerns with the regulations through the Marginal-to-Safety Program. Licensees are not currently taking advantage of all tne programmatic flexibility already afforded them in the fire protection area. Licensees should review their own programs to identify those details and prior commitments that are not based in the regulations and do not appear to be providing safety benefits commensurate wit' the costs. While current NRC guidance and standard operating license conditions allow licensees to exercise some flexibility in effecting program changes that do not reduce safety, the NRC could enhance flexibility by eliminating or relocating the current fire protection license condition and clarifying its intent to allow licensees to update their programs to take additional advantage of the present-day advances in technology, along with any lessons learned from historical performance data. The ultimate decision to exercise such flexibility then rests with licensee management. It was noted during staff comments that the Office of Nuclear Reactor Regulation performed a reassessment of the fire protection program. The following recommendations provide some insight into what further action the Review Group believes can be initiated or continued by the NRC to assist licensees in assessing the efficacy of implementing justifiable fire protection program revisions and remove any barriers to change which may be perceived to stand in the way of the realization of enhanced programmatic flexibility and should be evaluated in context within that reassessment.

V. RECOMMENDATIONS

- Eliminate the operating license condition governing fire protection programs. (This is consistent with a recommendation discussed in Volume Three of the Regulatory Review Group Report.)

  In accordance with the guidance of Generic Letters 86-10 and 88-12, this standard license condition has been deemed necessary to complement the relocation of fire protection requirements from the Technical Specification to licensee documents, controlled as part of the final safety analysis report. The intent of this NRC guidance is to further the goal of the Technical Specification Improvement process without reducing the level of fire safety. While specific NRC recommendations in this regard have, for the most part, enhanced licensee flexibility in this area, some reluctance to make major fire protection program changes that may be justified by present-day technology and plant-specific situations is evident. In practice, the subject license condition discourages major program revisions, even when the resultant net effect of the changes neither reduces safety nor violates any regulations. This situation can be attributed to an interpretation problem where any program reduction at all might be perceived to "adversely affect" safe shutdown during a fire. While such problems of interpretation could be addressed with clarification and/or additional guidance, a more direct approach would be the elimination altogether of the cause of the ambiguity. Consistent with the regulations, the need to obtain prior NRC approval of certain fire protection program revisions could be satisfied without the mandate of an operating license condition.

  The Review Group has developed a proposed rulemaking to allow licensees to change the fire protection plan without NRC approval if the proposed changes do not reduce the program below the regulatory requirements and would delete the need for a license amendment. The detailed proposed rulemaking is located in Appendix A to Volume One.

- Revise the current NRC guidance documents, including inspection procedures to clarify that while the various fire protection codes and standards referenced therein specify practices accepted by the NRC, other alternative methods of compliance with the regulations can be developed. The revised guidance should emphasize that the use of performance-based criteria may provide an equally acceptable, plant-specific way to meet the intended requirement.

Licensees currently have considerable flexibility to initiate changes to various aspects of their fire protection programs, but appear reluctant to exercise this flexibility if a deviation from an accepted code or standard (e.g., NFPA) is involved. The use of performance-based experience and criteria to revise program commitments, while not prohibited by the regulations, is not routinely adopted by licensees because of the perception that a departure from the accepted norm is not favored by the NRC staff, including field inspectors. While current NRC guidance may be intended to be value-neutral in terms of endorsement, it is not interpreted as such by the industry. Licensee efforts to relax unreasonable requirements by revising existing fire protection commitments are not likely to be pursued unless the NRC places additional emphasis upon the acceptability of soundly based alternative practices and until such a position is reinforced within the NRC staff.

- Continue to devote an appropriate level of NRC resources to the Marginal-to-Safety Program in an effort to address industry concerns with the current fire protection regulations.

  While it is recognized that certain requirements in 10 CFR 50, Appendix R (e.g., Section III.J) are prescriptive, other provisions (e.g., Section III.L) might be considered reasonably performance-based. A correct balance in the regulations between the appropriate technical criteria and a degree of performance-based flexibility can be achieved through a negotiated process between the industry and the NRC staff. Given the wide range of concerns expressed by several licensees relative to fire protection costs and efficiency issues, it would be most appropriate for the NRC to direct its resources to the review of those areas where regulatory change appears to have the greatest savings potential with little or no adverse safety impact. It is therefore incumbent upon licensees to communicate not only their dissatisfaction with the prescriptive nature of specific regulations, but also the positive potential and expected benefits believed to be achievable through the implementation of more flexible provisions which could supplant the current regulatory requirements.

ATTACHMENT

(Existing Operating License Condition)

Fire Protection

(Licensee) shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated (XX-XX-XX) as supplemented subject to the following provision:

> (Licensee) may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

2.3.5 FITNESS FOR DUTY

I. INTRODUCTION

This paper collates the miscellaneous observations and findings of the review of the rules concerning fitness for duty, 10 CFR Part 26. The fitness-for-duty part of the regulations is relatively new. Its relationship to safety is by inference (i.e., safety is enhanced by the elimination of individuals who are not "fit for duty"). The Part 26 rules contain several provisions that mandate reports, records, or other requirements that are not directly related to safety.

II. BEHAVIORAL OBSERVATION

Both Part 26 and Part 73 have requirements to identify and to deal with aberrant behavior in personnel granted unescorted access to the protected area of a nuclear power plant. Aberrant behavior may be induced by a variety of causes, such as illegal substances (e.g., various drugs), legal substances (e.g., alcohol and some medications), and personality or emotional problems (e.g., inability to cope with stress in a person's personal life). Section 26.22 establishes a requirement to train (and to retrain at nominal 12-month intervals) supervisors, managers, and escorts (inter alia) in behavioral observation. Section 73.56 makes behavioral observation a key element of access authorization. The drug testing part of these similar requirements was implemented first, resulting in the behavioral observation training being strongly focused on the Part 26 substance abuse problem.

The review group did note that there was a recent instance (documented in Inspection Report 50-445/92-50;50-446/92-50) in which the extensive inspection of an incident clearly indicated that the focus of a particular licensee's training was on aberrant behavior that was the result of substance abuse. In this incident, there were at least two precursor instances of aberrant behavior observed and reported. Neither instance was drug related, and essentially no remedial action was taken. The inspection report noted problems with the behavioral observation training (failure to train or retrain all required supervisors) and speculated that the licensee may have focused the behavioral observation training on substance abuse while not effectively instructing on other behavioral problems.

This is not viewed as a problem with the regulations. Since there was at least one licensee who did not provide balanced training and indications from the responsible NRR branch are that this problem may exist in other licensees' programs, the Review Group recommends that consideration be given to issuing an appropriate generic communication (e.g., an information notice) to bring this to the attention of all licensees.

III. REPORTING REQUIREMENTS

The recordkeeping section of Part 26 [26.71(d)] contains a reporting requirement to submit extensive data concerning drug testing at 6-month intervals. This requirement is in addition to the reporting requirements delineated in Section 26.73. NUMARC stated in a letter (J. Colvin to I. Selin, dated 12/23/92) that the 6-month reporting requirement was burdensome and suggested that an annual reporting frequency would be sufficient. NRC publishes the summarized data at annual intervals.

The Review Group recommends that the reporting frequency be changed from semi-annual to annual and that 10 CFR 26.71[d] be changed as follows:

> Collect and compile fitness-for-duty program performance data on a standard form for each calendar year, and submit this data to the Commission by March 1 of the following year. The data for each site (corporate and other support staff locations may be consolidated) must include: random testing rate; drugs tested for and cut-off levels, including tests using lower cutoff levels and tests for other drugs; workforce populations tested; numbers of tests and results by population and type [i.e., pre-access, random, for-cause, etc.]; substances identified; summary of management actions; and a list of events reported. The data must be analyzed and appropriate actions taken to correct program weaknesses. The data and analysis must be retained for 3 years. Any licensee choosing to suspend individuals temporarily under the provisions of 26.24[d] must report test results by process stage (i.e., onsite screening, laboratory screening, confirmatory test, and MRO determinations) and the number of temporary suspensions or other administrative actions taken against individuals based on onsite, unconfirmed screening positive tests for marijuana (THC) and for cocaine.

IV. SAMPLE FREQUENCY

The Review Group noted that there was a proposed rulemaking under development to change the rate of random testing from 100 percent to 50 percent for licensee employees while maintaining a 100 percent rate for contractors granted access. The proposed rulemaking was published for comment in the Federal Register. The comment period expired June 22, 1993. The staff is currently evaluating the comments and is scheduled to have a final rulemaking to the EDO in late 1993.

V. AUDITS

Audits of the fitness-for-duty program are required at nominal 12-month intervals by 10 CFR 26.80. It would appear that this was a reasonable requirement during the initial implementation of the program. However, with fitness-for-duty programs now operating at nuclear power plants (under prescriptive guidelines), it appears appropriate to reduce this requirement. The Review Group recommends that 10 CFR 26.80[a] be revised as follows:

> Each licensee subject to this Part shall audit the fitness-for-duty program as necessary but at intervals not to exceed 3 years. In addition, audits must be conducted, nominally every 18 months, of those portions of fitness-for-duty programs implemented by contractors or vendors. Licensees may accept audits of contractors or vendors conducted by other licensees and need not re-audit the same contractor or vendor for the same period of time. Each sharing utility shall maintain a copy of the audit report, to include findings, recommendations, and corrective actions. Licensees retain responsibility for the effectiveness of contractor and vendor programs and the implementation of appropriate corrective action.

VI. RECOMMENDATIONS

As a result of the review of Part 26 of the regulations and staff comments, the Review Group recommends:

- That regulations specifying reporting requirements of drug testing data be modified as delineated above.

- That audit frequency be allowed to be increased based on sustained satisfactory performance.

VII. POTENTIAL IMPROVEMENTS

That an information notice or other suitable generic communication be issued to remind licensees that their training and retraining in behavioral observation should not focus solely on aberrant behavior resulting from substance abuse. This information notice should also provide guidance on what the staff considers to be appropriate action for aberrant behavior that is not substance induced.

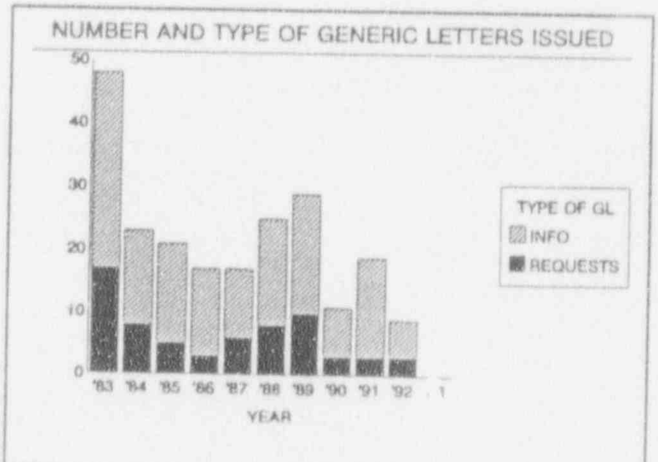
2.3.6 GENERIC COMMUNICATIONS

I. INTRODUCTION

The Review Group reviewed generic letters and bulletins to determine what types of actions were requested and whether the actions were in fact treated as regulatory requirements and therefore should have been codified in rulemakings. Generic letters, bulletins, and supplements issued from 1983 to 1993 were reviewed. Before 1987, generic letters were issued by the Office of Nuclear Reactor Regulation (NRR) and bulletins were issued by the Office of Inspection and Enforcement (I&E). In 1987, the two offices combined and since 1987 both types of generic communications have been issued by NRR.

II. GENERIC LETTERS

There were a total of 219 generic letters and supplements to generic letters reviewed. A brief description of each generic letter that contained a request is included as Enclosure 1. A complete list of reviewed generic letters is included as Enclosure 2. (If a supplement to a generic letter was issued, it was counted as occurring in the same year as the original generic letter.) Generic letters that set up meetings or workshops, discussed line-item Technical Specification improvements, provided information copies of NUREGs or reports, or identified voluntary programs were considered "information only" generic letters. Generic letters which request operator licensing examination schedules were also included as "information only" type communications because these do not impose a burden for licensees even though they request information. There were 153 of this type of generic letter issued in the 10-year time span. The other generic letters requested that actions be taken. These included requests to provide schedules of completed actions, requests for actions to be taken or programs to be developed. There were 66 of these generic letters.



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In direct response to comments on the process governing generic communications and the Commission's staff requirements memorandum, dated December 20, 1991, the process described initially in SECY-91-172 and later in SECY-92-224 was promulgated. The staff is currently developing guidance to implement and describe the procedure now being followed.

Under these procedures, each generic communication that goes to the Committee for Review of Generic Requirements (CRGR) is issued for public comment. The value/impact aspects of the proposed generic communication are included in the package sent to the public document room. There is also a required paragraph in the generic communication itself that requests comments on the burden (both in terms of staff hours and hardware cost) of the implementation of the request. When the package is sent to CRGR, the package is also sent to the Advisory Committee on Reactor Safeguards (ACRS). If the issue is of such urgent safety significance that the generic communication must be promulgated immediately, a paragraph is added to the generic communication stating that technical comments are welcome; a Federal Register notice is subsequently issued to notify the public. The staff must perform a backfit analysis or identify it as a compliance backfit to a specific regulation. This analysis is also discussed in the generic communication and the CRGR package. The cumulative impact of generic communications on licensee resources must also be considered in the cost-benefit analysis. After the public comments are incorporated into the final generic communication, an information paper is sent to the Commission informing them of the staff's intent to issue a new generic communication. The Commission has at least 1 week to comment. The staff also issues a biweekly letter listing all generic letters, bulletins, and information notices being worked on by the staff.

As seen in Figures 1 and 2, the scope of the problem in the 1983-1993 period has changed. The total number of generic letters and bulletins has been reduced and, more importantly, the number requesting action has been significantly reduced. However, under the current process there are three mechanisms to provide information to licensees--generic letters, bulletins, and information notices. Generic letters and bulletins convey information with broad implications, such as issuance of a NUREG, while information notices provide information that is more limited in scope, such as information on an event. There are two mechanisms to request information or action from licensees--generic letters and bulletins. As discussed in the introduction, generic letters were issued by NRR and bulletins by I&E. When the offices of NRR and I&E combined, all types of generic communications were issued by NRR. There appears to be no benefit from the use of a system with different types of generic communications that relay the same types of information and in fact may lead to confusion on the part of the staff--not knowing which mechanism to issue--and on the part of the licensees--not knowing which one will forward information or request action. A system that differentiates between communications that request actions and provide information or a simplified system of one type of generic communication to request information or action and one type to forward information may lead to better tracking of requests and actions and less confusion for the staff and licensees.

Based on experience from the Office of the General Counsel (OGC) reviews conducted over the last several years, the staff should ensure that, in developing generic communications, the distinction is drawn between backfits that involve compliance with existing staff interpretations of existing rules and those that involve new or different staff interpretations of existing rules. The former do not require a backfit analysis whereas the latter do. However, a finding that the former are involved must be documented in accordance with 10 CFR 50.109(a)(4). Moreover, when a generic modification or addition that is to be legally enforceable is proposed, it should be imposed by rulemaking. Otherwise, it must be imposed by a separate order to each individual licensee who did not voluntarily adopt it, and each licensee so ordered would have the right to a hearing.

V. RECOMMENDATION

The Review Group found that the improved procedures that request public comments on the technical and cost-benefit aspects of the generic communication adequately addressed the previous concerns regarding requests for information contained in generic communications. Therefore, although it is a topic to be evaluated in the future in the Marginal-to-Safety Program, 10 CFR 50.54(f), which governs requests for information, appears to be applied in an appropriate manner in issuing generic communications which merely request information and neither it nor the generic communication process needs to be changed for requesting information.

VI. POTENTIAL IMPROVEMENTS

Develop a generic communication system that differentiates between generic communications that request action or provide information. This would lead to a less complicated system for tracking requests and actions and would create less confusion among licensees and the staff. Office level guidance would implement this change in procedure.

It should be noted that NRR has implemented the use of an administrative letter to replace generic letters when only providing information or administrative in nature.

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VII. ANALYSIS OF PUBLIC COMMENTS

Several of the commenters disagreed with the Review Group's recommendation that 10 CFR 50.54(f) did not need revision. Alternatives to the current process were proposed and should be evaluated by the staff in the long-term to determine whether further improvements in the process can be achieved.

One commenter advocated that when the NRC's Committee for Review of Generic Requirements (CRGR) reviews a proposed rule or request for information (i.e., a draft Generic Letter) that all implementing documents such as Inspection Procedures, Temporary Instructions, etc., be reviewed simultaneously in order to assure they do not change, modify, expand, or otherwise enlarge the intent or requirements of the proposed rule or generic letter. The Review Group believes that this is a good recommendation. It appears reasonable to conclude that, if a problem is understood well enough to promulgate a rule or a regulatory position, then the planned inspection requirements should be identifiable. It is recommended that the validation or inspection requirements (e.g., Temporary Instructions) for generic letters and other new issues be reviewed b[illegible] CRGR simultaneously with the review of the generic letter or propo[illegible]

One of the public comment letters stated that th[illegible]kly generic communication listing could be issued monthly without impa[illegible] the industry while possibly reducing NRC burden.

Generic Letters That Requested Action or Information

GL 83-02 - NUREG-0737 Technical Specifications

Implement TS from NUREG-0737. These include limiting overtime, hydrogen penetrations, reporting valve failures, RCIC start and suction, isolation of HPCI and RCIC, interlock on recirculation pump loops, common reference level for all setpoints.

GL 83-08 - Modification of Vacuum Breakers on Mark I Containments

Provide a commitment to submit results of plant calculations which call for modifying vacuum breakers or a JCO and the schedules.

GL 83-10A - Resolution of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps

Currently approved model for small-break LOCAs is acceptable, submit plans and schedule of implementation.

GL 83-10B, GL 83-10C, GL 83-10D, GL 83-10E, GL 83-10F - same as above for different reactor types

GL 83-11 - Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions

When submitting licensee-performed reloads, submit code verification performed by the plant to ensure plant-performed reload is consistent with vendor's.

GL 83-15 - Implementation of Regulatory Guide 1.150, Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations, Revision 1

Staff recommends changing the TS to be consistent with the regulatory guide.

GL 83-18 - NRC Staff Review of the BWR Owners' Group (BWROG) Control Room Survey Program

Submit program referencing generic BWROG program, including the qualification of team members, deviations from the program, prioritization of HEDs, reporting of DCRDR results and implementation of control room enhancements, complete the checklist, prioritize actions and provide schedule, repeat part of the task analysis using EOPs, update the operating experience review.

GL 83-24 - TMI Task Action Plan Item I.G.1, Special Low-Power Testing and Training
Respond to letter stating adverse impact of station blackout test and that licensee will comply with BWROG recommendations.

GL 83-26 - Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests
Letter requested changes to surveillance TS.

GL 83-28 - Required Actions Based on Generic Implications of Salem ATWS Events
Submit the status of conformance to the positions, plans, and schedules.

GL 83-36 and GL 83-37 - NUREG-0737 Technical Specifications
Review TS to guidance, where there are deviations submit a TS change. These include: reactor coolant system vents, post-accident sampling, noble gas effluent monitors, sampling and analysis of plant effluents, containment high-range radiation monitor, containment pressure monitor, containment water level monitor, containment hydrogen monitor, control room habitability, and for PWRs, submit a long-term aux feedwater system evaluation.

GL 83-43 - Reporting Requirements of 10 CFR Part 50, Section 50.72 and 50.73 and Standard Technical Specifications
Requested licensees to change TS to be consistent with the rule.

GL 84-07 - Procedural Guidance for Pipe Replacement at BWRs
Even if replacement of recirculation pipe is not an unreviewed safety question, send in the radiation protection plan, including preplanning procedures, shielding, equipment, training, and estimated total cumulative dose of the replacement job.

GL 84-09 - Recombiner Capability Requirements of 10 CFR 50.44
Submit a report on whether criteria outlined are met.

GL 84-10 - Administration of Operating Tests Prior To Initial Criticality
If operators don't have sufficient experience, an exemption is required.

GL 84-11 - Inspection of BWR Stainless Steel Piping
If the outlined actions are followed it would constitute an acceptable response. Licensees requested to submit inspection and leak detection plans, schedule, surveillance methods, results of IEB 83-02 inspection, and remedial measures.

GL 84-14 - Replacement and Requalification Training Program
Requested licensees to include in FSAR update the current requalification program or reference the submittal of program to NRC.

GL 84-15 - Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability
Provide description of program, reliability of each diesel, and a description of program to maintain reliability.

GL 84-23 - Reactor Vessel Water Level Instrumentation in BWRs
Submit the plans to implement the recommended improvements (improvements that will reduce level errors, replace mechanical level indication equipment with analog).

GL 84-24 - Certification of Compliance to 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants
Requested licensees to submit certification that EQ program is in place and there is one path to safe shutdown using fully qualified equipment and that all other equipment is qualified or has a JCO.

GL 85-02 - Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity
Submit program for ensuring steam generator tube integrity and rupture mitigation is taken care of--staff will compare to the recommended actions.

GL 85-09 and GL 85-10 - Technical Specifications for Generic Letter 83-28, Item 4.3
Submit TS changes that are responsive to the guidance.

GL 85-12, 86-05 and GL 86-06 - Implementation of TMI Action Item II.K.3.5 Automatic Trip of Reactor Coolant Pumps
Implement criterion and schedules. If don't follow owner's group, submit plant-specific trip criteria or justifications.

GL 85-20 - Resolution of Generic Issue 69, High Pressure Injection/Make-Up Nozzle Cracking in Babcock and Wilcox Plants
Verify a valid stress analysis has been performed for HPI/MU nozzles and determine the cumulative usage.

GL 86-04 - Policy Statement on Engineering Expertise on Shift
Respond to GL stating a program for engineering expertise is in place, determine equivalency for engineering degree criteria.

GL 87-02 and Supplement 1 - Verification of Seismic Adequacy of Mechanical and Electric Equipment in Operating Reactors Unresolved Safety Issue (USI) A-46
Requested licensees to send in the schedule for implementation of seismic

verification program. Requested a response whether the licensee has committed to the SQUG and the SSER, the schedule and what procedures and criteria were used to generate response.

GL 87-03 - Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46

Backfit to review plants to current seismic criteria.

GL 87-05 - Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells

Provide plans for determining if drain lines are unplugged and functioning preventive maintenance and inspection program, and provide plans for UT.

GL 87-06 - Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves

Submit a list of all pressure isolation valves, and describe the tests and frequency.

GL 87-12 - Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled

Submit a description of operations during the approach to a partially filled RCS and the operations with a partially filled RCS.

GL 88-01 - NRC Positions on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping

Request information on current plans on piping replacement, inspection, repair, and leak detection and whether you plan to follow the staff positions. Staff positions include: long-term piping integrity (replacement), ISI, change to TS to say ISI will conform to the staff positions, TS change on leakage detection, and notify NRC of any flaws.

GL 88-03 - Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps

Provide assurance that a program has been implemented, including places where leaks could cause degradation of pressure boundary, identifying small leaks, how to conduct examinations and evaluations, and the corrective action - new staff position.

GL 88-05 - Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWRs

Provide assurance that a program has been implemented, including places where leaks could cause degradation of pressure boundary, identifying small leaks, how to conduct exams and evaluations, and corrective actions - new staff position.

GL 88-14 - Instrument Air Supply System Program Affecting Safety Related Equipment
Review NUREG and perform a design and operations verification, including air quality maintenance, training, and verify design will function under design basis conditions.

GL 88-17 - Loss of Decay Heat Removal
Description of actions taken to implement recommended actions, description of enhancements, plans and schedules for the enhancements.

GL 88-20, Supplements 1 and 4 - Individual Plant Examination for Severe Accident Vulnerabilities
Requested to perform an IPE and submit results. Submit program and schedule. IPE should also address external events.

GL 89-06 - Task Action Plan Item I.D.2 - Safety Parameter Display System
Requested licensees to assess SPDS and give status. Certify it is complete and meets NUREG-0797; if not describe compensatory measures.

GL 89-07 - Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs
Modify safeguards contingency plans to address land bombs. Confirm it is in plan. Determine safe standoff, review site features, short-term measures, and plans and procedures.

GL 89-08 - Erosion/Corrosion-Induced Pipe Wall Thinning
Provide information on whether or not a long-term program is in place, if not provide schedule.

GL 89-10 and Supplement 3 - Safety Related Motor Operated Valve Testing
Extends the scope of Bulletin 85-03 to all safety-related MOVs, position changeable MOVs, and as part of a good program it should include balance of plant MOVs that are important to safety. Assess capability of HPCI, RCIC, and RWCU. Submit criteria used, MOVs identified, schedule for corrective action, and inform NRC of changes.

GL 89-13 - Service Water System Problems Affecting Safety Related Equipment
Perform the actions outlined in GL or equally effective actions: identify biofouling, heat transfer of heat exchangers, inspection and maintenance, confirm maintenance ensures it will perform.

GL 89-16 - Installation of Hardened Wetwell Vent
Request licensees to volunteer to put it in; if not, submit cost so the staff can do a cost-benefit analysis to determine if a backfit is needed.

GL 89-19 - Request for Action Related to Resolution of Unresolved Safety Issue A-47, Safety Implication of Control Systems in LWR Nuclear Plants

Submit information whether the recommendations will be performed and the schedule. If not provide justification. Provide overfeed protection and TS changes.

GL 89-20 - Protected Area Long-Term Housekeeping

Submit report on site conditions: isolation zones, protected areas, construction, waste, scrap storage, vehicle storage, or other things that reduce illumination.

GL 89-21 - Request for Information Concerning Status of Implementation of Unresolved Safety Issue (USI) Requirements

Requested licensees to review their status of USIs and send in report.

GL 90-03 - Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2, Vendor Interface for Safety Related Components

Review and modify program to meet the GL; notify it has been done - relaxation.

GL 90-04 - Request for Information on the Status of Licensee Implementation of Generic Safety Issues Resolved with Imposition of Requirements or Corrective Actions

Review status of GSI and submit report.

GL 90-06 - Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability and Generic Issue 94, Additional Low Temperature Overpressure Protection for LWRs

Submit whether a commitment is made to the actions and TS changes.

GL 91-06 - Resolution of Generic Issue A-30, Adequacy of Safety Related DC Power Supplies, Pursuant to 10 CFR 50.54(f)

Request for information by completing a form.

GL 91-11 - Resolution of Generic Issues 48, LCOs for Class 1E Vital Instrument Buses, and 49, Interlocks and LCOs for Class 1E Tie Breakers Pursuant to 10 CFR 50.54(f)

Certify implementation of procedures or have a justification.

GL 91-13 - Request for Information Related to the Resolution of Generic Issue 130, Essential Service Water System Failures at Multi-Unit Sites Pursuant to 50.54(f)

Review TS and procedures, evaluate the applicability of changes, and submit TS changes if required.

GL 92-01, Revision 1 - Reactor Vessel Structural Integrity

Information requested on actions to comply with Appendix H and any exemptions, predicted energy, results of tests, heat treatments, and heat numbers, chemical composition, and effects of temperature surveillances.

GL 92-04 - Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs

Requested information on impact and short-term actions, and plans and schedules for corrective actions.

GL 92-08 - Thermo-Lag Fire Barriers

Confirm barriers have been qualified, ampacity derating factors are valid, barriers were installed with correct procedures and QA. Is thermo-lag relied upon to meet 50.48, whether qualified, whether installed meets qualification, whether as installed meets ampacity configuration, corrective actions and compensatory measures, list of barriers that can't be determined.

Enclosure 2

DIRECTORY OF NRC GENERIC LETTER FILES

I = Generic Letters which provide information

R = Generic Letters which request information

I	Generic Letter 83-01 - OPERATOR LICENSING EXAMINATION SITE VISIT
R	Generic Letter 83-02 - NUREG-0737 TECHNICAL SPECIFICATIONS
I	Generic Letter 83-04 - REGIONAL WORKSHOPS REGARDING SUPPLEMENT 1 TO NUREG-0737 REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY
I	Generic Letter 83-05 - SAFETY EVALUATION OF "EMERGENCY PROCEDURE GUIDELINES, REVISION 2," NED0-24934, JUNE 1982
I	Generic Letter 83-06 - CERTIFICATES AND REVISED FORMAT FOR REACTOR OPERATOR AND SENIOR REACTOR OPERATOR LICENSES
I	Generic Letter 83-07 - THE NUCLEAR WASTE POLICY ACT OF 1982
R	Generic Letter 83-08 - MODIFICATION OF VACUUM BREAKERS ON MARK I CONTAINMENTS
I	Generic Letter 83-09 - REVIEW OF COMBUSTION ENGINEERING OWNERS' GROUP EMERGENCY PROCEDURES GUIDELINE PROGRAM
R	Generic Letter 83-10A - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"
R	Generic Letter 83-10B - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"
R	Generic Letter 83-10C - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

R Generic Letter 83-10D - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

R Generic Letter 83-10E - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

R Generic Letter 83-10F - RESOLUTION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

R Generic Letter 83-11 - LICENSEE QUALIFICATION FOR PERFORMING SAFETY ANALYSES IN SUPPORT OF LICENSING ACTIONS

I Generic Letter 83-12 - ISSUANCE OF NRC FORM 398 - PERSONAL QUALIFICATIONS STATEMENT - LICENSEE

I Generic Letter 83-12A - ISSUANCE OF NRC FORM 398 - PERSONAL QUALIFICATIONS STATEMENT - LICENSEE

I Generic Letter 83-13 - CLARIFICATION OF SURVEILLANCE REQUIREMENTS FOR HEPA FILTER AND CHARCOAL ABSORBER UNITS IN STANDARD TECHNICAL SPECIFICATIONS ON ESF CLEANUP SYSTEMS

I Generic Letter 83-14 - DEFINITION OF "KEY MAINTENANCE PERSONNEL," (CLARIFICATION OF GENERIC LETTER 82-12)

R Generic Letter 83-15 - IMPLEMENTATION OF REGULATORY GUIDE 1.150, "ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS," REVISION 1

I Generic Letter 83-16 - TRANSMITTAL OF NUREG-0977 RELATIVE TO THE ATWS EVENTS AT SALEM GENERATING STATION, UNIT NO. 1

I Generic Letter 83-16A - TRANSMITTAL OF NUREG-0977 RELATIVE TO THE ATWS EVENTS AT SALEM GENERATING STATION, UNIT NO. 1

I Generic Letter 83-17 - INTEGRITY OF THE REQUALIFICATION EXAMINATION FOR RENEWAL OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR LICENSES

R Generic Letter 83-18 - NRC STAFF REVIEW OF THE BWR OWNERS' GROUP (BWROG) CONTROL ROOM SURVEY PROGRAM

I Generic Letter 83-19 - NEW PROCEDURES FOR PROVIDING PUBLIC NOTICE CONCERNING ISSUANCE OF AMENDMENTS TO OPERATING LICENSES

I Generic Letter 83-20 - INTEGRATED SCHEDULING FOR IMPLEMENTATION OF PLANT MODIFICATIONS

I Generic Letter 83-21 - CLARIFICATION OF ACCESS CONTROL PROCEDURES FOR LAW ENFORCEMENT VISITS

I Generic Letter 83-22 - SAFETY EVALUATION OF "EMERGENCY RESPONSE GUIDELINES"

I Generic Letter 83-23 - SAFETY EVALUATION OF "EMERGENCY PROCEDURE GUIDELINES"

R Generic Letter 83-24 - TMI TASK ACTION PLAN ITEM I.G.1, "SPECIAL LOW POWER TESTING AND TRAINING," RECOMMENDATIONS FOR BWRs

R Generic Letter 83-26 - CLARIFICATION OF SURVEILLANCE REQUIREMENTS FOR DIESEL FUEL IMPURITY LEVEL TESTS

I Generic Letter 83-27 - SURVEILLANCE INTERVALS IN STANDARD TECHNICAL SPECIFICATIONS

R Generic Letter 83-28 - REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

I Generic Letter 83-28 Sup. 1 - "REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS"

I Generic Letter 83-30 - DELETION OF STANDARD TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.8.1.1.2.d.6 FOR DIESEL GENERATOR TESTING

I Generic Letter 83-31 - SAFETY EVALUATION OF 'ABNORMAL TRANSIENT OPERATING GUIDELINES'

I Generic Letter 83-32 - NRC STAFF RECOMMENDATIONS REGARDING OPERATOR ACTION FOR REACTOR TRIP AND ATWS

I Generic Letter 83-33 - NRC POSITIONS ON CERTAIN REQUIREMENTS OF APPENDIX R TO 10 CFR 50

I Generic Letter 83-35 - CLARIFICATION OF TMI ACTION PLAN ITEM II.K.3.31

R Generic Letter 83-36 - NUREG-0737 TECHNICAL SPECIFICATIONS

R Generic Letter 83-37 - NUREG-0737 TECHNICAL SPECIFICATIONS

I Generic Letter 83-38 - NUREG-0965, "NRC INVENTORY OF DAMS"

I Generic Letter 83-39 - VOLUNTARY SURVEY OF LICENSED OPERATORS

I Generic Letter 83-40 - OPERATOR LICENSING EXAMINATIONS

I Generic Letter 83-41 - FAST COLD STARTS OF DIESEL GENERATORS

I Generic Letter 83-42 - CLARIFICATION TO GENERIC LETTER 81-07 REGARDING RESPONSE TO NUREG-0612, "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS"

R Generic Letter 83-43 - REPORTING REQUIREMENTS OF 10 CFR PART 50, SECTIONS 50.72 AND 50.73, AND STANDARD TECHNICAL SPECIFICATIONS

I Generic Letter 83-44 - AVAILABILITY OF NUREG-1021, "OPERATOR LICENSING EXAMINER STANDARDS"

I Generic Letter 84-01 - NRC USE OF THE TERMS, "IMPORTANT TO SAFETY' AND "SAFETY RELATED"

I Generic Letter 84-02 - NOTICE OF MEETING REGARDING FACILITY STAFFING

I Generic Letter 84-03 - AVAILABILITY OF NUREG-0933, "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

I Generic Letter 84-04 - SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

I Generic Letter 84-05 - CHANGE TO NUREG-1021, "OPERATOR LICENSING EXAMINER STANDARDS"

I Generic Letter 84-06 - OPERATOR AND SENIOR OPERATOR LICENSE EXAMINATION CRITERIA FOR PASSING GRADE

R Generic Letter 84-07 - PROCEDURAL GUIDANCE FOR PIPE REPLACEMENT AT BWRS

I Generic Letter 84-08 - INTERIM PROCEDURES FOR NRC MANAGEMENT OF PLANT-SPECIFIC BACKFITTING

R Generic Letter 84-09 - RECOMBINER CAPABILITY REQUIREMENTS OF 10 CFR 50.44 (c)(3)(ii)

R Generic Letter 84-10 - ADMINISTRATION OF OPERATING TESTS PRIOR TO INITIAL CRITICALITY (10 CFR 55.25)

R Generic Letter 84-11 - INSPECTIONS OF BWR STAINLESS STEEL PIPING

I Generic Letter 84-12 - COMPLIANCE WITH 10 CFR PART 61 AND IMPLEMENTATION OF THE RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) AND ATTENDANT PROCESS CONTROL PROGRAM (PCP)

I Generic Letter 84-13 - TECHNICAL SPECIFICATION FOR SNUBBERS

R Generic Letter 84-14 - REPLACEMENT AND REQUALIFICATION TRAINING PROGRAM

R Generic Letter 84-15 - PROPOSED STAFF ACTIONS TO IMPROVE AND MAINTAIN DIESEL GENERATOR RELIABILITY

I Generic Letter 84-16 - ADEQUACY OF ON-SHIFT OPERATING EXPERIENCE FOR NEAR TERM OPERATING LICENSE APPLICANTS

I Generic Letter 84-17 - ANNUAL MEETING TO DISCUSS RECENT DEVELOPMENTS REGARDING OPERATOR TRAINING, QUALIFICATIONS, AND EXAMINATIONS

I Generic Letter 84-18 - FILING OF APPLICATIONS FOR LICENSES AND AMENDMENTS

I Generic Letter 84-19 - AVAILABILITY OF SUPPLEMENT 1 TO NUREG-0933, "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

I Generic Letter 84-20 - SCHEDULING GUIDANCE FOR LICENSEE SUBMITTALS OF RELOADS THAT INVOLVE UNREVIEWED SAFETY QUESTIONS

I Generic Letter 84-21 - LONG TERM LOW POWER OPERATION IN PRESSURIZED WATER REACTORS

R Generic Letter 84-23 - REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs

R Generic Letter 84-24 - CERTIFICATION OF COMPLIANCE TO 10 CFR 50.49, ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

I Generic Letter 85-01 - FIRE PROTECTION POLICY STEERING COMMITTEE REPORT

R Generic Letter 85-02 - STAFF RECOMMENDED ACTIONS STEMMING FROM NRC INTEGRATED PROGRAM FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUES REGARDING STEAM GENERATOR TUBE INTEGRITY

I Generic Letter 85-03 - CLARIFICATION OF EQUIVALENT CONTROL CAPACITY FOR STANDBY LIQUID CONTROL SYSTEMS

I Generic Letter 85-04 - OPERATOR LICENSING EXAMINATIONS

I Generic Letter 85-05 - INADVERTENT BORON DILUTION EVENTS

I Generic Letter 85-06 - QUALITY ASSURANCE GUIDANCE FOR ATWS EQUIPMENT THAT IS NOT SAFETY-RELATED

I Generic Letter 85-07 - IMPLEMENTATION OF INTEGRATED SCHEDULES FOR PLANT MODIFICATIONS

I Generic Letter 85-08 - 10 CFR 20.408 TERMINATION REPORTS - FORMAT

R Generic Letter 85-09 - TECHNICAL SPECIFICATIONS FOR GENERIC LETTER 83-28, ITEM 4.3

R Generic Letter 85-10 - TECHNICAL SPECIFICATIONS FOR GENERIC LETTER 83-28, ITEMS 4.3 AND 4.4

I Generic Letter 85-11 - COMPLETION OF PHASE II OF "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS" NUREG-0612.

R Generic Letter 85-12 - IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

I Generic Letter 85-13 - TRANSMITTAL OF NUREG-1154 REGARDING THE DAVIS-BESSE LOSS OF MAIN AND AUXILIARY FEEDWATER EVENT

I Generic Letter 85-14 - COMMERCIAL STORAGE AT POWER REACTOR SITES OF LOW-LEVEL RADIOACTIVE WASTE NOT GENERATED BY THE UTILITY

I Generic Letter 85-15 - INFORMATION RELATING TO THE DEADLINES FOR COMPLIANCE WITH 10 CFR 50.49, "ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS"

I Generic Letter 85-16 - HIGH BORON CONCENTRATIONS

I Generic Letter 85-17 - AVAILABILITY OF SUPPLEMENTS 2 and 3 TO NUREG-0933, "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

I Generic Letter 85-18 - OPERATOR LICENSING EXAMINATIONS

I Generic Letter 85-19 - REPORTING REQUIREMENTS ON PRIMARY COOLANT IODINE SPIKES

R Generic Letter 85-20 - RESOLUTION OF GENERIC ISSUE 69: HIGH PRESSURE INJECTION/MAKE-UP NOZZLE CRACKING IN BABCOCK AND WILCOX PLANTS

I Generic Letter 85-22 - POTENTIAL FOR LOSS OF POST-LOCA RECIRCULATION CAPABILITY DUE TO INSULATION DEBRIS BLOCKAGE

I Generic Letter 86-01 - SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

I Generic Letter 86-02 - TECHNICAL RESOLUTION OF GENERIC ISSUE B-19-THERMAL HYDRAULIC STABILITY

I Generic Letter 86-03 - APPLICATIONS FOR LICENSE AMENDMENTS

R Generic Letter 86-04 - POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

R Generic Letter 86-05 - IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

R Generic Letter 86-06 - IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

I Generic Letter 86-07 - TRANSMITTAL OF NUREG-1190 REGARDING THE SAN ONOFRE UNIT 1 LOSS OF POWER AND WATER HAMMER EVENT

I Generic Letter 86-08 - AVAILABILITY OF SUPPLEMENT 4 TO NUREG-0933, "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

I Generic Letter 86-09 - TECHNICAL RESOLUTION OF GENERIC ISSUE NO. B-59-(N-1) LOOP OPERATION IN BWRs AND PWRs

I Generic Letter 86-10 - IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS

I Generic Letter 86-11 - DISTRIBUTION OF PRODUCTS IRRADIATED IN RESEARCH REACTORS

I Generic Letter 86-12 - CRITERIA FOR UNIQUE PURPOSE EXEMPTION FROM CONVERSION FROM THE USE Of HEU FUEL

I Generic Letter 86-13 - POTENTIAL INCONSISTENCY BETWEEN PLANT SAFETY ANALYSES AND TECHNICAL SPECIFICATIONS

I Generic Letter 86-14 - OPERATOR LICENSING EXAMINATIONS

I Generic Letter 86-15 - INFORMATION RELATING TO COMPLIANCE WITH 10 CFR 50.49, "ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS"

I Generic Letter 86-16 - WESTINGHOUSE ECCS EVALUATION MODELS

I Generic Letter 86-17 - AVAILABILITY OF NUREG-1169, "TECHNICAL FINDINGS RELATED TO GENERIC ISSUE C-8; BOILING WATER REACTOR MAIN STEAM ISOLATION VALVE LEAKAGE AND LEAKAGE TREATMENT METHODS"

I Generic Letter 87-01 - PUBLIC AVAILABILITY OF THE NRC OPERATOR LICENSING EXAMINATION QUESTION BANK

R Generic Letter 87-02 - VERIFICATION OF SEISMIC ADEQUACY OF MECHANICAL AND ELECTRICAL EQUIPMENT IN OPERATING REACTORS, UNRESOLVED SAFETY ISSUE

R Generic Letter 87-02 - VERIFICATION OF SEISMIC ADEQUACY OF MECHANICAL AND ELECTRICAL EQUIPMENT IN OPERATING REACTORS, UNRESOLVED SAFETY ISSUE (USI) A-46

R Generic Letter 87-02, Sup. 1 - SUPPLEMENT NO. 1 TO GENERIC LETTER (GL) 87-02 THAT TRANSMITS SUPPLEMENTAL SAFETY EVALUATION REPORT NO. 2 (SSER No. 2) ON SQUG GENERIC IMPLEMENTATION PROCEDURE, REVISION 2, AS CORRECTED ON FEBRUARY 14, 1992 (GIP-2)

R Generic Letter 87-03 - VERIFICATION OF SEISMIC ADEQUACY OF MECHANICAL AND ELECTRICAL EQUIPMENT IN OPERATING REACTORS, UNRESOLVED SAFETY ISSUE (USI) A-46

I Generic Letter 87-04 - TEMPORARY EXEMPTION FROM PROVISIONS OF THE FBI CRIMINAL HISTORY RULE FOR TEMPORARY WORKERS

R Generic Letter 87-05 - REQUEST FOR ADDITIONAL INFORMATION ASSESSMENT OF LICENSEE MEASURES TO MITIGATE AND/OR IDENTIFY POTENTIAL DEGRADATION OF MARK I DRYWELLS

R Generic Letter 87-06 - PERIODIC VERIFICATION OF LEAK TIGHT INTEGRITY OF PRESSURE ISOLATION VALVES

I Generic Letter 87-07 - INFORMATION TRANSMITTAL OF FINAL RULEMAKING FOR REVISIONS TO OPERATOR LICENSING - 10 CFR 55 AND CONFORMING AMENDMENTS

I Generic Letter 87-08 - IMPLEMENTATION OF 10 CFR 73.55 MISCELLANEOUS AMENDMENTS AND SEARCH REQUIREMENTS

I Generic Letter 87-09 - SECTIONS 3.0 AND 4.0 OF THE STANDARD TECHNICAL SPECIFICATIONS (STS) ON THE APPLICABILITY OF LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

I Generic Letter 87-10 - IMPLEMENTATION OF 10 CFR 73.57, REQUIREMENTS FOR FBI CRIMINAL HISTORY CHECKS

I Generic Letter 87-11 - RELAXATION IN ARBITRARY INTERMEDIATE PIPE RUPTURE REQUIREMENTS

R Generic Letter 87-12 - LOSS OF RESIDUAL HEAT REMOVAL (RHR) WHILE THE REACTOR COOLANT SYSTEM (RCS) IS PARTIALLY FILLED

I Generic Letter 87-13 - INTEGRITY OF REQUALIFICATION EXAMINATIONS AT NON-POWER REACTORS

I Generic Letter 87-14 - OPERATOR LICENSING EXAMINATIONS

I Generic Letter 87-15 - POLICY STATEMENT ON DEFERRED PLANTS

I Generic Letter 87-16 - TRANSMITTAL OF NUREG-1262, "ANSWERS TO QUESTIONS AT PUBLIC MEETING REGARDING IMPLEMENTATION OF TITLE 10, CODE OF FEDERAL REGULATIONS, PART 55 ON OPERATORS' LICENSES"

R Generic Letter 88-01 - NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

I Generic Letter 88-1 Sup. 1 - NRC POSITION ON INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) IN BWR AUSTENITIC STAINLESS STEEL PIPING

I Generic Letter 88-02 - INTEGRATED SAFETY ASSESSMENT PROGRAM II (ISAP II)

R Generic Letter 88-03 - RESOLUTION OF GENERIC SAFETY ISSUE 93, "STEAM BINDING OF AUXILIARY FEEDWATER PUMPS"

I Generic Letter 88-04 - DISTRIBUTION OF GEMS IRRADIATED IN RESEARCH REACTORS; SEE ALSO GENERIC LETTER 86-11, DATED JUNE 25, 1986

R Generic Letter 88-05 - BORIC ACID CORROSION OF CARBON STEEL REACTOR PRESSURE BOUNDARY COMPONENTS IN PWR PLANTS

I Generic Letter 88-06 - REMOVAL OF ORGANIZATION CHARTS FROM TECHNICAL SPECIFICATION ADMINISTRATIVE CONTROL REQUIREMENTS

I Generic Letter 88-07 - MODIFIED ENFORCEMENT POLICY RELATING TO 10 CFR 50.49, "ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS"

I Generic Letter 88-08 - MAIL SENT OR DELIVERED TO THE OFFICE OF NUCLEAR REACTOR REGULATION

I Generic Letter 88-09 - PILOT TESTING OF FUNDAMENTALS EXAMINATION

I Generic Letter 88-10 - PURCHASE OF GSA APPROVED SECURITY CONTAINERS

I Generic Letter 88-11 - NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS

I Generic Letter 88-12 - REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS

I Generic Letter 88-13 - OPERATOR LICENSING EXAMINATIONS

R Generic Letter 88-14 - INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY- RELATED EQUIPMENT

I Generic Letter 88-15 - ELECTRIC POWER SYSTEMS - INADEQUATE CONTROL OVER DESIGN PROCESSES

I Generic Letter 88-16 - REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM TECHNICAL SPECIFICATIONS

R Generic Letter 88-17 - LOSS OF DECAY HEAT REMOVAL

I Generic Letter 88-18 - PLANT RECORD STORAGE ON OPTICAL DISKS

I Generic Letter 88-19 - USE OF DEADLY FORCE BY LICENSEE GUARDS TO PREVENT THEFT OF SPECIAL NUCLEAR MATERIAL

R Generic Letter 88-20 - INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

R Generic Letter 88-20 Sup. 1 - INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

I Generic Letter 88-20 Sup. 2 - ACCIDENT MANAGEMENT STRATEGIES FOR CONSIDERATION IN THE INDIVIDUAL PLANT EXAMINATION PROCESS

I Generic Letter 88-20 Sup. 3 - COMPLETION OF CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM AND FORWARDING OF INSIGHTS FOR USE IN THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

R Generic Letter 88-20 Sup. 4 - INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES - 10CFR 50.54(f)

I Generic Letter 89-01 - IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR TO THE PROCESS CONTROL PROGRAM

I Generic Letter 89-01 Sup. 1 - NUREG-1301 - OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROLS FOR PRESSURIZED WATER REACTORS

I Generic Letter 89-02 - ACTIONS TO IMPROVE THE DETECTION OF COUNTERFEIT AND FRAUDULENTLY MARKETED PRODUCTS

I Generic Letter 89-03 - OPERATOR LICENSING NATIONAL EXAMINATION SCHEDULE

I Generic Letter 89-04 - GUIDANCE ON DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

I Generic Letter 89-05 - PILOT TESTING OF THE FUNDAMENTALS EXAMINATION

R Generic Letter 89-06 - TASK ACTION PLAN ITEM I.D.2 - SAFETY PARAMETER DISPLAY SYSTEM

R Generic Letter 89-07 - POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS

R Generic Letter 89-07 Sup. 1 - "POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS"

R Generic Letter 89-08 - EROSION/CORROSION-INDUCED PIPE WALL THINNING

I Generic Letter 89-09 - ASME SECTION III COMPONENT REPLACEMENTS

R Generic Letter 89-10 - SAFETY-RELATED MOTOR-OPERATED VALVE TESTING

I Generic Letter 89-10 Sup. 1 - RESULTS OF THE PUBLIC WORKSHOPS

I Generic Letter 89-10 Sup. 2 - "AVAILABILITY OF PROGRAM DESCRIPTIONS"

R Generic Letter 89-10 Sup. 3 - "CONSIDERATION OF THE RESULTS OF NRC-SPONSORED TESTS OF MOTOR-OPERATED VALVES"

I Generic Letter 89-10, Sup. 4 - CONSIDERATION OF VALVE MISPOSITIONING IN BOILING WATER REACTORS

I Generic Letter 89-11 - RESOLUTION OF GENERIC ISSUE 101, "BOILING WATER REACTOR WATER LEVEL REDUNDANCY"

I Generic Letter 89-12 - OPERATOR LICENSING EXAMINATIONS

R Generic Letter 89-13 - SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

I Generic Letter 89-13 Sup. 1 - SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

I Generic Letter 89-14 - LINE-ITEM IMPROVEMENTS IN TECHNICAL SPECIFICATIONS - REMOVAL OF THE 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS

I Generic Letter 89-15 - EMERGENCY RESPONSE DATA SYSTEM

R Generic Letter 89-16 - INSTALLATION OF A HARDENED WETWELL VENT

I Generic Letter 89-17 - PLANNED ADMINISTRATIVE CHANGES TO THE NRC OPERATOR LICENSING WRITTEN EXAMINATION PROCESS

I Generic Letter 89-18 - RESOLUTION OF UNRESOLVED SAFETY ISSUE A-17, "SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS"

R Generic Letter 89-19 - REQUEST FOR ACTION RELATED TO RESOLUTION OF UNRESOLVED SAFETY ISSUE A-47, "SAFETY IMPLICATION OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS"

R Generic Letter 89-20 - PROTECTED AREA LONG-TERM HOUSEKEEPING

R Generic Letter 89-21 - REQUEST FOR INFORMATION CONCERNING STATUS OF IMPLEMENTATION OF UNRESOLVED SAFETY ISSUE (USI) REQUIREMENTS

I Generic Letter 89-22 - POTENTIAL FOR INCREASED ROOF LOADS AND PLANT AREA FLOOD RUNOFF DEPTH AT LICENSED NUCLEAR POWER PLANTS DUE TO RECENT CHANGE IN PROBABLE MAXIMUM PRECIPITATION CRITERIA DEVELOPED BY THE NATIONAL WEATHER SERVICE

I Generic Letter 89-23 - NRC STAFF RESPONSES TO QUESTIONS PERTAINING TO IMPLEMENTATION OF 10 CFR PART 26

I Generic Letter 90-01 - REQUEST FOR VOLUNTARY PARTICIPATION IN NRC REGULATORY IMPACT SURVEY

I Generic Letter 90-02 - ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES IN DESIGN FEATURES SECTION OF TECHNICAL SPECIFICATIONS

I Generic Letter 90-02, Sup. 1 - ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES IN THE DESIGN FEATURES SECTION OF TECHNICAL SPECIFICATIONS

R Generic Letter 90-03 - RELAXATION OF STAFF POSITION IN GENERIC LETTER 83-28, ITEM 2.2 PART 2 "VENDOR INTERFACE FOR SAFETY-RELATED COMPONENTS"

I Generic Letter 90-03 Sup. 1 - RELAXATION OF STAFF POSITION IN GENERIC LETTER 83-28, ITEM 2.2 PART 2, "VENDOR INTERFACE FOR SAFETY-RELATED COMPONENTS"

R Generic Letter 90-04 - REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

I Generic Letter 90-05 - GUIDANCE FOR PERFORMING TEMPORARY NON-CODE REPAIR OF ASME CODE CLASS 1, 2, AND 3 PIPING

R Generic Letter 90-06 - RESOLUTION OF GENERIC ISSUE 70, "POWER-OPERATED RELIEF-VALVE AND BLOCK VALVE RELIABILITY," AND GENERIC ISSUE 94, "ADDITIONAL LOW-TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT-WATER REACTORS"

I Generic Letter 90-07 - OPERATOR LICENSING NATIONAL EXAMINATION SCHEDULE

I Generic Letter 90-08 - SIMULATION FACILITY EXEMPTIONS

I Generic Letter 90-09 - ALTERNATIVE REQUIREMENTS FOR SNUBBER VISUAL INSPECTION INTERVALS AND CORRECTIVE ACTIONS

I Generic Letter 91-01 - REMOVAL OF THE SCHEDULE FOR THE WITHDRAWAL OF REACTOR VESSEL MATERIAL SPECIMENS FROM TECHNICAL SPECIFICATIONS

I Generic Letter 91-02 - REPORTING MISHAPS INVOLVING LLW FORMS PREPARED FOR DISPOSAL

I Generic Letter 91-03 - REPORTING OF SAFEGUARDS EVENTS

I Generic Letter 91-04 - CHANGES IN TECHNICAL SPECIFICATION SURVEILLANCE INTERVALS TO ACCOMMODATE A 24-MONTH FUEL CYCLE

I Generic Letter 91-05 - LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

R Generic Letter 91-06 - RESOLUTION OF GENERIC ISSUE A-30, "ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES," PURSUANT TO 10 CFR 50.54(f)

I Generic Letter 91-07 - GI-23, "REACTOR COOLANT PUMP SEAL FAILURES" AND ITS POSSIBLE EFFECT ON STATION BLACKOUT

I Generic Letter 91-08 - REMOVAL OF COMPONENT LISTS FROM TECHNICAL SPECIFICATIONS

I Generic Letter 91-09 - MODIFICATION OF SURVEILLANCE INTERVAL FOR THE ELECTRICAL PROTECTIVE ASSEMBLIES IN POWER SUPPLIES FOR THE REACTOR PROTECTION SYSTEM

I Generic Letter 91-09 - EXPLOSIVES SEARCHES AT PROTECTED AREA PORTALS

R Generic Letter 91-11 - RESOLUTION OF GENERIC ISSUES 48, "LCOs FOR CLASS 1E VITAL INSTRUMENT BUSES," AND 49, "INTERLOCKS AND LCOs FOR CLASS 1E TIE BREAKERS" PURSUANT TO 10 CFR 50.54(f)

I Generic Letter 91-12 - OPERATOR LICENSING NATIONAL EXAMINATION SCHEDULE

R Generic Letter 91-13 - REQUEST FOR INFORMATION RELATED TO THE RESOLUTION OF GENERIC ISSUE 130, "ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES," PURSUANT TO 10 CFR 50.54(f)

I Generic Letter 91-14 - EMERGENCY TELECOMMUNICATIONS

I Generic Letter 91-15 - OPERATING EXPERIENCE FEEDBACK REPORT, SOLENOID-OPERATED VALVE PROBLEMS AT U.S. REACTORS

I Generic Letter 91-16 - LICENSED OPERATORS' AND OTHER NUCLEAR FACILITY PERSONNEL FITNESS FOR DUTY

I Generic Letter 91-17 - BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

I Generic Letter 91-18 - INFORMATION TO LICENSEES REGARDING TWO NRC INSPECTION MANUAL SECTIONS ON RESOLUTION OF DEGRADED AND NONCONFORMING CONDITIONS AND ON OPERABILITY

I Generic Letter 91-19 - INFORMATION TO ADDRESSEES REGARDING NEW TELEPHONE NUMBERS FOR NRC OFFICES LOCATED IN ONE WHITE FLINT NORTH

R Generic Letter 92-1, Rev. 1 - REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)

I Generic Letter 92-2 - RESOLUTION OF GENERIC ISSUE 79, "UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN"

I Generic Letter 92-3 - COMPILATION OF THE CURRENT LICENSING BASIS: REQUEST FOR VOLUNTARY PARTICIPATION IN PILOT PROGRAM

R Generic Letter 92-04 - RESOLUTION OF THE ISSUES RELATED TO REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs PURSUANT TO 10 CFR 50.54(F)

I Generic Letter 92-05 - NRC WORKSHOP ON THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP) PROGRAM

I Generic Letter 92-06 - OPERATOR LICENSING NATIONAL EXAMINATION SCHEDULE

I Generic Letter 92-07 - OFFICE OF NUCLEAR REACTOR REGULATION REORGANIZATION

R Generic Letter 92-08 - THERMO-LAG 330-1 FIRE BARRIERS

I Generic Letter 92-09 - LIMITED PARTICIPATION BY NRC IN THE IAEA INTERNATIONAL NUCLEAR EVENT SCALE

Bulletins That Requested Action or Information

IEB 83-01 - Failure of Reactor Trip Breakers (Westinghouse DB-50) to Open on Automatic Trip Signal

Requested licensees to perform a surveillance test, review the maintenance program, notify operators of the failure to trip event, and review EOPs. Provide a response to the NRC on the test results, conformance with the Westinghouse maintenance program and that operators had been notified.

IEB 83-02 - Stress Corrosion Cracking Large-Diameter Stainless Steel Recirculation System Piping At BWR Plants

Requested licensees to continue to perform the PCD program, perform an augmented ISI, demonstrate the effectiveness of UT, report the results of the tests, and retain the records.

IEB 83-03 - Check Valve Failures in Raw Water Cooling Systems of Diesel Generators

Requested consideration of all check valves in the IST program, licensees should determine if a verification procedure for valve internals is needed, perform an initial valve integrity check and submit a report.

IEB 83-04 - Failure of the Undervoltage Trip Function of Reactor Trip Breakers

Requested licensees to perform the same actions as IEB 83-01 but for GE breakers.

IEB 83-05 - ASME Nuclear Code Pumps and Spare Parts Manufactured by the Hayward Tyler Pump Company

Informed licensees that if the pumps were going to be used they should provide a list of applications, summary of IST, conduct a pump test, provide results of pressure tests, review the recommendations listed in the bulletin, and submit a report.

IEB 83-06 - Nonconforming Materials Supplied by Tube-Line Corporation Facilities At Long Island City, New York; Houston, Texas; and Carol Stream, Illinois

Requested a review of purchasing records, list of where equipment was installed, program to provide assurance it meets the code, basis for continued operation, and submittal of a report with results.

IEB 83-07 - Apparently Fraudulent Products Sold by Ray Miller, Inc.

Requested licensees to identify companies from whom they purchased material, whether fraudulent products are in the plant, evaluate the safety significance, discard fraudulent material or test, and provide a report.

IEB 83-08 - Electrical Circuit Breakers with an Undervoltage Trip Feature in Use in Safety-Related Applications Other Than the Reactor Trip System

Requested licensees to identify where the breakers are used. For each, review the design, surveillance program, operational experience, preventive measures and report to NRC verifying completion of requested actions.

IEB 84-01 - Cracks in Boiling Water Reactor Mark I Containment Vent Headers

Requested licensees to visually inspect the vent header for cracks, report results by phone within 8 hours, and provide written report within 7 days.

IEB 84-02 - Failures of General Electric Type HFA Relays in Use in Class 1E Safety Systems

Requested plans and schedules for replacing relays (until replacement, have to perform tests). Ensure the replacement relays are qualified - service life reliability, EQ.

IEB 84-03 - Refueling Cavity Water Seal

Requested licensees evaluate the potential for seal failure and the consequences. The evaluation should include: gross seal failure, leak rate, makeup capacity, time to cladding damage, EOPs - report to be submitted.

IEB 85-01 - Steam Binding of Auxiliary Feedwater Pumps

Licensees should develop procedures to monitor AFW fluid conditions, (temperature), recognize steam binding, and procedures on restoration (should remain in effect until hardware mods are complete). Submit a report.

IEB 85-02 - Undervoltage Trip Attachments of Westinghouse DB-50 Type Reactor Trip Breakers

Requests licensees perform a test, modify procedures to add a margin test, provide written instructions to operators to read the bulletin, declare the breaker inoperable if it doesn't pass either test, notify the NRC of inoperability, and submit a report.

IEB 85-03 - Motor Operated Valve Common Mode Failure During Plant Transients due to Improper Switch Setting

For MOVs in high-pressure systems, licensees were requested to develop and implement programs to ensure switches are set and maintained. Licensees should review the design basis for each valve, the maximum differential pressure, perform a test after changing switch setting, revise procedures, and send in a report verifying actions are complete.

IEB 86-01 - Minimum Flow Logic Problems That Could Disable RHR Pumps
Requested licensees determine whether a single failure vulnerability exists, instruct all shifts how to deal with failure, and provide a report on the short- and long-term actions.

IEB 86-02 - Static "O" Ring Differential Pressure Switches
Requested licensees to determine whether switches are in a system that has TS LCOs, notify the operators, conduct operational tests, and provide long-term corrective actions. Submit a report on where the switches are installed, if operators were informed, conduct special tests, report failures, implement interim performance criteria, provide the margin and basis for switch actuation, provide description of long-term corrective actions, the schedule, and the impact.

IEB 86-03 - Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line
Requested licensees to determine whether the plant has single failure vulnerability that could cause failure of more than one ECCS train, instruct the shifts what actions to take, report to NRC on short- and long-term modifications.

IEB 87-01 - Thinning of Pipe Walls in Nuclear Power Plants
Requested licensees to provide information on monitoring program including codes and standards, scope of program, acceptance criteria, results of all inspections, any plans for revising program.

IEB 87-02 - Fastener Testing to Determine Conformance with Applicable Material Specifications
Requested licensees provide information on receipt inspection and procedure control, select 10 fasteners for mechanical and chemical testing. Send in a report with results and discussing any further action to be taken.

IEB 87-02, Supplement 1
Requests licensees to provide a list of manufacturers.

IEB 87-02, Supplement 2
Rescinds actions requested in Supplement 1. Licensees should provide a list of manufacturers of safety-related fasteners and non-safety-related fasteners. Clarifies information requested.

IEB 88-01 - Defects in Westinghouse Circuit Breakers

Requested licensees perform short-term and long-term inspections. If can't meet the schedule, provide an alternative. Maintain records. If no circuit breakers of this type in the plant, submit a letter. If there are these circuit breakers present, send in a letter confirming the requested actions are complete.

IEB 88-02 - Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

Requested licensees to submit a report on the status of compliance with actions. Requested a review of steam generator inspection data, performance of inspection (schedule), implementation of monitoring program.

IEB 88-03 - Inadequate Latch Engagement in HFA Type Latching Relays Manufactured by General Electric (GE) Company

Requested licensees measure distance between the contact and relay, check the latch, replace defective relays, inspect spare relays, and provide a report confirming the requested actions were taken. The letter should include the number and type of relay inspected and the number that required corrective action.

IEB 88-04 - Potential Safety-Related Pump Loss

Requested licensees determine whether there are any pumps that might have the problem discussed, evaluate the system, evaluate the adequacy of the minimum flow bypass. Provide a response that summarizes problems and affected systems, short- and long-term modifications, the schedule, and any JCO's. Maintain evaluations for 2 years.

IEB 88-05 - Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey

Requested licensees review purchasing records, identify material that was out of conformance, provide assurance that purchased material meets the code, replace questionable flanges, and submit a report describing the actions taken and verifying completion.

IEB 88-05, Supplement 1

Reduces scope of review to only fittings and flanges. Test only those already installed, provide a JCO for inaccessible ones, and include results of tests in response.

IEB 88-05, Supplement 2

Temporarily suspends actions of previous bulletins and outlines exceptions.

IEB 88-07 - Power Oscillations in Boiling Water Reactors (BWRs)

Requested licensees to brief operators on the event, verify adequacy of procedures and training instruction, and submit a report confirming completion.

IEB 88-07, Supplement 1

Requested licensees to implement GE interim stability criteria, ensure training and procedures are in place and adequate. Submit a report stating whether actions have been taken.

IEB 88-08 - Thermal Stresses in Piping Connected to Reactor Coolant Systems

Requested licensees to review systems connected to RCS to find unisolable sections subject to stress and not previously evaluated. If subject to stress, NDE welds and zones, implement a program to provide assurance that pipe won't be subject to stress. Submit a letter stating actions are complete.

IEB 88-09 - Thimble Tube Thinning in Westinghouse Reactors

Requests licensees to establish an inspection program including wear acceptance criteria, inspection frequency, and inspection methodology. Maintain records and submit a report.

IEB 88-10 - Nonconforming Molded-Case Circuit Breakers

Requests licensees to identify all spare molded case CBs. Verify traceability. If circuit breakers can't be traced, submit a JCO analysis. If over 80% of the total number of circuit breakers are traceable, test those that aren't traceable - replace failures, keep records. From now on, molded case CBs in safety-related applications will be procured under an Appendix B program and be traceable. Provide a report with the results of tests and the number and type identified.

IEB 88-10, Supplement 1

Requested licensees to review submittals and ensure the provisions of the bulletin are met. Retain documentation.

IEB 88-11 - Pressurizer Surge Line Thermal Stratification

Requests licensees establish and implement a program to confirm pressurizer surge line integrity: visual inspection, demonstrate it meets the code, if not - plant-specific data is needed, update stress and fatigue analysis, evaluate piping modifications, monitor for surge line stratification, keep documents and report any stress. Submit the schedule for action and verification of completion of actions.

IEB 89-01 - Failure of Westinghouse Steam Generator Tube Mechanical Plugs

Requested licensees to verify information, estimate lifetime, and remedial actions taken.

IEB 89-01, Supplement 1

Expanded the scope to all Westinghouse steam generator tube mechanical plugs fabricated from thermally treated Inconel 600.

IEB 89-01, Supplement 2

Requested same actions except different heat numbers.

IEB 89-02 - Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Mode S350W Swing Check Valves or Valves of Similar Design

Requested licensees to identify valves, disassemble and inspect. If susceptible to SCC, replace. Send in a report verifying completion of actions - numbers and locations of valves identified.

IEB 89-03 - Potential Loss of Required Shutdown Margin During Refueling Operations

Requests licensees to ensure adequate shutdown margin by identifying and evaluating intermediate fuel assembly configuration, ensuring fuel loading procedures don't violate shutdown margin, and providing staff training. Submit a report on whether the requested actions have been taken. (Adequate protection backfit)

IEB 90-01 - Loss of Fill-Oil in Transmitter Manufactured by Rosemount

Requested licensees to identify the types of transmitters used, identify those having problems, review records to see if any have exhibited symptoms of loss of fill-oil, develop and implement enhanced surveillance program, and document a JCO. Submit a report stating the requested actions have been taken. (Compliance backfit)

IEB 90-01, Supplement 1

Supersedes original bulletin. Requests a review of records to identify certain models of transmitters and requests replacements or monitoring. Perform an evaluation of the monitoring program to ensure it can identify problems. Provide a response whether the requested actions will be taken, the schedule and notify NRC when complete. Evaluate the actions not being taken. (Compliance backfit)

IEB 90-02 - Loss of Thermal Margin Caused by Channel Box Bow

Requested licensees who use channel boxes more than once to verify TS are met. Send in a report including the number of these channel boxes and the disposition and the methods used to determine the effects on the box.

IEB 92-01 - Failure of Thermo-Lag 330 - Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage

Requested licensees to identify areas of plant with thermo-lag for small conduit or wide trays, implement compensatory measures if found inoperable, and within 30 days send in a report to NRC whether or not thermo-lag is in plant. If yes, state whether the actions were taken and what's being done to restore operability.

IEB 92-01, Supplement 1

Expanded the scope of the original bulletin to all conduit and trays, including walls, ceilings, and equipment enclosures.

2.3.7 INSERVICE TESTING (IST)

I. INTRODUCTION

The requirements for pump and valve tests are outlined in 10 CFR 50.55a(f) and state that ASME Code Class 1, 2, and 3 safety-related pumps and valves must meet the applicable requirements of the ASME Code, Section XI. However, the rule allows licensees some flexibility in meeting those requirements. Subpart 4(iv) states that licensees may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in the rule, but not addressed in the licensee's 10-year IST program, subject to the limitations outlined in the rule being followed, all related requirements of the respective editions and addenda being met, and Commission approval. Subpart 5(iii) states that, if a licensee determines conformance with Code requirements is impractical, it shall notify the Commission and provide the basis for the determination no later than 12 months after the expiration of the initial 120-month period of operations (first 10-year interval). The regulation goes on to state that the Commission may grant relief from the Code requirements and may approve licensee-proposed alternative testing that provides an acceptable level of quality and safety.

II. BACKGROUND

IST is a very important part of maintaining public health and safety in that it demonstrates that a component is not degrading over time. However, in the past, testing described in the Code has not been as performance based as it could be. Additionally, in the past there has been a large backlog of IST programs and relief requests for review. Many of the relief requests were common to a number of licensees, but were being reviewed and relief granted on a plant-specific basis. In order to reduce the backlog and grant relief for requests that were generic in nature, the staff issued Generic Letter (GL) 89-04. The GL approved IST relief requests if the alternative methods of testing outlined in the GL were adopted by plants. The most recent Code referenced in 10 CFR 50.55a will result in a reduction in the number of relief requests submitted by licensees. The staff is developing guidance on implementation of the recent Code prior to the 10-year program update.

III. DISCUSSION

There are still a number of relief requests that are being submitted that request permission to use the updated Code testing requirements. The staff has developed guidelines that will allow the staff to be proactive in approving the use of new editions and addenda so that licensees will not send in individual relief requests. This will allow the time and effort of both the industry and the staff to be concentrated on the more unusual relief requests and preparing for the future use of risk-based techniques to support relief requests. The guidelines are currently a draft NUREG that will offer generic approval of provisions contained in the most recently approved addenda and editions to the Code per 50.55a(f)(4)(iv). This would allow licensees to perform testing that is already approved by the Code and the regulations, although not specifically identified in their 10-year program. The staff anticipates this will eliminate approximately two-thirds of the relief requests currently being submitted. It appears that an appropriate tool to promulgate the approval would be the use of a generic letter informing licensees of the availability of the NUREG and that licensees may take advantage of positions therein.

The staff also plans to revise the regulations such that they are consistent with the Code. One example of this would be that the Code allows extension of the 120-month interval, but the regulations remain silent on the issue. The improved Standard Technical Specifications are more consistent with the Code on the subject of test frequency.

The ASME Code is also undergoing changes. The Code is beginning to address design basis testing and test frequency based on risk techniques. The Code now provides for testing of a component to verify functionality under non-specified conditions that may not verify functionality under design basis conditions. The Code is also developing risk-based testing guidelines that will base test frequency on risk.

As part of the staff comments it was noted that risk-based technology is being used also for establishing the scope and frequency of inservice inspection (ISI). An initiative is already underway by an ASME Code Committee and should, in the next several years, along with appropriate revisions to the regulations, save industry and NRC resources that are currently being spent on low risk issues.

IV. RECOMMENDATIONS

The Review Group believes the staff is approaching the issue of improved Code testing in a proactive and effective manner. The industry and staff should continue to build a consensus view in the Code committees to revise the Code based on risk techniques.

V. POTENTIAL IMPROVEMENTS

The staff should also continue work on the IST Program Guidelines. These guidelines would allow licensees to take advantage of the generic approval of using the most recent addenda and editions rather than what was originally committed to in the IST programs. The Review Group recommends issuing a GL informing licensees of the NUREG and the flexibility allowed if licensees take advantage of it. The Review Group also recommends that Inspection Procedure 73756 be reviewed for continued applicability in light of the issuance of the NUREG and that the inspection procedure be revised, if appropriate.

VI. ANALYSIS OF PUBLIC COMMENTS

As a result of public comments it appears design basis testing may not be warranted and is viewed as a new requirement.

2.3.8 LICENSEES PERFORMING THEIR OWN SAFETY ANALYSES

I. BACKGROUND

In Generic Letter 83-11, the NRC staff encouraged utilities to perform their own safety analyses for licensing actions such as reload applications and Technical Specification (TS) amendments. However, because of the complexity in the computer codes in use and the impact of user options on the results, the staff was concerned that some licensees would not demonstrate their ability to use the code by performing their own code verification. Therefore, the NRC stated that licensees who intended to use a safety analysis computer code to support licensing actions should demonstrate their proficiency in using the code by submitting to the NRC the code verification performed by the licensee.

In Generic Letter 88-16, the staff offered an alternative to reactor physics parameters located in the Technical Specifications that need to be amended each fuel cycle. The staff provided guidance to licensees regarding relocating cycle-specific parameter limits to a core operating limits report (a licensee-controlled document). The generic letter proposed three voluntary amendments to the TS to facilitate this change: (1) the addition of the definition of the formal report (core operating limits report), (2) an administrative reporting requirement to submit the core operating limits report for information, and (3) modification of Section 6.9 of the TS to note that cycle-specific parameters shall be maintained within the limits provided in the core operating limits report. The modified TS would also state that the analytical methods used to determine the limits shall be previously reviewed and approved by NRC in a topical report identified by number, date, or SER approving the plant-specific methodology. These approved methodologies are listed in Section 6.9 of the TS.

II. DISCUSSION

Over the past 10 years the industry has responded to GL 83-11, and many licensees have submitted code verification information to allow them to perform their own reload analyses. Considerable experience has been gained in using safety analysis codes; however, the NRC staff during the review of code verifications still identifies areas where licensees have gone beyond the limitations set forth in the NRC-approved generic methodology topical report or have developed their own codes for use in reload analyses. Reviews of the entire code verification package or packages that deviate from the generically approved topical reports are resource intensive on the part of the staff. The reports are usually assigned a low priority, often requiring over 2 years to be approved for licensee use. Scheduling delays in reviews also occur when licensees submit topical reports with no previous staff knowledge that the report was being submitted and when licensees submit major changes to their topical reports. Due to the low priority, the uncertainty associated with the availability of contractor funding, and the lack of prior interactions with the staff, the schedules for approval are sometimes uncertain and licensees may have to place a contract with the vendor to perform the reload analysis on an exigent basis if the licensee methodologies are not approved by the NRC in time for reload. This can cost the utilities a great deal of money.

To help alleviate the lengthy review and approval process, the NRC staff has begun a process of developing guidance for licensees to use when preparing a code verification package for NRC review and approval. The staff plans to issue guidance to licensees on what is needed in order for approval to be obtained as quickly as possible. It is envisioned the guidance will streamline the process for both the licensee and the staff.

Licensees, however, can also simplify the process by using NRC-approved generic methodologies and adhering to the limitations set forth in the staff's approval. A complete staff review is not performed in these cases. Usually benchmarking or an audit is performed. These types of reviews take less time than a complete review.

Another improvement to the process might include the voluntary TS revisions associated with GL 88-16. GL 88-16 recommends TS 6.9 be revised to state, "The analytical methods used to determine the [core] operating limits shall be those previously reviewed and approved by NRC in [identify the Topical Reports(s) by number title, and date, or identify the staff's safety evaluation report for a plant specific methodology by NRC letter and date]." It appears the GL revisions would require licensees to submit a license amendment whenever a revision or new topical report is approved by the NRC and used in a reload analysis. A more flexible approach might be to have the approved topical report revision referenced in the core operating limits report that is required by TS 6.9, as revised per GL 88-16, to be submitted to the NRC for information. This would eliminate the need for a license amendment to change the reference to the most recently approved topical report and still allow the staff to be informed of what approved methodologies are used in the reload analysis.

III. POTENTIAL IMPROVEMENTS

As noted above, the guidance for GL 88-16, which states that the TS reference the specific-approved topical report number and date used in the analysis, could be replaced by a generic and more flexible statement such that a license amendment would not be needed to reference the most current NRC-approved topical report used in performing the analysis. The specific-approved topical report revision would still be required to be referenced in the core operating limits report.

NRC staff comments indicate guidance is being developed to address this issue.

IV. ANALYSIS OF PUBLIC COMMENTS

Several respondents agreed with the improvements and proposed further actions: (1) allow all core limit values to be placed in the core operating limits report, (2) scope and depth of staff review should distinguish between topical reports which are implementing new methodologies and those which are implementing already approved methodologies, and (3) eliminate submittal of code verification packages entirely.

2.3.9 MATTERS PLACED UNDER LICENSEE CONTROL

I. INTRODUCTION

The charter of the Review Group directs that a review of regulations affecting power reactor licensees be conducted. This review was to identify areas in which regulations go beyond that necessary for safe operation, are overly prescriptive, or are in need of clarification. The review of the Part 50 regulations indicated that there are several regulated activities that are turned over to licensee control, but under different constraints. These areas include:

- Changes made to the facility or procedures **that neither change the Technical Specifications nor involve an unreviewed safety question** (10 CFR 50.59);

- Changes made to the quality assurance plan **that do not reduce commitments in the program description previously accepted by the NRC** (10 CFR 50.54(a));

- Changes made to security procedures **that do not decrease the safeguards effectiveness of the security plan, the safeguards contingency plan, and the guard training and qualification plan** (10 CFR 50.54(p));

- Changes made to the emergency plans **that do not decrease the effectiveness of the plans and the plans as changed still meet the standards of 10 CFR 50.47(b)** (10 CFR 50.54(q)); and

- Changes to the fire protection plan should be kept as a record until the Commission terminates the license and superseded procedures for 3 years (10 CFR 50.48(a)).

II. DIFFERENCES IN REPORTING REQUIREMENTS

As delineated above, there are differences in the handling of changes to the plans. The potential safety significance of a change made in each of the five areas varies. For example, the ability to make changes in the facility as described in the final safety analysis report (FSAR) under 50.59 is a broad authority. There are restrictions on this authority: (1) the change cannot result in a change in the Technical Specifications, which is definitive, and (2) the change cannot introduce "an unreviewed safety question," which is the granting of broad discretion to the licensee in that the NRC does not review and approve the change prior to implementation. Arguably, this authority has great potential safety significance, but it is under the control of the licensee who is charged to make a summary report at a frequency not to exceed 2 years. Similarly, changes to the quality assurance plan are reported to the NRC annually. The pervasive nature of quality assurance on many phases of operations and maintenance makes the potential significance of this authority also broad.

In contrast, the other three areas (fire, emergency planning, and safeguards) are essentially contingency plan areas. In two of these areas (safeguards and emergency planning), reporting requirements are far more restrictive (2 months and 30 days, respectively), but the third area (fire), which has a large potential risk from a safety perspective, has no reporting requirement. It is also noted that for reports made under 10 CFR 50.59, regulations require an annual report or "...along with the FSAR updates as required by 50.71(e)." However, 10 CFR 50.71(e)(4) states:

> "Subsequent revisions must be filed annually or 6 months after each refueling outage provided that the interval between successive updates to the FSAR does not exceed 24 months."

This gives licensees a choice in the reporting frequency for changes made under 10 CFR 50.59.

III. REGULATORY COHERENCE

The differences in the frequency of the reporting periodicity does not reflect the safety significance of the subjects addressed by the regulations. These differences are accentuated by the potential safety significance of each of the five categories, for the three with the greatest apparent potential from a safety perspective have the least restrictive reporting requirements.

The Improved Standard Technical Specifications place even more items in the FSAR and under administrative control similar to that provided by 10 CFR 50.59. This emphasizes the need to reach a regulatory determination as to what control is needed in order to provide adequate control of safety-related functions delegated to licensees.

The change to 10 CFR 50.71(e) discussed above established the report periodicity as the refueling cycle (not to exceed 24 months). This would lead to the conclusion that the requirements of 10 CFR 50.54(p) and (q) should be changed.

There is an issue of definitions also buried in the various requirements. Changes to the safeguards and emergency plans are allowed if they do not reduce the effectiveness of the licensee's commitments, but what does this mean and what constitutes the licensee's commitment? We believe that the commitment is to the basic articulation of the requirement and not to the method by which the licensee initially stated that the commitment would be met. This should mean that the licensee may make changes as long as these changes do not preclude the licensee from meeting (as a minimum) the requirements of 10 CFR 73.55 for security and of 10 CFR 50.49 (and Appendix E) for emergency planning. We believe that on occasion licensees have been held to their methods instead of their commitments. The regulatory issue is the potential for each licensee to be regulated to a different set of standards.

There is a similar issue involved in the quality assurance area. Licensees may implement changes in the quality assurance plan without prior Commission approval if these changes do not reduce their commitments. Because of the broad, performance-based nature of Appendix B to Part 50, the differentiation between the commitment to it and the method of carrying out this commitment has been difficult for reviewers and inspectors to separate. In addition, quality assurance plans in whole or in part have been incorporated into Chapter 17 of the FSAR. If the FSAR is taken as a commitment, then why are not the controls of 10 CFR 50.59 sufficient? The process has also led to what appears to be a variation from plant to plant in commitments. We recommend that a clear and consistent definition for commitment be developed (see Section 2.3.2) such that licensees can meet the intent of Appendix B. In addition, the authority to make changes already provided in 10 CFR 50.54(a) is broad, and the staff should not place unnecessary restriction on licensees through the inspection or review process.

IV. RECOMMENDATION

The Review Group believes the change mechanisms for the five areas discussed should be consistent. Because of the importance of this recommendation, the Review Group developed a proposed rulemaking to allow licensees to make changes to their QA, emergency, fire protection, and security plans without NRC approval provided the changes do not reduce the plans' contents below that necessary to meet the requirements in the regulations. The Review Group received comments from the NRC offices and revised the proposed rulemaking. The detailed proposed rulemaking is located in Appendix A to Volume One.

2.3.10 MISCELLANEOUS FINDINGS AND RECOMMENDATIONS

This is a summation of miscellaneous comments concerning regulations and related items. They represent a compilation of items found during the review that were not considered worthy of separate papers. Generally, they were items that were not found to be particularly burdensome but either were anomalies or otherwise deemed worthy of comment.

I. USE OF CONSTRUCTION PERMITS FOR FACILITY ALTERATIONS

The Review Group identified several areas in the regulations that refer to the need for a construction permit for an "alteration" of a production or utilization facility. Although there is no definition in 10 CFR 50.2 for alteration, this term has apparently been interpreted by the staff to mean a modification to a facility that results in a change to the design basis.

The specific paragraphs referencing a facility alteration are listed below:

- 10 CFR 50.23 states, "A construction permit for the alteration of a production or utilization facility will be issued prior to the issuance of an amendment of a license, if the application for amendment is otherwise acceptable, as provided in 50.91."

- 10 CFR 50.45 states, "An applicant for a license or an amendment of a license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit, if the application is in conformity with and acceptable under the criteria of 50.31 through 50.38 and the standards of 50.40 through 50.43."

- 10 CFR 50.56 states, "Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be."

- 10 CFR 50.92(a) states, "If the application involves the material alteration of a licensed facility, a construction permit will be issued before the issuance of the amendment to the license."

The Review Group recommends the following definition be added to 10 CFR 50.2 and the above regulations be modified to be consistent with the definition:

> "Material Alteration" means any modification to a facility that changes the design bases of the facility.

It appears this definition is consistent with discussions on the need for a construction permit in the context of plant life extension and in the December 13, 1991, Federal Register notice.

II. NUCLEAR "WHISTLEBLOWER" PROTECTION

The Energy Policy Act of 1992, P.L. 102-486, directs the NRC not to postpone the investigation of safety concerns during the pendency of Department of Labor (DOL) investigations of "whistleblower" allegations; the NRC presently does not postpone its safety investigations. Additionally, the Act extends the statute of limitations for a "whistleblower" to file a claim with DOL from 30 days to 6 months; expands the definitions of both whistleblower (to include three more classes of employees) and employer; and changes the burden of proof for whistleblowers and for employers. The regulation (10 CFR 50.7) should be revised to the extent necessary to reflect these changes.

III. SPACE PROVIDED FOR RESIDENT INSPECTION STAFF

The regulations [10 CFR 50.70(b)(2)] require licensees to provide space for NRC inspectors. The rule states:

> "For a site with a single power reactor or fuel facility licensed pursuant to Part 50, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel ... For sites containing multiple reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspector(s)."

As noted in the public comments, licensees provide sufficient space for residents; however, the rule should be updated to address the current policy of two full-time inspectors at single unit sites.

IV. CONFLICTS WITHIN AREAS OF REGULATIONS

As a result of the recent CRGR regulatory review effort, 10 CFR 50.71(e)(4) was amended to allow revisions to the FSAR be filed either annually or 6 months after each refueling outage provided the interval between successive updates to the FSAR does not exceed 24 months. The requirements in 10 CFR 50.54(a)(3) state that changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC at least annually in accordance with the requirements of 10 CFR 50.71; however, the quality assurance plan is often in the FSAR. The requirements of 54.37(b) state that the annual FSAR update required by 50.71(e) must include any structures, systems, and components newly identified as important to license renewal. It appears there is a conflict and inconsistency in these regulations in that the requirements of 10 CFR 50.71(e) allow an FSAR submittal annually or 6 months after each refueling outage. In order to be consistent and eliminate confusion about whether an exemption to 10 CFR 50.54 is needed for submittal of changes to the quality assurance program description on a refueling outage basis, it appears 10 CFR 50.54(a)(3) should be amended to delete the words "at least annually." The requirement would then read, "Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC in accordance with the requirements of 10 CFR 50.71(e)." This would give licensees the flexibility of submitting the changes either annually or 6 months after each refueling outage. The requirements of 10 CFR 54.37(b) should also be changed to be consistent with 10 CFR 50.71(e) by deleting the word "annual" so that it would read "The FSAR update required by 50.71(e) must include any structures, systems and components newly identified as important to license renewal...."

The Review Group initiated a proposed rulemaking package to amend these two sections of the regulations. The proposed changes are of the type the EDO has been delegated authority to promulgate. The Review Group published a proposed rulemaking on May 14, 1993, for a 30-day comment period and has prepared a final rulemaking package.

V. REVISED AND ADDITIONAL DEFINITIONS

Section 50.2 of NRC regulations provides definitions of some of the terms used in 10 CFR 50. Other parts of Chapter 10 that affect power reactors also provide definitions of terms used. For example, 10 CFR 21.3 includes definitions of "dedication" and "commercial grade." (See the discussion of these two terms in Section 2.3.1 of this volume.)

During the audits and workshop conducted by the staff in connection with the preparation of SECY-92-314, the staff found, among other things, that there is not universal agreement among licensees or within the NRC staff as to the meanings of certain terms used in 10 CFR 50. For example, although the term "current licensing basis" is used in 10 CFR 50.54(f), that term is not defined in 10 CFR 50. The definition in 10 CFR 54.3 applies only to renewal of operating licenses. In addition, the staff found in its audits and workshop that there is no clear understanding of the scope and depth of the term "design bases" as that term is defined in 10 CFR 50.2 and as discussed in the Policy Statement on the Availability and Adequacy of Design Bases Information at Nuclear Power Plants.

The Review Group recommends a definition be developed for the term "current licensing basis" and that the scope and depth of the term "design bases" be clarified.

In recognition of the potential rulemaking on Part 54, the definitions developed in that effort should be consistently applied in Part 50.

VI. CONTROL OF MATERIAL REMOVED FROM TECHNICAL SPECIFICATIONS

Under the original system for specifying and for changing Technical Specifications that was established in June 1962 (27 FR 5491), the Commission specified in great detail in the original Appendix A to 10 CFR 50 the types of matters that it generally expected to be covered by Technical Specifications. For licenses issued prior to the effective date of those amendments to 10 CFR 50.36, the entire hazards summary report (later designated the safety analysis report) was deemed to be the Technical Specifications. A new 10 CFR 50.59 was added to the regulations at that time to allow changes in the facility and procedures as described in the hazards summary report and to allow the conduct of tests and experiments not described in the hazards summary report--unless the change, test, or experiment involved a change in the Technical Specifications or an unreviewed safety question. The term "unreviewed safety question" did not at that time include a reference to "technical specifications bases" because that concept had not yet come into being.

The recognition of the need for a carefully prepared safety analysis report (SAR) came into being in December 1968 (33 FR 18610). This change (1) eliminated the original Appendix A to 10 CFR 50 because experience had shown that the degree of detail contained in Technical Specifications prepared in accordance with that Appendix A was not necessary for purposes of public safety; (2) added a definition of the term "design bases" to 10 CFR 50.2; (3) revised 10 CFR 50.34 to emphasize the need for analysis and evaluation, to require submittal of a preliminary SAR at the construction permit stage and a final SAR at the operating license stage, and to require that those SARs include information describing the facility, explaining its design bases and the limitations on its operation and showing by evaluation that its safety functions will be accomplished; and, (4) redefined the term "unreviewed safety question" to address the reduction of the margin of safety as defined in <u>technical specifications bases</u>. The Commission believed that the system for Technical Specifications that it was implementing in December 1968 was better adapted to focus the attention of both licensee management and the Commission on those features and characteristics of the facility that are important to safety. Since it placed increased emphasis on systematic analysis and evaluation of a facility in order to provide a sound basis for each Technical Specification, the preparation of Technical Specifications required a <u>carefully prepared SAR</u>.

The December 1968 system for preparation of Technical Specifications now is to be supplanted by revised improved Technical Specifications. That system as described in the "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated February 6, 1987 (52 FR 3788), is expected to produce safety improvements through the development of more operator-oriented Technical Specifications, improved Technical Specification bases, reduced action-statement-induced plant transients and more efficient use of NRC and industry resources. This new system involves a complete rewriting and streamlining of existing Technical Specifications. It also results in many requirements being transferred from Technical Specifications to other documents, such as the FSAR, that are licensee-controlled.

Because the system for improved Technical Specifications places great reliance on the FSAR and 10 CFR 50.59, the 1987 interim policy statement describing this system contemplated rule changes, a complete and accurate updated FSAR, and NRC endorsement of an industry standard for the conduct of 10 CFR 50.59 reviews (NSAC-125) to be necessary. However, the NRC staff has not endorsed NSAC-125 as presently written. Moreover, the NRC staff found in connection with its preparation of SECY-92-314 that updated FSARs contain or reference only a small portion of the additions to the licensing basis that have been made since the TMI accident. This has resulted from the interpretation by some licensees of 10 CFR 50.71(e), which states that the FSAR is to be updated "to include <u>the effects of</u>: all changes made in the facility or procedures as described in the FSAR." Many licensees have interpreted "the effects of" to mean that the updated FSAR need only include information that if not included would cause the FSAR to be in error with respect to information currently included. Therefore, the less originally included, the less there is to update. To the extent that the facility or procedures are not described in much detail (or not at all) in the FSAR, 10 CFR 50.59 does not operate as a control on a licensee's ability to change them without at least reporting the changes to the NRC. In addition, to the extent that details removed from the Technical Specifications and put into the FSAR do not describe the facility or procedures, 10 CFR 50.59 <u>does not provide</u> control of changes to such material.

The regulations (10 CFR 50.71(e), 10 CFR 50.59, 10 CFR 50.36, and pertinent portions on 10 CFR 50.54) should be reviewed and revised as necessary to ensure that the system for improved Technical Specifications will maintain appropriate control of changes to material that is removed from Technical Specifications and placed in licensee-controlled documents.

VII. ANALYSIS OF PUBLIC COMMENTS

Seismic Hazard

One respondent pointed out both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) had completed significant work which indicated that - for nuclear power plants in the eastern United States - the probability of hazard from a seismic event is relatively low. The letter further stated that despite the apparent decreased seismic hazard, the staff has continued to pursue information for older vintage plants to satisfy Unresolved Safety Issue A-46 and to evolve toward a seismic IPE (Individual Plant Examination). The commenter recommended that this topic be included in the report in the belief that "...a consensus (LLNL and EPRI results) that this design basis accident is, in fact, a severe accident of very low probability."

The Review Group did not specifically review this issue and accordingly does not have a technical position relative to it. However, the comment indicates at the very least a perception of significant burden without a commensurate increase in safety. It is therefore recommended that this issue be reviewed by the staff to determine if requests for information, studies, etc., in the area can be justified or should be made.

10 CFR Part 20

One respondent commented on Review Group data sheets for 10 CFR 50.34a, 50.36a, 50.36b, and Appendix I to Part 50. The Review Group data sheets indicated that these regulations were not candidates for further review. The respondent stated that 10 CFR 50.34a, 50.36a, 50.36b, and Appendix I to Part 50 should be revised because of the new Part 20 and that failure to revise these would result in unnecessary burden.

While the indication on the data sheets related only to the intention of the Review Group, the issue raised previously in a letter dated December 21, 1992, is valid. The NRC recently stated in a letter dated June 30, 1993, that it was acceptable to the staff for licensees to retain their existing level of effluent control as implementing the ALARA requirement after January 1, 1994. The letter also confirmed that the staff was preparing a generic letter. This proposed generic letter would provide model technical specification wording and would not require technical specification changes to be submitted before the new Part 20 regulations become effective on January 1, 1994. The Review Group also was told that changes to the affected regulations would be promulgated at a later date.

2.3.11 POLICY STATEMENTS

The NRC periodically issues policy statements. Those affecting power reactor licensees were reviewed.

I. INTRODUCTION

The Attorney General's Manual on the Administrative Procedure Act (APA) states that policy statements are issued by an agency to advise the public prospectively of the manner in which the agency proposes to exercise a discretionary power. Various courts in trying to distinguish among the attributes of policy statements and substantive or legislative rules--those having the force and effect of law--have stated that:

> [a] general statement of policy...does not establish a "binding norm." It is not finally determinative of the issues or rights to which it is addressed. The agency cannot apply or rely upon a general statement of policy as law....
>
> [t]o the extent that the directive merely provides guidance to agency officials in exercising their discretionary powers while preserving their flexibility and their opportunity to make "individualized determination[s]" it constitutes a general statement of policy. However, to the extent it narrowly limits administrative discretion or establishes a "binding norm" so that...upon application one need only determine whether a given case is within the directive's criterion it is a substantive rule.
>
> A policy statement genuinely leaves the agency and its decisionmakers free to exercise discretion.

II. DISCUSSION

Many NRC policy statements express the Commission's expectations of industry performance or direct industry to carry out certain actions. To the extent that these requests are treated by the staff as requirements, the policy statements are being applied as if they were substantive rules.

In addition to questions concerning the application of policy statements, there are also questions regarding the updating process. If a policy statement is issued on a topic and then a rule is promulgated that addresses the same issue, is there a requirement to delete that policy statement from the list of current and applicable policy statements? Additionally, there is no requirement to solicit public comment on policy statements.

NRC practice generally has been to issue proposed policy statements for comment, but neither the APA nor NRC procedures require this practice. The Administrative Conference of the United States has urged the use of the notice-and-comment procedure in connection with the issuance of policy statements even though such is not required by the APA. The Conference has urged also that policy statements should make clear that they are not legally binding and inform persons affected of the manner for challenging policy statements.

III. SPECIFIC POLICY STATEMENTS

The following policy statements recommend or endorse actions that may directly or indirectly impose regulatory burdens on licensees. They have been grouped according to possible actions that could be taken for each group.

A. The following policy statements have been superseded by rulemakings or are no longer applicable. These policy statements should be deleted. However, as noted in the staff comments, before eliminating any policy statement, the staff will need to carefully look at how such a deletion would affect individual licensees if licensees have incorporated the policy statement into license conditions, Technical Specifications, etc., or if the policy statement is more restrictive or prescriptive than the regulations. The NRC will need to convey its expectations to licensees.

1. <u>Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents</u>

   This policy statement endorses the use of guidance contained in a task force report on emergency planning. The report recommended that two emergency planning zones (EPZs) be established (surrogates that have now been interpreted as rules) around nuclear power plants and recommended time values to implement protective action. The policy statement states that, although the guidance may have significant response impacts for many local jurisdictions, it believes implementation of the guidance is nevertheless needed to improve emergency planning. The policy statement states that the guidance will be incorporated into existing documents and rules.

   This policy statement has been superseded by rulemaking.

2. <u>Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel</u>

   This policy statement endorses the INPO Training Accreditation Program and outlines essential elements that comprise an acceptable training program.

This policy statement has been superseded by the training rule which was issued April 26, 1993.

3. Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants

This policy states that it is widely recognized that plant-specific PRAs have yielded valuable insight to unique plant vulnerabilities to severe accidents and licensees of each operating reactor will be expected to perform a limited-scope accident safety analysis designed to discover outliers of vulnerability.

The portion of this policy statement affecting operating reactors has been superseded by rulemakings and Generic Letter 88-20. The portion of the statement addressing advanced reactors should be retained and reevaluated after a firm decision is made on the form and content of 10 CFR Part 52 certifications.

4. Commission Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel

The policy statement recognized the industry initiatives in this area and provided a summary of the Commission expectations of licensee programs for fitness for duty.

This policy statement has been superseded by Part 26.

5. Nuclear Power Plant Access Authorization Program; Policy Statement

This policy statement requests comments on whether the Commission should adopt industry-developed guidelines or promulgate a rule codifying access authorization provisions.

This has been superseded by 10 CFR 73.56.

6. Maintenance of Nuclear Power Plants; Revised Policy Statement

The policy statement describes the Commission's expectations during the 18-month period that rulemaking is held in abeyance.

This policy statement has been superseded by 10 CFR 50.65.

7. <u>Below Regulatory Concern; Policy Statement</u>

The policy statement establishes a framework within which the Commission will formulate rules or make decisions to exempt regulatory controls involving small quantities of radioactive material.

This policy statement was revoked by Congress in the Energy Policy Act.

8. <u>Policy Statement of Information Flow</u>

The policy statement emphasizes to licensees their responsibility to provide the Commission with timely and accurate information during the course of an incident or event. The policy statement outlines particular situations where communication is expected.

This reporting burden is redundant to the requirements in the regulations.

B. The following policy statements are voluntary in nature and, if licensees endorse them, will result in a regulatory burden.

1. <u>Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors</u>

This interim policy statement encourages licensees to implement a voluntary program to update their Technical Specifications to be consistent with revised vendor-specific Standard Technical Specifications (STS).

For licensees who volunteer, there will be a significant burden changing over to the new STS. The staff has issued a final policy statement.

2. <u>Integrated Schedules; Policy Statement</u>

The policy statement outlines the procedures the Commission plans to use to promote voluntary implementation of licensee-integrated schedules for regulatory requirements and other activities. Integrated schedules will provide a systematic method of coordinating, managing, and scheduling major modification and activities initiated by the NRC and licensees.

The decision to volunteer is licensee-specific and is based on a licensee-performed cost-benefit analysis.

C. The following policy statements impose a limited regulatory burden. However, the agency has exercised discretion in this area and believes the burden is outweighed by the Commission's commitment to openness to the public and will allow the public to observe and, in some cases, participate in the regulatory process.

1. Two-Year Trial Program for Conducting Open Enforcement Conferences

This policy statement discusses the 2-year trial program to allow selected enforcement conferences to be open to the public.

This may cause some indirect impact on licensees if the open meetings cause licensees to inhibit their participation due to the public forum. Because this is a trial program the regulatory burden on licensees will be assessed before a decision is made on a final policy statement.

2. Cooperation with States at Commercial Nuclear Power Plants and Other Nuclear Production or Utilization Facilities; Policy Statement

The policy statement allows State representatives to observe NRC inspections and entrance and exit meetings. NRC will consider State participation in inspections and entrance and exit meetings.

This could cause a burden to licensees who have very active State oversight. In this case it may be similar to a Federal regulator and a State regulator although the policy statement is careful to state that this is not the intention.

D. The following policy statements address conduct of operations at nuclear power plants. The policy statements should be reviewed and evaluated to determine whether they should be combined or go into rulemaking or whether the portions of the policy statements superseded by rulemakings should be deleted.

1. Nuclear Power Plant Staff Working Hours

This policy states that licensees shall establish controls to prevent situations where fatigue could preclude the ability of operating personnel to keep the reactor in a safe condition. Enough personnel should be employed to maintain adequate shift coverage without requiring heavy use of overtime.

Portions have been incorporated into 50.54(m) and generic letters.

2. Commission Policy Statement on Engineering Expertise on Shift

In this policy, the Commission continues to stress the importance of providing engineering expertise on shift. The intent of the policy statement may be satisfied by either of the options described, but the Commission prefers Option 1--a combined senior reactor operator/shift technical advisor (SRO/STA) position. In the long term, the Commission would prefer that the STA be combined with the shift supervisor in the dual role position.

This appears to be more than policy; it appears to require licensees to adopt one of the options and states the Commission's preferred option. The staff issued SECY 93-193 on July 13, 1993, to address this issue.

3. Policy Statement on Conduct of Nuclear Power Plant Operations

The policy statement outlines the Commission's expectation of utility management and license operators establishing and maintaining a professional working environment. The policy statement defines safety culture and provides criteria that reflect the expectations. This is similar to the objectives of INPO.

This appears to state more than policy in that it provides performance criteria to licensees.

4. Education for Senior Reactor Operators and Shift Supervisors at Nuclear Power Plants; Policy Statement

The Commission believes safety is enhanced by having on each shift a team of licensed professionals that combine technical and academic knowledge with plant-specific training and operating experience. The policy statement encourages licensees to sustain and increase the professionalism of the operators. The policy statement also restates the Commission's preference that all licensees continue to move toward the dual role (SRO/STA) position.

It appears this policy statement is a request for licensees to implement actions without promulgating a rule.

E. The following policy statement appears to have been interpreted as a requirement and imposes a regulatory burden on licensees. The policy statement should be reviewed and evaluated to determine whether revisions are necessary or rulemaking should be undertaken. No change in the intent would occur as a result of rulemaking; it would codify, as regulatory requirements, the informal guidance contained in the policy statement.

1. Availability and Adequacy of Design Basis Information at Nuclear Power Plants; Policy Statement

The policy statement establishes the Commission's expectations with regard to the availability of design basis information. The Commission believes that all power reactor licensees should assess the accessibility and adequacy of their design basis documentation. The licensee should decide whether a design basis reconstitution program is necessary. The Commission expects that licensees will have current design documents and adequate technical bases to demonstrate that the plant is consistent with the design basis and that the plant is being operated consistent with the design basis. The NUMARC guidelines are a useful framework. The policy statement also states that a generic letter will be issued requesting licensees to describe their program, use the responses to prioritize inspections, modify SALP (Systematic Assessment of Licensee Performance) to explicitly address licensee programs, and encourage self-identification.

This is an enormous burden on licensees who have to reconstitute their design bases. It appears to threaten licensees with inspections and incorporation into the SALP process based on the licensee's response to the Commission's expectations with respect to the availability of design basis information.

ANALYSIS OF PUBLIC COMMENTS

There appeared to be a consensus among the respondents that the industry has expended a considerable amount of time and effort regarding this policy statement and any further NRC action would be considered disruptive to those efforts.

2.3.12 PROGRAMS IDENTIFIED

The Review Group reviewed ongoing staff efforts to determine the areas that the staff was already addressing to relieve regulatory burden and streamline the regulatory process. A list and short description of the efforts is included below. The list represents ongoing efforts to the knowledge of the Review Group; all ongoing efforts may not be listed.

FITNESS FOR DUTY - Part 26

SECY-92-271 - Recommends the Commission approve reducing the testing rate to 50 percent for licensee employees and keep 100 percent testing for contractors. The Commission approved this option, a notice of proposed rulemaking was published for comment, and the staff is currently evaluating the comments.

SECY-92-308 - Identifies a number of proposed amendments to the fitness-for-duty (FFD) rule based on the lessons learned. This has been before the Commission since September, but the Commission has not voted; they have concerns regarding the backfit issue and have asked RES to do a study on the question.

COMSECY-92-018 - Requests the staff to look into the scope of testing under the FFD rule (should secretaries with no access to vital areas have to be tested). The staff is continuing its review of this issue.

SECY-91-293 - Proposes changes to the FFD rule. The staff withdrew some of its information collection requirements that requested licensees to provide test results by process stage and management actions on appeals and their resolutions.

The Review Group supports the efforts being conducted to determine a realistic scope and testing rate for the FFD program (see Section 2.3.5 of this report).

CRGR PRESIDENTIAL REVIEW

SECY-92-141 - A special review of existing regulations promulgated after 1981 was performed that resulted in eight areas of regulation changes, six of which affect nuclear power plants. They include: (1) changing the FSAR submittal frequency to once per refueling cycle, (2) changing the 50.59 report frequency to once per refueling cycle, (3) allowing receipt back of low-level waste, (4) changing the radiological effluent report frequency, (5) reducing the number of event reports required, and (6) deleting the need to request exemptions to use a certain type of cladding in fuel bundles.

Areas not addressed under this program were referred to the Marginal-to-Safety Program. The Marginal-to-Safety Program is discussed in Section 2.3.17 of this volume.

MARGINAL-TO-SAFETY PROGRAM

SECY-92-263 - Describes the staff plans for the first 3-year period for eliminating requirements that are marginal to safety but impose a burden on licensees. The staff identified three areas to initiate rulemaking (Appendix J, Appendix R, and 10 CFR 50.44, combustible gas control), identified two areas where license requirements could be relaxed (MSIV leak control system, containment leakage rate), and identified four areas to analyze further (Appendix B, EQ, security, and post-accident sampling system). A marginal-to-safety workshop was held on April 27-28, 1993, to discuss these issues.

The Review Group's recommendations regarding the Marginal-to-Safety Program are discussed in Section 2.3.17 of this report.

PRA WORKING GROUP

SECY-92-273 - The working group's first status report summarized the general characteristics of the staff's approach, addressed the use of external experts (seeking outside help to use PRA in developing generic approaches), and reviewed risk evaluations jointly with ACRS.

SECY-92-428 - The working group's second status report provided information on the present staff use of PRA and an assessment of the limitation on its uses. The group looked at experience and training of staff, the guidance the staff uses, and the methods and scope of the use of PRA. The group has three remaining tasks: developing guidance for PRA uses, assessing training and staffing needs, and assessing PRA methods development needs.

TECHNICAL SPECIFICATIONS

Risk-Based Technical Specifications

The staff has no formal program, but is reviewing an extensive proposal by the South Texas Project to decrease surveillance frequency and increase allowed outage times (AOTs) based on their PRA as a measure of the effect redundancy has beyond the Standard Technical Specification basis document considerations.

San Onofre has a pilot program that is comparing the AOTs and surveillance frequencies determined from the PRA to those in their Technical Specifications to see if their Technical Specifications are overly restrictive from a risk perspective. Efforts are also under way at River Bend and Grand Gulf to incorporate risk-based techniques into plant operation.

The staff is participating in the low-power/shutdown risk study.

The December 23, 1992 memo from Taylor to Commission provides a discussion of risk insights in Technical Specifications.

Improved Standard Technical Specifications

The improved Standard Technical Specifications (STS) have been issued, and the staff is waiting for licensees to adopt them; the first lead plants are under way. Reliefs were based on risk analyses that were proposed in topical reports and generally accepted by the staff. Reliefs were granted on AOTs and surveillance frequency; the changes were reasonable from a risk perspective. Approximately 40 percent of the old Technical Specifications were moved to licensee-controlled documents.

The December 13, 1991 memo from Taylor to Commission provides an update of the status of the improved Standard Technical Specifications.

A final policy statement for STS has been issued.

Line-Item Technical Specification Improvements

The staff is still developing and approving line-item improvements. These improvements are available independent of the new STS. Licensees may adopt these through the license amendment process.

(See December 13, 1991 memo from Taylor to Commission.)

The Review Group endorses the staff efforts regarding the improved STS development and addresses risk-based Technical Specifications in Volume Four of the report. The Review Group also endorses line-item Technical Specification improvements as discussed in Volume Three of the report.

OPERATOR LICENSING - Part 55

SECY-92-432 - Because of industry comments on the inconsistencies of the operator licensing process, the staff is embarking on a study to look at centralization versus decentralization of examiners for operator licensing. The study should be completed in July 1993.

SECY-92-430 - Proposed rulemaking to allow licensees to perform their own requalification exams after a determination by the NRC that they are qualified to do so.

The Review Group is supportive of these efforts.

REGULATORY AGENDA

NUREG-0936 - Quarterly reports on all final rules, proposed rules, advance notices of proposed rulemaking, petitions for rulemaking, and unpublished rules.

The Review Group recommendations regarding the Regulatory Agenda are contained in Section 2.3.17 of this volume.

STANDARD REVIEW PLAN UPDATE

The staff is performing a limited update of the standard review plan (SRP) to ensure it contains the latest codes and standards references. Based on the SRP update, NRR will identify those regulatory guides that are still referenced and used in the SRP.

The Review Group supports the SRP update program. See Section 2.3.15 of this volume for a discussion on updating of the regulatory guides.

MAINTENANCE/B-56 - 10 CFR 50.65

SECY-91-385 - Describes the staff plans for revising and issuing the inspection procedures to be used for inspection of licensee maintenance activities. In the interim period from the present until the rule becomes effective (1996), the staff will revise the inspection procedures and make them performance based. The staff will also revise the inspection procedures to accommodate the implementation of the new rule.

SECY-92-229 - The staff plans to endorse NUMARC's document for the implementing guidance for the new rule. The Commission has approved this approach.

SECY-92-385 - Provides the Commission with the revised Inspection Procedure 62703 for use until the rule is effective in 1996.

The staff is considering whether to resolve B-56 by encompassing it in the maintenance rule. The staff is also exploring the feasibility of allowing licensees to take credit for activities under the maintenance rule to meet the license renewal rule.

REPORTING REQUIREMENTS

The staff is currently preparing data on a limited number of regulations that contain reporting requirements and identifying the type of report; the purpose of the report; the organizations receiving, reviewing, and using the report; the potential actions taken as a result of the report; resources to produce and review the report; any similar reporting requirements; the potential reduction in public health and safety that would result if the requirement were eliminated; and the potential savings. The staff is preparing a Commission paper to characterize an approach for assessing agency reporting requirements and provide the data.

The Review Group has addressed reporting requirements in Section 2.3.16 of this report.

REGULATORY IMPACT SURVEY

SECY-91-172 - Provided an evaluation of the results of the surveys conducted to determine utility views on the effect of the NRC on the operation of nuclear power plants. A number of improvements to the regulatory process were identified.

The staff is continuing to obtain feedback from licensees on the regulatory process. During site visits, Division Directors and Associate Directors for Projects met with their licensee counterparts; during the regional assessments, members of the team visited two licensee sites; and during the management team visits, members of the team met with licensees. The staff will provide an annual report to the Commission discussing the findings.

The Review Group endorses these feedback mechanisms.

REVISION OF SOURCE TERM

The staff is reassessing the source term used in order to incorporate new technology and knowledge. (See NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms.") The staff is also updating the seismic portion of Part 100--currently out for public comment on how to assess the seismic design (probabilistic or deterministic).

CURRENT LICENSING BASIS

SECY-92-314 - discussed the findings of audits performed on licensee programs. A task force has been set up to revisit some of the plants discussed in the Commission paper and to address Commissioner Curtiss' questions and concerns on SECY 92-314. A Commission paper is scheduled for September 1993.

The Review Group discusses this in Section 2.3.10 of this report.

10 CFR 50.59

The final policy statement on the improved Standard Technical Specifications has been issued and addresses how items that have been relocated will be controlled. A letter has been sent to NUMARC discussing the industry's guidance on performing safety evaluations, NSAC-125. The letter is not an endorsement, but acknowledges industry effort.

INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS

NMSS is currently working on rewriting the requirements for independent spent fuel storage installations, including the security requirements to make the NRR and NMSS approaches alike.

The Review Group supports this effort.

TRAINING - 50.120

The Commission issued a final rule on April 26, 1993, that requires each licensee to establish, implement, and maintain programs that consider all modes of operation for the training of nuclear power plant personnel. The rule requires training programs be derived from a systems approach to training. (Very performance-based rule). The staff foresees no change in already existing programs. Training programs accredited and implemented consistent with the INPO accreditation program would be in compliance with the rule.

The staff has developed performance-based inspection criteria. The rule supersedes the policy statement previously discussed in Section 2.3.11 of this report.

The Review Group supports this effort.

ENGINEERING EXPERTISE ON SHIFT

There are a lot of diverse opinions on the use of a dual role STA/SRO. The Commission recommendation for a dual role STA was prepared at the time when the Commission was encouraging degreed operators (this resulted in a degreed person in the control room). It was also intended as an interim measure until all the NUREG-0737 and Supplement 1 actions were implemented.

The staff issued a Commission paper (SECY 93-193) on July 13, 1993, further discussing the STA/SRO dual role.

As discussed in Section 2.3.11 of this volume, the Review Group recommends that policy statements not be used as surrogates for rulemaking.

SHIFT STAFFING

Whatever happens related to the STA situation, the staff has found that perhaps the minimum staffing number in 50.54(m) may not be enough for plants to meet all their obligations during an event. There have been a number of events where a plant has had minimum staffing duty, and an event has occurred where the capability to perform certain regulatory or plant-specific actions was not available. In reality, all licensees have increased their staff above the regulatory minimum requirements.

Another area where licensees have gone beyond the regulatory requirement is in the composition of the fire brigade. The fire brigade requires a person with operational expertise--some licensees have interpreted this to mean one or more SROs, ROs, or STAs on the fire brigade, which may take away from the number of trained people in the control room should an event occur.

There are a lot of similarities in this issue with the issue of security staffing. The rules are very clear as to the minimum or regulatory requirement, but for whatever reasons (safety or external pressures from the NRC) the licensees have chosen to go beyond the rule.

The Review Group recommends licensees determine how many staff are needed to perform their safety functions and, if necessary, request relief from their current license requirements. The NRC staff should be receptive to changes in licensing amendment requests that still meet the requirements of the regulations.

PROCUREMENT

The staff is preparing a revision to the inspection procedure regarding procurement programs (38701) to address some of the problems encountered during these types of inspections. It is expected that a draft will be put out for comment in mid-1993.

The staff also held workshops to discuss these issues with the industry.

The Review Group discusses the procurement issue in Sections 2.3.1 and 2.3.13 of this volume.

REGULATORY ANALYSIS GUIDELINES

The staff is revising the guidelines it uses to evaluate any regulatory initiative (rules, generic letters, bulletins, etc.) to incorporate risk-based guidelines, incorporate the safety goal[1], encourage performance-based initiatives, develop guidelines for evaluating the benefit and impact, with the final goal being development of a well-justified initiative. A draft Commission paper is being prepared and will go out for public comment.

The Review Group recommends this effort be continued.

GENERIC ISSUES

The staff is revising the threshold for identifying something as a generic issue so that it uses a more risk-based approach and narrowing the scope of what is called a generic issue.

The Review Group recommends this effort be continued such that it may lead to quicker resolution of generic issues.

REVISION OF $1000/PERSON-REM

In 1973 this number, which is used for determining the cost benefit for reducing radiation doses, was identified as an interim number. The agency would go through rulemaking to determine the correct number. The staff is currently working on developing a new number and current plans indicate that a policy paper may be completed by the end of 1993.

The Review Group recommends this effort be continued to develop a more realistic value for cost-benefit purposes.

2.3.13 QUALITY ASSURANCE AND PROCUREMENT

I. INTRODUCTION

The charter of the Review Group called for a detailed review of regulations and implementing practices that appeared to go beyond that which is required for adequate protection. Specifically, the review was to focus on the feasibility of substituting flexibility in the form of performance-based rules for prescriptive ones and to take advantage of risk insights.

In the area of quality assurance (QA), it was found that the regulations are performance-based, but that the industry has not taken advantage of this flexibility.

All operating nuclear power plants have QA programs implemented. The requirement to discuss the QA program in the FSAR stems from the regulations [10 CFR Part 34(b)(6)(ii)], and program content is outlined in Appendix B to Part 50. Appendix B, as written, is performance based; it provides licensees with apparent flexibility. Moreover Appendix B itself contains words that indicate that the QA program should be applied in a graded manner. Specifically, Criterion II of Appendix B to Part 50 states (in part):

> "The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety."

The graded approach is not, however, found in practice. What is found is a full application of all aspects of QA to every phase of plant operations that can be considered safety-related. This appears to exceed the regulatory requirement.

When applied to some activities, procurement, for example, the extensive documentation of QA for all systems, components, and parts has not necessarily produced increased reliability of safety-related equipment. What it has produced is a system for obtaining spare parts (specifically "commercial grade" parts) that consumes vast amounts of engineering, QA, and management time without evidence of an equivalent increase in the assurance of reliability or safety.

II. STRUCTURE OF QUALITY ASSURANCE

Figure 1 shows the structure of the NRC's QA requirements and supporting guidance. This structure and process were initially put in place to ensure that the construction of nuclear power plants was of high quality in an economic environment conducive to allowing an excess burden to be assumed. This same program structure was brought forward to cover the operations phase of reactors in a now significantly different economic environment in which an unnecessary cost burden is far less tolerable. In review of the governing regulations and supporting documents, the review group found that significant flexibility exists in 10 CFR 50, Regulatory Guide 1.33, and ANS 3.2/ANSI 18.7. As a consequence, two licensee QA plans were reviewed. As a result of these reviews, the group concluded that the lack of flexibility was introduced at the plant procedure level. This creates a unique problem with its roots at the specialist level, both at utilities and in the NRC. QA specialists within both institutions have caused the implementation of quality assurance to develop into a burden without a corresponding increase in safety.

The QA programs at operating nuclear power plants are generally the logical outgrowths of the QA programs that were implemented for the construction of the facility. Appendix B is the common regulatory root for both construction and operations QA programs. Appendix B has performance-based wording, which lends itself well to dual usage.

The documents that support QA (see Figure 1) envision a procurement situation in which there is a broad base of industrial manufacturers each with its own QA program in place, which meets Appendix B. This is not the case today, because the current market for commercial nuclear material is not for the structures, systems, and components that are needed to build a plant, but rather it is for a limited number of replacement parts. Manufacturers apparently do not find that maintaining an Appendix B QA program is in their economic interest for this relatively small market. This results in procurement's having become an activity focused on buying "commercial grade" parts and material and dedicating these commercial parts and material to safety-related applications. Part 21 of the regulations defines terms such as commercial grade and dedication of parts. (See Section 2.3.1 of this report.)

The combination of "pedigree" requirements and the lack of Appendix B manufacturers contributed to a situation in which fraudulent material was introduced into the supply system. Nuclear power plants were not the only affected industry, for the military and other industries (e.g., the airlines) faced significant safety issues with fraudulent repair parts. The cost of documenting, testing, and inspecting "pedigreed" material as contrasted to producing identical material without "pedigree" provided the environment for parts with fraudulent pedigrees to enter the market place. Fraudulently documented and misrepresented material by its nature poses a threat to system or equipment reliability (and hence to safety). But the actual impact may be small. For example, low pressure systems (e.g., the service water system) built with pipe that is not as strong as that ordered by the designer may perform adequately throughout a plant's life or may have any accelerated degradation which would be detected through normal surveillance programs. The loss in this instance would appear to be economic and not one of public protection.

In summary, licensees carried forward successful QA programs from construction to operations when regulatory burden and economic conditions were less of a factor. The question of applying QA in a graded manner per Criterion II of Appendix B was not a serious consideration.

III. DISCUSSION

When past NRC communications to licensees, such as generic letters, are reviewed it appears obvious that the staff never thought in terms of a graded approach to QA. For example, Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related," stated in part:

> "The QA controls in Appendix B to 10 CFR 50 describe one form of a comprehensive management control system for a complex task. While Appendix B describes only one such system, licensees and applicants have expressed a desire to minimize proliferation of different kinds of management systems for their plants. The NRC staff concurs with this desire not to establish new and separate management control systems for non-safety-related ATWS equipment."

There is a recognition that QA is a management control system. The generic letter concurs with the industry that a second or alternate QA system for non-safety-related equipment is not desirable, but remains silent on the use of a graded QA system.

Review of the standard review plan reveals that the fire prevention and protection system was also informally brought under the Appendix B umbrella. In fact, when the standard review plan was subjected to a word search for the term "quality assurance," there were 811 instances identified in 91 different sections. It appears that QA has became a focal point for many problems and issues with the staff encouraging its broad application.

All power plants have a "Q-List," which is the list of all parts, components, and material used in safety-related applications. The origin of the Q-List is Regulatory Guide 1.29, "Seismic Design Classification." This regulatory guide provides a broad list of structures, systems, and components that are designated as Seismic Category I and to which Appendix B should be applied. As discussed above, the Q-Lists have grown to include other systems and components over which either licensee management or the NRC has seen the need to exercise greater control.

IV. QUALITY ASSURANCE AS A MANAGEMENT TOOL

QA is a management tool. Its goal is reasonable assurance that a structure, system, or component will function as designed when called upon to do so. This can be termed reliability. It is this reliability that provides confidence that there is no undue risk to the health and safety of the public.

As with any tool, QA should be used appropriately and effectively to achieve the desired result...reliability. Review of the various Appendix B criteria reveal that there are in fact several QA tools, thus QA may be considered a tool box or collection of tools. For example, there are various methods (or tools) that can be used to reach a reasonable assurance of the reliability of structures, systems, and components. One method is to test (Criterion XI). Another method is to "pedigree" the material, component, or part (Criteria III, IV, VI, VII, & VIII). The use of all of the QA tools for each task may or may not be appropriate. The importance to safety of the structure, system, or component and the ability to achieve a reasonable assurance of reliability are the factors that should dictate which tool or tools are to be used.

V. SAFETY-RELATED AND IMPORTANCE TO SAFETY

Appendix A to 10 CFR 50 provides the General Design Criteria for Nuclear Power Plants. The introduction to Appendix A states (in part):

> "The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety."

Appendix B to 10 CFR 50 provides Quality Assurance Criteria for Nuclear Power Plants. It states (in part):

> "Nuclear power plants...include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public."

and

> "The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components..."

and

"...quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service."

Generic Letter 84-01 dealt with the NRC use of terms **"important to safety"** and **"safety-related."** It pointed out that QA would need to be applied (in specific cases) to equipment that was outside of Appendix B (safety-related), but was important to safety. The specific quote is:

"While previous staff licensing reviews were not specifically directed towards determining whether, in fact, permitees or licensees have developed quality assurance programs which adequately address all structures, systems and components important to safety, this was not because of any lack of regulatory requirements for this class of equipment. Rather our practice was based upon the staff view that normal industry practice is generally acceptable for most equipment not covered by Appendix B within this class. Nevertheless, in specific situations in the past where we have found that quality assurance requirements beyond normal industry practice were needed for equipment "important to safety," we have not hesitated in imposing additional requirements commensurate with the importance to safety of the equipment involved. We intend to continue that practice."

If QA is viewed as a "yes" or "no" function, the addition of important to safety would apparently greatly increase the scope activities that in effect are under Appendix B. But if QA is viewed as a set of tools, then application of the right tool or tools appropriate to each application allows QA to be effective and efficient as a means of achieving reliability in a wide variety of structures, systems, and components (safety-related, important to safety, or because management believes that QA provides an appropriate control mechanism). The use of a graded approach to QA provides such a tool.

VI. APPLICATION OF QA TOOLS

It is apparent that all structures, systems, and components do not have an equal safety value (whether they be **safety-related** or only **important to safety**). There is only one reactor vessel, and its potential failure would be so serious an accident that all reasonable measures to protect its integrity are warranted (the use of every possible QA tool is justified). In contrast, where there are redundant and diverse means to perform a function such as to pump cooling water, the relative importance of a single pump is not as great as is the integrity of the reactor vessel.

Similarly, the function of each part of a system or component may not be of equal importance to safety. For example, the pump discussed above performs a safety function, and its reliability to perform this function is important to safety. However, a small line connecting the pump's discharge pipe to a local gage is not as important. It could fail, but this would only result in a small loss of flow from the pump and would probably not affect the pump's ability to perform its safety function adequately. The pump's reliability is not challenged significantly by the tubing used to connect the discharge to the local gage. Therefore, the consequences of inadvertently installing the wrong tubing do not have safety significance, and the application of QA measures for procuring the tubing could be made more appropriate to the risk.

The above examples suggest that Q-List items could be assigned a relative safety significance. It is also apparent that no more QA should be applied to a specific task than is needed to achieve the reasonable assurance of reliability.

The regulations would appear to allow this, but neither the historical development of QA at nuclear plants nor staff practice has encouraged a graded approach to QA.

The inspection process has generally not encouraged the use of a graded approach to QA. Inspectors ask for records and written proof upon which to base their findings. Somewhere, the recognition and acceptance of reasonable engineering judgment has been lost. For example, Generic Letter 89-02 ("Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products") stated in part:

> "Involvement of a licensee's engineering staff in an effective procurement process would normally include (1) development of specifications to be used for the procurement of products to be used in the plant, (2) determination of the critical characteristics of the selected products that are to be verified during product acceptance, (3) determination of specific testing requirements applicable to selected products, and (4) evaluation of test results. The extent of engineering involvement is dependent on the nature and use of the products involved."

This clearly implies a graded approach in the application of the QA tools. It places reliance and credence in engineering judgment. What it does not resolve is the traditional approach of the QA purist who would demand the use of all QA tools in every application. The answer may be that, despite the efforts over the last several years to shift the QA focus to be performance based, the traditional reliance on paper and pedigree as a surrogate for quality and reliability has been so ingrained that the system has effectively resisted change.

VII. VENDOR QA PROGRAMS

As discussed in Section II of this paper, a broad base group of vendors and suppliers with Appendix B programs does not exist. This results in the use of the dedication of commercial grade parts for procurement. If a vendor does not have an Appendix B QA program and licensees cannot accept the quality controls that the vendor has, then the licensee must use the dedication process. This does not appear to be the optimal use of the QA tools.

Licensees used to audit vendors who had QA programs that met the requirements of Appendix B and if the audit results justified it, accept the vendor products. The requirement for vendor QA stems from Appendix B, Criterion IV, states in part:

> "To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix."

This requirement does **not** say a full Appendix B QA program is needed in every case. It does say that a vendor must have a program that provides appropriate QA. Past practice has tended to put the onus on the vendor to demonstrate why he does not have a one-for-one match with Appendix B. Safety would lead one to conclude that the emphasis should have been on determining if the vendor had a program that delivered hardware that met the procurement specifications. There are many quality assurance programs in use: The system of military specifications (MilSpecs) is one example. Another is the QA standards initially developed in Europe (and based on Appendix B), which are gaining wide acceptance throughout the world. It would seem that the narrow focus on Appendix B programs instead of a focus on reasonably measuring an effective system for controlling quality has contributed to the potential for the use of the burdensome dedication process.

VIII. USE OF RISK INSIGHTS IN QA

Risk technology provides quantification to plant accident and operational scenarios. While one may perhaps argue about the validity of specific determinations, the relative significance of certain event sequences and equipment failures is generally accepted as correct. It would therefore appear logical to use risk insights as one means to determine where the quality assurance effort should be expended. In other words, this concept appears to have the potential for the grading of the Q-list for structures, systems, and components that are safety-related, important to safety, or captured on the Q-List for other reasons.

IX. INFLUENCE OF INSPECTION ACTIVITIES ON QUALITY ASSURANCE

The NRC's inspection program has had an effect on quality assurance over the years. The Inspection Manual index (dated August 21, 1992) lists 33 inspection procedures in the Quality Assurance (35000 series) section. Additionally, there are three more inspection procedures listed in the Procurement (38000 series) section.

The Review Group looked at the application in the inspection process in procurement inspections. The applicable inspection procedures are 38701 ("Procurement Program") and 38703 ("Commercial Grade Procurement Inspection"). The inspection procedures were reviewed and found to be typical and acceptable. It should be noted, however, that Inspection Procedure 38703 states the following in the inspection guidance section:

> "Highlighted for your information below, are four methods for accepting commercial grade items....
>
> "Method 1 - Special Tests and Inspections...
>
> "Method 2 - Commercial Grade Survey or Evaluations of Supplier...
>
> "Method 3 - Source Verification...
>
> "Method 4 - Acceptable Supplier/Item Performance Record..."

An inspection report for procurement was reviewed; it emphasized the importance of the paper trail and found fault with licensee performance that did not provide a perfect document package. Each individual finding was technically correct and justified under the regulations and licensee commitments. There was no mention of performance or other testing to demonstrate that the parts procured would perform reliably in service. Thus the Review Group concluded that the regulatory message being heard by licensees was that the paper trail--a surrogate--was the path to follow. What appeared to be missing was a message that linked procurement (especially the commercial grade dedication process) to safety and reliability. The Review Group also noted that the basic findings of unsatisfactory performance in the arena concluded that the material or parts procured were of "questionable quality." There were no inspection results found during the review that reached a determination that the affected equipment or system would not perform reliably.

X. ONGOING ACTIVITIES

There are two ongoing NRC staff activities in the area of QA and of procurement which are potentially significant. These are: (1) a workshop was held in April 1993 to solicit public comments on the draft revision of the inspection procedure for commercial grade procurement, and (2) the update and revision of Regulatory Guide 1.33. Both of these activities offer the opportunity for public participation and change.

Additionally, by letter dated July 20, 1993, NUMARC informed the NRC that it has formed a new working group to access and improve current QA practices associated with the implementation of 10 CFR Part 50, Appendix B requirements.

XI. SUMMARY

Quality assurance is a vital part of safety at nuclear power plants. It is a management tool for achieving reliability and a means to an end in attaining safety. The regulations are written in a performance-based manner. In practice, QA is not performance based. There is a long-term tendency to use paper (the pedigree) as a substitute for engineering judgment.

This does not necessarily provide increased reliability or promote an improved level of safety. In fact, there are some who believe that an overemphasis on pedigree may detract from reliability. This occurs when the pedigree becomes so all-consuming an effort that engineering judgment and experience are not used.

In order for QA to be transformed from the prescriptive, paper-dominated system into which it has developed to a potentially more effective, appropriately documented program for safety, a significant change in both licensee and staff thinking would be required. Implementation of such change would probably be evolutionary. It would appear that a staff-endorsed articulation of expectations followed by an industry-phased implementation of specific measures would be the practical way to start changing QA to a performance-based conception.

XII. RECOMMENDATIONS

The Review Group did not identify the need to revise the regulations for quality assurance (Appendix B to Part 50) in the near term, but did conclude that some of the implementing documents and guidance will need to be revised in order to implement Appendix B in a performance-based and graded manner. It should be noted that, if implementing guidance is modified, this will cause licensees to revise their individual QA plans. This could result in some staff action (under the provisions of 10 CFR 50.54(a)) but not in license amendments.

XIII. ANALYSIS OF PUBLIC COMMENTS

One respondent commented that the acceptance criteria of the Standard Review Plan (SRP) Sections 17.1 and 17.2 may provide a barrier to adopting "Total Quality Management" and performance-based quality systems. The respondent recommended that the staff consider parts of the Department of Energy's (DOE) Order 5700.6c. The comment and recommendation also stated that SRP Section 17.3 and the DOE Order 5700.6c were similar and noted that the DOE order had only 10 criteria instead of the 18 criteria of the SRP 17.3 (i.e., of Appendix B to Part 50). Finally, it was noted that there was a section after each of the criteria in DOE Order 5700.6c entitled "Discussion," which provided insight into the intent and expectation of the specific criterion. It was suggested that this might provide the basis for SRP acceptance criteria for the existing Appendix B.

The recommendation merits consideration. Although the Review Group did not recommend the rewrite or restructuring of Appendix B to Part 50 at this time, the group understood why some commenters recommended a complete review and restructuring of Appendix B and hold the opinion that this may be necessary and desirable. If such a rewrite/restructure were directed (or even studied), all options should be considered. This could include an approach similar to the one used in DOE Order 5700.6c. Moreover, the restructuring or repackaging of quality assurance criteria to a different form is a recognized option. As a part of this, the addition of a "Discussion" section to delineate intent and expectations (as a bridge to the SRP) might be appropriate, although it would appear that this purpose could also be served by the SRP itself or by a regulatory guide.

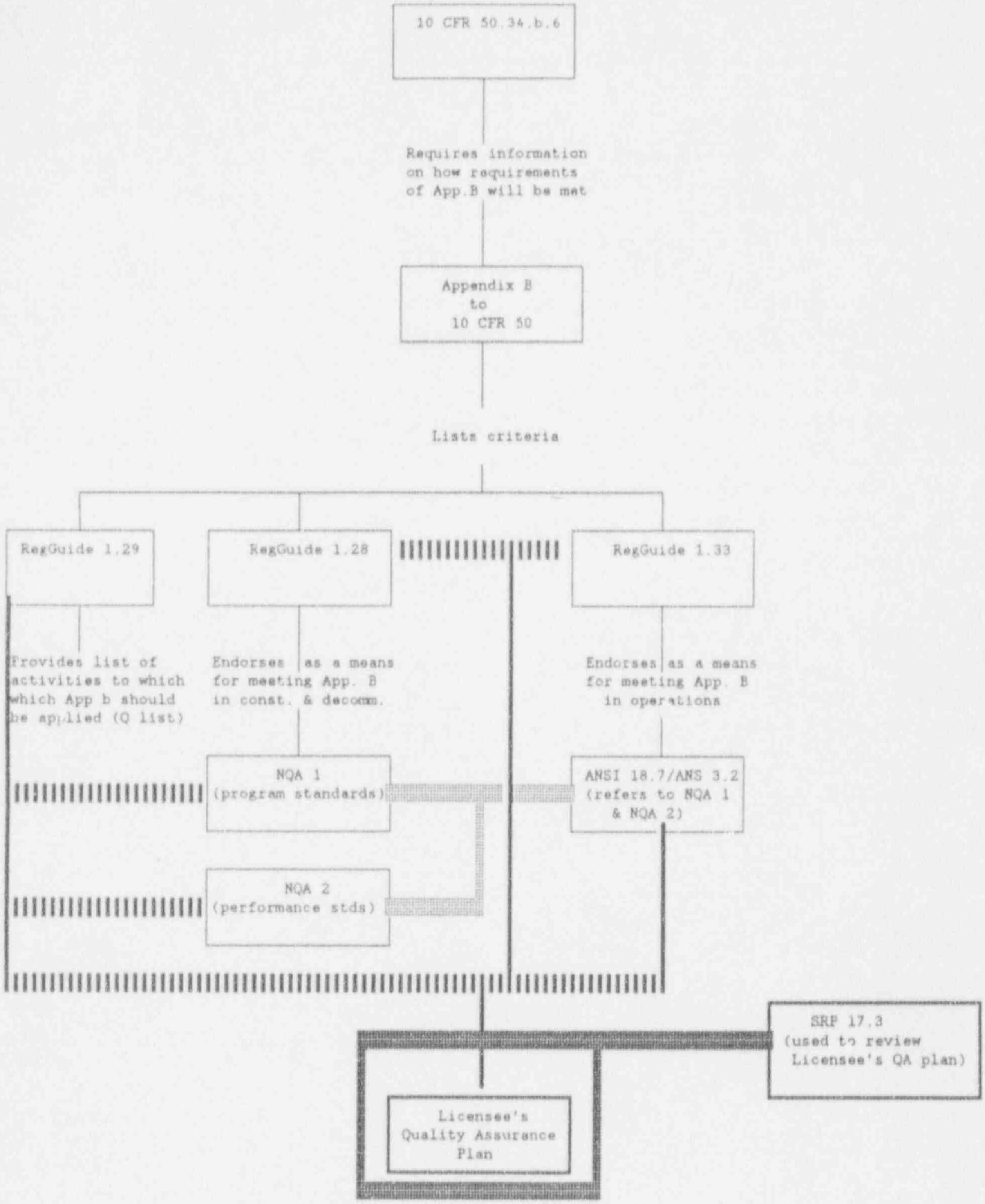
10 CFR 50.34.b.6
Requires information on how requirements of App.B will be met
Appendix B to 10 CFR 50
Lists criteria
RegGuide 1.29
RegGuide 1.28
RegGuide 1.33
Provides list of activities to which which App b should be applied (Q list)
Endorses as a means for meeting App. B in const. & decomm.
Endorses as a means for meeting App. B in operations
NQA 1 (program standards)
ANSI 18.7/ANS 3.2 (refers to NQA 1 & NQA 2)
NQA 2 (performance stds)
SRP 17.3 (used to review Licensee's QA plan)
Licensee's Quality Assurance Plan

Figure 1

2.3.14 QUALITY ASSURANCE PROGRAMS FOR FIRE PREVENTION AND PROTECTION SYSTEMS

The regulations (10 CFR 50.48) require that all licensees have a fire protection plan that satisfies Criterion 3 of the General Design Criteria (Appendix A to Part 50).

I. INTRODUCTION AND BACKGROUND

Appendix R to Part 50 establishes the features required to satisfy Criterion 3 of the General Design Criteria only for plants licensed prior to January 1, 1979. The fire protection plan has to describe the features necessary to implement the program. Branch Technical Position 9.5-1 and Appendix A thereto are referenced in a footnote to the rule as guidance documents. The appendix to the branch technical position states:

> "The quality assurance (QA) programs should ensure that the guidelines...for the fire protection systems for safety-related areas are satisfied...The QA program should be under the management control of the QA organization. This control consists of (1) formulating a fire protection QA program...The QA program for fire protection should be part of the overall plant QA program."

Another footnote, which is in both 10 CFR 50.48 and Appendix R, cites four documents that provide clarification and guidance with respect to permissible alternatives to satisfy Appendix A to the branch technical position. These documents address supplementary guidance for fire protection evaluation, sample technical specifications, manpower requirements, and fire protection functional responsibilities, administrative control, and quality assurance. Although it appears that these documents were used as guidance to licensees and the staff to develop and evaluate the fire protection plan, they were committed to by many licensees, and they were not developed as formal regulatory guides (no number or public comment).

The last document listed was sent out to applicants by a letter, which stated that it was being used by the NRC as supplemental guidance for the review and evaluation of the organization and administrative aspects of the fire protection system. The document states much of what is in the branch technical position, but also states that applicants and licensees can meet the fire protection quality assurance program criteria of Appendix A to Branch Technical Position 9.5-1 or Regulatory Guide 1.120 either by implementing the fire protection quality assurance criteria as part of their Appendix B [to Part 50] program or by providing a separate fire protection quality assurance program. The branch technical position identifies "10" of the "18" Appendix B criteria as criteria which should be satisfied. Regulatory Guide 1.120 states much of what is in the appendix to the branch technical position.

II. DISCUSSION

After the rule was promulgated and implemented, the staff provided further guidance on the "QA program requirements for fire protection" in Generic Letter 86-10. This generic letter states that fire protection systems are not safety-related and therefore not within the scope of Appendix B [to Part 50] unless the licensee has committed to include the system under their Appendix B quality assurance program based on the guidance contained in the branch technical position or the 1977 quality assurance document. There is no apparent requirement in either 10 CFR 50.48 or Appendix R that states a specific fire protection quality assurance plan is required. The fire protection system should be covered under the General Design Criterion 1 and Appendix B quality assurance programs commensurate with the safety significance of the equipment and system. Interpretation, however, of the informal guidance has led to the situation where some licensees either develop specific fire protection quality assurance plans or apply the entire Appendix B quality assurance program to the entire fire protection system. The Review Group believes that some form or degree of quality assurance should be applied to the fire prevention and protection system, for it would appear that this system is important to safety (but not safety-related).

III. RECOMMENDATION

The Review Group endorses continued licensee use of quality assurance in a graded manner to their fire prevention and protection system in accordance with the provisions of Criterion II of Appendix B to Part 50 (or develop a separate quality assurance system in keeping with the system's importance to safety).

1975, this guide parallels some license requirements for reporting as delineated in the Administrative Section of Technical Specifications. It includes specific guidance on how to prepare some reports (e.g., monthly operating reports, reportable occurrences). These reports were required under a different regulatory oversight from what exists today. They need to be updated to reflect the needs of an expanded licensing inspection process with resident inspectors and improved communications. Most of the data elements of the monthly operating report are unnecessary, and the few that are used could be collected far less frequently. Since performance indicator data are generally computed on a quarterly basis, it would be appropriate to reduce the frequency of collection of those elements remaining in the monthly operating report submittal. (See Section 2.3.16 of this report.)

There are several regulatory guides that address quality assurance. In some instances, these regulatory guides endorse consensus standards, such as the ANSI 45.2 and daughter series, to which many licensees are committed. The technical content of the ANSI 45.2 series was adequate, but it has been superseded as guidance by NQA-1 and NQA-2 for newer plants. It would appear that revision of the regulatory guides endorsing consensus standards and providing NRC expectations in the area of quality assurance is warranted. It would also appear that the regulatory process would be best served if the number of regulatory guides affecting quality assurance were reduced, if possible to a single guide.

Section 2.3.13 in this volume addresses quality control and procurement. Without discussing the issues contained therein, it is noted that the current guidance used by the industry for commercial grade dedication is contained in an EPRI document, and the staff comments on and endorsement of this important process were by generic letter, not regulatory guide. In revising and condensing the regulatory guides affecting quality control, current practice in procurement and commercial grade dedication should be included.

There is recent experience in the termination of licensees, and there is a prospect that more licenses may be terminated. Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," was issued by the Atomic Energy Commission in 1974. This regulatory guide is a prime candidate for review and revision to reflect current experience and expectations. (See Section 2.3.3 in this report.)

III. RECOMMENDATIONS

The major recommendation of the Review Group with regard to regulatory guides is that significant staff resources not be devoted to a wholesale revision or update of them. This recommendation also implies that revisions should be made only on a case-by-case basis when good cause is demonstrated. A complete list of the regulatory guides that were reviewed is included in Appendix A to Volume Two.

2.3.15 REGULATORY GUIDES

I. INTRODUCTION

This paper discusses the review of the Division 1 regulatory guides, which are the regulatory guides affecting power reactors (except in the area of plant protection).

II. DISCUSSION

Based on a review of the "Regulatory Guide List," there are 122 Division 1 regulatory guides currently in effect. Over the years, more than 20 regulatory guides have been withdrawn. Additionally, there are 21 draft regulatory guides that were issued between 1979 and 1991. Based on the age and lack of industry or staff interest in the completion of these draft regulatory guides, all of them older than 18 months should be withdrawn.

Review and classification of the active Division 1 regulatory guides indicated that less than half support power plant operations. The majority address design and construction issues. For those Division 1 guides currently effective, the issue dates (or the revision in effect) range from 1970 to 1992, with a mean issue date of 1978. There were four revisions issued in 1992. Three of these were updates to show ASME Code Case acceptability. The other was Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors." In fact, only 25 regulatory guides or revisions have been issued during the last 13 years (from 1980 through the present).

Additional review of the guides indicated that 44 of them endorsed an industry standard (e.g., IEEE Std.). However, the mean date of issue of these was 1980. These primarily were concerned with design and construction issues.

The facts presented above can be used to derive two conclusions. The first is that regulatory guides themselves are not a prime source of new burden on licensees. The second conclusion is that there is a small group of regulatory guides that need to be updated or revised on a regular basis. The best examples of this are those guides that list the acceptability of various ASME Code cases. Other more recent (last 6 years) guides that were issued addressed areas on which there has been significant regulatory activity (e.g., 1.155, "Station Blackout," and 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants").

While not imposing a new burden, the older guides such as Regulatory Guide 1.16, "Reporting of Operating Information-Appendix A Technical Specifications (for Comment)," appear to support the continuation of unnecessary burden. Issued in August

1975, this guide parallels some license requirements for reporting as delineated in the Administrative Section of Technical Specifications. It includes specific guidance on how to prepare some reports (e.g., monthly operating reports, reportable occurrences). These reports were required under a different regulatory oversight from what exists today. They need to be updated to reflect the needs of an expanded licensing inspection process with resident inspectors and improved communications. Most of the data elements of the monthly operating report are unnecessary, and the few that are used could be collected far less frequently. Since performance indicator data are generally computed on a quarterly basis, it would be appropriate to reduce the frequency of collection of those elements remaining in the monthly operating report submittal. (See Section 2.3.16 of this report.)

There are several regulatory guides that address quality assurance. In some instances, these regulatory guides endorse consensus standards, such as the ANSI 45.2 and daughter series, to which many licensees are committed. The technical content of the ANSI 45.2 series was adequate, but it has been superseded as guidance by NQA-1 and NQA-2 for newer plants. It would appear that revision of the regulatory guides endorsing consensus standards and providing NRC expectations in the area of quality assurance is warranted. It would also appear that the regulatory process would be best served if the number of regulatory guides affecting quality assurance were reduced, if possible to a single guide.

Section 2.3.12 in this volume addresses quality control and procurement. Without discussing the issues contained therein, it is noted that the current guidance used by the industry for commercial grade dedication is contained in an EPRI document, and the staff comments on and endorsement of this important process were by generic letter, not regulatory guide. In revising and condensing the regulatory guides affecting quality control, current practice in procurement and commercial grade dedication should be included.

There is recent experience in the termination of licensees, and there is a prospect that more licenses may be terminated. Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," was issued by the Atomic Energy Commission in 1974. This regulatory guide is a prime candidate for review and revision to reflect current experience and expectations. (See Section 2.3.3 in this report.)

III. RECOMMENDATIONS

The major recommendation of the Review Group with regard to regulatory guides is that significant staff resources not be devoted to a wholesale revision or update of them. This recommendation also implies that revisions should be made only on a case-by-case basis when good cause is demonstrated. A complete list of the regulatory guides that were reviewed is included in Appendix A to Volume Two.

The Review Group further suggests that maintenance of the following regulatory guides be considered within the context of the above recommendation:

- Regulatory Guides 1.84, 1.85, and 1.147, which delineate ASME Code Case acceptability, should be maintained current;

- The regulatory guides that delineate NRC expectations for quality assurance (1.26, 1.28, 1.29, 1.30, 1.33, 1.37, 1.38, and 1.39[1]) should be reduced to the fewest possible number (i.e., one preferably, but no more than three) and these regulatory guides should be maintained current and include the NRC position on commercial grade procurement and dedication; and

- Regulatory Guide 1.86, which delineates termination of operating licenses, should be reviewed and revised.

IV. POTENTIAL IMPROVEMENTS

Regulatory Guide 1.16 should be reviewed and revised to eliminate unnecessary reporting requirements.

V. ANALYSIS OF PUBLIC COMMENTS

While there were few public comments related to the Review Group analysis of regulatory guides and the resulting recommendations listed above, the comments received were widely divergent. One respondent commented that since regulatory guides are used as de facto requirements during inspections, the guidance should be updated for consistency with the regulations and current NRC staff positions. Another respondent instead recommended voiding outdated regulatory guides to eliminate a source of confusion. The need for regulatory positions in those cases where consensus standards (e.g., NQA-1) have been endorsed was also questioned by yet another respondent.

Regulatory guides are recognized by the NRC as providing information on acceptable methods for implementing NRC requirements, rather than substitutes for the regulations. Compliance with the regulatory positions is not required except in those isolated cases in which a regulatory guide is incorporated into the license. The Review Group believes that the existing regulatory guides should not represent a significant burden to licensees and that only a small subset of the current guides need to be updated on a regular basis.

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[1] There are other regulatory guides that relate to quality assurance, but these generally refer to process control instead of program elements.

With regard to inspection practices, NRC personnel routinely use regulatory guides as background information in the evaluation of licensee programs and their implementation. Where deviations from specific licensee commitments to regulatory positions are identified, NRC followup is expected so that the discrepancies can be reconciled. However, licensees are not held accountable in the enforcement arena for compliance with regulatory positions to which they have not committed, unless further inspection identifies unacceptable alternative practices in violation of the regulatory requirements. Such an inspection policy provides no additional burden on licensees who are in compliance with NRC regulations and is consistent with the Review Group recommendations that significant staff resources not be devoted to the wholesale revision of the body of the existing regulatory guides.

2.3.16 REPORTING REQUIREMENTS

I. INTRODUCTION

This paper discusses the findings of the Review Group regarding reporting requirements for nuclear power plants. It is based on the results both of the sub-group assigned to review regulations and of the sub-group assigned to review selected licenses. Because an NRC task force to assess reporting requirements for power reactor licensees currently exists, the reporting requirements addressed in this section are limited to those routine reports required by the regulations and those contained in the plant operating license, including the Technical Specifications and the environmental plan. Other situational reporting requirements to the NRC, including those contained in documents administratively controlled by the licensee, are not addressed in this evaluation.

II. ASSESSMENT APPROACH

Each of the reporting requirements evaluated was placed into one of the following categories:

Appropriate (A) - The reporting requirement appears to be reasonable and to have a sound regulatory basis; no further review of this item is deemed necessary.

Further Evaluation (E) - The reporting requirement appears to contain elements that would make it a candidate for a reduction in regulatory burden; additional review is necessary to determine if the item requires the transmission of information that is redundant, is of no further utility to the NRC, or has a reporting frequency that is inappropriate.

Redundant (R) - The reporting requirement appears to duplicate provisions for reports or the transmission of data and information that is specified elsewhere in NRC regulations, Technical Specifications, or license conditions.

Not Required (N) - The reporting requirement appears to have little regulatory basis or only marginal utility; elimination of this item as a regulatory requirement merits strong consideration.

The categorization for each of these reporting requirements was determined by considering the answers to the following 11 questions.

1. Is the information reported used by the NRC in an ongoing evaluation process upon which the NRC depends to carry out its mission?

2. Is the information collected only for use in the event of an incident or emergency?

3. Is the information collected, evaluated, and/or published by the NRC in order to keep the public informed?

4. Is NRC guidance available that permits elimination of this reporting requirement?

5. Is there NRC guidance that allows a reduction in frequency of submittal of this report?

6. Is the information reported used in the determination of generic or plant-specific safety decisions?

7. Is the reporting frequency or timeliness of the information critical to the NRC's use of the information?[1]

8. Would extending or eliminating the reporting requirement have a detrimental effect on the NRC's ability to execute its safety mission?

9. Is the information reported readily available in other forms or reports that would suffice to satisfy the NRC's needs?

10. For requirements delineated in licenses, are these redundant to reporting requirements in regulations?

11. Is the information reported used for background or trending?

The questions were written in such a manner that the responses to the questions would determine the categorization of each reporting requirement.

III. DISCUSSION OF REPORTS REQUIRED BY REGULATIONS

NUREG-1460, "Guide to NRC Reporting and Recordkeeping Requirements," provides a listing of reporting requirements that are based on the regulations. It delineates over 60 reports required by 10 CFR Part 50; however, this number is somewhat misleading for two reasons.

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[1] For information that is "non-critical," reporting frequencies of "annual" or "refueling cycle" are considered appropriate.

First, most of the reporting requirements listed for 10 CFR Part 50 are situational[2] in nature rather than periodic (routine).[3] In fact, only six repetitive (routine) reports are delineated in 10 CFR Part 50. Second, there are 10 separate line items referring to 10 CFR 50.72 or 50.73. This tends to inflate the report total of NUREG-1460.

Similarly, NUREG-1460 lists some 18 line items from 10 CFR Part 26 that require reports; however, only one of these is a routine, periodic report--the semiannual fitness-for-duty performance data, which is discussed in Section 2.3.5.

All the reports listed for Part 21 are situational. There are 61 report line items in NUREG-1460 listed under 10 CFR Part 73, but only 11 of these apply to nuclear power plants, and just one of the 11 is periodic (the quarterly safeguards log submittal is discussed in Section 2.3.18 of this Volume). Another of the 11 reports (the submittal of fingerprint cards) occurs so frequently that it could be counted among the periodic reports.

The periodic (routine) reports were analyzed using the 11 questions to determine their categorization. The results of the analysis are contained in Table 1.

The eight periodic reports evaluated are:

- The annual effluent release report [10 CFR 50.36a(a)(2)],
- The annual report of changes to the quality assurance program that do not reduce commitments in the program [10 CFR 50.54(a)(3)],
- The annual report of insurance and financial security [10 CFR 50.54(w)(3)],
- The annual report of changes, tests, and experiments made without prior Commission approval [10 CFR 50.59(b)(2)],
- The annual financial report [10 CFR 50.71(b)],
- The update of the FSAR [10 CFR 50.71(e)(4)],

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[2] "Situational" means a report is required whenever predetermined threshold or trigger level for initiation has been exceeded.

[3] "Periodic" means a report is required at a specified frequency.

- The quarterly submittal of safeguard events logs [10 CFR 73.71(b)(2)], and
- The semiannual fitness for duty performance data [10 CFR 26.71(d)].

Five of the eight 10 CFR 50 reporting requirements were determined to be "appropriate" with regard to both the need for the reporting requirement for the NRC to perform its function and frequency of submittal. It should be noted that for the reporting requirements associated with the effluent release report, the FSAR updates and revisions, and the report of changes, tests, and experiments, this decision was at least partially based on the fact that the frequency of the reporting requirements related to these regulations was recently changed from either semiannual to annual or from annual to refueling cycle (not to exceed 2 years). It was determined that the annual financial report, which is the fifth reporting requirement listed, requires "further evaluation."

The evaluation also categorized the quarterly submittal of safeguard events logs as "Not Required" and the semiannual submittal of fitness-for-duty performance data as requiring "further evaluation." With site resident inspectors and inspectors with security expertise in each regional office who can access these logs at the site, the submittal of the safeguard events logs is no longer necessary. Since the NRC is only publishing the extensive data related to drug testing submitted by licensees on an annual basis, the need for the submittal of this performance data on a semiannual basis requires further evaluated.

IV. DISCUSSION OF REPORTS REQUIRED BY TECHNICAL SPECIFICATIONS

Part of the Review Group's charter related to the assessment of operating licenses called for the identification of requirements in the license that could be categorized as actions that may not be needed for protection of the health and safety of the public. During the course of review of four operating reactors, certain reporting requirements were categorized as such items. Since few differences in the reporting requirements between the BWRs and PWRs were identified, it was decided to assess Seabrook and Surry (an older and newer PWR). The reporting requirements contained in the operating licenses, which includes the Technical Specifications and environmental protection plan, for these two plants are presented in Tables 2 and 3.

There were 46 and 22 reporting requirements identified in the Seabrook and Surry operating licenses, respectively. The Seabrook Technical Specifications still contain the Radiological Environmental Technical Specifications (RETS), which could be removed if the licensee requested an amendment to the Technical Specifications in accordance with Generic Letter 89-01. In addition, there are also reporting requirements in the Seabrook license associated with the environmental protection plan (the Surry license does not contain an environmental protection plan). Since RETS can be removed from the

Technical Specifications and if the four environmental protection plan reporting requirements are not considered, the number of reporting requirements for Seabrook would be 33.

The categorization of each of the reporting requirements for Seabrook and Surry is contained in Tables 2 and 3, respectively. The evaluation performed to determine the categorization of each of these reporting requirements is contained in Tables 4 and 5. A summary of the number of reporting requirements in each category is as follows:

	Seabrook	Surry
Appropriate	7[4]	3
Further Evaluation	22[5]	9
Redundant	6	6
Not Required	11	4

The larger number of reporting requirements for Seabrook in comparison to Surry is directly attributable to the additional reporting requirements placed on newer plants. It is interesting to note that most of the reporting requirements added in the Seabrook license were situational. In addition, there is a significant variation of between 10 and 90 days in the time permitted to submit reports to fulfill situational reporting requirements in the Seabrook license. There also does not always appear to be a correlation between the time required to submit a report and the safety significance of the reporting requirement. For example, a report is required within 14 days if the radiation monitoring instrumentation is inoperable and 90 days if an emergency core cooling system (ECCS) actuation and injection of water into the reactor coolant system occurs.

The Seabrook license contains 22 situational reporting requirements compared to only two in the Surry license. Of these 22 reporting requirements, 20 are either related to RETS or are in the "not required" or "redundant" category. If the licensee pursued the line item improvements available through generic letters and the improved Standard Technical Specifications, most of these situational reporting requirements could be eliminated from

[4] Includes nine RETS and three environmental protection plan reporting requirements.

[5] Includes one environmental protection plan reporting requirement.

the Seabrook Technical Specifications. With the exception of the situational reporting requirements, the number of and the type of reporting requirements are about the same for Seabrook and Surry.

The number of reporting requirements evaluated that have been deemed "appropriate" is a small fraction of the total number. Some of the reporting requirements that have been the classified as appropriate, such as the annual report, still require further evaluation to determine whether the individual reporting requirements contained in the report are still needed.

The largest number of reporting requirements were characterized as requiring "further evaluation." This means that further evaluation is required to determine if either the frequency of the reporting requirement can be reduced or if the reporting requirement can be eliminated in its entirety. For example, the need for the licensee to submit a monthly operating report and the contents of this report need to be assessed by the staff. Additionally, although Generic Letter 89-01 permits the relocation of the procedural details of the RETS into the licensee-controlled Offsite Dose Calculation Manual or the Process Control Program, which would remove it from the Technical Specifications, the reporting requirements would still exist. Therefore, the reporting requirements associated with RETS for Seabrook were categorized as requiring "further evaluation" to assess both their need and frequency rather than categorizing them as "not required."

Almost all the "redundant" reporting requirements identified were redundant to the reporting requirements contained in 10 CFR 50.72 and 10 CFR 50.73 so there is no significant increase in regulatory burden in retaining these requirements in the license if the licensee chooses to do so as an aid to the operators. However, in some instances there is an inconsistency between the license and the regulations in the time allowed to submit the report. For example, if an ECCS activation and injection of water should occur, the Seabrook Technical Specification permits 90 days to submit a report even though the licensee would have to submit a Licensee Event Report within 30 days in accordance with 10 CFR 50.73.

The reporting requirements identified as "not required" were ones that could be pursued by the licensee for elimination as a line-item improvement under a generic letter or the improved Standard Technical Specifications. Therefore, no further staff action is required for the reporting requirements that are in this category.

In conclusion, it appears that almost all the reporting requirements in the Seabrook and Surry licenses are probably typical of most operating plants. Although the reporting requirements in these licenses may not be all inclusive, they include all those that are addressed by the regulations and probably most, if not all, of those that are generic. Most of the reporting requirements contained in the Seabrook and Surry operating licenses fall into one of two categories: either they require further staff evaluation with regard to frequency or need; or they are either redundant to regulatory requirements or are no longer required as a result of a generic letter or the improved Standard Technical Specifications and, therefore, could either be eliminated or reduced in frequency if the licensee pursued a license amendment.

V. RECOMMENDATIONS

The Review Group has the following recommendations for staff followup relative to the reporting requirements associated with the regulations and the license for operating plants:

- Eliminate 10 CFR 73.71(b)(2), which requires the submittal of quarterly of safeguards events logs.

- Change 10 CFR 26.71(d) to permit the submittal of the fitness-for-duty performance data on an annual basis rather than semiannually.

- License amendments to delete reporting requirements requested by licensees of the type identified as "not required" in the Seabrook and Surry licenses should be acted upon by the staff.

- Evaluate the need and/or the frequency for the generic reporting requirements identified as requiring "further evaluation" in the Surry and Seabrook licenses.

- Evaluate the need and/or the frequency for routine and situational generic reporting requirements that are contained in documents that are administratively controlled by the licensee.

- Evaluate the need for each of the individual reporting requirements contained in periodic reports.

TABLE 1 - EVALUATION OF ROUTINE, REPETITIVE REPORTS REQUIRED BY THE CODE OF FEDERAL REGULATIONS

REGULATORY REQUIREMENT	QUESTIONS											FREQUENCY	ASSESSMENT
	1	2	3	4	5	6	7	8	9	10	11		
Effluent Release Report [10 CFR 50.36a(a)(2)]	Y	N	Y	N	Y	N	N	Y	N	Y	Y	Annual	A
Annual Report on Changes in the Quality Assurance Program [10 CFR 50.54(a)(2)]	Y	N	N	N	N	N	N	Y	N	NA	Y	Annual	A
Annual Report of Insurance and financial security [10 CFR 50.54(w)(3)]	Y	N	N	N	N	N	N	Y	N	NA	N	Annual	A
Report of Changes, Tests, and Experiments [10 CFR 50.59(b)(2)]	Y	N	N	N	N	N	N	Y	N	NA	N	Refueling cycle (<2 yrs.)	A
Annual Financial Report [10 CFR 50.71(b)]	N	N	N	N	N	N	N	N	N	NA	N	Annual	E
FSAR Updates & Revisions [10 CFR 50.71(b)(4)]	Y	N	N	N	N	N	N	Y	N	NA	N	Refueling cycle (<2 yrs.)	A
Fitness for Duty Program Performance Data 10 CFR 26.71(d)]	Y	N	Y	N	N	N	N	N	N	NA	Y	Semi-Annual	E (for freq of rpt'g)
Quarterly Submittal of Security Logs [10 CFR 73.71(b)]	N	N	N	N	N	N	N	N	N	NA	N	Quarterly	N

TABLE 2: SEABROOK LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
OL 2.C.4	Transfer of Managing Authority	(3)	A
OL 2.G	Violations of Section 2.C of License	Notification, 1 Hour Report, 30 Days	A
TS 3.1.1.3	Monitor Temperature Coefficient more Positive than Beginning of Life Limit	Report, 10 Days	N
TS 3.3.3.1	Radiation Monitoring Instrumentation Inoperable	Report, 14 Days	N
TS 3.3.3.3	Seismic Instrumentation Inoperable	Report, 40 Days	N
TS 4.3.3.3.2	Results of Seismic Event Upon Facility	Report, 14 Days	N
TS 3.3.3.4	Meteorological Tower Instrumentation Inoperable	Report, 30 Days	N
TS 3.3.3.6	Accident Monitoring Instrumentation Inoperable	Report, 14 Days	N
TS 3.3.3.9	Radioactive Liquid Effluent Monitoring Instrumentation Inoperable	(4),(9)	N
TS 3.3.3.10	Radioactive Gaseous Effluent Monitoring Instrumentation Inoperable	(4),(9)	N
TS 4.4.5.5	Steam Generator Tubes Plugged and Results of C-3 Sample Inspections	(5)	E
TS 3.4.9.3	Overpressure Protection Systems - PORV's or RHR Suction Relief Valves, or RCS Vents Used to Mitigate Reactor Coolant System Transient	Report, 30 Days	E
TS 3.5.2	Emergency Core Cooling System Actuated and Injects Water into Reactor Coolant System	Report, 90 Days	R
TS 3.5.3.1	ECCS Actuation and Injection of Water into the Reactor Coolant System	Report, 90 Days	R(8)
TS 4.6.1.6	Abnormal Degradation of Containment Vessel Detected by Visual Inspection During Shutdown for Integrated Leak Rate Test	Report, 15 Days	R(8)

TABLE 2: SEABROOK LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
TS 4.6.5.3	Abnormal Degradation of Containment Enclosure Building Detected by Visual Inspection During Shutdown for Integrated Leak Rate Test	Report, 15 Days	R(8)
TS 3.7.5	Portable Tower Makeup Pump System Inoperable	Notification, 1 Hour	N
TS 4.7.8.3	Sealed Source or Fission Detector Exceed Removable Contamination Limits	Report, Annual	N
TS 3.7.10	Area Temperature Monitoring Inoperable	Report, 30 Days	N
TS 4.8.1.1.3	All Diesel Generator Failures	Report, 30 Days	E
TS 3.11.1.2	Release of Radioactive Materials in Liquid Effluents Exceeds Technical Specification Limit	Report, 30 Days (10),(13)	E
TS 3.11.1.3	Radioactive Liquid Waste Discharged without Treatment Exceeds Technical Specification Limit and Portion of Treatment System Inoperable	Report, 30 Days (10),(13)	E
TS 3.11.1.4	Quantity of Radioactive Material in Temporary Unprotected Outdoor Tank Exceeds Technical Specification Limit	(4),(9),(10)	E
TS 3.11.2.2	Radioactive Noble Gases in Gaseous Effluent Exceeds Technical Specification Limit	Report, 30 Days (10),(13)	E
TS 3.11.2.3	Release of Iodine-131 and 133, Tritium and Radionuclides in Particulate Form in Gaseous Effluents Exceeds Technical Specification Limit	Report, 30 Days (10),(13)	E
TS 3.11.2.4	Radioactive Gaseous Waste Discharged Without Treatment Exceeds Technical Specification Limit	Report, 30 Days (10),(13)	E
TS 3.11.4	The Annual Dose to Any Member of the Public from Release of Radioactive Materials in Liquia and Gaseous Effluent Exceeds the Technical Specification Limit	Report, 30 Days (10)	E

TABLE 2: SEABROOK LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
TS 3.12.1	Radiological Environmental Monitoring Program (REMP) not Conducted as Specified or Radioactivity in Sampling Medium at Specified Location Exceeds Reporting Levels of REMP	(6),(10)	E
TS 3.12.2	Land Use Census Location Yields Dose Higher than Currently Used and Resulted in Addition of New Locations in the Offsite Dose Calculation Manual	(4),(9),(10)	E
TS 4.12.2	Land Use Census Performance	(7)	E
TS 3.12.3	Interlaboratory Comparison Program Analysis on all Radioactive Materials not Performed as Required	(7)	E
TS 6.5	Reportable Events	(8)	R(8)
TS 6.6	Safety Limit Violation	Notification, 1 Hour Report, 14 Days	R(8)
TS 6.8.1.1	Startup Report	Report, 90 Days	E
TS 6.8.1.2	Annual Reports - Personnel Exceeding Dose Limits, Primary Coolant Exceeded Technical Specification Limits, Challenges Power Operated Relief and Safety Valves	Report, Annual	A(11)
TS 6.8.1.3	Radiological Environmental Operating Report	Report, Annual (Prior to May 1)	A(11)
TS 6.8.1.4	Radiological Effluent Report	Report, Semi-Annual	A(9),(11)
TS 6.8.1.5	Operating Report	Report, Monthly	E
TS 6.8.1.6	Core Operating Limits Report	Prior to Each Reload Cycle	A
TS 6.12.2.a	Changes to Process Control Program	(4),(9)	E
TS 6.13.2	Changes to Offsite Dose Calculation Manual	(4),(9)	E

TABLE 2: SEABROOK LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
TS 6.14.1	Major Changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems	(4),(9)	E
EP 3.2	Changes to the NPDES Permit or State Certification	Report, 30 Days	E
EP 4.1	Unusual or Important Environmental Events	Notification, 1 Hour Report, 30 Days	A
EP 5.4.1	Environmental Operating Report - Noncompliances, Changes to Station Design or Operation, Summaries and Analyses of Environmental Activities	Report, Annual (Prior to May 1) (12)	E
EP 5.4.2	Nonroutine Reports Related to Occurrence of a Nonroutine Event	Report, 30 Days	E

FOOTNOTES

(1) OL - Operating License, TS - Technical Specification, EP - Environmental Plan.

(2) Appropriate (A) - the reporting requirement appears to be reasonable and to have a sound regulatory basis; no further review of this item is deemed necessary.

Further Evaluation (E) - the reporting requirement appears to contain elements that would make it a candidate for a reduction in regulatory burden; additional review is necessary to determine if the item requires the transmission of information that is redundant or of no further utility to the NRC.

Redundant (R) - the reporting requirement appears to duplicate provisions for reports or the transmission of data and information that is specified elsewhere in NRC regulations, other Technical Specifications and/or license conditions.

Not Required (N) - the reporting requirement appears to have little regulatory basis or only marginal utility; elimination of this item as a Technical Specification or license condition merits strong consideration.

TABLE 2: SEABROOK LICENSE REPORTING REQUIREMENTS TO THE USNRC

(3) Various notifications, reports, and changes to the Joint Ownership Agreement related to the NRC issuance of a license amendment approving transfer of management authority to North Atlantic Energy Service Company from Public Service Company of New Hampshire.

(4) Semi-annual Radiological Effluent Report.

(5) Report within 15 days following completion of Inservice Inspection, the number of tubes plugged and report within 30 days and prior to resumption of plant operation the results of C-3 sample inspection.

(6) Report in annual REMP if program not as specified and submit report in 30 days if reporting level of REMP exceeded in sampling medium.

(7) Annual Radiological Environment Operating Report.

(8) Report in accordance with requirements of 10 CFR 50.72 and 50.73 or current reporting requirement redundant to these reporting requirements.

(9) Regulation 10 CFR 36a has been changed to permit submitting of a Radiological Effluent Report from semi-annually to annually. Licensee has to submit an amendment to the Technical Specifications requesting this change to the reporting requirement.

(10) In accordance with Generic Letter 89-01 the procedural details of the Radiological Effluent Technical Specifications can be relocated to the Offsite Dose Calculation Manual or the Process Control Program and associated reporting requirements submitted in the annual Radiological Effluent Report if the licensee submitted an amendment to the Technical Specifications requesting the change.

(11) Report, although appropriate, contains multiple items that are reported. The reporting requirements for each of these items should be individually evaluated.

(12) Redundant to Technical Specification 6.8.1.3.

(13) Reporting requirement in 10 CFR 50.73(a)(VIII)(A) and (B) associated with exceeding Part 20, Appendix B limits for airborne and liquid effluent releases, respectively.

TABLE 3: SURRY LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
OL 3.C	States Reports Made in Accordance with the Technical Specification	(3)	R
OL 3.G	Steam Generator Repair	(4)	N
TS 3.1.D.4	Specific Activity of Reactor Coolant Exceeds Specified Limit	Report, Annual	E
TS 3.1.G.3	Reactor Coolant System Overpressure Mitigating System Used to Mitigate a Transient	Report, 30 Days	E
TS 3.7.E.2	Number of Explosive Gas Monitoring Instrumentation Channels Inoperable Less than Required	(5)	N
TS 3.12.B.7	Flux Power Tilt not Corrected and Design Hot Channel Factors are not Exceeded	(6)	N
TS 4.4.E	Containment Tests	(7)	R(14)
TS 4.10	Reactivity Anomalies	(8)	E
TS 4.19.F	Steam Generator Tubes Plugged and Results of ISI and C-3 Sample Inspections	(9)	E
TS 6.2.A	Reportable Events	(3)	R(3)
TS 6.2.B	Immediate Notification	(3)	R(3)
TS 6.3.A	Safety Limit Violation	(10)	R(3)
TS 6.6.A.1	Startup Report	Report, 90 Days	E
TS 6.6.A.2	Annual Reports - Personnel Exceeding Dose Limits, Primary Coolant Exceeded Technical Specification Limits	Report, Annual	A(15)
TS 6.6.A.3	Operating Reports	Report, Monthly	E
TS 6.6.B.1	Inservice Inspection Evaluation	Report, After 5 Years Operation	N
TS 6.6.B.2	Radiological Environmental Operating Report	Report, Annual (Prior to May 1)	A(15)

TABLE 3: SURRY LICENSE REPORTING REQUIREMENTS TO THE USNRC

LICENSE SECTION (1)	SUBJECT OF REPORTING REQUIREMENT	TYPE/TIME	ASSESSMENT (2)
TS 6.6.B.3	Radiological Effluent Report	(11)	A(12)(15)
TS 6.6.B.4	Containment Leak Rate Test, Periodic Test Results	(13)	R(14)
TS 6.6.C	Special Report - Reactor Vessel Overpressure Mitigating System	Report, 30 Days	E
TS 6.8.B.3	Changes to Offsite Dose Calculation Manual	(11),(12)	E
TS 6.9.A.1	Major Changes to Radioactive Waste Treatment Centers	(11) or UFSAR Update	E

FOOTNOTES

(1) OL - Operating License, TS - Technical Specification, EP - Environmental Plan.

(2) Appropriate (A) - the reporting requirement appears to be reasonable and to have a sound regulatory basis; no further review of this item is deemed necessary.

Further Evaluation (E) - the reporting requirement appears to contain elements that would make it a candidate for a reduction in regulatory burden; additional review is necessary to determine if the item requires the transmission of information that is redundant or of no further utility to the NRC.

Redundant (R) - the reporting requirement appears to duplicate provisions for reports or the transmission of data and information that is specified elsewhere in NRC regulations, other Technical Specifications and/or license conditions.

Not Required (N) - the reporting requirement appears to have little regulatory basis or only marginal utility; elimination of this item as a Technical Specification or license condition merits strong consideration.

(3) Report in accordance with requirements of 10 CFR 50.72 and 50.73 or current reporting requirement redundant to these reporting requirements.

(4) Identifies obsolete reporting requirements not removed from license.

(5) Special report if inoperable more than 30 days, no time specified.

(6) Special report if after further 24 hours tilt not corrected, no time specified.

FOOTNOTES

(7) Inspection and reporting in accordance with Appendix J.

(8) Report in accordance with Section 6.6, station reporting requirements of Technical Specifications. However, it is not specified in Section 6.6 under which reporting requirement this should be reported.

(9) Report within 15 days, following completion of Inservice Inspection (ISI) the number of tubes plugged, report on annual basis, for period in which inspection completed, results of ISI, and report prior to assumption of plant operation the results of C-3 sample inspection.

(10) Safety limit violation report submitted to NRC within 14 days of violation.

(11) Semi-annual Radiological Effluent Report.

(12) Regulation 10 CFR 50.36 has been changed to permit submitting of a semi-annual Radiological Effluent Report annually. Licensee has to submit an amendment to the Technical Specifications requesting this change to the reporting requirement.

(13) Report initial Type A test results approximately 3 months after test, periodic tests that meet acceptance criteria in periodic operating report, results that fail acceptance criteria in separate summary report.

(14) Redundant to 10 CFR 50, Appendix J, Section V.

(15) Report, although appropriate, has multiple reporting requirements that need to be individually assessed.

TABLE 4 - EVALUATION OF SEABROOK LICENSE REPORTING REQUIREMENTS

License Reporting Requirement	Questions											Assessment
	1	2	3	4	5	6	7	8	9	10	11	
OL 2.C.4	N	N	N	Y	N	Y	N	Y	N	N	N	A
OL 2.G	Y	N	N	N	N	Y	Y	Y	N	N	N	A
TS 3.1.1.3	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.3.3.1	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.3.3.3	N	N	N	Y	N	N	N	N	N	N	N	N
TS 4.3.3.3.2	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.3.3.4	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.3.3.6	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.3.3.9	N	N	N	N	Y	N	N	N	N	N	N	N
TS 3.3.3.10	N	N	N	N	Y	N	N	N	N	N	N	N
TS 4.4.5.5	Y	N	N	Y	N	Y	N	N	N	N	Y	E
TS 3.4.9.3	N	N	N	N	N	N	Y	N	N	N	Y	E
TS 3.5.2	N	N	N	N	N	Y	N	N	Y	Y	Y	R
TS 3.5.3.1	N	N	N	N	N	Y	N	N	Y	Y	Y	R
TS 3.6.1.6	N	N	N	N	N	Y	N	N	Y	Y	N	R
TS 3.6.5.3	N	N	N	Y	N	Y	Y	N	Y	Y	N	R
TS 3.7.5	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.7.10	N	N	N	Y	N	N	N	N	N	N	N	N
TS 4.7.8.3	N	N	N	Y	N	N	N	N	N	N	N	N
TS 4.8.11.3	N	N	N	N	N	Y	N	N	N	Y	Y	E

TABLE 4 - EVALUATION OF SEABROOK LICENSE REPORTING REQUIREMENTS

License Reporting Requirement	Questions											Assessment
	1	2	3	4	5	6	7	8	9	10	11	
TS 3.11.1.2	N	N	N	N	Y	N	N	N	N	Y	Y	E
TS 3.11.1.3	N	N	N	N	Y	N	N	N	N	Y	Y	E
TS 3.11.1.4	N	N	N	N	N	N	N	N	N	N	Y	E
TS 3.11.2.2	N	N	N	N	Y	N	N	N	N	Y	Y	E
TS 3.11.2.3	N	N	N	N	Y	N	N	N	N	Y	Y	E
TS 3.11.2.4	N	N	N	N	Y	N	N	N	N	Y	Y	E
TS 3.11.4	N	N	N	N	Y	N	N	N	N	N	Y	E
TS 3.12.1	N	N	Y	N	N	N	N	N	N	N	Y	E
TS 3.12.2	N	N	Y	N	N	N	N	N	N	N	Y	E
TS 4.12.2	N	N	Y	N	N	N	N	N	Y	N	Y	E
TS 3.12.3	N	N	N	N	N	N	N	N	Y	N	Y	E
TS 6.5	Y	N	N	N	N	Y	Y	Y	N	Y	Y	R
TS 6.6	Y	N	N	N	N	Y	Y	Y	N	Y	Y	R
TS 6.8.1.1	N	N	N	N	N	N	N	N	N	N	Y	E
TS 6.8.1.2	Y	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.8.1.3	N	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.8.1.4	Y	N	Y	N	Y	N	N	N	N	N	Y	E
TS 6.8.1.5	Y	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.8.1.6	N	N	N	N	N	Y	Y	N	N	N	N	A
TS 6.12.2.a	N	N	N	N	N	N	N	N	N	N	N	E

TABLE 4 - EVALUATION OF SEABROOK LICENSE REPORTING REQUIREMENTS

License Reporting Requirement	Questions											Assessment
	1	2	3	4	5	6	7	8	9	10	11	
TS 6.13.2	N	N	N	N	N	N	N	N	N	N	Y	E
TS 6.14.1	N	N	N	N	N	N	N	N	Y	N	Y	E
EP 3.2	N	N	N	N	N	N	N	N	N	N	N	E
EP 4.1	Y	N	N	N	N	N	Y	Y	N	N	N	A
EP 5.4.1	N	N	Y	N	N	N	N	N	Y	N	Y	E
EP 5.4.2	N	N	N	N	N	N	N	N	N	N	N	E

TABLE 5 - EVALUATION OF SURRY LICENSE REPORTING REQUIREMENTS

License Reporting Requirement	Questions											Assessment
OL 3.C	N	N	N	N	N	N	Y	Y	Y	Y	Y	R
OL 3.G	N	N	N	N	N	N	N	N	Y	N	N	N
TS 3.1.D.4	N	N	Y	N	N	N	N	N	N	N	Y	E
TS 3.1.G.3	N	N	N	N	N	N	Y	N	N	N	Y	E
TS 3.7.E.2	N	N	N	Y	N	N	N	N	N	N	N	N
TS 3.12.B.7	N	N	N	Y	N	N	N	N	N	N	N	N
TS 4.4.E	N	N	N	Y	N	N	N	N	Y	Y	N	R
TS 4.10	N	N	N	N	N	N	N	N	N	N	N	E
TS 4.19.F	Y	N	N	Y	N	Y	N	N	N	N	Y	E
TS 6.2.A	Y	N	N	N	N	Y	Y	Y	N	Y	Y	R
TS 6.2.B	Y	N	N	N	N	Y	Y	Y	N	Y	Y	R
TS 6.3.A	Y	N	N	N	N	Y	Y	Y	N	Y	Y	R
TS 6.6.A.1	N	N	N	N	N	N	N	N	N	N	Y	E
TS 6.6.A.2	Y	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.6.A.3	Y	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.6.B.1	N	N	N	N	N	N	N	N	Y	N	N	N
TS 6.6.B.2	N	N	Y	N	N	N	N	N	N	N	Y	E
TS 6.6.B.3	Y	N	Y	N	Y	N	N	N	N	N	Y	E
TS 6.6.B.4	N	N	N	N	N	Y	N	N	Y	Y	N	R
TS 6.6.C	N	N	N	N	N	N	Y	N	N	N	Y	E

TABLE 5 - EVALUATION OF SURRY LICENSE REPORTING REQUIREMENTS

License Reporting Requirement	Questions											Assessment
TS 6.8.B.3	N	N	N	N	N	N	N	N	N	N	Y	E
TS 6.9.A	N	N	Y	N	Y	N	N	N	N	Y	N	E

2.3.17 RULEMAKING PROCESS

I. BACKGROUND

The Review Group reviewed the rulemaking process to determine what types of controls and procedures govern the rulemaking process and how the process is implemented. The Review Group reviewed the four volumes of the Regulatory Agenda (NUREG-0936) for 1992. The Regulatory Agenda is issued by the Office of Administration and is a quarterly compilation of all rules upon which the NRC has recently completed action, proposed action, or is considering action; and it also lists all petitions for rulemaking the NRC has received and are pending disposition. The Regulations Handbook (NUREG/BR-0053, Revision 2), the EDO Procedures Manual (NUREG/BR-0072), and guidance contained in memoranda to the staff were also reviewed to identify the current procedures and guidance in place for rulemaking activities.

II. DISCUSSION

The Regulations Handbook provides an overview of the rulemaking process. Originally the procedures were outlined in memoranda from the Executive Director for Operations (EDO) to office directors. These were later codified in the Regulations Handbook. According to the procedures, when an office determines a rulemaking is necessary, the rulemaking must be approved by the EDO before resources can be expended and it is added to the Regulatory Agenda. This is done through an initiation for rulemaking memorandum to the EDO. If approved, the EDO procedures require that a schedule be established and a final rulemaking be complete within 2 years of the EDO's initial approval of the action. The Regulatory Agenda is updated quarterly to provide the latest status of the rulemaking.

The Office of Nuclear Regulatory Research (RES) performs most of the rulemaking activities. Twice a year RES reviews the list of rulemakings requested by other offices and discusses the priority with the responsible office. RES then sends the list of rulemakings with the priority to the EDO. RES has five categories of rulemaking actions. High-priority items are scheduled for completion within 2 years of the EDO's approval of the rulemaking. Medium-priority rulemakings are assigned as secondary rulemakings that are developed as time allows by the same person who has a high-priority rulemaking. Planned rulemakings are developed on an "as resources are available" basis. Potential rulemakings are rulemakings on hold because of unforeseen circumstances (waiting for another agency to develop companion regulations, etc.). The drop category is used for rulemakings no longer needed. The prioritization of rulemakings is a continuous process. Rulemakings shift from category to category as circumstances necessitate. However, from the review of the Regulatory Agenda, it was difficult to determine which category a rulemaking was in and why the schedule was undetermined. An understanding of the prioritization process and an identification of which category the rulemaking is in would provide the users a more complete picture of the rulemaking process and an explanation for schedule delays.

Additionally, the EDO reviews the Regulatory Agenda annually to reapprove rulemaking actions until they are issued as final actions. Other than the annual EDO review, it appears there is no independent, critical review of all the rulemakings as a whole. The Review Group concluded that this is needed and serves a useful purpose in determining whether rulemaking activities should continue in light of changing industry and regulatory initiatives. This review should be the responsibility of a separate entity, not the organization initiating the rulemaking.

As part of the rulemaking process, a regulatory history of the rulemaking must be compiled. The office responsible for originating the rulemaking is responsible for compiling the regulatory history. According to the procedures, the regulatory history should include prior drafts of the rulemaking transmitted for interoffice review, formal office comments, source documents, supporting documents (such as the regulatory analysis), environmental assessment, OMB package, and public comments (including comments from the CRGR and ACRS), Commission papers, staff requirements memoranda, and Federal Register notices. The office must ensure that the appropriate documents are included in the public document room (including accession numbers) and must compile an index of the documents that comprise the regulatory history. The index is then forwarded to the Office of Administration, which is responsible for maintaining the indices.

The procedures discussed above apply to all offices and all rulemaking activities. NRR has developed procedures for requesting RES to develop rulemakings. NRR Office Letter 400, dated February 28, 1992, describes how NRR requests RES to begin work on a rulemaking package through a user needs request memorandum. Currently, there are approximately 10 outstanding user needs requests memoranda requesting RES to develop rules.

Figure 1 provides a breakdown of the types of rulemaking the staff worked on in calendar year 1992. As seen from the figure, there are approximately 80 rulemaking activities being worked on at one time and of those, approximately 27 are proposed rules and approximately 38 are unpublished rules. Proposed rules have been published in the Federal Register and have received or are receiving comments from the public, and unpublished rules are still being developed by the staff. A portion of the proposed and unpublished rules do not have schedules for proposed action or final action. As discussed earlier, this could be a result of which category or priority the rulemaking is in. Schedules should be established in accordance with the guidance in the Regulations Handbook and the memoranda from the EDO on control of rulemaking if possible. Schedules for proposed and unpublished rules should include the date on which the EDO provided initial approval of the rulemaking to determine how long the rulemaking has been in the process and ensure the 2-year timeliness goal is met.

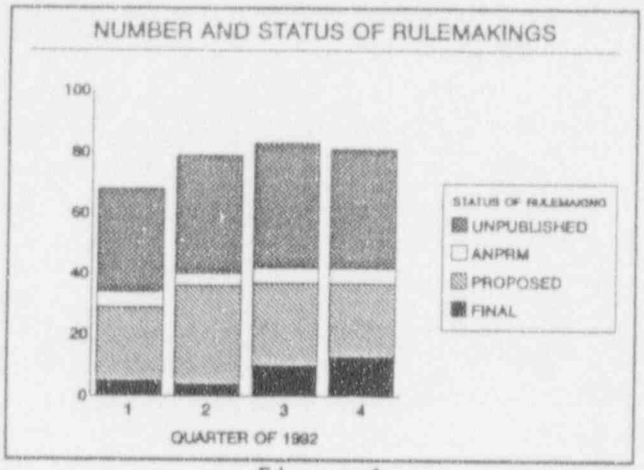

Figure 1

The following four charts provide a distribution by age of the proposed rulemakings for each quarter of 1992. The age was determined from the date on which it was published in the Federal Register, not the date on which it was approved by the EDO to which the 2-year limit applies. As seen from the charts, the majority of the rulemakings are less than 2 years old; however, a considerable number are over 4 years old. In those cases where the priority of the rulemaking is so low that staff resources are not anticipated to be available for several years, the rulemaking should be dropped. In those cases where the schedule delays are because of circumstances beyond the NRC staff's control or the result of NRC's established priority system, this should be stated clearly in the Regulatory Agenda.

NUMBER AND AGE OF PROPOSED RULEMAKINGS

FIRST QUARTER 1992

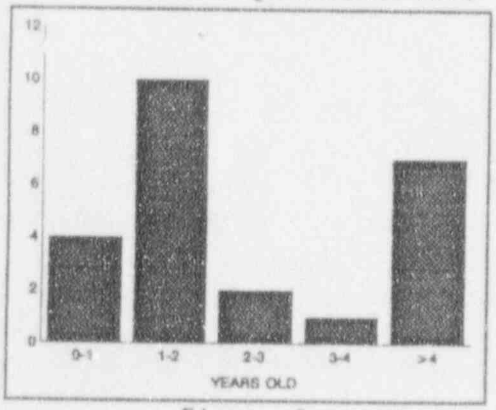

Figure 2

SECOND QUARTER 1992

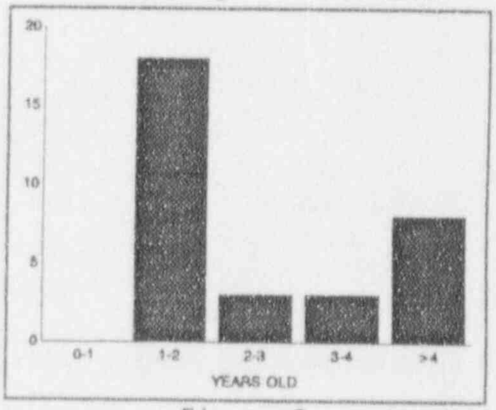

Figure 3

NUMBER AND AGE OF PROPOSED RULEMAKINGS

THIRD QUARTER 1992

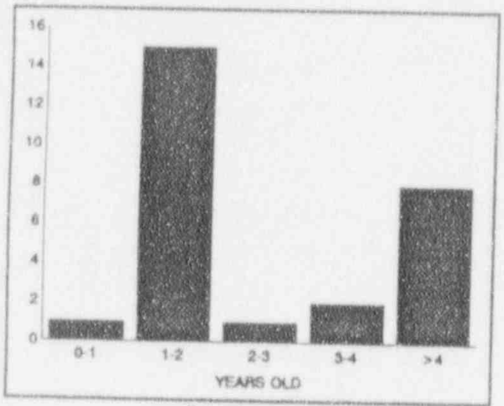

Figure 4

FOURTH QUARTER 1992

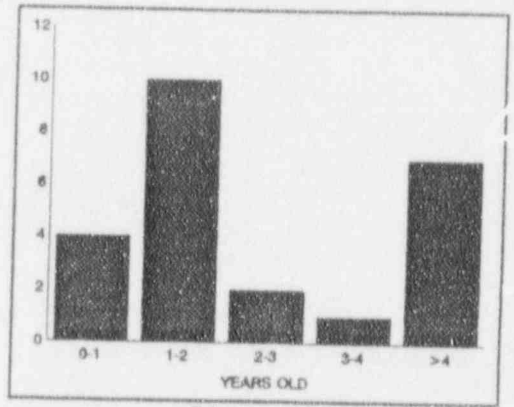

Figure 5

During the review of the Regulatory Agenda, it was identified that some of the abstracts lacked details and the latest information regarding the development of the rule and the schedule. It would be helpful to the people who rely on the Regulatory Agenda for information regarding rulemaking activities if the information were as up to date as possible concerning the status and reasons for changes to the schedule.

Procedures have also been established for petitions for rulemaking. Petitions for rulemaking are to be resolved within 12 months of the Federal Register notice notifying receipt of the petition. Petitions for rulemaking may be granted (all or in part), denied, or withdrawn by the petitioner. If a petition for rulemaking is technically sound and complete, nothing in the NRC procedures inhibits the staff from performing a timely review, publishing the petition as a proposed rule, and requesting comments. Submittal of a complete and technically sound petition for rulemaking provides the industry a method for directly decreasing the perceived regulatory burden. This would speed the rulemaking process and allow the industry a voice in developing rulemaking other than the traditional route via commenting on proposed rules developed by the staff.

Figure 6 provides a breakdown of the status of petitions for rulemaking for calendar year 1992. As seen from the chart, there are an average of 18 petitions for rulemaking with the NRC staff at any one time. Most are under review by the staff. Approximately one quarter of these do not have a resolution timetable associated with them. As discussed above, schedules should be established for resolution of these petitions in accordance with the procedures in the Regulations Handbook, the EDO Procedures Manual, and the EDO memorandum on timely resolution of petitions for rulemaking.

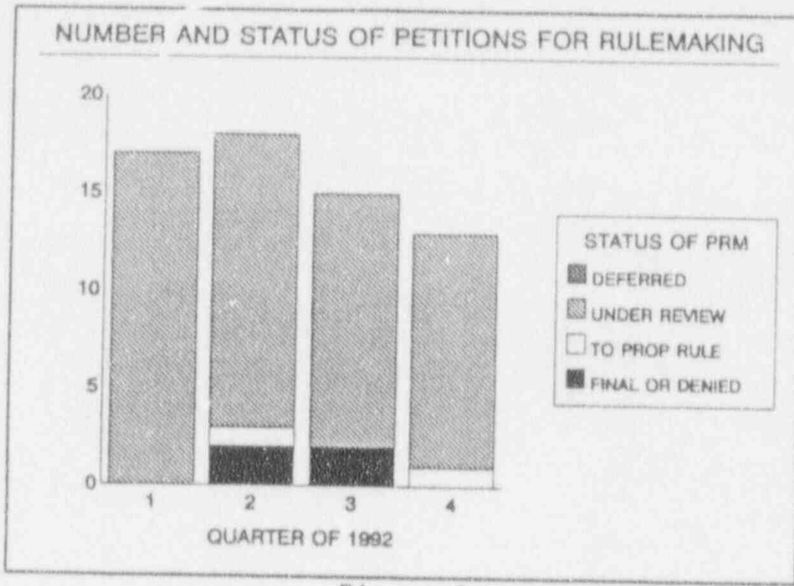

Figure 6

In 1991, the staff informed the Commission of the results of a program conducted to identify, assess, and eliminate regulatory requirements that have marginal importance to safety and impose a burden on licensees (known as the Marginal-to-Safety Program). In 1992, the Commission requested the staff to respond to a presidential request to initiate a special review of existing regulations by the CRGR to identify and eliminate regulations that have no impact on safety. As a result of public comments received from these two efforts, the staff has identified a number of regulations that have a marginal impact on safety. The staff plans to address these issues under the Marginal-to-Safety Program and has proposed an effort to address the issues through a systematic review of a number of the issues every 3 years. As part of the first 3-year period, the staff identified nine areas to eliminate, relax, or study further. The staff held a workshop in April 1993 to discuss and solicit comments on these topics. The staff identified 16 additional areas that will be reviewed during the next 3-year period.

The items identified for inclusion in the Marginal-to-Safety Program originated from broad public comments. This has resulted in a long list of issues that have little detail associated with them. In order to act on the issues, the staff has the burden of developing the issues into full rulemaking packages and determining how best to solve the problem offered up by the industry. The Review Group believes it is the industry that knows best what regulations are a burden and has the details to support changes to those regulations. The Review Group believes the Marginal-to-Safety Program should be redirected to focus on and be responsive to very specific and detailed petitions for rulemaking submitted by the industry.

The procedures in 10 CFR 2.802 describe what a petition for rulemaking should include. The regulation states that each petition set forth: (1) a general solution to the problem, (2) a statement in support of the petition that shall set forth the specific issues involved, and (3) relevant technical scientific or other data involved that are reasonably available to the petitioner. The Review Group believes this low level of detail and higher NRC staff burden appropriate for a petition that is being filed on the basis of undue risk to the health and safety of the public. However, the Review Group believes that, to maximize the Commission's responsiveness to a petition that is being proposed to eliminate regulatory burden, very detailed technical analyses and data are required to provide a high level of detail and lower staff burden. The Review Group recommends potentially revising 10 CFR 2.802 to draw a distinction between these two types of petitions for rulemaking (one that is being made for public health and safety reasons and one that is being made to eliminate burden). The accompanying guidance should outline the acceptable threshold of information and detail needed for each type of petition. The level of detail for a petition requesting relief should be such that the staff can evaluate the petition and determine whether it is substantive in a short period of time. In evaluating the adequacy of the petition, the staff will not only examine the change being sought and determine whether it is substantive (as is currently the case--is it a reasonable request), but the staff will also determine whether the content of the petition is substantive (is the level of detail sufficient to support the actions being sought). The staff should also be advised of the restrictions regarding negotiated rulemaking.

As we encourage the public and the industry to develop requests to eliminate regulations and burdensome implementing practices, it is also necessary to provide guidance on the definition and interpretation of the term "marginal" if it is to continue to be used in the safety context.

A definition that includes quantitative risk criteria for "marginal" requirements or practices at first glance appears to be acceptable in terms of evaluation, compliance, enforcement, etc. Two factors that affect the quantification of the term "marginal" are the limit of resolution associated with the current state of the art in probabilistic risk assessment (PRA) and the perception of risk by the public. Most experts would caution against focusing on more than one significant figure in understanding PRA results. This would suggest in the eyes of the PRA analyst, given the uncertainties involved, a 50 percent increase in predicted core damage frequency (e.g., from 2×10^{-5} to 3×10^{-5}) would be barely detectable, a "marginal" change having little real meaning. In contrast, many members of the public would regard a 50 percent increase in an estimate of risk as quite significant, rather than "marginal." This perception will not be easily changed in the near future. A more acceptable position could lead to a definition of "marginal" as a 10 percent increase in calculated risk. This would represent the smallest value a PRA expert might feel had some significance while still having some appeal to the public as a rational limit to the overall increase in risk associated with regulatory changes.

However, as these two examples suggest, developing an acceptable level of increase in risk is difficult at best and may not even be necessary. The Review Group believes the marginal-to-safety effort should not focus on a specific risk number, below which the effort is deemed "marginal to safety," but should focus on qualitative and performance-based criteria. Future rulemakings should be performance-based, where applicable, thereby allowing the evaluation of an issue with the general criterion that the net effect of the proposed change does not change the level of safety of a plant as currently licensed.

Requests for amendments to licenses could include, as one amendment, a number of modifications that overall have no effect on the current level of safety at the plant. In addition to demonstrating a neutral effect on safety, the proposed changes would need to be evaluated against existing deterministic criteria, such as the General Design Criteria contained in Appendix A to 10 CFR 50, to ensure the overall philosophy of defense in depth is preserved. In both the rulemaking and licensing areas, the burden and obligation would fall on licensees to submit technically sound proposals. The data and analysis must be sufficiently complete to justify the proposal. Nothing in the Atomic Energy Act precludes the Commission from making a no significant hazards finding on an amendment that proposes more than one change as long as the amendment as a whole meets the criteria of 10 CFR 50.92(c).

Encouraging the submittal of petitions for performance-based regulations and evaluating licensing actions in an integral manner with an overall acceptance criterion of neutral safety impact is a significant departure from current NRC policy and practice. The Review Group recommends the appropriate vehicle to advise licensees and the public prospectively of the manner in which the agency will exercise its discretionary power in this area is a policy statement. In keeping with Recommendation 92-2 of the Administrative Conference of the United States (as discussed in Section 2.3.11), the proposed policy statement should be issued for public comment and should make clear the policy statement is of general applicability, would provide guidance to agency personnel, and would not constitute a standard where noncompliance may form an independent basis for actions.

III. RECOMMENDATIONS

- The Marginal-to-Safety Program should be renamed to demonstrate a focus on and be responsive to specific and detailed petitions for rulemaking that are performance-based, propose to eliminate regulatory burden, and are safety neutral.

- Recognize and advise the staff and public (through guidance) of the availability of an integral approach to licensing actions such that a number of issues may be proposed in a license amendment request if the level of safety remains the same.

- The industry should take advantage of the petition for rulemaking process and submit complete, technically sound petitions in accordance with the NRC Regulatory Analysis Guidelines that the staff could review and publish expeditiously as proposed rules to request public comments.

- Revise 10 CFR 2.802 to clearly distinguish between a petition for rulemaking proposed for public health and safety reasons and one that is made to eliminate burden. The accompanying guidance should clearly state the level of detail needed for each type of petition such that the request can be evaluated by the staff in a timely manner. The staff should be advised of the restraints of negotiated rulemaking.

IV. POTENTIAL IMPROVEMENTS

The Regulatory Agenda is an excellent compilation of the rulemaking activities in the agency. The Review Group has several recommendations that would enhance its usefulness to both the industry and the NRC staff.

- Provide a discussion of the RES prioritization system in the preface of the Regulatory Agenda. Identify in which category each rulemaking is located.

- Schedules should be established for all rulemaking activities in the Regulatory Agenda. The schedules should include the date on which the action was originally approved by the EDO to improve tracking of the action.

- The abstract information regarding the rule should be current.

- Rulemakings and petitions for rulemaking whose schedules are significantly beyond the 2-year and 1-year resolution guidelines should be brought to resolution promptly. In those cases where the priority of the rulemaking is so low that staff resources are not anticipated to be available for several years, the rulemaking should be dropped.

V. ANALYSIS OF PUBLIC COMMENTS

Although one commenter recommended a wholesale change to the rulemaking process, based on most comments it appears a rule change to 10 CFR 2.802 is not warranted, but guidance should still be considered to discuss the level of detail needed for petitions that reduce burden. Additionally, one commenter requested a public comment period before very low priority rulemakings are dropped unilaterally.

2.3.18 SECURITY

This paper addresses the review of security regulations affecting nuclear power plants and supplements the memorandum from F. P. Gillespie to J. H. Sniezek, dated February 2, 1993, SUBJECT: NEAR TERM ACTIONS WHERE SAFEGUARDS PRACTICES AS PROMULGATED BY THE NRC APPEAR TO GO BEYOND 10 CFR 73.55].

I. INTRODUCTION

The regulations for security and associated matters at nuclear power plants are contained in 10 CFR Part 73. As a group they have several prescriptive elements, especially when compared to the body of other regulations affecting nuclear power plants.

The Commission requirements for security at nuclear power plants are delineated in 10 CFR 73.55. Paragraph (a) of this rule gives the general performance objective, which is to provide a high degree of assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The physical protection system is designed to protect against the design basis threat of radiological sabotage. Section 73.55(a) also states that the onsite physical security system and organization must include, but not necessarily be limited to, the capabilities to meet the specific requirements contained in paragraphs 73.53(b) through (h). Paragraphs 73.53(b) through (h) are considered performance-based.

10 CFR 50.40 requires each licensee to provide physical protection against radiological sabotage and the theft of special nuclear material in accordance with security plans approved by the Commission.

II. DISCUSSION OF METHODS TO MEET PLAN COMMITMENTS

Review of a sample of security plans reveals that they vary greatly from licensee to licensee. The regulations [10 CFR 50.54(p)] allow licensees to change their security plans **if such changes do not decrease the effectiveness of their plan.** The "effectiveness of the plan" is not defined. The consequence of this is the regulation of each licensee to a different standard. A licensee who acquiesced to the pressure to prepare a more comprehensive security plan may well be regulated to a stricter standard than another licensee who did not. The level of commitment would appear to be driven by the reviewers' and inspectors' influence during and following licensing rather than on an agency position based on the design basis threat.

If 10 CFR 73.55(a) delineates the Commission's expectation for physical security based on the design basis threat, licensees should be free to change their implementation methods **as long as they meet or exceed the defined standard.**

III. DISCUSSION OF PRESCRIPTIVENESS

The prescriptiveness of some of the rules related to security at power plants is in striking contrast with the rules in many other areas. For example, Appendix B to Part 73 provides guidance at a level of detail generally not found in Part 50. In areas such as quality assurance, the rules (i.e., Appendix B to Part 50) are written as performance-based. Even codes often allow licensees discretion in choosing the method to meet important requirements.

The Review Group recognized that security requirements at power plants may be subject to a critical reassessment because of events both inside and beyond NRC jurisdiction. The group concluded that, if the anticipated reassessment were to lead to the re-writing of all or part of the security rules, this would afford the opportunity to recast the prescriptive sections of the rules in a more performance-based approach.

IV. REGULATORY DIFFERENCES BETWEEN SECURITY AND OTHER AREAS

There are differences between the regulatory approach used in security inspection and other inspected activities. For example, Technical Specifications have built in time limits (i.e., allowed outage times in limiting conditions for operation (AOT/LCO)) for a licensee to act when a component or system is found to be inoperable. Security generally works on the basis of immediate response (or no AOT/LCO). While it is appropriate to respond to a threat immediately or during an initial evaluation of equipment failure or degradation, it is also appropriate, after verification that there is no immediate threat, to allow a reasonable time to respond to the problem or inoperability. The attached memorandum (F. Gillespie to J. Sniezek, dated February 2, 1993) highlighted an aspect of this issue in which a licensee had been "ratcheted" into carrying several extra security officers on shift in order to provide an immediate response capability. This is in contrast to the general allowance for a licensed operator (including the shift supervisor) to be absent because of illness for up to 4 hours before a replacement must be on site.

This same type of difference in the regulatory approach existed in response to the failure of components found by surveillance testing. In all areas other than security, inoperability identified as a result of surveillance testing was addressed in accordance with the Technical Specifications; but in security it had been common practice to issue a violation if an inspector witnessed a test in which a piece of security equipment failed to demonstrate operability. This dichotomy was addressed as a result of the June 1992 NRC Senior Management Meeting, and security surveillance testing results are now handled in a manner similar to the method used for inoperability found as the result of safety system surveillance testing.

V. REPORTING REQUIREMENTS

The regulations (10 CFR 73.71) require licensees to submit copies of all safeguards event logs not previously submitted at 3-month intervals. This requirement applies to various types of licensees including power plants. The industry (through NUMARC) has requested that this requirement be deleted [letter J. Colvin to I. Selin, dated December 21, 1992]. The Review Group could find no compelling reason for this quarterly log submittal. This was based on the facts that power plants have resident inspectors assigned and receive routine inspections from physical security specialists on a regular basis. Thus, the need to submit logs on a quarterly basis for review could not be validated by the Review Group.

The Review Group also noted that, in the same letter referenced above, NUMARC questioned the need for 1-hour reports to the NRC Operations Center. The stated NUMARC rationale was that all the facts might not be known in 1 hour and that therefore incomplete or inaccurate reports might be made and must later be supplemented, revised, or in some cases withdrawn. The Review Group did not agree with the NUMARC proposal. This was based on the viewpoint that, if an event were sufficiently significant to warrant an immediate report, the report should be made based on the facts known to the licensee at the time. Later additions or corrections are always acceptable and are, in fact, expected during events. This is the same rationale applied to the reporting of operating events and does not seem to cause the same concerns in the operational arena. Of course, the threshold of events that trigger the 1-hour report can be debated, but this was not the thrust of the NUMARC argument. The Review Group concluded that this was a manifestation of the issue discussed in Paragraph IV above, i.e., treating security in a philosophically different manner than other regulated activities.

VI. RECOMMENDATIONS

The Review Group noted that there were several initiatives affecting security. These were delineated in SECY-92-272, which is being reevaluated by the staff. There are also security issues discussed in the December 12, 1992 NUMARC letter. The recent security event at Three Mile Island One and the bombing of the World Trade Center in New York are expected to lead to a critical review of security. Assuming such a review occurs, the Review Group recommends that it include consideration of the points made in this analysis. These points are:

- Review existing security requirements (particularly Appendix B to Part 73) to determine if they should be expressed in a more performance-based manner.

- Eliminate the requirement for submittal of quarterly security logs. This would eliminate a small burden that appears to have no benefit.

- Revise existing guidance to provide an approach in security similar to that used for safety systems for compensatory measures.

- Additionally, the Review Group developed a proposed rulemaking to allow licensees to make changes to the security plan without NRC approval provided the changes do not reduce the plan's content below that necessary to implement the requirements of the regulations. The Review Group received comments from the

NRC offices and revised the proposed rulemaking. The detailed proposed rulemaking is located in Appendix A to Volume One.

VII. ANALYSIS OF PUBLIC COMMENTS

Public comments indicated a desire that the staff consider NUMARC's Alternate Protection System in parallel with the staff review of the design basis threat.

Attachment: Memo F. Gillespie to J. Sniezek, dated February 2, 1993, SUBJECT: NEAR TERM ACTIONS WHERE SAFEGUARDS PRACTICES AS PROMULGATED BY THE NRC APPEAR TO GO BEYOND 10 CFR 73.55

February 2, 1993

MEMORANDUM FOR: James H. Sniezek, Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research

FROM: Frank P. Gillespie RRG/EDO

SUBJECT: "EAR TERM ACTIONS WHERE SAFEGUARDS PRACTICES AS PROMULGATED THE NRC APPEAR TO GO BEYOND 10 CFR 73.55

In the last two weeks we have discussed the disparity in approaches taken in implementing safeguards compensatory measures versus the approach taken relative to safety systems in technical specifications. This area was further highlighted in the letter yo' received from Mr. Perry of Illinois Power dated January 12, 1993. Based on 'eview and his letter, we have broken the issue into two pieces.

- The staffing necessary to meet both the need for compensatory measures and the requirements of 10 CFR 73.55(h), response requirements.
- The gradation of the actions with time, based on an assessment of the cause of the system degradation, consistent with the performance objective of 10 CFR 73.55(a), ".....do not constitute an unreasonable risk to the public."

On the issue of staffing as raised by Mr. Perry, we have reviewed all the relevant documents in order to assess exactly what is required by rule and to understand how the actual requirements were augmented with time. The following regulatory requirements apply:

10 CFR 73.55(h)(3) - The total number of guards, and armed, trained personnel immediately available at the facility to fulfill these response requirements shall nominally be ten (10), unless specifically required otherwise on a case by case basis by the Commission; however, this number may not reduced to less than five (5) guards.

10 CFR 73.55(e)(1) - All alarms required pursuant to this part must annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station not necessarily on-site, so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm.

10 CFR 73.55(b)(2) - At least one full time member of the security organization who has the authority to direct the physical protection activities of the security organization shall be on-site at all times.

10 CFR 73.55(g)(1) - All alarms, communication equipment, physical barriers, and other security related devices or equipment shall be maintained in operable condition. The licensee shall develop and employ compensatory measures including equipment, additional security personnel and specific procedures to assure that the effectiveness of the security system is not reduced by failure or other contingencies affecting the operation of the security related equipment or structures.

The words of 10 CFR 73.55(h)(3) clearly state that the minimum response force is five. The literal reading of the words; "to fulfill these response requirements shall nominally be ten (10)", leads one to the dictionary to understand the meaning of "nominally".

Definition: nominal (nom a nal) adj. 1. a. Of like, pertaining to, or consisting of a name or names; b. Bearing a person's name; nominal shares. 2. Existing in name only and not in actuality. 3. Insignificantly small; trifing: a nominal sum. 4. Gram. Of or pertaining to a noun or a word group that functions as a noun. 5. According to plan; a nominal flight check - n. Gram. A word or group of words that functions as a noun. (Lat. nominalis (nomen, name) - nom i nal ty adv.

usage: Nominal in one of its senses means "in name only." Hence a nominal payment is a token payment, bearing no relation to the real value of what is being paid for. The word is often extended in use, especially by sellers to describe a low or bargain price.

nominal value n. The stated, par, or book value of a share of stock as opposed to the actual or market value.

To what does this lead? The minimum response force is five and with the expectation that normally, ten would be available. We also noted that this section said **available**, not **dedicated**. This would mean, that these guards could be performing other duties, as they typically do, as long as those duties can be terminated to make them available. This leads us to the conclusion that five of the ten could perform duties. Until required as as part of the response force. The expectation is that compensatory measures for system failures that occur on shift generally can not be anticipated and therefore fit the intent of the Commission as reflected in the context of 10 CFR 73.55(h)(3). We also question the idea that a guard performing compensatory is not available in the short term as part of the response force. Once a penetration is found and response made, all compensatory measures except those on the intrusion path, would certainly be curtailed with the resources being brought to bear on the incursion. This would potentially make available the full ten guards "nominally required" to be available to fill a dual role. The exception to this would be for prolonged compensatory measures since this would shift the need from the category of an unanticipated event to one which is planned.

The other references listed state the need for both required alarm stations to be manned continuously which accounts for two guards, who obviously could not have dual roles, and the need for a licensee individual authorized to direct the guard force. While not specifically prohibiting or allowing the individual in charge to be considered part of the response force, he could not be in overall charge if he were allowed to be considered part of the response force referred to in 10 CFR 73.55(h)(3).

Two documents issued by the NRC are Information Notice No. 86-88 and NUREG-1045 (Enclosures 1 and 2 respectively), these bear significantly on the past interpretation of the rules discussed. "Guidance on the Application of Compensatory Safeguards Measures for Power Reactor Licensees," NUREG-1045, discusses what are referenced as typical compensatory measures for failures or degradation of the safeguards system. By their very nature, these are generally pointed at making up for failures of equipment. This document does not address the use of security personnel in the dual role of compensatory measures and being considered as part of the response force. It also does not make any distinction between **short term** and **prolonged** needs or the impact and possible mitigation of the need for the recommended measure when the cause of the failure is identified as not being related to an intrusion but rather to a random equipment failure. The response in NUREG-1045 appears completely appropriate while in the process of assessing the cause of any failure. The NUREG goes on to describe compensatory measures which do not take into consideration any assessment or knowledge relative to the cause of the failure or 10 CFR 73.55(a) which states the general design objective of the safeguards system to be:

- General performance objective and requirements. The licensee shall establish and maintain an on-site physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

The use of the words "Unreasonable Risk" leaves the option open to treat security failures in the same way we treat safety system failures in the Technical Specifications with the application of AOT's and LCO's.

The second document referenced by Mr. Perry, Information Notice No. 86-88, would appear to have been applied out of context. Without assessing blame as to why or how, we discuss the exact words of the notice. First, the title of the Information Notice was "Compensatory Measures For **Prolonged Periods** Of Security System Failures." The meaning of the title was amplified under the heading "Purpose" of the notice which stated:

> "This notice is provided to alert addressees to increased vulnerability of their sites when compensatory measures are implemented for prolonged or indefinite periods. It is suggested that recipients review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems from occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required."

The title and purpose of the notice are very clear on the context in which it was written. As accurately reflected in Mr. Perry's letter, the notice states: "When security personnel are employed as compensatory measures, licensees are reminded that as a general policy security personnel cannot be considered simultaneously available for both compensatory measures and response force duties". Apparently this sentence has been applied, at least in the Clinton case, without considering the Prolonged Or Indefinite Context. Within the context of the notice the application of this sentence is completely compatible with the earlier discussion of the applicable rules and the short term assignment of dual responsibility.

In summary, the minimum staffing level required by 10 CFR 73.55 appears to be thirteen plus those necessary to compensate for prolonged compensatory measures. This would appear to be less than what exists at Clinton specifically and other plants based on informal discussions.

The concept of time and randomness of events as used in safety considerations appears to be absent in the security area. This would allow a gradation in compensatory measures employed once the cause of a failure was determined until the failure was determined to be prolonged, without decreasing the effectiveness of the safeguards system to protect the public from undue risk as stated in 10 CFR 73.55(a).

Before recommending an action to clarify the staffing requirements to the utilities, one additional point must be covered, the applicability of 10CFR 50.54(p) to any reduction in staffing which may take place. This rule states:

"(1) The licensees shall prepare and maintain safeguards contingency plan procedures in accordance with Appendix C of Part 73 of this chapter for effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may make no change which would decrease the effectiveness of a security plan or guard training and qualification plan, prepared pursuant to 50.34(c) or Part 73 of this chapter or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base Responsibility Matrix) contained in a licensee safeguards contingency plan prepared pursuant to 50.34(d) or Part 73 of this chapter, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to the licensee's license pursuant to 50.90."

"(2) The licensee may make changes to the plans referenced in paragraph (p) (1) without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of three years from the date of the change, and shall submit, as specified in 50.4, a report containing a description of each change within two months after the change is made."

The question which needs to be addressed is, would a reduction in security staffing reduce the effectiveness of a security plan? This raises a second question, effectiveness to do what? We believe the answers to these questions are clear and are given in 10 CFR 73.55(a), "General performance objective and requirements", which state:

(a) General performance objective and requirements. The licensee shall establish and maintain an on-site physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The physical protection system shall be designed to protect against the design basis threat of radiological sabotage as stated in Section 73.1(a). To achieve this general performance objective, the on-site physical protection system and security organization must include, but not necessarily be limited to, the capabilities to meet the specific requirements contained in paragraphs (b) through (h) of this section. The Commission may authorize an applicant or licensee to provide measures for protection against radiological sabotage other than those required by this section if the applicant or licensee demonstrates that the measures have the same high assurance objective as specified in this paragraph and that the overall level of system performance provides protection against radiological sabotage equivalent to that which would be provided by paragraphs (b) through (h) of this section and meets the general performance requirements of this section.

The measure of effectiveness is the ability to meet the objectives and requirements as stated. The words are permissive relative to doing more than is necessary where it states "must include but not necessarily be limited to the capability to meet the specific requirements contained in paragraphs (b) through (h) of this section." Therefore, we believe that as long as the requirements of 73.55(b) through (h) are effectively being met, changes to the security plan need not be submitted for prior review and that the burden is on the NRC to show that the licensee is in violation of 10 CFR 73.55. This may be a different process than now exists relative to the wording of 50.54(p) being a built-in ratchet. Further support for this position exists in 10 CFR 50.34(c) which states:

"(c) Physical security plan. Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part 1 shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 (and Part I) of this chapter, if applicable, including the identification and description of jobs as required by 11.11(a), at the proposed facility). Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable."

The security plan is required. However, the plan is a demonstration of how the applicant will comply with 10 CFR 73 and is not therefore a requirement in and of itself. Changes to the plan that do not reduce the licensee's ability to meet 10 CFR 73.55 effectively but actually reduce the resources should be acceptable without prior approval. The burden is then on the staff to establish not that the plan commits less resources, but that 10 CFR 73 is not being effectively met. In summary, meeting the objective of 10 CFR 73.55(b) through (h) is the measure of effectiveness, changes beyond these requirements and objectives can clearly be made at the complete discretion of the licensee.

Because this may be viewed as a significant departure from current practice, the analysis and position should probably be formalized and issued as a generic letter so that the expectations will be understood by both the industry and the staff. As a minimum, the comment process now used for generic letters would be one way of getting feedback from both the industry and public on the positions and arguments presented.

Frank P. Gillespie
RRG/EDO

Enclosures:
1. IE Information Notice No. 86-88
2. NUREG-1045
3. Letter from J.S. Perry dated 1/12/93

cc: J. Taylor
T. Murley
R. Bernero
E. Beckjord
E. Jordan
J. Scinto

2.3.19 USE OF RISK TECHNOLOGY IN UNREVIEWED SAFETY QUESTION DETERMINATIONS

The charter for the Review Group directs that the group determine how an integral analysis (probabilistic risk assessment [PRA]) can be used to provide more flexibility in the regulations and the implementation of regulations. This paper discusses the possible application of PRA (or risk technology) in making a determination of whether or not there is an unreviewed safety question raised by a proposed change to the facility or procedures as described in the final safety analysis report (FSAR) or the conduct of a test or experiment not described in the FSAR.

I. INTRODUCTION

Regulation 10 CFR 50.59 states in part:

> "The holder of a license authorizing operation of a production or utilization facility may (i) makes changes in the facility as described in the safety analysis report, (ii) make changes in procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question."

The same regulation describes an unreviewed safety question as follows:

> "A proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report; or (iii) the margin of safety as defined in the basis for any technical specification is reduced."

The question is: Does risk technology offer a viable method for making a determination of whether or not a proposed change creates an unreviewed safety question?

II. DISCUSSION

Risk technology is relatively new, but it is recognized as a means to gain insights into safety not necessarily available through deterministic techniques. There have always been questions about the accuracy of the precise numbers produced in a PRA. This has resulted in a historical tendency by some to discount the insights resulting from the application of risk technology, but this approach may not take advantage of what a PRA can produce reliably.

PRAs produce precise numbers simply because any mathematical application will result in a number after completing an operation. The accuracy of the result obviously depends on the accuracy of the numbers that were used as input data. Thus, absolute faith in the resultant precise number has been shown to be misplaced. However, when PRA is applied to an unreviewed safety question determination, it would appear that the result needed is not an absolute determination of risk but is a measure of the change in risk or in other words the "relative risk."

Relative risk could potentially make PRA a useful tool for unreviewed safety question determinations. For if risk can be calculated both before and after the proposed change and if many of the uncertainties appear in both the before (i.e., risk denominator) and the after (i.e., risk numerator), the question about whether or not the probability of occurrence has changed can be answered. PRA may also provide insights into another of the unreviewed safety question tests, which is whether or not a different type of accident or malfunction will be introduced by the change. It may also be argued that the use of PRA in the relative sense may provide insights into the final unreviewed safety question as to whether or not the margin of safety is reduced by the proposed change.

A reading of the regulation does not indicate by what means the unreviewed safety question issue should be answered. Therefore, the rule appears to neither bar nor encourage the use of risk technology in this application.

III. RECOMMENDATION

Industry and the NRC staff are encouraged to pursue the development of guidelines (if applicable) of PRA methods for use in this application (see Section 4.5 for a detailed discussion).

2.3.20 SAFETY GOAL

The Commission's objective in publishing the Safety Goal Policy Statement was "...to define an acceptable level of radiological risk from nuclear power plant operation" (memorandum from Samuel J. Chilk to James M. Taylor on SECY-89-102 - Implementation of the Safety Goals, dated June 15, 1990). With regard to direct application of the Safety Goals, the Commission guidance contained in this staff requirements memorandum further states:

> "These arguments clearly established that there is a level of safety that is referred to as 'adequate protection.' This is the level that must be assured without regard to cost and, thus, without invoking the procedures required by the Backfit Rule. (Footnote omitted.) Beyond adequate protection, if the NRC decides to consider enhancements to safety, costs must be considered, and the cost-benefit analysis required by the Backfit Rule must be performed. The Safety Goals, on the other hand, are silent on the issue of cost but do provide a definition of 'how safe is safe enough' that should be seen as guidance on how far to go when proposing safety enhancements, including those to be considered under the Backfit Rule." (Emphasis added.)

Thus, the determination of "adequate protection" is a case-by-case finding evaluating a plant and site combination considering the body of the regulations.[2] With regard to proposed safety enhancements, the Safety Goals provide a structure for the disciplined examination of proposed new requirements for nuclear power plants. They set a limit on where or when enhancements can be considered, i.e., if the Safety Goals are satisfied with proper considerations of the uncertainties involved, no additional requirements are justifiable for implementation, even if cost beneficial. Since the original regulations were promulgated, a number of rules and staff actions (e.g., bulletins, generic letters) have been imposed as enhanced safety, under the provisions of 10 CFR 50.109, the Backfit Rule. Thus, with time, the level of safety embodied in the regulations has continued to approach the level of safety associated with the Commission's Safety Goals. This process is illustrated conceptually in Figure 1. (Individual plants may be found to have a level of safety that is above or below that associated with the Safety Goals, since the Safety Goals address the overall population of plants and individual plants may have added features that improve safety at the licensee's discretion. The comparison of five plants with the Safety Goals in NUREG-1150 indicates that the five plants considered in that study had

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[2] It should be noted, however, that adequate protection does not necessarily require compliance with the body of the regulations since certain regulations have been issued to enhance safety under 10 CFR 50.109. Moreover, the regulations only presumptively assure adequate protection. But in the absence of a redefinition of "adequate protection," that presumption can be overcome only by significant new information or a showing that the regulations do not address some significant safety issue. (See 50 SC 37, 53 FR 20603 et seq., June 1990 for additional details.)

met the Safety Goals. Programs such as the Accident Sequence Precursor program show that in certain events that have occurred at individual plants, however, the level of safety may not have been consistent with the Safety
Goals.)

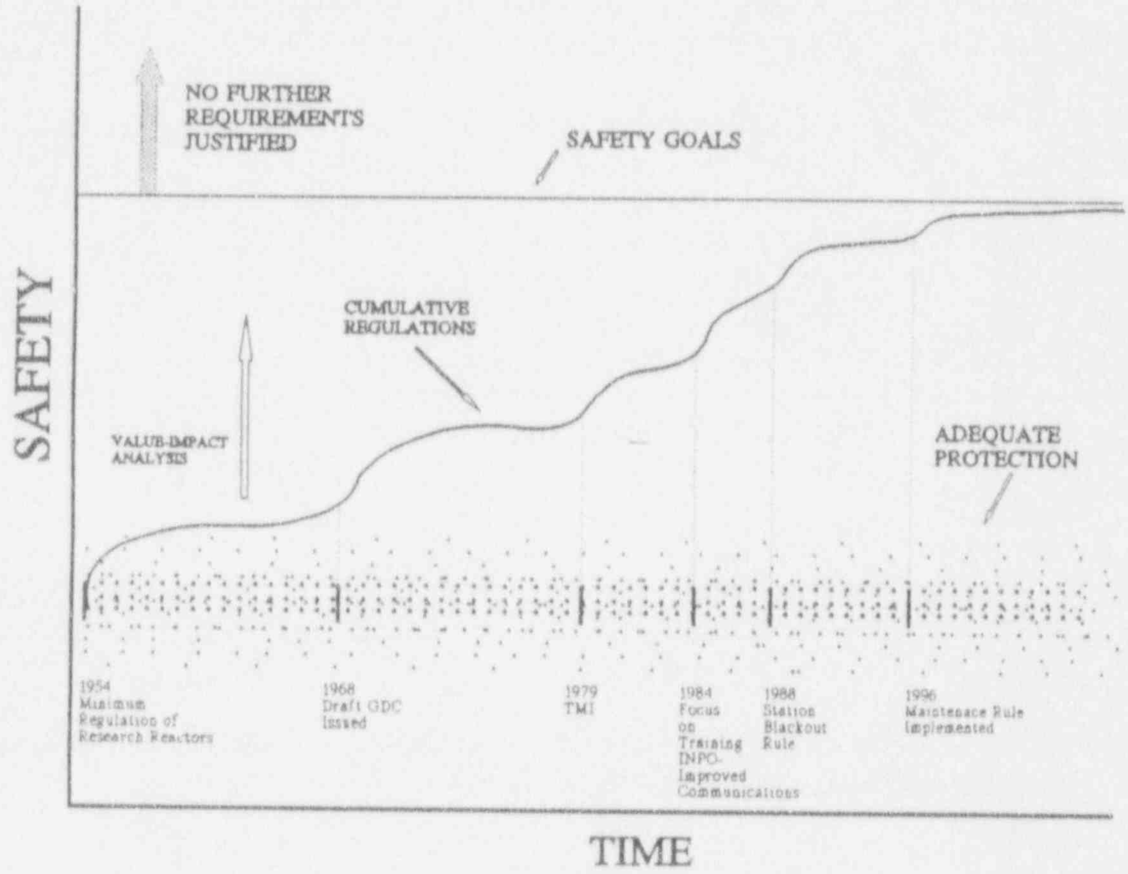

Figure 1

This approach outlined above is embodied in the Regulatory Analysis Guidelines (and the associated Regulatory Analysis Handbook) recently approved by the Commission for issuance for public comment. In these documents, the decision of whether to proceed with a value-impact analysis when considering a potential regulatory requirement is based on a comparison of the impact of the issue on the subsidiary objectives identified by the Commission, viz., the potential change in the frequency of severe core damage and the probability of containment failure and subsequent release of fission products. In practice, the decision criteria for proceeding to value-impact analysis are set an order of magnitude below the subsidiary objectives in recognition of the uncertainties involved and to preclude the inadvertent screening out of a potentially important issue that deserves further investigation before a final decision is made.

In addition to this initiative, efforts also are under way to examine the regulatory fabric in a more global way. Each Individual Plant Examination submitted in response to Generic Letter 88-20 identifies any plant-specific vulnerability found in the analysis. Vulnerabilities that are identified will be evaluated to determine their generic significance in light of the level of safety associated with the Safety Goals and the regulations; first, by assessing whether adequate protection and compliance with the regulations are maintained; second, by a comparison with the Safety Goals; and third, (if deemed appropriate), by a detailed backfit analysis as required by 10 CFR 50.109.

The Safety Goals will also play an important role in the evaluation of a generic change in regulatory practices intended to reduce burden on licensees without any significant reduction in plant safety. The Backfit Rule, 10 CFR 50.109, addresses only the steps required of the NRC staff to add provisions to the NRC rules (or impose a staff position interpreting those rules) for licensed plants. It is not applicable to changes in requirements associated with reductions of burden. Administrative processes established through the CRGR charter require that any request for relaxation in requirements be accompanied by a demonstration that adequate protection is still provided but are silent on comparisons to the safety goal. Further guidance is given in the proposed Regulatory Analysis Guidelines discussed in SECY-93-167, which require that the following conditions be satisfied:

- The public health and safety and the common defense and security would continue to be adequately protected if the proposed reduction in requirements or positions were implemented, and

- The cost savings attributed to the action would be substantial enough to justify taking the action, and the savings would clearly outweigh any reduction in benefit.

If the premise that the industry is operating at a level approaching the Safety Goals is accepted, then the Review Group recommends that alternative approaches to current requirements should be allowed based on analyses that demonstrate that (1) the two conditions noted above are satisfied, and (2) provide confidence that the relaxation will not significantly affect the safety margin of U.S. plants in light of the Commission's Safety Goals. In practice, this would effectively mean that the existing level of safety would be maintained overall.

Volume Three

Regulatory Review Group

Operating Licenses

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3.1 INTRODUCTION

3.1.1 Background

The Regulatory Review Group (referred to hereinafter as Review Group) charter calls for the assessment of operating licenses by selecting several licenses issued at various times, determining how the regulations and regulatory guidance were incorporated into the licenses, determining how much inherent flexibility the licensees have in making changes to their plants or operations, and determining what in the regulatory process may be inhibiting the use of the inherent flexibility. In addition, the Review Group considered areas where enhanced flexibility could potentially be provided.

The following sections describe the selection of the plants whose operating licenses were assessed and the approach that was used to assess the licenses.

3.1.2 Selection of Plants (Licenses)

Four plants (licenses) were selected for the assessment. This number was based on the number judged necessary to accomplish the objectives of the Review Group's charter and the number needed to be representative of a significant number of plants (licenses).

A substantial number of criteria were considered in the selection of the four plants. However, it was the view of the Review Group that the following criteria were the most important (listed in order of importance) for the purposes of this activity:

- Recent and early licenses
- BWR and PWR plants (licenses)
- Representativeness of significant number of plants
- Availability of PRA/IPE (for possible interface with the PRA Technology Subgroup)

Using the above criteria, Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2 were selected from among all the plants currently licensed to operate.

Seabrook was selected because it is one of the most recently licensed PWRs; it is a Westinghouse four-loop plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE that has been reviewed by the NRC.

Surry was selected because it is one of the earliest licensed PWRs; it is a Westinghouse three-loop plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE whose review by the NRC is nearly complete. Surry 1 is also one of the plants evaluated in WASH-1400 and NUREG-1150.

Perry was selected because it is one of the most recently licensed BWRs; it is a General Electric BWR-6, Mark III containment plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE that is under review by the NRC.

Peach Bottom was selected because it is one of the earliest licensed BWRs; it is a General Electric BWR-4, Mark I containment plant and is, therefore, representative of a significant number of plants (licenses); and, although the NRC has not completed the review of its IPE, it is one of the plants evaluated in WASH-1400 and NUREG-1150.

3.1.3 Assessment Approach

The assessment approach is summarized in Table 1. The approach involved the assessment of items of the operating license, either individually or collectively. For the purposes of this assessment, an item is defined as any license condition or Technical Specification definition, safety limit, limiting safety system setting, limiting condition for operation, design feature, or administrative control that is designated alphanumerically in the license. Technical Specification bases were excluded since they are not part of the Technical Specifications and, hence, the license. Except for the applicability section, a Technical Specification limiting condition for operation and its associated surveillance requirement were counted as a single item.

A typical operating license contains several hundred items. To facilitate the assessment and to ensure adequate consideration of all types of license requirements, the items were reviewed and assigned to one of the seven categories described in Table 2. Where an item could be assigned to more that one category, it was assigned to the most dominant category.

The categories were defined to optimize the assessment effort and to ensure adequate consideration of all types of license requirements. First, categories were established that would allow all the items in as many categories as possible to be assessed collectively. This meant that all the items in the category had to have similar characteristics. Secondly, where it was not possible to assess the items collectively and the items had to be assessed individually, the categories were established to allow the items to be representative of as many of the others in the same category as possible.

The items were reviewed to determine which categories contained items with similar enough characteristics to be assessed collectively. The items in the remaining categories were considered to determine the percentage that could be assessed individually. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category.

The items that were to be assessed individually were selected from the remaining categories. The items were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest.

Although not every item of the license was assessed, the categorization of the items and the selection of a significant number of representative items for assessment from each category ensured adequate coverage of the license. The selection of items for assessment from subsequent license(s) was based on validating the findings from the license(s) already assessed and expanding both the number and scope of the items assessed.

The items were assessed either collectively or individually as appropriate by considering the answers to specified questions presented in Table 3. The questions were designed to determine whether the item has a sound regulatory basis, is related to public health and safety, inherently allows the licensee flexibility in making changes to the plant or operations, or could be modified to provide increased flexibility to the licensee. The questions were written in such a manner that a "no" response would elicit additional review. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contained overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items that were selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

Table 1

SUMMARY OF ASSESSMENT APPROACH

1. Review each operating license item and assign it to a category.

2. Determine which categories contain items that are appropriate to be assessed collectively.

3. Determine which items from the remaining categories will be assessed individually.

4. Assess items in accordance with specified questions; analyze items as necessary.

5. Prepare assessment summaries.

6. Validate results from assessment(s) of previous license(s).

7. Integrate overall results, and develop findings and recommendations.

Table 2

CATEGORIES OF ITEMS

A. Technical Requirements - items that impose requirements based upon plant design, operational, or other technical constraints (e.g., limiting conditions for operation).

B. Non-Technical License Conditions - items exclusive of the Technical Specifications that discuss broad management/issue considerations, generally of a non-engineering nature (e.g., financial conditions, organizational constraints).

C. License Conditions That Rely on Other Documents for Requirements - items that refer to other documents (e.g., physical security plan, NPDES permit) for the required actions or constraints.

D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements) - items in the Technical Specifications that impose non-technical organizational and programmatic requirements (e.g., station staff, committees, training), exclusive of specific reporting and recordkeeping provisions.

E. Reporting and Recordkeeping Requirements - items that discuss licensee reports and records, or impose related requirements (e.g., routine and annual reports and record retention and distribution).

F. Unique Plant Features - items that describe a design feature of the plant and its environs or define plant system/component configuration details (e.g., site characteristics and reactor and containment design parameters).

G. Other - items that impose conditions that are not covered by any of the other categories (e.g., legal provisions, exemptions, definitions, statements).

Table 3

ASSESSMENT QUESTIONS

1. <u>Regulatory Bases</u>

   A. Are the items supported by documented regulatory bases (e.g., regulatory guidance or requirements)?

   B. Are the regulatory bases supported by a legal requirement (e.g., Atomic Energy Act, Commission regulation or order)?

   C. If not legally required, have regulatory guidance and/or licensee commitments been appropriately used to impose the items?

2. <u>Safety Relevance</u>

   A. Are the items necessary to ensure public health and safety (e.g., are they needed for adequate protection, defense in depth)?

   B. Are the items in the group generally consistent, coherent, and commensurate with safety significance?

   C. Are the items, as implemented, reasonably within their original intent?

   D. Are surrogate items (e.g., quantitative requirements) both necessary and appropriately used to meet the safety objective?

3. <u>Inherent Flexibility</u>

   A. Does an inherent flexibility exist that allows the licensee a tradeoff of items without a reduction in overall safety?

   B. Are other means, besides a license amendment, available to the licensee for revising the items?

   C. Can the change/revision be made without NRC pre-approval?

   D. If yes, can the change/revision be made without an NRC post-implementation review?

TABLE 3 (Continued)

ASSESSMENT QUESTIONS

4. Enhanced Flexibility Potential

A. If prescriptive language appears in the items, is it needed to convey the intended requirement?

B. Would the use of performance-based criteria be inappropriate to add flexibility to item implementation?

C. If specific factors that limit flexibility are identified, are all these factors beyond the control of the NRC?

D. Would further NRC review of this area for enhanced flexibility be unproductive (i.e., the licensee doesn't need or isn't likely to use any resulting initiatives)?

E. Are there NRC programs currently ongoing or under evaluation for implementation that would provide enhanced flexibility to the licensee?

3.2 ASSESSMENT OF OPERATING LICENSES

3.2.1 Licenses

As discussed in Chapter 1 of this report, the Review Group assessed the operating licenses of four plants--Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2. The licenses as they were reviewed were as follows:

- The Seabrook Unit 1 operating license was issued on March 15, 1990. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the environmental protection plan, which is Appendix B to the license. The license as reviewed has been amended through Amendment 11, dated May 29, 1992.

- The Surry Unit 1 operating license was issued on May 25, 1972. The operating license consists of the license itself and the Technical Specifications, which are Appendix A to the license. The license as reviewed has been amended through Amendment 170, dated June 1, 1992.

- The Perry Unit 1 operating license was issued on November 13, 1986. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; the environmental pro' ction plan, which is Appendix B to the license; and the antitrust conditions, which are Appendix C to the license. The license as reviewed has been amended through Amendment 43, dated May 28, 1992.

- The Peach Bottom Unit 2 operating license was issued on December 14, 1973. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Technical Specifications, which are Appendix B to the license. The license as reviewed has been amended through Amendment 168, dated July 6, 1992.

The Review Group assessed the Seabrook and Surry licenses separately. The assessment of the Seabrook license, which was performed first, resulted in seven recommendations. The assessment of the Surry license, which was performed next, largely validated the Review Group's assessment of the Seabrook license and resulted in only three additional recommendations. Based on the results of these two assessments, and its knowledge of and experience with other licenses, the Review Group did not expect to find significant information in its reviews of the Perry and Peach Bottom licenses that would result in a substantial number of additional recommendations. Therefore, the Review Group assessed the Perry and Peach Bottom licenses together. The combined assessment was performed using the same methodology as that used previously for the individual plant assessments and resulted in one additional recommendation.

3.2.2 Assessment of Licenses

The four operating licenses contain a total of 1,127 items. Each item was reviewed and assigned to one of the categories in Table 2. The numbers of items in each of the operating licenses by category are shown in Table 4.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on their regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 297 items or approximately 26 percent of the total number of items.

The number of items in the remaining categories that were assessed individually was determined to be approximately 10 percent for Seabrook and Surry, and, since they were reviewed together, 5 percent for each of Perry and Peach Bottom. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or five of the 50 Category D items in the Perry license were selected for further assessment. With the 297 items that were assessed collectively, this meant that 358 or approximately 32 percent of the 1,127 total items were assessed either collectively or individually.

The items to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. The items that were assessed for each plant are listed in Table 5 of the plant assessment reports (Appendixes A, B, and C for Seabrook, Surry, and Perry and Peach Bottom, respectively).

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table 3. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted. The numbers of items in each of the operating licenses by group are shown in Table 5.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

3.2.3 Results of Assessment

The assessment summaries for the items assessed for each of the licenses are contained in the attachments of the respective plant assessment reports. The findings and recommendations for each of the licenses are presented in the respective plant assessment reports. The separate assessment reports for Seabrook and Surry, and the combined assessment report for Perry and Peach Bottom are provided in Appendixes A, B, and C to this report, respectively.

The overall assessment results, which are based on the findings and recommendations of the assessments of all four of the licenses, are presented in Section 3.3 of this report.

Table 4

OPERATING LICENSE ITEMS BY CATEGORY

	Category	Seabrook	Surry	Perry	Peach Bottom	Totals
A.	Technical Requirements	136	93	136	75	440
B.	Non-Technical License Conditions	4	1	7	2	14
C.	License Conditions That Rely on Other Documents for Requirements	32	16	23	19	90
D.	Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	50	20	48	59	177
E.	Reporting and Recordkeeping Requirements	36	19	39	29	123
F.	Unique Plant Features	10	9	11	10	40
G.	Other	63	34	65	81	243
	Totals	331	192	329	275	1,127

Table 5

FINDINGS (NUMBERS OF ITEMS) BY GROUP

Group	Seabrook	Surry	Perry	Peach Bottom	Totals
Items that appear to exceed applicable regulatory requirements	7	0	0	2	9
Items that should be considered for possible reduction in regulatory burden	4	5	4	4	17
Items that provide inherent flexibility	6	5	3	3	17
Items that should be considered for enhanced flexibility	6	4	7	3	20
Items considered or being considered in other programs	7	4	6	5	22
Items for which no further consideration is warranted	36	19	39	29	123

3.3 ASSESSMENT RESULTS

3.3.1 Introduction

As discussed in the previous chapters of this report, the Review Group assessed the operating licenses of four plants--Seabrook, Surry, Perry, and Peach Bottom. The assessment reports for theses licenses are presented in the appendixes to this report. The overall results of the individual license assessments are integrated in this chapter, which discusses the Review Group's overall observations, findings, recommendations, and conclusions.

3.3.2 Observations

In its assessment of the four licenses, the Review Group made a number of general observations about the licenses themselves. The observations relate to the overall condition of the licenses, the numbers and types of items in the licenses, and the license conditions and Technical Specifications. In addition, some of the observations relate to the plant (license) selection criteria and selection of plants to be assessed, as described in Chapter 1 of this report. The Group's observations are as follows:

3.3.2.1 Overall Condition of Licenses

The recently issued licenses are generally crisper and cleaner than the ones issued earlier. That is because a concerted effort has been made by the NRC to limit the license conditions to those explicitly required by the Atomic Energy Act or the Commission's regulations. One exception, which is also valid for the older plants, is the incorporation of "contemporary" issues, that is issues that were "hot" at the time the license was issued. An example of this is the TDI diesel-generator license condition in the Perry license. In contrast, the earlier issued licenses appear to contain many more license conditions that address plant-specific issues. Handling of these issues as commitments in licensee-controlled documents would not only increase the flexibility available to the licensee, but also would reduce the regulatory burden on both the licensee and the NRC. In addition, the older licenses contain a number of conditions that overlap or have been superseded, e.g., physical security conditions. Such conditions could lead to confusion and mistakes.

3.3.2.2 Numbers and Types of Items in Licenses

The average number of items that represent requirements (Categories A through E items in Table 4) in the recently issued licenses (256) is substantially greater than that in the licenses issued earlier (167). However, the percentage of items that represent technical requirements (Category A items in Table 4) in the recently issued licenses (53 percent) is not substantially different from that in the licenses issued earlier (51 percent).

The average number of items that represent requirements (Categories A though E items in Table 4) in the pressurized water reactor plant licenses (204) is not substantially different from that in the boiling water reactor plant licenses (219). Also, the percentage of items that represent technical requirements (Category A items in Table 4) in the pressurized water reactor plant licenses (56 percent) is not substantially different from that in the boiling water reactor plant licenses (63 percent).

The distributions of the items within the categories shown in Table 4 and within the groups shown in Table 5 are not substantially different from one license to another, regardless of the age of the license or the type of reactor. Also as illustrated in Tables 4 and 5, the distributions of the categories of items within each group are not substantially different from one license to another regardless of the age of the license or the type of reactor. These observations imply that no part of an operating license, e.g., the license itself, the Technical Specifications, or a specific Technical Specification section, appears to contain a disproportionate number of items that do not have either a sound regulatory basis or provide inherent flexibility. They also imply that these items may be found in any part of the license.

No significant differences in the Review Group's findings could be attributed to reactor type. Where design and other technical differences based on reactor type were found, no meaningful correlation of these differences with regulatory basis, safety relevance, inherent flexibility, or enhanced flexibility potential was identified.

3.3.2.3 License Conditions and Technical Specifications

The Technical Specifications of the recently issued licenses were based on the Standard Technical Specifications whereas the Technical Specifications of the early licenses were "custom." This could complicate line-item improvements for the older plants under the Technical Specification Improvement Program as indicated below.

- A number of the Technical Specifications (as well as license conditions) of the early licenses are less restrictive than their later counterparts.

- The Technical Specifications of the early plants are written in a narrative fashion whereas those of the newer plants are written in more coherent format.

- The Technical Specifications of the early plants tend to be more interdependent on others. That is, they frequently cross-reference other Technical Specifications.

- The Technical Specifications exhibit differences in philosophy. A number of the older plant Technical Specifications tend to be system based whereas the newer plant Technical Specifications tend to be function based.

Finally, a number of errors and inconsistencies were identified in the bases of the Technical Specifications, especially those of the older plants. The errors and inconsistencies were often related to Technical Specification revisions for which the bases had not been updated. Given the importance and expanded use of the bases in the Improved Standard Technical Specifications, the existence of errors and inconsistencies, while not an operating license compliance issue, is a problem that nevertheless could have safety impact. Particularly when used as guidance or reference during operation and training, the Technical Specification bases should provide plant operators with accurate and coherent information. In addition, the bases assist both the licensees and the NRC in the interpretation of Technical Specification requirements that might be otherwise ambiguous.

3.3.2.4 Applicability of Assessment Results

As discussed in Section 3.1 of this report, each of the four licenses selected for review is representative of a significant number of licenses of similar reactor type and containment design configurations. The use of representativeness as a plant selection criterion, along with plant type and license age criteria, is consistent with the Review Group's assessment approach for achieving broad results and generic insights through the review of a limited number of licenses. Further, the findings developed from a specific license review were subjected to additional evaluations during subsequent license reviews to validate the results. Upon the completion of all four plant license assessments, an analysis of the resulting data was conducted. This included efforts to correlate the items supporting specific findings with the operating license categories that had been assigned for those items by the review methodology. The results enumerated in Tables 4 and 5 are typical of the data analyzed in this process.

This overall approach to an integrated assessment process was developed with the intent to evaluate the generic applicability of the findings and conclusions reached as a result of the review of the four specific plant licenses. The consistency of the comparative plant-to-plant data compiled in Tables 4 and 5, along with the proportionate distribution of findings within the operating license categories defined in Table 2, provide evidence of the representative nature of the findings. The generic applicability of the findings was also confirmed by the validation process. Based on these results and the insights gleaned from the assessment of four typical plant operating licenses, it is concluded, therefore, that the recommendations provided in this report are generally relevant and directly applicable to most of the operating licenses.

3.3.3 Findings

The item assessment summaries were reviewed to determine which of the items appear to exceed the applicable regulatory requirements, given their safety significance and regulatory bases; which of the items should be considered for possible reduction in regulatory burden; which of the items provide at least some inherent flexibility, and why licensees may not be taking full advantage of that flexibility; and which of the items should be considered for enhanced flexibility. The items that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, the physical security operating license condition generally appears in three groups. The item appears to have potential for reduction in regulatory burden; it has at least some inherent flexibility; and it appears to have potential for enhanced flexibility.

3.3.3.1 Items That Appear To Exceed Applicable Regulatory Requirements

Nine, or approximately 3 percent, of the 358 items assessed appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the licenses. Almost half of the instances in which items were found to exceed the applicable regulatory requirements involved situations where the plant design was different from that assumed in the Standard Technical Specifications used to develop the plant-specific Technical Specifications. In most cases, a system that has a design not in conformance with a standard plant design appeared to have additional requirements and/or less flexible provisions imposed in its Technical Specifications compared to the Standard Technical Specifications.

A number of the instances in which items were found to exceed the applicable regulatory requirements involved the elevation of provisions of Commission policy statements, regulatory guides, and other non-requirements to the status of legal requirements. While it is recognized that 10 CFR 50.50 authorizes the Commission to include in licenses such conditions as it deems appropriate, the inclusion of these non-requirements into licenses effectively elevates their status to requirements. In many instances, these non-requirements have not had the benefit of the rigorous regulatory review normally associated with the promulgation of requirements.

The remaining instances in which items were found to exceed the applicable regulatory requirements involved cases where the licensee had not taken advantage of the opportunity to eliminate requirements that are no longer required and at least one case where a licensee voluntarily incorporated a non-requirement into the license (e.g., the Peach Bottom licensee apparently voluntarily adopted the ISEG function, which was not generically required of pre-TMI accident licensees) and incorporated the requirement into the Technical Specifications.

3.3.3.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

Seventeen, or approximately 5 percent, of the 358 items assessed appear to have the potential for possible reduction in regulatory burden. Approximately half of these items are Technical Specification requirements that, when compared to the Improved Standard Technical Specifications, are unduly prescriptive or no longer required. Such requirements are, therefore, potential candidates for line-item improvements under the Technical Specification Improvement Program.

A few of the items that should be considered for possible reduction in regulatory burden are license conditions, such as physical security or fire protection, which require license amendments for changes that decrease safeguards effectiveness or the ability to achieve cold shutdown in the event or a fire, respectively. The Review Group believes it is appropriate that such changes be approved by the NRC before they are implemented; however, it sees no benefit in having to amend the license in addition.

A number of the items that have the potential for possible reduction in regulatory burden involve duplicative or otherwise unnecessary reporting requirements. These reporting requirements are (1) surrogates for more appropriate corrective actions, (2) duplicative of those required by the Commission's regulations, especially 10 CFR 50.73, or (3) not essential, at least at the frequency specified, to the accomplishment of the agency's mission. While surrogate and duplicative reporting requirements constitute an unnecessary regulatory burden, those that are redundant may have some value as reminders that reports to the NRC may be required, e.g., if a Technical Specification is violated. For these situations, however, the use of notes that reference the applicable regulations may be more appropriate.

3.3.3.3 Items That Provide Inherent Flexibility

Seventeen, or approximately 5 percent, of the 358 items assessed appear to have at least some inherent flexibility. These items are generally (1) performance-based requirements, which establish desired objectives without prescriptive details, (2) requirements that are prescriptive only at a high level and allow the implementation details to be specified in licensee-controlled documents, or (3) requirements that allow specified changes to be made without prior NRC approval. A number of items reviewed have demonstrated that performance-based requirements can ensure that adequate safety is provided and at the same time provide the licensee with considerable flexibility in meeting the requirement. Such requirements also reduce the regulatory burden on both the licensee and the NRC. The licenses contain only a relatively small number of exemptions from the Commission's regulations. The most common of these exemptions are from Appendix J to 10 CFR 50. However, some exemptions are not reflected in the licenses; the most common of these exemptions are from Appendix R to 10 CFR 50. The limited areas in which exemptions have been granted and, except for Appendix R to 10 CFR 50, the relatively small number of exemptions that have been granted imply that the regulations afford sufficient flexibility to accommodate wide spectra of plant designs and operations. The Review Group notes that both Appendixes J and R to 10 CFR 50 are presently being considered in the Marginal-to-Safety Program.

3.3.3.4 Items That Should Be Considered For Enhanced Flexibility

Twenty, or approximately 6 percent, of the 358 items assessed appear to have enhanced flexibility potential. The greatest potential for enhancing flexibility, at least in the short term, appears to lie with the Technical Specification Improvement Program. The program provides the opportunity for licensees not only to totally convert their Technical Specifications to the Improved Standard Technical Specifications but also to pursue generically approved line-item improvements. The Review Group believes that the utility of the program can be greatly enhanced by making available line-item improvements for individual licensees.

The next greatest potential for enhancing flexibility appears to be the expanded use of either performance-based requirements or requirements that are prescriptive only at a high level and allow the implementation details to be specified in licensee-controlled documents. As discussed in Section 3.3.3 of this report, these types of requirements have been shown to ensure that adequate safety is provided and at the same time provide the licensee with considerable flexibility in meeting the requirement. Such requirements also reduce the regulatory burden on both the licensee and the NRC.

The third greatest potential for enhancing flexibility appears to be the increased use of risk assessment methodology in both establishing and implementing requirements. Examples include extending completion times and offering a graded approach to safety such as that already allowed by Appendix B to 10 CFR 50.

The use of allowed outage times, such as those provided for safety-related equipment in the Technical Specifications, could provide additional flexibility in areas such as physical security and fire protection. Also, greater flexibility could be provided by redefining the baselines from which changes to physical security plans, emergency plans, quality assurance plans, and fire protection can be made without prior NRC approval. The underlying regulatory requirements would serve as more appropriate baselines for future changes than the plans themselves.

3.3.3.5 Items Considered or Being Considered in Other Programs

Twenty-two, or approximately 6 percent, of the 358 items assessed have already been or are being considered in other programs. Items were included here if the other programs offer the potential for reduced regulatory burden or enhanced flexibility. The program in which most of these items have been considered is the Technical Specification

Improvement Program. Other programs in which items have already been or are being considered include generic letters, rulemaking efforts, and reporting requirement re-evaluation.

3.3.3.6 Items for Which No Further Consideration Is Warranted

Three hundred and thirty nine, or approximately 95 percent, of the 358 items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is also included here.

3.3.4 Recommendations

In its assessment of the four licenses, as documented in Appendixes A, B, and C to this report, the Review Group identified 11 recommendations. In the process of integrating the results of the individual plant assessments and preparing this report, the Review Group identified an additional recommendation. The 12 recommendations are presented below in the order in which they were identified. As discussed in Section 3.3.2.4 of this report, the Review Group believes that the recommendations generally apply to all licenses, therefore, their order of presentation is not important. References to the sections in this report in which the recommendations were identified are provided in parentheses.

To improve the clarity and specificity of the recommendations, many of them have been slightly reworded from the way in which they appear in the individual plant assessments. Also, subsequent to the identification of the recommendations, actions have been taken or are being taken to address a number of them. Where such actions have been or are being taken, it is so noted.

It is important to note that the recommendations have the potential to reduce the regulatory burden on and enhance the flexibility available to the licensees. None of the recommendations needs to be implemented to ensure safety. However, since the implementation of the recommendations would result in a reduction in licensee manpower requirements, this excess manpower could indirectly benefit safety if it were redirected to safety-significant work.

- Eliminate the past practice of treating certain Commission policy statements, regulatory guides, and other non-requirements as legal requirements by generically including them in the licenses without following the appropriate disciplined process for establishing regulatory requirements. (Sec. A.2.2.1).

- Consider developing NRC staff guidance that considers Technical Specification requirements for design features that provide safety margin in excess of NRC requirements, for example, systems that provide additional redundancy. (Sec. A.2.2.1).

- Provide a consistent approach for making changes to "plans," such as the fire protection, physical security, emergency response, and quality assurance plans, within their proper regulatory and safety contexts. Eliminate the regulatory requirement that compliance with physical security plans be imposed by a license condition. (Sec. A.2.2.2).

The Review Group has developed a proposed rule change which addresses this recommendation. A Commission Paper containing the rule change is located in Appendix A to Volume 1.

- Reevaluate the information/data the NRC needs from nuclear power plant licensees in order to accomplish its mission of protecting the health and safety of the public (taking into consideration the efforts of the CRGR and the Reporting Requirements Task Force). Information/data requirements without a clear nexus to that mission and duplicative reporting requirements should be eliminated. (Sec. A.2.2.2).

Subsequent to its identification of this recommendation, the Review Group reevaluated the reporting requirements contained in a number of the Commission's regulations and several operating licenses. That reevaluation and its results are discussed in volume two of the Review Group's report.

- Invite the industry to provide the NRC with candid insights on licensees' reasons for not taking more advantage of the inherent flexibility afforded them. (Sec. A.2.2.3).

In its report on its assessment of the Seabrook license (Appendix A to this report), the Review Group found that many licensees have not taken advantage of the considerable flexibility that is already available to them. The Group listed possible reasons for this and recommended that the industry provide its views on the subject. The Review Group believes the opportunity for the public to comment on this report provides that occasion. The Review Group will consider any comments received on this subject and, therefore, considers that this recommendation will be adequately addressed.

- Provide additional flexibility in the implementation of the physical security plans, such as providing Technical-Specification-type allowed outage times. (Sec. A.2.2.4).

- Adopt a graded approach to limiting conditions for operation and surveillance requirements wherever practicable, and to the implementation of specific review committee functions, e.g., station onsite review committee procedure and design change reviews. The appropriate application of risk assessment methodology could be valuable in establishing both the bounds and direction of such an approach. (Sec. A.2.2.4).

- Eliminate the practice of including fire protection plans and the provisions for making changes thereto as license conditions. (Sec. B.2.2.2).

- Expand the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan. Eliminate the inconsistencies in the change requirements for these plans. (Sec. B.2.2.2).

The Review Group has developed a proposed rule change which addresses the two previous recommendations. A Commission Paper containing the rule change is located in Appendix A to Volume 1.

- Permit line-item improvements in accordance with the Improved Standard Technical Specifications to be made available to individual licensees (on a plant-specific basis) in addition to lead and subsequent plant licensees. (Sec. B.2.2.4).

The Office of Nuclear Reactor Regulation (NRR) plans to establish an organizational structure to deal with individual plant Technical Specification improvement requests. Therefore, the Review Group considers that appropriate action has been initiated to address this recommendation.

- Expand the use of performance-based requirements to supplant prescriptive criteria in license conditions and technical specifications. In items exhibiting inherent flexibility, the functional requirement is distinguishable from the technical details needed to implement that requirement. As evidenced in the Technical Specification Improvement Program, licensee-controlled programs that govern such implementation details can provide both flexibility and the requisite assurance of system functionality. (Sec. C.2.2.4).

- The licensees should conduct a comprehensive and thorough assessment of their own licenses to identify any items that have the potential for reducing regulatory burden or enhancing flexibility without decreasing the current level of safety. The licensees should inform NRR of any license changes that they would likely pursue and the schedules on which they would pursue them. NRR should consider this information in view of its other regulatorily-mandated work before it decides whether to redirect additional resources to this effort.

3.3.5 Other Considerations

The Review Group's recommendations, provided in Section 3.3.4 of this report, have the potential to both reduce the regulatory burden on and enhance the flexibility available to licensees without reducing the current level of safety. Implementation of these recommendations would require that NRR redirect some of its resources not only to deal with the license amendment requests that would be needed to revise the licenses but also to modify some of its own processes. Some of these processes would require rulemaking before license amendments could be issued.

Because NRR's workload has for some time exceeded its resources, it has established a system for setting priorities for its work. License amendments such as those that would result from the Review Group's recommendations and that are not needed to ensure safety are of the lowest priority. Therefore, in order for NRR to implement many of the Review Group's recommendations, it would have to redirect some of its resources away from what is now higher priority work.

The Review Group's recommendations were made without regard to either the magnitude of potential licensee requests for such changes or NRR's ability to accommodate the requests. It would be imprudent for NRR to consider redirecting its resources to this endeavor without first knowing the expected licensee response. Likewise, it would be imprudent for the licensees to proceed with their license amendment requests without the assurance that NRR would be able to process them.

Since the licensees would be the primary beneficiaries of the license changes, the Review Group believes that before NRR considers whether to implement its recommendations, the burden is properly on the licensees to inform NRR of the license changes that they would likely pursue and the schedules on which they would pursue them. In addition, the licensees have an obligation to provide submittals of a quality necessary to support the requests.

These aspects are discussed in more detail in the following sections.

3.3.5.1 Licensee Burden

The Review Group's recommendations were based on its assessment of a selected number of items in a representative sample of licenses. To determine whether such license changes would be worthwhile, each licensee should conduct a comprehensive and thorough assessment of its own license. Some of the methodology used by the Review Group may prove useful in any such endeavor.

The Review Group found that many licensees have not taken advantage of the considerable flexibility that is already available to them. It is not clear whether they are unaware of this flexibility potential or whether they have chosen to not pursue the changes for other reasons. Nevertheless, it is up to the individual licensee to determine whether changes should be pursued.

In order for NRR to be able to make an informed decision on whether to redirect some of its resources, it must have a reasonable idea of the changes that would be requested by the licensees, including those that would require rulemaking to implement. The information needed includes not only the number of licensees that are expected to pursue changes but also the numbers and types of changes and the schedules on which they would be requested. The Review Group believes the collection of this information could be accomplished most efficiently through a representative licensee organization.

It is imperative that the licensees' amendment requests be of a quality sufficient to avoid the need for NRR to request additional information. Quality submittals can be ensured in several ways. First, the licensees should clearly address the pertinent regulatory requirements and the safety relevance of their request. Secondly, the licensees should try to anticipate the NRC's information needs; they should not just provide a minimal amount of information assuming that if NRR needs more, it will ask for it. Thirdly, the licensees could establish a clearinghouse-like process in which license change requests that have already been approved are made known and readily available to others. A representative licensee organization could effectively provide this function. Finally, pre-application dialogue with the cognizant NRR projects and review personnel can provide valuable insights that can help ensure complete submittals.

3.3.5.2 NRC Resources

The license changes that would result from the Review Group's recommendations could result in potential resource savings to the licensees. However, substantial upfront investments must be made by both the licensees and the NRC before these savings can be realized. As discussed previously, the resource impact on both the NRC and the licensees must be considered in establishing an efficient process for handling license amendment requests as well as in implementing many of the Review Group's recommendations, particularly those that would require rulemaking. The successful implementation of such changes and process modifications is, therefore, dependent on the effective communication of the expected response of the licensees to the NRC. In addition, if substantially more flexibility is provided to the licensees by allowing implementing details to be relocated to licensee-controlled documents, NRR would have to augment its inspection program, perhaps in the form of additional performance-based inspections as part of its core inspection program, to ensure that the current level of safety is maintained.

Since NRR would have to redirect some of its resources to this effort, it will have to consider not only how many resources would be needed to implement the Review Group's recommendations but also the impact that resource redirection would have on its currently higher priority work. NRR is currently devoting approximately 75 percent of its headquarters resources to operating reactor support and 25 percent to areas such as advanced reactors and plant license renewal. Of those resources that are being devoted to operating reactor support, only about one-fifth are being spent on processing operating license amendment requests. In assessing both the need for and impact of future resource allocations, NRR will have to consider the sometimes competing interests of the licensees and other segments of the industry as well as its own regulatory mandate. The Review Group believes that comments on this aspect from both the licensees and other segments of the industry, and the public in general would be beneficial.

Finally, the Review Group makes no recommendations concerning the details of how its recommendations should be implemented, whether NRR should redirect some of its resources to this effort, and, if so, where those resources should come from and how they should be used. These details are properly the prerogative of NRR.

3.3.6 Summary and Conclusions

The Review Group assessed the licenses of four plants that it believes are representative of a substantial fraction of the total population of licenses. The Review Group found that the licenses provide not only considerable flexibility to the licensees, but also the potential for further reducing the regulatory burden and enhancing the flexibility for licensees without adversely affecting the current level of safety. With a few exceptions, operating license conditions were noted to generally have a clear relevance to safety and sound regulatory bases. This observation is supported by the Review Group's finding that the number of license items that have either inherent flexibility or enhanced flexibility potential are much larger than the number of items that exceed the applicable regulatory requirements. Also, this suggests that the greatest potential for reducing regulatory burden lies in pursuing additional flexibility instead of making changes to the underlying regulatory requirements.

The Review Group also found that many licensees have not taken advantage of the flexibility that is already available to them. Achieving this additional flexibility through the adoption of the Technical Specification Improvement Program initiatives and other generic guidance appears to represent a significant benefit that is readily attainable at the present time. However, the expenditure of substantial resources devoted to the submittal and processing of license amendments is the upfront cost to both licensees and the NRC. The implementation of such a large effort would also likely adversely impact NRR's other currently higher priority work.

Before making the necessary investment in resources to adequately support such a project, NRR must know whether licensees would take advantage of the enhanced flexibility if the process for achieving it were made more readily available to them. Since the licensees are the primary beneficiaries of amendments that would add flexibility to the licenses, the burden is properly on them to inform NRR of their intentions in this regard. If NRR decides to redirect some of its resources to this effort, it is also incumbent on the licensees to provide license amendment requests of a quality necessary to support the changes.

Finally, while all the Review Group's recommendations have the potential to reduce the regulatory burden and enhance the flexibility for licensee implementation of its license requirements, action to adopt certain of these recommendations would entail NRC process modifications, some of which would require rulemaking. Since none of these recommendations are necessary to ensure safety, the resource impact and other implementing ramifications should be assessed by the affected NRC staff organizations to determine if the perceived benefits to the industry are cost effective. The Review Group notes that action has been initiated or already taken to address a number of the recommendations. While it is believed that the recommendations are generally relevant and directly applicable to most of the existing plant operating licenses, the Review Group recognizes that any decision to initiate actions on the remaining recommendations will be made based on other cogent considerations that could outweigh the findings and conclusions presented in this report.

APPENDIX A

ASSESSMENT OF OPERATING LICENSE

SEABROOK UNIT 1

U. S. Nuclear Regulatory Commission

Regulatory Review Group

February 1993

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A.1 ASSESSMENT OF SEABROOK OPERATING LICENSE

A.1.1 Seabrook License

The Seabrook Unit 1 operating license was issued on March 15, 1990. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Protection Plan, which is Appendix B to the license. The license as reviewed had been amended through Amendment 11, dated May 29, 1992.

A.1.2 Assessment of License

The Seabrook operating license contains 331 items. Each of the items was reviewed and assigned to one of the categories in Table 2 of Section 3.1. The numbers of items in the Seabrook operating license by category are shown in Table A.1.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on the items' regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 77 items or approximately 23 percent of the total number of items.

The number of items in the remaining categories that would be assessed individually was determined to be approximately 10 percent or 25 of the 254 remaining items. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or five of the 50 items in Category D would be selected for further assessment. With the 77 items that would be assessed collectively, this meant that 102 or approximately 31 percent of the 331 total items would be assessed either collectively or individually.

The items that were to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. The items assessed are listed in Table A.2.

Each item was assessed either collectively or individually as appropriate by considering the answers to specified questions presented in Table 3 of Section 3.1. The questions were designed to determine whether the item has a sound regulatory basis, is related to public health and safety, inherently allows the licensee flexibility in making changes to the plant or operations, or could be modified to provide increased flexibility to the licensee. The questions were written in such a manner that a "no" response would elicit additional review. The items were analyzed as necessary to ensure an adequate understanding of the items' regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. The results of each of the assessments were integrated and summarized, and the overall findings and recommendations were developed. Finally, those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed to determine what in the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

The overall results were integrated and the recommendations developed.

A.1.3 Results of Assessment

The summaries of the assessments of each of the items are provided in the attachment to this appendix. The summaries are presented in the order of the categories into which each of the items was assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; next, in the Technical Specifications (TS); and, finally, in the Environmental Protection Plan (EP).

The overall findings and recommendations are presented in Section A.2.

Table A.1

SEABROOK OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	136
B. Non-Technical License Conditions	4
C. License Conditions That Rely on Other Documents for Requirements	32
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	50
E. Reporting and Recordkeeping Requirements	36
F. Unique Plant Features	10
G. Other	63
Total	331

Table A.2

INDEX OF SEABROOK OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category A (13 of 136)**	
TS 2.1.2	Reactor coolant system pressure	A-21
TS 3.0.3	General limiting condition for operation	A-22
TS 3.1.2.7	Isolation of unborated water sources	A-23
TS 3.3.3.3	Seismic instrumentation	A-25
TS 3.4.6.2	Operational leakage	A-26
TS 3.5.4	Refueling water storage tank	A-27
TS 3.6.1.7	Containment ventilation system	A-28
TS 3.7.1.2	Auxiliary feedwater system	A-30
TS 3.7.4	Service water system	A-32
TS 3.8.2.1	D.C. electrical power system	A-34
TS 3.9.4	Containment building penetrations	A-35
TS 3.12.2	Land use census	A-36
TS 5.6.3	Spent fuel storage pool capacity	A-37
	Category B (4 of 4)***	
OL 2.B.7	Sale and leaseback condition	A-38
OL 2.H	Financial protection condition	A-38
OL 2.I	Marketing of energy condition	A-38
OL 2.J	Effective date and expiration condition	A-38
	Category C (3 of 32)**	
OL 2.E	Physical security condition	A-39
TS 3.4.10	Structural integrity	A-40
TS 6.2.2.e	Station staff working hours	A-42

Table A.2 (Continued)

INDEX OF SEABROOK OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category D (5 of 50)**	
TS 6.2.2.a	Minimum shift crew composition	A-44
TS 6.2.3.2	ISEG composition	A-45
TS 6.4.1.7	SORC responsibilities	A-46
TS 6.7.3	Temporary changes of procedures	A-48
EP 3.1	Changes in design and operation	A-50
	Category E (4 of 36)**	
OL 2.G	Violation reporting condition	A-51
TS 3.3.3.4	Meteorological instrumentation	A-52
TS 6.4.1.8	SORC records	A-54
TS 6.8.1.5	Monthly operating reports	A-55
	Category F (10 of 10)***	
OL 2.A	Applicability condition	A-56
TS 5.1.1	Exclusion area	A-56
TS 5.1.2	Low population zone	A-56
TS 5.1.3	Unrestricted areas	A-56
TS 5.2.1	Containment configuration	A-56
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OL = Operating license condition
TS = Technical Specification
EP = Environmental Protection Plan condition

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** Numbers in parentheses indicate the number of the total number of items in the category that were assessed.

*** Items that were assessed collectively; all others were assessed individually.

A.2 ASSESSMENT FINDINGS AND RECOMMENDATIONS

A.2.1 Introduction

The item assessment summaries were reviewed to determine which of the items appear to exceed the applicable regulatory requirements, given their safety significance and regulatory bases; which of the items should be considered for possible reduction in regulatory burden; which of the items provide at least some inherent flexibility, and why licensees may not be taking full advantage of that flexibility; and which of the items should be considered for enhanced flexibility. The items that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Item 2E, the physical security operating license condition, appears in three groups. The item appears to have potential for reduction in regulatory burden, it has at least some inherent flexibility, and it appears to have potential for enhanced flexibility.

A.2.2 Findings and Recommendations

A.2.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

Findings: Seven of the items assessed appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the Seabrook operating license. It is recognized that 10 CFR 50.50 authorizes the Commission to include in licenses such conditions as it deems appropriate. The Review Group was not able to review the entire body of underlying regulatory guidance for all these items. Therefore, although all the items appear to prescribe conditions or require actions that exceed applicable regulatory requirements, there may indeed be additional regulatory bases for their presence as license conditions.

The items that appear to exceed the applicable regulatory requirements are as follows:

TS 3.1.2.7	Isolation of unborated water sources
TS 3.7.1.2	Auxiliary feedwater system
TS 3.7.4	Service water system
TS 3.8.2.1	D.C. electrical power system
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.8.1.5	Monthly operating reports

Technical Specification 3.1.2.7 exceeds the provisions of both the Standard and Improved Standard Technical Specifications in that they contain no provisions for isolation of unborated water sources in the shutdown modes.

Although generally similar in design to other Westinghouse four-loop plants, some of Seabrook's systems are unique, both in meeting applicable regulatory guidance and in providing component and system redundancy that exceeds regulatory requirements. Technical Specification 3.7.1.2 appears to elevate the interpretation of branch technical position guidance to the status of a general design criterion resulting in the imposition of additional requirements somewhat inconsistent with the original plant design. Technical Specifications 3.7.4 and 3.8.2.1 appear to ignore the extra redundancy afforded by the design of the original systems and either impose additional provisions on the systems or require that the extra components receive the equivalent Technical Specification controls mandated for other Westinghouse four-loop plants without spare equipment. The licensee, in effect, appears to have been penalized for providing this additional redundancy, and therefore increased safety margin, and for its attempt to use unique design applications.

The problems with Technical Specifications 3.1.2.7, 3.7.1.2, 3.7.4, and 3.8.2.1 appear to be in their implementation in the Seabrook operating license. Since the problems are plant-specific in nature, they can be pursued directly by the Seabrook licensee. However, these and similar types of Technical Specification provisions may exist at other plants. Therefore, consideration should be given to providing additional guidance for accommodating the governing criteria of systems with extra component redundancy and unique design applicability.

Technical Specifications 6.2.2.a, 6.2.2.e, and 6.8.1.5 elevate provisions of Commission policy statements, regulatory guides, and other non-requirements to the status of legal requirements. Technical Specification 6.8.1.5 elevates a regulatory guide reporting provision for which there is questionable safety justification to the status of a legal requirement.

Recommendations: Based on the foregoing, the Review Group recommends the following:

- Reconsider the practice of elevating Commission policy statements, regulatory guides, and other non-requirements to the status of legal requirements without following the disciplined rulemaking process.

- Evaluate the adequacy of existing guidance for reviewing design features that exceed regulatory requirements or provide alternative means of compliance. Such guidance should encourage flexibility in the Technical Specifications for those design features for which the review concludes that increased safety margin is provided.

A.2.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

Findings: Four of the items assessed appear to have the potential for possible reduction of regulatory burden. They are as follows:

OL 2.E	Physical security condition
TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 6.8.1.5	Monthly operating reports

The physical security license condition, OL 2.E, essentially repeats the 10 CFR 50.54(p) requirement to obtain a license amendment to make changes to the physical security plans that decrease their safeguards effectiveness. Similar plans, e.g., the emergency response plan and the quality assurance plan, do not require a license amendment to make such changes. Although required by the regulations, this higher-level change process does not appear to be justified in terms of the physical security plans' safety significance relative to that of the other plans. Also, consideration should be given to providing enhanced flexibility in the implementation of the physical security plans. This aspect is addressed in Section A.2.2.4 of this report.

Two items--Technical Specifications 3.3.3.3, seismic instrumentation, and 3.3.3.4, meteorological instrumentation--impose reporting requirements as surrogates for corrective actions. Further analysis, however, revealed that these Technical Specifications do not appear in the Improved Standard Technical Specifications and, therefore, can be considered for line-item elimination.

Technical Specification 6.8.1.5 imposes a regulatory guide reporting provision for licensees to submit monthly operating reports. This appears to be a significant burden for the licensees without a commensurate return in safety. Although the Committee to Review Generic Requirements (CRGR) and the Reporting Requirements Task Force have evaluated a number of specific reporting requirements, a broader approach that considers all the information needed by the NRC to satisfy its regulatory mandate may be appropriate.

Recommendations: Based on the foregoing, the Review Group recommends the following:

- Evaluate the efficacy of a consistent approach for accommodating changes to the physical security, emergency response, and quality assurance plans within their proper regulatory and safety contexts; reconsider the current requirement for physical security plans to be included in a license condition.

- Conduct a comprehensive reevaluation of the information/data the NRC needs from nuclear power plant licensees in order to accomplish its mandate of protecting the health and safety of the public (recognizing the efforts of the CRGR and the Reporting Requirements Task Force), information/data requirements without a clear nexus to that mandate and duplicative reporting requirements should be eliminated.

A.2.2.3 Items That Provide Inherent Flexibility

Findings: Six of the items assessed were found to have at least some inherent flexibility. That an item has at least some inherent flexibility does not preclude it from consideration for enhanced flexibility or reduction in regulatory burden. The items with inherent flexibility are as follows:

OL 2.E	Physical security condition
TS 3.4.10	Structural integrity
TS 3.9.4	Containment building penetrations
TS 3.12.2	Land use census
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours

The nature of the inherent flexibility provided by these items varies from item to item. For example, the physical security and land use census items provide inherent flexibility by specifying the conditions under which changes to their respective programs can be made without prior NRC approval. The item governing the structural integrity of ASME Code components derives its flexibility not only from the ASME Code component classification process, but also from the relief request process used to exempt impractical Code requirements. Further flexibility has been provided by NRC guidance, such as Generic Letter 91-18, which is an example of a regulatory enhancement to flexibility with no adverse impact on safety.

The inherent flexibility of the containment building penetrations item is recognized in the options provided for compliance with the operability criteria. The minimum shift crew composition item specifies just minimums; licensees may exceed the minimums without NRC approval. The station staff working hours item provides the licensee essentially unlimited flexibility in setting the staff's working hours without NRC approval provided the appropriate procedures are followed.

Of the six items with inherent flexibility, one item--physical security--was also judged to have potential for reduction in regulatory burden and enhanced flexibility. These aspects are addressed in Sections A.2.2.2 and A.2.2.4 of this report. Another item--land use census--has been eliminated from the Improved Standard Technical Specifications and, therefore, could be considered by licensees for line-item elimination from their Technical Specifications. The remaining four items revealed no bases for further consideration.

Although licensees appear to be taking advantage of much of the inherent flexibility afforded them, a significant amount of that flexibility is not being exercised. Possible reasons include (1) the lack of awareness on the part of the licensees that the flexibility exists; (2) the flexibility afforded by an item is not needed; (3) the cost in time or resources to take advantage of the flexibility outweighs its benefits; (4) potential for public hearing if exercise of the flexibility requires a license amendment or prior NRC approval; (5) fear of second-guessing by NRC reviewers or inspectors if the change is subject to post-implementation scrutiny; (6) fear of ratcheting by NRC reviewers or inspectors during the change process; (7) negative perception of the licensee's actions by State regulatory bodies, the NRC, or the public; (8) complacency on the part of the licensee; and (9) reluctance of a licensee to assume the lead in pursuing changes to license requirements, e.g., line-item improvements in accordance with the Technical Specification Improvement Program.

Recommendation: Based on the foregoing, the Review Group recommends the following:

- Invite the industry to provide the staff with candid insights on licensees' reasons for not taking more advantage of the inherent flexibility afforded them.

A.2.2.4 Items That Should Be Considered for Enhanced Flexibility

Findings: Six of the items assessed appear to have enhanced flexibility potential. They are as follows:

OL 2.E	Physical security condition
TS 3.0.3	General limiting condition for operation
TS 3.6.1.7	Containment ventilation system
TS 6.2.3.2	ISEG composition
TS 6.4.1.7	SORC responsibilities
TS 6.7.3	Temporary changes of procedures

The physical security license condition, OL 2.E, provides flexibility in making changes to the physical security plans; however, additional flexibility could be provided in the implementation of the plans. For example, compensatory measures are generally prescriptive and may not always be in the best interest of overall plant security. Allowed outage times are not permitted as they are for safety-related equipment in the Technical Specifications. In addition, the baselines from which changes can be made without prior NRC approval are set by the provisions of the plans themselves, not by the regulations.

Technical Specification 3.0.3 may be unduly prescriptive in that it requires that the plant be shut down within specified completion times when the other Technical Specification limiting conditions for operation and their associated action statements are not met. It does not consider the risk of extending the completion times relative to that of shutting down the plant. This is an area that could be made more performance based and in which the application of risk assessment methodology could be considered.

Technical Specification 3.6.1.7 appears to be unduly prescriptive in ensuring the intended containment isolation requirement. More performance-based options for ensuring that valves are "locked-closed" or "sealed-closed" are needed. In addition, flexibility in the surveillance requirements, especially for the smaller diameter penetrations, may be appropriate, particularly if properly coordinated with the provisions of 10 CFR 50, Appendix J. This is an area in which the application of risk assessment methodology could be considered.

Technical Specification 6.2.3.2 was initially identified for consideration for enhanced flexibility but it has been replaced in the Improved Standard Technical Specifications by a substantially more flexible requirement. Therefore, it may be considered for a line-item improvement.

Technical Specification 6.4.1.7 appears to be unduly prescriptive in that it requires the SORC to provide the same level of consideration to required procedures and all proposed changes to station systems or equipment that affect nuclear safety. A more performance-based or graded approach that takes into account the relative safety significance of the different areas and items under review would provide additional flexibility. Such implementation flexibility would likewise affect the conduct of Technical Specification 6.7.3 activities, as the need for controls over temporary procedure changes could be conditioned on the safety significance of the affected procedures. These are areas in which the application of risk assessment methodology could be considered.

Recommendations: Based on the foregoing, the Review Group recommends the following:

- Consider providing additional flexibility in the implementation of the physical security plans, such as providing Technical-Specification-type allowed outage times.

- Evaluate the feasibility of employing a graded approach to the applicability of the technical provisions of certain limiting conditions for operation and surveillance requirements and in the implementation of specific review committee functions, e.g., SORC procedure and design change reviews. The appropriate application of risk assessment methodology could be valuable in establishing both the bounds and direction of such an approach.

A.2.2.5 Items Considered or Being Considered in Other Programs

Findings: Seven of the items assessed have already been or are being considered in other programs. They are as follows:

TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 3.12.2	Land use census
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.2.3.2	ISEG composition
TS 6.8.1.5	Monthly operating reports

Technical Specifications 3.3.3.3, 3.3.3.4, 3.12.2, and 6.2.3.2 have already been considered and eliminated by the Technical Specification Improvement Program. Therefore, these items can be considered for possible elimination from plant-specific Technical Specifications as line-item improvements.

The subjects of minimum shift crew composition and station staff working hours are being considered for possible modification by the Office of Nuclear Reactor Regulation. This effort could result in changes to their underlying Commission policy statements and regulations and, consequently, the Improved Standard Technical Specifications.

Technical Specification 6.8.1.5 is being considered by the Reporting Requirements Task Force.

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

A.2.2.6 Items for Which No Further Consideration Is Warranted

Findings: Ninety-three of the items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is also included here.

The items for which no further consideration is warranted are as follows:

OL 2.G	Violation reporting condition
TS 2.1.2	Reactor coolant system pressure
TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 3.4.6.2	Operational leakage
TS 3.4.10	Structural integrity
TS 3.5.4	Refueling water storage tank
TS 3.9.4	Containment building penetrations
TS 3.12.2	Land use census
TS 5.6.3	Spent fuel storage pool capacity
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.2.3.2	ISEG composition
TS 6.4.1.8	SORC records
TS 6.8.1.5	Monthly operating reports
EP 3.1	Changes in design and operation
Cat. B items	Non-technical license conditions (4 items)
Cat. F items	Unique plant features (10 items)
Cat. G items	Other (63 items)

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

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ATTACHMENT TO APPENDIX A - ITEM ASSESSMENT SUMMARIES

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 2.1.2**

Seabrook Technical Specification 2.1.2, reactor coolant system pressure, requires that the reactor coolant system pressure not exceed 2,375 psig. This item was chosen because it is representative of the Seabrook Technical Specification safety limits.

The regulatory bases for this Technical Specification are 10 CFR 50.36 and 10 CFR 50.55a. The former requires that the Technical Specifications include "... limits on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity"--in this case, the reactor coolant pressure boundary. The latter requires that pressurized reactor coolant pressure boundaries meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The Technical Specification is relevant to safety in that it is needed to ensure the integrity of the reactor coolant pressure boundary, one of the plant's multiple barriers against the release of reactivity.

The Technical Specification provides no inherent flexibility to the licensee; it prescribes the maximum limit for the reactor coolant system pressure. That degree of prescriptiveness is not inappropriate in view of its safety significance. There appears to be no enhanced flexibility potential for this requirement.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: A Item: TS 3.0.3

Seabrook Technical Specification 3.0.3, general limiting condition for operation, specifies what action must be taken when other limiting conditions for operation action statements are not met. This item was chosen because of its potential for enhanced flexibility.

10 CFR 50.36(c)(2) requires that when a Technical Specification limiting condition for operation, the lowest functional capability or performance level required for safe operation, is not met, the licensee shall follow any remedial action permitted by the Technical Specifications or shut down the reactor until the condition can be met. Technical Specification 3.0.3 delineates the completion times for shutting down the reactor when the limiting conditions for operation and their associated action statements are not met.

The requirement is relevant to safety in that the Technical Specification limiting conditions for operation and their associated action statements cannot cover all possible situations. Such a requirement is needed to cover those circumstances in which the other requirements are not met. The Technical Specification provides no inherent flexibility to the licensee.

It is not clear that the Technical Specification could not be made more flexible. Since not all limiting conditions for operation have the same safety significance, the completion times allowed for achieving hot standby, hot shutdown, and cold shutdown could possibly be made more performance oriented, e.g., by considering situation-specific factors. Further, it may not always be safer to change operational modes. For example, if there is reasonable assurance that the situation could be rectified within 1 hour after the completion time for changing modes expires, it might be safer to maintain the reactor in its present mode for that additional period of time than to change modes.

Based on the above considerations, it is concluded that the Technical Specification is appropriate; however, it may be unduly restrictive. Therefore, it is recommended that further consideration be given to this item for possible enhanced flexibility. This might be an area where risk assessment methodology could be applied to compare the relative risks of extending the completion times and shutting down the plant.

SUMMARY OF SEABROOK ASSESSMENT

Category: A Item: TS 3.1.2.7

Seabrook Technical Specification 3.1.2.7, isolation of unborated water sources, requires isolation of the reactor coolant system from unborated water sources in the shutdown modes. This limiting condition for operation (LCO) ensures that the boron dilution flow rates cannot exceed the value assumed in the plant transient analysis. This item was selected for review because it is representative of the requirements for reactivity control systems and also provides the opportunity to evaluate shutdown provisions.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A, and has safety relevance in providing reactivity controls (i.e., precluding boron dilution) that ensure acceptable fuel design limits are not exceeded. While some flexibility is allowed by providing the licensee options on component manipulations, there appears to be little overall inherent flexibility in this item. It is prescriptive in the LCO provisions as well as the action requirements. This Technical Specification might also be considered a surrogate item in that it requires non-safety-related systems to be maintained in an inoperable state as a means of ensuring that an acceptable shutdown margin is maintained, whereas the capability to provide adequate boration during shutdown modes is redundantly ensured by other Technical Specification requirements.

It is noted that both the Standard and Improved Standard Technical Specifications do not specify a comparable requirement to this item for the isolation of unborated water sources during shutdown conditions. Also, an inconsistency between the LCO and the documented bases in the Seabrook Technical Specifications was identified in that the bases imply that the isolation provisions are needed in Mode 3 (i.e., hot standby) but the LCO as written is not applicable in Mode 3.

Based upon the above discussion, it is not clear whether either the prescriptive language of this item or the item itself is a needed Technical Specification requirement. The potential for delaying core alterations (e.g., refueling operations) if the LCO is not met exists. However, any change to enhance the flexibility of the item may not [illegible] worth the effort, because the overall requirements are not considered onerous.

This item appears to be unique to the Seabrook Technical Specifications. While having a regulatory-based safety intent, this item is prescriptive and appears to go beyond the regulatory requirements that provide the equivalent assurance of acceptable reactivity controls for similar reactors. More review is required to determine whether revision or elimination of this item from the Seabrook Tech....al Specifications is warranted. This item appears to illustrate how prescriptive technical requirements may be added as license conditions without a clear and consistent rationale for either the prescriptiveness or the lack of equivalency.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.3.3.3**

Seabrook Technical Specification 3.3.3.3, seismic instrumentation, requires that the seismic monitoring instrumentation, delineated as a specific listing of components, be operable at all times. This capability is deemed necessary to permit a comparison of the measured response to any earthquake to the design basis of the plant. Selection of this item for review was based upon the desire to evaluate a technical provision that prescribes the submittal of a report as the only action requirement.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 100, Appendix A, with reference to 10 CFR 50, Appendix A criteria and describes seismic instrumentation intended to meet the recommendations of Regulatory Guide 1.12. Safety relevance is established by the need for data to determine if the plant can continue to be operated safely following an earthquake. While there is no inherent flexibility in meeting the limiting condition for operation or the action and surveillance requirements, continued operation is permissible with the seismic instrumentation inoperable. The prescriptive language in the surveillance requirements appears warranted to meet the safety intent of maintaining operable instruments and of analyzing seismic data following an earthquake. However, the prescriptive action requirement to submit a special report to the NRC if one or more seismic instruments is inoperable for more than 30 days appears to represent an example of a report being substituted as a surrogate item to the actual goal, i.e., timely repair of the instrument.

Reduction in regulatory burden could be provided by the elimination of the surrogate special report. It is recommended that all Technical Specification action items that require only a report to the NRC be reviewed further for appropriate usage. If the reporting requirement is only a surrogate for corrective action, a more direct and flexibly worded action statement or the elimination of the item altogether may be better. It is noted that seismic monitoring instrumentation is not included in the Improved Standard Technical Specifications.

SUMMARY OF SEABROOK ASSESSMENT

Category: A Item: TS 3.4.6.2

Seabrook Technical Specification 3.4.6.2, operational leakage, states that the reactor coolant system leakage shall be limited to the following: no pressure boundary leakage, 1 gpm unidentified leakage, 1 gpm total reactor-to-secondary leakage through the steam generators and 500 gpd through any one steam generator, 10 gpm identified leakage, 40 gpm controlled leakage, and reactor coolant system pressure isolation valve leakages as prescribed by formula and the referenced table. This item was chosen because it is representative of a technical requirement that does not provide flexibility.

The legal requirement for this item is contained in 10 CFR 50, Appendix A, General Design Criterion 30, which states that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant system leakage. The guidance for achieving this requirement is contained in Regulatory Guide 1.45.

Maintaining the integrity of the reactor coolant system pressure boundary is a primary safety concern. Consistent with that philosophy, it is necessary to maintain the prescriptive requirements related to the leakage limits currently contained in the Technical Specifications. The only requirement where some flexibility may be permissible is related to the 10 gpm identified leakage limit provided that it could be demonstrated that there would be no reduction in the margin of safety if this limit were increased (i.e., the sensitivity of the leakage detection system was not degraded).

There are many surrogate methods of detecting reactor coolant system leakage; however, most do not provide a quantitative measurement. Regulatory Guide 1.45 contains several acceptable alternative methods and the Instrument Society of America Standard ISA-S67.03 also identifies alternative methods of leakage detection. Although these surrogates are available, it is questionable that they would provide the sensitivity required to satisfy the primary requirement of this Technical Specification or if these alternatives would be any easier to operate or maintain.

Based on the above considerations, it is concluded that the current Technical Specification requirements are appropriate to ensure primary reactor coolant system integrity.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.5.4**

Seabrook Technical Specification 3.5.4, refueling water storage tank, requires that the refueling water storage tank contain a minimum volume of borated water, a minimum boron concentration, and a minimum and maximum solution temperature. This item was chosen because it is an example of a Technical Specification requirement that has the potential to provide additional flexibility.

The legal bases for this requirement is contained in 10 CFR 50, Appendix A, General Design Criterion 27, which requires that the reactivity control systems be designed with the capability of adding poison to the reactor through the emergency core cooling system to ensure that reactivity changes can be controlled under accident conditions. Standard Review Plan Section 4.3 provides the guidance related to this requirement.

This requirement is important to safety since it provides a second independent method of reactivity control during accident conditions. This requirement is also prescriptive and affords little flexibility. The poison injection systems for boiling water reactors can use different combinations of poison concentration and flow rates provided the solution in the tank is maintained at a temperature that ensures the poison remains in solution. Since this approach has been found acceptable and used for boiling water reactors, it may also be applicable to pressurized water reactors. However, there may not be any significant benefit for PWRs since the minimum volume of borated water in the refueling water storage tank is dictated by emergency core cooling system considerations.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.6.1.7**

Seabrook Technical Specification 3.6.1.7, containment ventilation system, requires that each containment purge supply and exhaust isolation valve be operable to ensure primary containment isolation capability. The large, 36-inch-diameter containment purge isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a loss-of-coolant accident (LOCA) or steam line break accident. The selection of this item for review was based upon its representativeness of Technical Specifications where administrative controls (e.g., locking closed valves) are implemented to comply with the limiting conditions for operation (LCO).

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in the primary containment isolation criteria of 10 CFR 50, Appendix A, and the radiation dose criteria of 10 CFR 100. The surveillance requirements of this item are also related to 10 CFR 50, Appendix J, but are more prescriptive in their provisions. A clear and coherent safety relevance has been established in the LCO action, and surveillance requirements; however, no inherent flexibility exists within the item. This is evidenced by the fact that even with blind flanges installed in the shutdown purge and exhaust pipe lines, no relief from the routine valve surveillances is inherently available. The blind flanges were installed to meet the quantitative local leak rate criteria for the valves.

The prescriptive language of this item does not appear to be necessary to convey the primary containment isolation functional requirements. For example, an asterisked note regarding verification of valve position monthly could be interpreted to require visual checks upon containment entries even though the circuit breakers for these fail-closed valves are locked open and valve position indication is available in the control room. The enhanced flexibility potential for this item is, therefore, great. However, a Technical Specification revision would be required to clarify the existing language and expand the licensee's options to comply with the intended requirement. As a result of NRC inspection activities regarding Technical Specification compliance in this area, the Seabrook licensee is currently working with the Office of Nuclear Reactor Regulation and Region I on the interpretation and possible revision of this item.

While this item has a sound regulatory basis and safety relevance, the overall language is prescriptive and precludes the use of flexibility to meet the intended containment isolation requirement. The use of standard convention (e.g., what options exist to maintain a valve "locked-closed" or "sealed-closed") may add flexibility. Risk assessment methodology could be used to further evaluate the prescriptive requirements applied to all valves that are used to isolate the containment atmosphere. The results may indicate that smaller diameter penetrations require less rigorous surveillance requirements or administrative controls.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.7.1.2**

Seabrook Technical Specification 3.7.1.2, auxiliary feedwater (AFW) system, requires at least three independent steam generator auxiliary feedwater pumps and associated flow paths to be operable. This capability ensures that the reactor coolant system can be cooled down to the point when the residual heat removal system may be placed into operation, in the event of loss of offsite power. This item was selected for review because it represents a case in which a Seabrook safety system, such as the AFW system, design differs from the Westinghouse standard design.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in several General Design Criteria of 10 CFR 50, Appendix A, and is required to meet Branch Technical Position ASB 10-1 regarding diverse power sources in the application of the Standard Review Plan to the acceptability of the AFW design. While this item is safety relevant, the Seabrook AFW system design is unique (i.e., one 100% electric motor-driven pump in one AFW train instead of two 50% pumps to go along with the steam turbine-driven pump). This unique design has resulted in the addition of the non-safety-related startup feedwater pump to the AFW system Technical Specification as a third pump capable of being powered by an emergency electrical power supply upon manual operator action. The treatment of the startup feedwater pump as an AFW system Technical Specification requirement appears to go beyond the regulations and be otherwise based on a conservative interpretation of Branch Technical Position ASB 10-1, along with the apparent intent that the Westinghouse Standard Technical Specifications, which requires three AFW pumps, be mimicked.

This item has little inherent flexibility. The action requirement for an inoperable startup feedwater pump is the same as for either of the other two safety-related emergency feedwater pumps. Only when two pumps are declared inoperable and one of the pumps happens to be the startup feedwater pump is the action time extended. Given that the startup feedwater pump is located in the turbine building (i.e., a non-safety, non-seismic structure) and is normally powered by non-Class 1E (i.e., non-safety electric power), it appears that enhanced flexibility could be provided to the Seabrook licensee by at least allowing for a greater outage time for the startup feedwater pump than would be justified for either of the other two safety-related emergency feedwater pumps.

While the prescriptive language in this item was found to be needed to clearly delineate the requirements, the technical basis for incorporating all the startup feedwater pump requirements into this Technical Specification is neither consistent nor coherent. For example, two startup feedwater pump flow paths, via both the normal, non-safety main feedwater flow path and the emergency feedwater header, are required to be demonstrated operable; whereas each emergency feedwater pump requires only its normal flow path to the steam generators. This surveillance requirement, in effect, adds an additional requirement that would not have been imposed if a third emergency feedwater pump had been designed into the AFW system.

While the above discussion reveals a unique Seabrook AFW question, it may be an example of a more generic issue. Plants whose system designs meet the regulations but differ from Standard Review Plan guidance or Standard Technical Specification format may be penalized for their unique applications. As a generic coherency question, this issue may warrant further review.

SUMMARY OF SEABROOK ASSESSMENT

Category: A Item: TS 3.7.4

Seabrook Technical Specification 3.7.4, service water system, requires at least two independent service water loops to be operable with three operable pumps in each loop. The operability of the service water system ensures that sufficient cooling capacity is available for the continued operation of safety-related equipment during normal and accident conditions. This item was selected for review because the limiting condition for operation restrictively dictates the number of pumps in each service water loop that must be operable.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A, and has safety relevance in its functional capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink. However, while General Design Criterion 44 requires that "suitable redundancy" in components shall be provided, assuming a single failure, this item goes beyond the regulation by prescribing action if any one of six 100% pumps (or any combination thereof) is inoperable. Furthermore, this item requires more prescriptive actions than specified in the Standard Technical Specifications. In effect, it appears that, in this case, the Seabrook licensee is being penalized for having a spare pump installed in each service water loop.

There exists no inherent flexibility in this item. The safety-related cooling tower on site is designed with two independent cooling loops and provides an adequate ultimate heat sink option to the normal service water bay cooling path. Additionally, with two 100% capacity pumps in each loop of the service water cooling path, the loss of one pump in each loop would still provide redundant cooling capability to the normal ultimate heat sink, i.e., the Atlantic Ocean. However, given the above scenario (i.e., cooling tower totally available and each service water path functional, but one pump in each loop out of service), the Seabrook plant is placed in a 3-day action requirement to shutdown. By comparison, a plant upon which the Standard Technical Specification requirement was imposed would only have to take similar action if just one service water loop were operable (versus the four available Seabrook loops posed for the above scenario).

The foregoing discussion illustrates that an enhanced flexibility potential is great for items where the licensee has chosen to design "spare" components into the safety-related plant systems. This upfront conservatism could be viewed by risk assessment methodology and/or performance-based system criteria as an enhancement to system availability. However, if the Technical Specification requirements do not recognize the inherent redundancy of the installed "spare" components, both the flexibility and the consistent application of safety significance are diminished.

As a generic issue, plant designs that use installed "spare" components to increase system reliability should be encouraged and not penalized by the addition of prescriptive Technical Specification requirements. While such spare components (e.g., pumps) must be safety-related and should be governed by Technical Specification surveillance requirements, the NRC should evaluate the need for imposing shutdown actions on plants with fully functional and redundant loops available to perform the system safety function. The Seabrook licensee is currently reviewing this item, and other similar items whose system design employs spare equipment, and plans to submit Technical Specification revisions to address total-loop versus component operability.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.8.2.1**

Seabrook Technical Specification 3.8.2.1, D.C. electrical power system, identifies the D.C. electrical power sources that are required to be operable and energized when the plant is not shut down. This item was selected because it is an example of a Technical Specification that appeared to exceed the applicable regulatory requirements.

The primary regulatory requirement for this item identified in Standard Review Plan Section 8.3.2 is General Design Criterion 17, which states that the D.C. power system must be capable of performing its safety function assuming a single failure. The acceptance criteria for this requirement are contained in various regulatory guides and IEEE Standards.

This requirement can be satisfied by having two independent D.C. battery banks, one on each independent electrical train (i.e., Trains A and B). Seabrook Technical Specification 3.8.2.1 requires the licensee to have two operable 125-volt D.C. battery banks in each electrical train, which is twice the number required by the regulations. In addition, although the extra batteries are not required, the Technical Specifications contain an action statement that requires the plant to be shut down if one of the battery banks in one of the trains is inoperable for 30 days and requires the surveillances to be performed on these batteries to demonstrate operability. Other plants have installed backup battery banks and the NRC has required them to be included in the Technical Specifications because they are safety-grade systems that are used in place of the primary battery system. However, the NRC imposed no operability requirements on these backup battery systems. The surveillance requirements are only applicable to these batteries when they are used in place of the primary batteries and no plant shutdown requirements are imposed if the batteries are inoperable when not in use (performing the backup function). Although the licensees generally maintain these batteries in accordance with the surveillance requirements, they are not subject to Technical Specification violations. This affords the licensees flexibility that is not permitted in the Seabrook Technical Specifications.

Based on the above considerations, it is concluded that the Seabrook Technical Specification requirement related to D.C. battery sources goes beyond the regulatory requirements. Although this item reveals a plant-specific issue, it may be representative of a more generic concern. Therefore, it is recommended that the incorporation of requirements that go beyond the regulatory bases into plant-specific Technical Specifications be evaluated further.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.9.4**

Seabrook Technical Specification 3.9.4, containment building penetrations, requires all containment building penetrations to meet a specified status during core alteration activities such as refueling. These requirements ensure that a release of radioactive material within containment will be restricted from leakage to the environment. This item was selected for review because it is representative of the Technical Specifications governing refueling operations.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in both 10 CFR 50, Appendix A, and 10 CFR 100. It has safety relevance and the provisions appear commensurate with a postulated radioactive material release, i.e., a fuel element rupture with the containment is at atmospheric pressure. Inherent flexibility in both the limiting condition for operation and surveillance requirements exists since options are provided for complying with the stated operability criteria. Additionally, the action statement is consistent with the safety intent by requiring only a suspension of core alterations or the movement of irradiated fuel in the containment building, which represent the only applicable ongoing activities that relate to the postulated fuel element rupture event.

While a certain prescriptiveness exists in the Technical Specification, such language appears to be necessary to convey the intended technical details. Therefore, the enhancement flexibility potential for this item is considered low, particularly since the action statement is logical and not onerous. Further review of this area for enhanced flexibility is likely to be unproductive.

Overall, this item, even though limited in applicability to general refueling operations, appears to be technically sound and well directed to its safety intent, while at the same time allowing the licensee some flexibility of compliance activities. A direct correlation exists between the wording of this item and the language of the corresponding section of the Standard Technical Specifications. No additional review of this Technical Specification appears warranted.

SUMMARY OF SEABROOK ASSESSMENT

Category: A Item: TS 3.12.2

Seabrook Technical Specification 3.12.2, land use census, requires that a land use census be conducted and identify within a distance of 5 miles in each of the meteorological sectors the location of the nearest milk animal, the nearest residence, and the nearest garden greater than 500 square feet producing broad-leaf vegetation. This item was chosen because it was representative of requirements contained in the radiological environmental monitoring section of the Technical Specifications.

The legal requirement for this Technical Specification is contained in 10 CFR 50, Appendix I, and the regulatory bases for the implementation of Appendix I are contained in Regulatory Guide 1.109.

This requirement is relevant to safety in that it is necessary to protect the health and safety of the public. Maintaining doses as low as reasonably achievable is consistent with that philosophy. The land use census provides the information needed to identify a location that yields an exposure to the public from routine releases of plant radioactive effluent that are greater than at a location from which samples are currently being obtained.

This requirement has a great deal of inherent flexibility with regard to how and when this census is taken. Only the requirement that the survey be conducted at least once per 12 months during the growing season and the time limitations on incorporating new locations into the radiological monitoring program are prescriptive.

The one area where reduction might be possible is related to the frequency of the land use census; however, this would be dependent on the significance of the regulatory burden and on whether data were available to support a reduction in this requirement. It is noted that this item has been removed from the Improved Standard Technical Specifications and placed under the administrative control of the licensee. Therefore, this change could be considered by the licensee.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would probably be unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 5.6.3**

Seabrook Technical Specification 5.6.3, spent fuel storage pool capacity, states that the spent fuel storage capacity is designed and shall be maintained with a capacity limited to no more than 1,236 fuel assemblies. This item was chosen because it is representative of a design feature Technical Specification.

There is no specific legal requirement for this item. The regulatory bases for this requirement are identified in SRP Section 9.1.2, Subsection III.1, which states that the minimum storage capacity in the spent fuel storage pool shall be in accordance with ANS 57.2 Paragraph 5.1.15 (equal to or exceed one full core discharge plus the maximum normal fuel discharge for a single unit facility). This requirement is important to safety in that General Design Criterion 17 states that the system shall be designed with the capability to permit periodic inspection and testing of components important to safety. Therefore, it is necessary to have the capability to offload the core.

Although there is no flexibility in the spent fuel storage capacity, this limit can be changed by a Technical Specification amendment based on design considerations, e.g., criticality, rack size, and heat load limitations.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would be unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: B **Items: All**

The Seabrook license contains four items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. They deal with sale and leaseback transactions, financial protection, marketing of energy from the plant, and the effective date and expiration date of the license. Specifically, the Category B items are as follows:

OL 2.B.7	OL 2.H
OL 2.I	OL 2.J

The financial protection license condition is based on Section 170 of the Atomic Energy Act and 10 CFR 140. The effective and expiration dates license condition is required by Section 103 of the Atomic Energy Act and 10 CFR 50.51. The other two license conditions, the sale and leaseback transaction and marketing of energy license conditions, are not regulatory requirements but are authorized by 10 CFR 50.50, which provides that the license may contain such conditions as the Commission deems appropriate. Given Seabrook's unique financial and ownership situation, these conditions do not appear to be inappropriate.

None of the items is directly related to safety. Although the license conditions are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the non-technical license conditions are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: C Item: OL 2.E

Seabrook License Condition 2.E, physical security condition, requires the licensee to implement and maintain in effect all provisions of its approved physical security, guard training and qualification, and safeguards contingency plans and all amendments and revisions to the plans made pursuant to 10 CFR 50.90 and 10 CFR 50.54(p). This item was chosen because it allows the plans to go beyond the requirements specified by the regulations and thereby provides opportunity for ratcheting. It also elevates the baseline from which changes can be made without prior NRC approval to that higher level. In addition, it is similar to a number of other plans, such as the emergency response plan, quality assurance plan, and environmental protection plan, which are required by the regulations or the license.

The physical security plans are required by 10 CFR 50.34 and 10 CFR 73. Changes to the plans that do not decrease their safeguards effectiveness may be made without prior NRC approval in accordance with 10 CFR 50.54(p). Changes to the plans that decrease their safeguards effectiveness must receive prior NRC approval in accordance with 10 CFR 50.90. The plans are safety relevant in that they ensure protection of the plant against radiological sabotage and the potential resulting release of radioactive materials.

The regulatory process provides flexibility in developing and revising the plans However, additional flexibility could be provided in the implementation of the plans. For example, the generally assumed compensatory measure for loss of a plant perimeter alarm system is the immediate placement of guards within line of sight of each other around the perimeter. The placement of the guards around the perimeter could call unnecessary attention to the fact that the perimeter alarm system is not operable and, therefore, may not be in the best interest of overall plant security. No allowed outage times are permitted as they are for safety-related equipment in the Technical Specifications. Given the likelihood of a threat during relatively short periods of inoperability of the perimeter alarm system and the effectiveness of other security barriers, e.g., access to the plant buildings and vital areas, it seems that Technical-Specification-type allowed outage times would provide additional flexibility without reducing the overall safeguards effectiveness.

Based on the above considerations, it is concluded that the requirement is appropriate; however, it may be unduly restrictive. Therefore, it is recommended that further consideration be given to standardizing the change processes for these and similar plans and providing additional flexibility in their implementation, e.g., by providing Technical-Specification-type allowed outage times.

SUMMARY OF SEABROOK ASSESSMENT

Category: C Item: TS 3.4.10

Seabrook Technical Specification 3.4.10, structural integrity, requires that the structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with the inservice inspection (ISI) and inservice testing (IST) programs for the plant in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. This item ensures that the structural integrity and operational readiness of the piping and pressure boundary components governed by the ASME Code are maintained at an acceptable level throughout the life of the plant. This item was selected because of its reliance on other documents (e.g., the ASME Code) for technical requirements. In addition to the requirements contained in Technical Specification 4.0.5, this Technical Specification contains specific surveillance provisions for the reactor coolant pump flywheel that reference Regulatory Guide 1.14 (Revision 1) related to flywheel inservice inspection.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.55a and 10 CFR 50, Appendix A. While Regulatory Guide 1.14 is not a legal requirement, it also has basis in 10 CFR 50, Appendix A, and the guidance that is referenced in Technical Specification 3.4.10 appears to be consistent with other ISI program requirements. Since the reactor coolant pump flywheel is not a pressure boundary component, this regulatory guidance provides technical details unavailable in the ASME Code.

This item has safety relevance and appropriately uses a graded approach to the action requirements, dependent upon the ASME Code Class of the affected component. Reliance upon a regulatory guide to provide the reactor coolant pump flywheel inspection details also is appropriate, given the missile impact hazard and the lack of other standard technical criteria. Inherent flexibility does exist since this item refers to Technical Specification 4.0.5, which allows relief from the pertinent code requirements, if granted by the NRC, in accordance with 10 CFR 50.55a(f)(6)(i). Such relief requests are generally used to exempt code requirements that are impractical to a specific plant design or configuration. While the overall ISI/IST programs, which are submitted to the NRC for review and safety evaluation, may represent surrogate items to the intended goal (i.e., acceptable structural integrity of the pressure boundaries and associated components), the use of these surrogate items appears both technically sound and appropriate from a regulatory standpoint.

While prescriptive language is used in this Technical Specification and its referenced documents, i.e., the ASME Code and Regulatory Guide 1.14, such details are needed to provide the appropriate technical criteria. Performance-based criteria are already incorporated into the ASME Code Section XI requirements upon which the plant ISI/IST programs are based. Any attempt to use additional performance-based criteria, beyond the ASME Code provisions, would unnecessarily complicate this Technical Specification. Further NRC review of this area for enhanced flexibility does not appear warranted from a regulatory standpoint. However, from a research and technical standpoint, continued NRC liaison with the ASME Code Section XI committees will continue to provide for program revisions and additional flexibility, if appropriate. It is noted that, with Generic Letter 91-18, further flexibility in the form of NRC Inspection Manual Technical Guidance was provided in this area by allowing continued operation with nonconforming piping/support components until the next refueling outage if certain referenced analytical criteria (e.g., Appendix F of Section III of the ASME Code, NRC Bulletins 79-02 and 79-14) are met. Given that such guidance for continued operation can be supported by quantitative analysis, this Technical Specification currently establishes reasonable and acceptable controls. While no further review of this item is warranted, the use of a Generic Letter 91-18 to add flexibility to this area appears to have been beneficial and this approach could be explored further in other areas.

SUMMARY OF SEABROOK ASSESSMENT

Category: C   Item: TS 6.2.2.e

Seabrook Technical Specification 6.2.2.e, station staff working hours, requires that the licensee develop and implement administrative procedures that limit the working hours of station staff who perform safety-related functions. The Technical Specification further requires that the amount of overtime worked by such personnel "... be limited in accordance with the NRC Policy Statement on Working Hours." This item was chosen because it is an example of a Commission policy statement that has become a de facto requirement by its incorporation by reference in the plant's Technical Specifications.

The Commission's original "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" was issued on February 18, 1982 (47 FR 7352) and was forwarded to applicants and licensees by Generic Letter 82-02, "Nuclear Power Plant Staff Working Hours." The policy statement itself contains a request for applicants and licensees to include in their Technical Specifications administrative procedures regarding working hour restrictions that conform to those in the policy statement. The policy statement was revised slightly on June 1, 1982 (47 FR 23836) and was forwarded to applicants and licensees by Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours." The requirement is also contained in the Improved Standard Technical Specifications.

The requirement is relevant to safety in that personnel working in a fatigued condition could have reduced mental alertness or decisionmaking ability. It is noted that limiting working hours is used as a surrogate for limiting fatigue. Other surrogates have been considered but have been rejected.

The requirement has a great deal of inherent flexibility. Although there is no flexibility in the requirement for the licensee to have an administrative procedure, the policy statement and, hence, the Technical Specification, provides essentially no limit on the amount of overtime an individual can work. It only specifies that the overtime be given deliberate consideration and authorized in writing.

The Office of Nuclear Reactor Regulation is considering this issue for possible rulemaking and, to that end, has requested the Office of Nuclear Regulatory Research to proceed with the development of a rulemaking package.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive. Although the Commission clearly intended that this policy statement become a de facto requirement by its incorporation in plants' Technical Specifications, such is not the case for policy statements in general. Therefore, it is recommended that the elevation of non-requirements, such as policy statements, into requirements and the regulatory status of policy statements in general be given further consideration.

SUMMARY OF SEABROOK ASSESSMENT

Category: D Item: TS 6.2.2.a

Seabrook Technical Specification 6.2.2.a, minimum shift crew composition, specifies the minimum on-duty shift crew size and composition for the various operational modes. This item was chosen because it not only repeats the minimum licensed operator shift staffing requirements of 10 CFR 50.54(m) but also adds minimum shift staffing requirements for auxiliary operators and the shift technical advisor.

10 CFR 50.54(m) specifies minimum licensed operator shift staffing requirements for the various operational modes. The Technical Specification is consistent with that regulation for licensed operators. The NRC has no minimum shift staffing requirements for auxiliary operators or the shift technical advisor. The shift technical advisor is the embodiment of the Commission's policy statement on engineering expertise on shift. The policy statement, not a legal requirement, provides that engineering expertise on shift may be provided by either a dedicated shift technical advisor or by a senior reactor operator serving in a dual role. Technical Specification 6.2.2.a also provides that flexibility. In summary, the Technical Specification repeats an existing legal requirement and elevates a policy statement and non-requirement to a de facto legal requirement. This requirement is also contained in the Improved Standard Technical Specifications.

The requirement is relevant to safety in that it prescribes the minimum shift staffing requirements for the plant. It is noted that the shift technical advisor is a surrogate for engineering expertise on shift.

The requirement, although prescriptive, offers inherent flexibility in that it only prescribes the minimum staffing requirements. The licensee is free to exceed these minimum requirements and, in practice, usually does. However, the Technical Specification appears to have little if any potential for enhanced flexibility.

The Office of Nuclear Reactor Regulation is reevaluating the Commission's policy statement on engineering expertise on shift, including the need for and use of shift technical advisors, and the broader issue of minimum shift staffing requirements.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive. However, it is recommended that the elevation of non-requirements, such as policy statements, to the status of requirements and the regulatory status of policy statements in general be given further consideration.

SUMMARY OF SEABROOK ASSESSMENT

Category: D Item: TS 6.2.3.2

Seabrook Technical Specification 6.2.3.2, Independent Safety Engineering Group (ISEG) composition, states that the ISEG shall be composed of at least five dedicated, full-time engineers located on site with a science or engineering degree and at least 2 years of experience in the degreed field and 1 year of experience in the nuclear field. This item was chosen because of its very prescriptive nature with regard to manpower requirements.

This requirement is based on TMI Action Plan Item I.B.1.2 contained in NUREG- 0737. This particular item was required of applicants for operating licenses only. The purpose of ISEG is to perform independent reviews and audits of plant activities, review other appropriate internal and external information available, and provide recommendations to management where useful improvements can be made. Other than the scope of issues that ISEG reviews, the licensee has no control over the utilization of the five dedicated plant staff assigned to this function. The Improved Standard Technical Specifications permit the ISEG function to be performed under the review and audit program. This permits more flexible methods of performing the ISEG function (i.e., by a standing committee or by assigning qualified individuals capable of conducting these reviews and audits).

A survey performed on a limited number of plants licensed after TMI determined that some licensees have already requested license amendments that incorporate the provisions of the Improved Standard Technical Specifications into their Technical Specifications. In addition, some of the older plants' Technical Specifications were also surveyed, and it was determined that a few have adopted the ISEG approach while others have adopted the Improved Standard Technical Specification approach. The remaining older plants surveyed have incorporated variations of these approaches. It is not clear at this time why some of the older plants surveyed have incorporated the ISEG function into their Technical Specifications since it was not required by NUREG-0737. However, it appears that it was included on a voluntary bases.

Based on the above considerations, it is concluded that the Seabrook Technical Specification requirement related to the composition of ISEG provides little flexibility. However, a Technical Specification change can be submitted adopting the Improved Standard Technical Specification approach; that would provide considerable flexibility in the implementation of this requirement. Based on the viable alternative available, it is concluded that further consideration of this requirement would be unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: D Item: TS 6.4.1.7

Seabrook Technical Specification 6.4.1.7, Station Operation Review Committee (SORC), requires the SORC to make specific written recommendations to the Station Manager, render written determinations whether certain items constitute unreviewed safety questions, and provide written notification of disagreements between the SORC and the Station Manager. This administrative control implements a continuing monitoring activity that is considered to be an integral part of the routine supervisory function. This item was selected as a representative review activity of a committee required by the Technical Specifications.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.40(b) as it relates to the licensee being technically qualified to engage in licensed activities. The guidance provided by ANSI Standard N18.7 (ANS 3.2), as endorsed by Regulatory Guide 1.33, conveys additional regulatory criteria for the required review activities of an onsite operating organization. While the SORC monitoring activities have safety relevance in providing a timely oversight of routine and revised plant operations, the details of exactly what SORC is responsible to review, document, and report in writing have little basis in the regulation and relate more specifically to Standard Review Plan (NUREG-0800) provisions. The language in this item resembles the wording of the applicable section of the Improved Standard Technical Specifications.

While a certain degree of inherent flexibility exists for the implementation aspects of this item (e.g., telephone meetings, agenda), there is no inherent flexibility in what this Technical Specification requires the SORC to accomplish (e.g., recommend approval or disapproval of changes to any procedures required by the Technical Specifications; reference Technical Specification 6.7). This prescriptiveness does not appear to be either consistent or commensurate with the intended safety impact because not all the referenced procedures carry the same safety significance. While the use of SORC subcommittees can add some additional flexibility in workload allocation, a rigid interpretation of many of the SORC requirements, e.g., recommend in writing approval or disapproval of "<u>all</u> proposed changes or modifications to station systems or equipment that <u>affect nuclear safety</u>" (emphasis added) appears onerous given the various levels of safety significance that are inherent in nuclear power plant system and component designs.

It should be noted that the SORC has only advisory authority in that it recommends and renders determinations; the Station Manager has the responsibility for the resolution of any disagreements on overall station operation. Thus, the language in this item to convey the administrative control of SORC requirements appears to be overly prescriptive and could be flexibly enhanced by the use of performance-based criteria or a graded approach to safety-significant review activities. Use of risk assessment methodology could provide valuable input into the prioritization of SORC efforts and the determination of where limited review time could be most effectively directed.

This Technical Specification is prescriptive yet broadly scoped such that interpretation is required to define implementation details. Such a reliance on interpretation can lead to misapplication of this license condition in the inspection and enforcement area. While the safety intent of the SORC as an overview and advisory authority is soundly based, achieving enhanced flexibility in the administrative control of the SORC functions would be a worthwhile initiative. The Improved Standard Technical Specifications, while reducing the overall SORC review responsibilities, do not significantly alter the plant review function directed by this item. It is recommended that further review of this item beyond what is already in progress in the NSAC-125/10 CFR 50.59 area be conducted to evaluate not only the need for the current prescriptive language of Technical Specification 6.4.1.7, but also the prospects for enhanced flexibility by supporting more of a graded safety approach to the SORC review and recommendation functions.

SUMMARY OF SEABROOK ASSESSMENT

Category: D Item: TS 6.7.3

Seabrook Technical Specification 6.7.3, temporary changes of procedures, allows temporary changes to the procedures required by other Technical Specifications if the change is accomplished in accordance with specified provisions. These provisions include the requirement that the "intent" of the original procedure not be altered and other approval conditions. This item was selected as a representative administrative control governing plant procedures and programs.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.40(b) as it relates to the contribution of the administrative procedures to the technical qualification of the licensee; and also to 10 CFR 50.54(l), which requires that designated individuals be responsible for directing the licensed activities of plant operators. By reference, an association with Regulatory Guide 1.33 and the endorsed ANSI Standard N18.7 (ANS-3.2) also exists. Additionally, 10 CFR 50, Appendix B, delineates general quality assurance criteria for procedures and, in conjunction with Regulatory Guide 1.33, provides regulatory measures governing safety-related procedural controls. The safety relevance of this item is clearly established by the above regulatory references and by the need for procedural changes to properly reflect the appropriate safety-related requirements.

Some inherent flexibility can be found in this item both in the plant management staff options for review and in the judgment allowed for the determination of whether an original procedure intent has been altered. However, once a temporary procedural change is determined to be appropriate, this Technical Specification is generally prescriptive as to the controls that are required prior to and after implementation. While the prescriptive language in this item may not be necessary in that other review and approval processes could provide equivalent temporary procedural change controls, the existing requirements appear not only to incorporate standard industry guidelines but also to represent a sound practice that is not particularly burdensome.

One area where enhanced flexibility might be beneficial for this item is the possible reduction of the total number of procedures for which the full review and approval conditions must be applied. Since not all safety-related and Technical-Specification-required procedures carry the same safety significance, a "non-intent" temporary change to a procedure governing activities of lesser safety relevance may not need the full review dictated for temporary changes of greater impact. Performance-based criteria could be used to distinguish the safety significance of different levels of procedural controls. In turn, a graded approach to the review and approval process for procedural changes could thus be applied. However, development of such a hierarchical process of controls may not be worth the effort, especially if the simplicity and conservatism in the existing Technical Specification provisions are not considered onerous by the licensee.

Overall, this item has a sound regulatory basis and is coherent in the application of a logical review process to the procedural controls of safety-related activities. While little inherent flexibility exists, initiatives to enhance flexibility may overcomplicate the practice and not provide any tangible benefits. Also, since temporary procedure changes represent a contingency option to the formal procedure revision process, the need for additional flexibility may be neither great nor practical. No further NRC review of the Technical Specification is recommended. However, the use of a graded approach to procedure safety significance as discussed in the Summary Assessment for Technical Specification 6.4.1.7 would likewise provide implementation flexibility in the controls of temporary procedural changes.

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: EP 3.1**

Seabrook Environmental Protection Plan Section 3.1, changes in design and operation, specifies that before engaging in additional construction or operational requirements that may significantly affect the environment, the licensee shall prepare an environmental evaluation of such activity to determine if the activity involves an unreviewed environmental question. Section 3.1 also requires the licensee to provide a written evaluation of any activity that involves an unreviewed environmental question and to obtain NRC approval and maintain records of the changes associated with these activities. This item was selected because it an example of an administrative control.

The legal bases for this requirement are contained in 10 CFR 50.36b, which requires that conditions to protect the environment should be incorporated into an attachment to the license that is made a part of the license. The requirement provides protection to the health and safety of the public by ensuring that changes to the plant design or operation that could significantly affect the environment are evaluated prior to implementation. This requirement provides limited flexibility for items that do not constitute an unreviewed environmental question.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: E **Item: OL 2.G**

Seabrook Operating License Condition 2.G, violation reporting condition, states that the licensee shall report any violations of the requirements contained in Section 2.C of the license initially via the Emergency Notification System and with written followup within 30 days in accordance with procedures described in 10 CFR Part 50.73(b). This item was chosen because it is representative of a license condition that contains reporting requirements.

There does not appear to be a legal requirement or a regulatory basis for this license condition. This reporting requirement was put in the operating license to provide assurance that the licensee was fulfilling all its commitments identified under Section C of the license.

This reporting requirement does not have a great deal of flexibility and is judged to have little potential for any increased flexibility. However, there is one aspect of this license condition that some licensees may be misinterpreting that could in increased reporting requirements. Section 2.G of the license, as currently written, does not clearly define the licensee responsibilities for reporting violations of the Technical Specifications identified in Section 2.C(2) of the license. The wording in Section 2.G can be interpreted as requiring additional reporting requirements beyond those specified within the Technical Specifications. The wording in Appendix A to the license specifically states that violations of the Technical Specifications will be reported in accordance with the requirements of 10 CFR 50.72 and 50.73. It therefore appears it was not the intent of the operating license to require reports that go beyond these requirements. In addition, Section 2.G of some of the newer licenses specifically excludes Technical Specifications (Section 2.C(2) of the license) from the reporting requirements of Section 2.G.

A license amendment specifically excluding Section 2.C(2) of the license from this reporting requirement would eliminate any possible misinterpretation of the Technical Specification reporting requirement contained in Section 2.G. It is concluded that, beyond a plant-specific license amendment, consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: E Item: TS 3.3.3.4

Seabrook Technical Specification 3.3.3.4, meteorological instrumentation, requires that the specified meteorological monitoring instrumentation be operable at all times. This requirement ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This item was selected as a representative Technical Specification where the only action is a reporting requirement.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in both 10 CFR 100.10(c)(2) and 10 CFR 50.36a(a)2. The detailed requirements provide a capability to evaluate the need for initiating protective measures under certain plant conditions to protect the health and safety of the public and are consistent with the recommendations of Regulatory Guide 1.23. The support to radiological dose assessment capabilities provided by the details of this Technical Specification is therefore also connected to 10 CFR 50, Appendix E, and 10 CFR 20.

The safety relevance of this item is clearly established by the significance that correct and timely meteorological information has in proper dose assessments and emergency planning decisions. However, the consistency and safety significance of the action requirement of this Technical Specification is not readily evident. Given the inoperability of certain meteorological monitoring instrumentation, the action statement requires the licensee to submit a special report to the NRC outlining the cause of the malfunction and the plans for restoration. Such a reporting requirement within a 10-day deadline after an allowable outage time of 7 days appears to be inconsistent with the fact that, in accordance with the Seabrook Station Emergency Response Manual, an Unusual Event would have to be declared if certain categories of meteorological data (e.g., wind speed) became unavailable.

There is no inherent flexibility in the provision for the aforementioned report submittal when the conditions and timing trigger this requirement. The function of such a special report could be questioned, particularly if its purpose is only to encourage the licensee to take prompt corrective action. Such an intent would make the special report nothing more than a surrogate for timely restoration of the instrumentation. Given the existence of the Seabrook Station Radiological Emergency Plan, written in compliance with 10 CFR 50.34(b) and 10 CFR 50, Appendix E, and the potential for entrance into an Emergency Action Level (i.e., Unusual Event) upon loss of meteorological data, the need for such reporting appears even less consistent and significant. As discussed from a regulatory basis, the meteorological instrumentation has safety relevance. However, a more meaningful action, upon loss of some monitoring capability, would be an evaluation of the inoperable equipment in the context of any diminished capacity of the overall Emergency Response Plan.

This item is prescriptively worded and similar to the language in the Standard Technical Specifications. It is noted that the Improved Standard Technical Specifications do not include meteorological monitoring instrumentation. Therefore, enhanced flexibility could be provided by either eliminating the item or directing an action more consistent with the unique Seabrook Radiological Emergency Plan. This item also warrants further review to determine the function and utility of the special report currently directed by this Technical Specification action. It is recommended that this item, along with any other Technical Specifications that require reports as the only actions (see also Summary of Seabrook Assessment for Technical Specification 3.3.3.3), be evaluated further for appropriateness and/or improved coordination with existing plant programs that already address corrective response measures.

SUMMARY OF SEABROOK ASSESSMENT

Category: E Item: TS 6.4.1.8

Seabrook Technical Specification 6.4.1.8, Station Operation Review Committee (SORC) records, specifies the recordkeeping requirements of the committee. It requires that the SORC maintain written minutes of each meeting that document the results of all Technical-Specification-required SORC activities and that the SORC provide copies of the minutes to the Executive Director Nuclear Production and the Nuclear Safety Audit Review Committee. This item was chosen because it is representative of a number of Technical Specification administrative controls that impose reporting requirements.

The stated regulatory requirement for this item is 10 CFR 50.40(b), which requires that the licensee be technically qualified to engage in the licenses activities.

The requirement is relevant to safety in that it ensures that the offsite review committee and the corporate-level individuals responsible for the safe operation of the plant are kept informed of the Technical-Specification-required activities of the SORC.

The requirement provides no inherent flexibility to the licensee; it prescribes minimum requirements for content and distribution of the report. That prescriptiveness does not appear to be inappropriate. In view of its nature and safety significance, there appears to be no enhanced flexibility potential for this requirement.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: E **Item: TS 6.8.1.5**

Seabrook Technical Specification 6.8.1.5, monthly operating reports, requires the licensees to submit routine reports of operating statistics and shutdown experience to the NRC on a monthly basis. The guidance for submitting these reports is contained in Regulatory Guide 1.16. This Technical Specification was chosen because it is an example of a reporting requirement that is inflexible and whose safety significance is questionable. In addition, it appears that this is an example of NRC staff guidance that has been made a legal requirement.

Although the regulatory bases for this Technical Specification are contained in Regulatory Guide 1.16, there does not appear to be any direct regulatory requirement. The staff provides these reports to other agencies, e.g., Environmental Protection Agency, Department of the Interior, National Institute of Standards and Technology, pursuant to memoranda of understanding. In addition, some of the data in these reports is used by AEOD to evaluate performance indicators, e.g., critical hours, and also by users outside the NRC, e.g., public utility commissions, intervenors, consultants. The information from these reports is also used in the preparation of NUREG-0020 (Gray Book), which may be used by the industry to track the performance of other licensees.

This requirement provides no flexibility with regard to either reporting or the frequency of reporting. Since the usefulness of the information contained in these reports has not been determined, it is difficult to assess the merits of requiring that the licensees continue to provide these reports on a monthly basis. Whether the reports could be provided less frequently or could be totally eliminated should also be considered. The determination of the usefulness of the information provided should include an assessment of its need by other agencies, the industry, public interest groups, and the general public, in addition to the need of the NRC.

The task force formed to evaluate reporting requirements for power reactors is also evaluating the need for this requirement.

Based on the above considerations, it is concluded that although this and other specific reporting requirements are currently being evaluated, a broader approach that determines all the information needed by the NRC to accomplish its safety mission may be appropriate and result in a possible reduction of regulatory burden.

SUMMARY OF SEABROOK ASSESSMENT

Category: F **Items: All**

The Seabrook operating license contains ten items in Category F, "Unique Plant Features." These items were deemed appropriate to be assessed collectively. They identify the plant and its location and delineate the plant's major design features. Specifically, the Category F items are as follows:

OL 2.A	TS 5.1.1
TS 5.1.2	TS 5.1.3
TS 5.2.1	TS 5.2.2
TS 5.3.1	TS 5.3.2
TS 5.4.2	TS 5.5.1

These items are basically statements of facts. They generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the unique plant features items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: G **Items: All**

The Seabrook operating license contains 62 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include legal provisions, including exemptions, definitions, and statements of fact. Specifically, the Category G items are as follows:

OL 1.A	OL 1.B	OL 1.C	OL 1.D
OL 1.E	OL 1.F	OL 1.G	OL 1.H
OL 1.I	OL 2.B.1	OL 2.B.2	OL 2.D
TS 1.1	TS 1.2	TS 1.3	TS 1.4
TS 1.5	TS 1.6	TS 1.7	TS 1.8
TS 1.9	TS 1.10	TS 1.11	TS 1.12
TS 1.13	TS 1.14	TS 1.15	TS 1.16
TS 1.17.a	TS 1.17.b	TS 1.17.c	TS 1.18
TS 1.19	TS 1.20	TS 1.21	TS 1.22
TS 1.23	TS 1.24	TS 1.25	TS 1.26
TS 1.27	TS 1.28	TS 1.29	TS 1.30
TS 1.31.a	TS 1.31.b	TS 1.31.c	TS 1.32
TS 1.33	TS 1.34	TS 1.35	TS 1.36
TS 1.37.a	TS 1.37.b	TS 1.38	TS 1.39
TS 1.40	TS 1.41	TS 1.42	TS 1.43
EP 1.0	EP 4.2.2	EP 4.2.3	

These items generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

APPENDIX B

ASSESSMENT OF OPERATING LICENSE

SURRY UNIT 1

U. S. Nuclear Regulatory Commission

Regulatory Review Group

March 1993

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B.1 ASSESSMENT OF SURRY OPERATING LICENSE

B.1.1 Surry License

The Surry Unit 1 operating license was issued on May 25, 1972. The operating license consists of the license itself and the Technical Specifications, which are Appendix A to the license. The license as reviewed has been amended through Amendment 170, dated June 1, 1992.

B.1.2 Assessment of License

The Surry operating license contains 192 items. Each of the items was reviewed and assigned to one of the categories in Table 2 of Section 3.1. The numbers of items in the Surry operating license by category are shown in Table B.1.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on the items' regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 44 items or approximately 23 percent of the total number of items.

The number of items in the remaining categories that would be assessed individually was determined to be approximately 10 percent or 15 of the 148 remaining items. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or two of the 20 items in Category D would be selected for further assessment. With the 44 items that would be assessed collectively, this meant that 59 or approximately 31 percent of the 192 total items would be assessed either collectively or individually.

The items that were to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. All the items that were assessed are listed in Table B.2.

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table 3 of Section 3.1. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the Surry license, certain items that were assessed previously in the Seabrook license were compared to the corresponding items in the Surry license in order to validate the results of the Seabrook assessment. The items that were selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

B.1.3 Results of Assessment

The assessment summaries for each of the items are provided in the attachment to this appendix. The summaries are presented in the order of the categories into which each of the items was assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; then, in the Technical Specifications (TS).

The overall findings and recommendations are presented in Section B.2.

Table B.1

SURRY OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	93
B. Non-Technical License Conditions	1
C. License Conditions That Rely on Other Documents for Requirements	16
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	20
E. Reporting and Recordkeeping Requirements	19
F. Unique Plant Features	9
G. Other	34
Total	192

Table B.2

INDEX OF SURRY OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category A (9 of 93)**	
OL 3.K	Secondary water chemistry monitoring program	B-19
TS 3.2.D	Chemical and volume control system	B-21
TS 3.4.C	Spray systems	B-23
TS 3.5.B	Residual heat removal system	B-25
TS 3.8.A	Containment integrity and operating pressure	B-27
TS 3.16.B	Emergency power system	B-28
TS 3.19	Main control room bottled air system	B-29
TS 4.0.2	General surveillance requirement	B-30
TS 5.4.C	Fuel storage	B-31
	Category B (1 of 1)***	
OL 4	Effective date and expiration condition	B-32
	Category C (2 of 16)**	
OL 2.C	Nuclear materials condition	B-33
OL 3.I	Fire protection condition	B-34
	Category D (2 of 20)**	
TS 6.1.C.2	Management Safety Review Committee	B-36
TS 6.4.K	Systems integrity	B-38
	Category E (2 of 19)**	
TS 3.12.B.7	Power distribution limits	B-40
TS 6.3.A	Action to be taken if a safety limit is exceeded	B-42

Table B.2 (Continued)

INDEX OF SURRY OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category F (9 of 9)***	
OL 1	Applicability condition	B-44
TS 5.1	Site	B-44
TS 5.2.A	Containment structure	B-44
TS 5.2.B	Containment penetrations	B-44
TS 5.2.C	Containment systems	B-44
TS 5.3.A	Reactor core	B-44
TS 5.3.B	Reactor coolant system	B-44
TS 5.4.A	Fuel storage - structures	B-44
TS 5.4.D	Fuel storage - draining	B-44
	Category G (34 of 34)***	
OL a	Finding - construction completion	B-45
OL b	Finding - conformance with requirements	B-45
OL c	Finding - reasonable assurance	B-45
OL d	Finding - technical and financial qualification	B-45
OL e	Finding - financial protection	B-45
OL f	Finding - issuance of license	B-45
OL 2.A	Authorization - possess, use and operate	B-45
TS 1.0	Technical Specification definitions (26 items)	B-45
TS 4.0.1	General surveillance requirement	B-45

OL = Operating license condition
TS = Technical Specification

* Page number of assessment summary in the attachment to this appendix.

** Numbers in parentheses indicate the number of the total number of items in the category that were assessed.

*** Items that were assessed collectively; all others were assessed individually.

B.2 ASSESSMENT FINDINGS AND RECOMMENDATIONS

B.2.1 Introduction

The item assessment summaries were reviewed to determine which of the items appear to exceed the applicable regulatory requirements, given their safety significance and regulatory bases; which of the items should be considered for possible reduction in regulatory burden; which of the items provide at least some inherent flexibility, and why licensees may not be taking full advantage of that flexibility; and which of the items should be considered for enhanced flexibility. The items that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Item TS 3.2.D, chemical and volume control system, appears in three groups. The item appears to have the potential for possible reduction of regulatory burden; it appears to have enhanced flexibility potential; and it has already been or is being considered in another program.

B.2.2 Findings and Recommendations

B.2.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

Findings: None of the items assessed appears to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the Surry operating license. However, the restrictive provision of TS 3.2.D, chemical and volume control system, may exceed the regulatory intent. While the boration capability of the chemical and volume control system design is appropriately based in 10 CFR 50, the specification of rigid operability requirements on all components in the boration flow paths appears to have less foundation. Given that the definition of "operable" in TS 1.D includes support system functionality, the imposition of shutdown provisions for inoperable heat tracing circuits appears to be not only redundant but possibly unwarranted since the intended safety function can be fulfilled by other means.

In its assessment of the Seabrook license, the Review Group found seven items that appear to exceed the applicable regulatory requirements. To validate the Seabrook results, these seven items were also reviewed for the Surry license. Four of the Seabrook items--TS 3.1.2.7, isolation of unborated water sources; TS 6.2.2.a, minimum shift crew composition; TS 6.2.2.e, station staff working hours; and TS 6.8.1.5, monthly operating reports--appear to exceed the applicable regulatory requirements for Surry in the same manner as for Seabrook. Therefore, for these four items, the Seabrook findings also apply to Surry. The three remaining Seabrook items--TS 3.7.1.2, auxiliary feedwater system; TS 3.7.4, service water system; and TS 3.8.2.1, D.C. electrical power system--do not appear to exceed the applicable regulatory requirements for Surry.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license.

B.2.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

Findings: Five of the items assessed appear to have the potential for possible reduction of regulatory burden. They are as follows:

OL 3.I	Fire protection condition
TS 3.2.D	Chemical and volume control system
TS 3.8.A	Containment integrity and operating pressure
TS 3.12.B.7	Power distribution limits
TS 6.3.A	Action to be taken if a safety limit is exceeded

Although the Surry fire protection license condition, OL 3.I, is atypical, licenses generally contain fire protection conditions that require license amendments for changes to the fire protection plans that adversely affect the ability to achieve and maintain shutdown in the event of a fire. While the need to obtain NRC approval for such changes is evident, it is not clear why a license amendment is necessary. Consideration should be given to eliminating the practice of including fire protection plans and the provisions for making changes thereto as license conditions. In addition, consideration should be given to expanding the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan, and eliminating the inconsistencies in the change requirements for these plans.

Technical Specification 3.2.D represents a surrogate item for core reactivity control. Redundancies were identified in the handling of certain chemical and volume control system components in accordance with the additional requirements of the safety injection system. Such redundancy is not required by the Improved Standard Technical Specifications. Likewise for TS 6.3.A, a redundancy and conflict in the reporting details was identified between this item and the 10 CFR 50.72 and 50.73 reporting requirements. Not only to eliminate any dual reporting but also to provide consistency with 10 CFR 50.36 requirements for safety limit violations, the language of Technical Specification 6.3.A could be modified to reference the above regulations and organized in a more coordinated manner with the safety limit sections of the Technical Specifications.

Technical Specification 3.8.A retains a listing of containment isolation valves. Generic Letter 91-08 provides guidance for preparing a license amendment to remove component lists, such as containment isolation valves, from the Technical Specifications. Therefore, this list could be considered for possible removal.

Technical Specification 3.12.B.7, quadrant power tilt ratio, imposes reporting requirements without time limitations in addition to corrective actions. Further evaluation revealed that this reporting requirement does not appear in the Standard Technical Specifications. Therefore, this requirement could be considered for possible elimination. In its assessment of the Seabrook license, the Review Group found four items that appear to have the potential for possible reduction in regulatory burden. To validate the Seabrook results, these four items were also reviewed for the Surry license. Two of the Seabrook items--OL 2.E, physical security condition, and TS 6.8.1.5, monthly operating reports--should be considered for possible reduction in regulatory burden in the same manner as for Seabrook. Therefore, for these two items, the Seabrook findings also apply to Surry. The two remaining Seabrook items--TS 3.3.3.3, seismic instrumentation, and TS 3.3.3.4, meteorological instrumentation--do not apply to Surry.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license and, in addition, recommends the following:

- Reconsider the practice of including fire protection plans and the provisions for making changes thereto as license conditions.

- Expand the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan. Eliminate the inconsistencies in the change requirements for these plans.

B.2.2.3 Items That Provide Inherent Flexibility

Findings: Five of the items assessed were found to have at least some inherent flexibility. That an item has at least some inherent flexibility does not preclude it from consideration for enhanced flexibility or reduction in regulatory burden. The items with inherent flexibility are as follows:

OL 3.I	Fire protection condition
OL 3.K	Secondary water chemistry monitoring program
TS 3.5.B	Residual heat removal system
TS 4.0.2	General surveillance requirement
TS 6.4.K	Systems integrity

License Condition OL 3.I provides inherent flexibility in that except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is required for changes to the fire protection plan. However, it is noted that the Surry fire protection license condition is atypical in that licenses generally contain fire protection conditions that require license amendments for changes to the fire protection plans that adversely affect the ability to achieve and maintain shutdown in the event of a fire.

License Condition OL 3.K allows the tailoring of secondary water chemistry programmatic controls to site-specific details and industry guidelines. Major program changes can be handled by the 10 CFR 50.59 process.

Inherent flexibility is also found in TS 3.5.B, given both the allowable outage time and the conditional nature of the need for an operable RHR system depending upon the existing plant situation.

The requirement to perform surveillances at specified time intervals as contained in TS 4.0.2 provides flexibility by permitting some adjustment of these time intervals to accommodate normal test schedules consistent with the guidance contained in Generic Letter 89-14 and the Improved Standard Technical Specifications.

Technical Specification 6.4.K allows the licensee to establish its own program of compliance, coordinated with the ASME Code Section XI provisions, and is not constrained by prescriptive leakage criteria.

In its assessment of the Seabrook license, the Review Group found six items that have at least some inherent flexibility. To validate the Seabrook results, these six items were also reviewed for the Surry license. Five of the Seabrook items--OL 2.E, physical security condition; TS 3.4.10, structural integrity; TS 3.9.4, containment building penetrations; TS 6.2.2.a, minimum shift crew composition; and TS 6.2.2.e, station staff working hours--have at least some inherent flexibility for Surry in the same manner as for Seabrook. Therefore, for these five items, the Seabrook findings also apply to Surry. The remaining Seabrook item--TS 3.12.2, land use census--does not apply to Surry.

Recommendation: Based on the foregoing, the Review Group reaffirms its recommendation in this area from its assessment of the Seabrook license.

B.2.2.4 Items That Should Be Considered for Enhanced Flexibility

Findings: Four of the items assessed appear to have enhanced flexibility potential. They are as follows:

TS 3.2.D	Chemical and volume control system
TS 3.5.B	Residual heat removal system
TS 3.16.B	Emergency power system
TS 6.1.C.2	Management safety review committee

Technical Specification 3.2.D, as a surrogate for reactivity control, could be made more flexible with respect to required boration activities by addressing the shutdown margin aspect without prescribing conditions on the boration flow paths and components.

For Technical Specification 3.5.B, the insights provided by the application of risk assessment methodology to the evaluation of residual heat removal system operability and shutdown constraints would appear to be valuable in avoiding unnecessary plant transients.

Technical Specification 3.16.B, emergency power, also appears to be an area where the application of risk-based methodology could be used to evaluate the relative risk associated with continued plant operation for limited periods of time compared to shutting down and starting up.

Technical Specification 6.1.C.2, Management Safety Review Committee, appears to be overly prescriptive in that it requires the MSRC to provide the same level of consideration to required procedures and all proposed changes to station systems or equipment that affect nuclear safety. A more performance-based or graded approach that takes into account the relative safety significance of the different areas and items under review would provide additional flexibility. This is also an area in which the application of risk assessment methodology could be considered.

In its assessment of the Seabrook license, the Review Group found six items that have enhanced flexibility potential. To validate the Seabrook results, these six items were also reviewed for the Surry license. Five of the Seabrook items--OL 2.E, physical security condition; TS 3.0.3, general limiting condition for operation; TS 3.6.1.7, containment ventilation system; TS 6.4.1.7, SORC responsibilities; and TS 6.7.3, temporary changes of procedures--have enhanced flexibility potential for Surry in the same manner as for Seabrook. Therefore, for these five items, the Seabrook findings also apply to Surry. The remaining Seabrook item--TS 6.2.3.2, ISEG composition--does not apply to Surry.

Although the review of the Surry license validated the Seabrook results for five out of the six items for which enhanced flexibility potential was identified, significant differences in the age, organization, and functional structure of the Technical Specifications for these two plants were found. These differences could have a direct impact upon the potential for success of any enhanced flexibility initiatives. Similarly, for the four additional Surry items that were assessed in this group, the regulatory philosophy that underlies the Surry

Technical Specifications reflects an approach to component and system controls quite different from that of both the Improved and Standard Technical Specifications. Therefore, even though the Technical Specification Improvement Program provides a mechanism for facilitating increased flexibility and reduced plant operational restrictions, effecting a transition to the Improved Standard Technical Specifications would be more difficult for Surry than for Seabrook.

However, for individual line items, the Surry licensee may be able to justify some of the enhancement options provided by the Technical Specification Improvement Program. Attempts to gain further flexibility by means of significant revisions to the Technical Specifications would have to be weighed by the licensee against the constraints of the plant-specific design and system-based limitations that exist for Surry, as well as for other older plants.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license and, in addition, recommends the following:

- Consider making the Technical Specification Improvement Program available to individual licensees, in addition to lead and subsequent plant licensees, by allowing line-item improvements to be made on a plant-specific basis in addition to the present generic basis.

B.2.2.5 Items Considered or Being Considered in Other Programs

Findings: Four of the items assessed have already been or are being considered in other programs. They are as follows:

TS 3.2.D	Chemical and volume control system
TS 3.4.C	Spray systems
TS 3.8.A	Containment integrity and operating pressure
TS 6.3.A	Action to be taken if a safety limit is exceeded

Technical Specifications 3.2.D, 3.4.C, and 6.3.A have been replaced by better organizationally and technically structured items in the Improved Standard Technical Specifications. The noted revisions eliminate redundancy and establish a more coordinated functional relationship between the safety intent of each Technical Specification and the conditions and surveillances required to meet that intent.

The list of containment isolation valves contained in Technical Specification 3.8.A has already been considered and can be eliminated in accordance with the guidance contained in Generic Letter 91-08.

Recommendation: Based on the foregoing, the Review Group has no recommendations in this area.

B.2.2.6 Items for Which No Further Consideration Is Warranted

Findings: Fifty-two of the items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is also included here.

The items for which no further consideration is warranted are as follows:

OL 2.C	Nuclear materials condition
OL 3.K	Secondary water chemistry monitoring program
TS 3.4.C	Spray systems
TS 3.8.A	Containment integrity and operating pressure
TS 3.19	Main control room bottled air system
TS 4.0.2	General surveillance requirement
TS 5.4.C	Fuel storage
TS 6.4.K	Systems integrity
Cat. B item	Non-technical license conditions (1 item)
Cat. F items	Unique plant features (9 items)
Cat. G items	Other (34 items)

Recommendation: Based on the foregoing, the Review Group has no recommendations in this area.

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ATTACHMENT TO APPENDIX B - ITEM ASSESSMENT SUMMARIES

SUMMARY OF SURRY ASSESSMENT

Category: A   Item: OL 3.K

Surry License Condition 3.K, secondary water chemistry monitoring program, requires the licensee to implement a monitoring program of secondary side water to inhibit steam generator tube degradation. General provisions are listed in OL 3.K regarding what this monitoring program shall include. This item was selected for review because it represents a technical requirement incorporated into the operating license, rather than the Technical Specifications, by license amendment.

This item has regulatory basis in the General Design Criteria of Appendix A to 10 CFR 50, which address the integrity of the reactor coolant pressure boundary (e.g., steam generator tubes) and discuss design margins to limit leakage during all operating and accident conditions. The use of secondary water chemistry controls to prevent degradation of the steam generator tubes, representing the barrier between primary and secondary coolant, is a surrogate item. In turn, a secondary water chemistry monitoring program can be viewed as a surrogate for the maintenance of adequate chemistry parameters. In the mid-1970's, such secondary water chemistry parameters were incorporated directly into plant Technical Specifications as limiting conditions for operation with associated surveillance requirements. However, such restrictive requirements were considered to be hindrances to operational flexibility without the realization of commensurate benefits in limiting steam generator tube degradation. Hence, the development of plant-specific secondary water chemistry monitoring and control programs was judged to provide a more effective approach to the same overall goal. The inclusion of such surrogate programs as license conditions appears to have been technically sound as an alternative regulatory approach.

The goal of minimizing steam generator, including reactor coolant pressure boundary, degradation is clearly relevant to safety. Branch Technical Position MTEB 5-3 of the Standard Review Plan (NUREG-0800) provides guidance for secondary water chemistry monitoring and indicates that the NRC will review the individual monitoring program for each plant. It also recommends that steam generator vendor recommendations should be incorporated in the technical requirements of each individual program. This allows a licensee to tailor its programmatic controls to the site-specific design features and needs, while at the same time using industry, e.g., Electric Power Research Institute (EPRI), guidelines where appropriate. Therefore, this Surry license condition has a measure of inherent flexibility, not in the stipulation that a secondary water chemistry monitoring program is required, but rather in the licensee's own development of the implementation details.

Both the Standard and Improved Standard Technical Specifications have incorporated language similar to this Surry item into the appropriate administrative controls section. For Surry, this programmatic requirement could also be more appropriately delineated as an administrative control (similar to item TS 6.4.K) rather than an operating license condition.

It should be noted that individual plant secondary water chemistry control and monitoring programs written to comply with Westinghouse and EPRI guidelines require plant shutdowns if certain action level chemistry limits are exceeded. Such shutdown provisions are not NRC conditions or Technical Specification requirements, but licensee self-imposed controls to not only ensure safe operation, but also maximize overall long-term plant reliability. While major changes to the plant secondary water chemistry program would require processing in accordance with 10 CFR 50.59 provisions, licensee ownership of the implementation details of this operating license condition allows sufficient flexibility for licensees to adequately develop and manage the required programs.

Based on the above considerations, it is concluded that this operating license condition is both appropriate and not unduly restrictive. Although this condition might be more appropriately characterized as an administrative control of the Technical Specifications, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.2.D**

Surry Technical Specification 3.2.D, chemical and volume control system (CVCS), requires reactor shutdown if specific conditions related to system and component inoperability are not met within stated times. This item applies to one and two-unit operation and takes into consideration the availability of shared systems and common components. This Technical Specification was selected for review because it represents a system-oriented approach to delineating reactivity controls, somewhat different from the functional (i.e., boration) orientation of newer Technical Specifications.

This item has regulatory basis in the reactivity control and limits criteria of 10 CFR 50, Appendix A. It has clear safety relevance. Given the dual function of certain components (e.g., charging pumps) to perform both emergency core cooling and CVCS roles, a Technical Specification redundancy is recognized and addressed by the provisions of this item. While some inherent flexibility is afforded the licensee by the allowances for inoperable equipment, particularly where components from the opposite unit are available, the item is restrictive not only in its action statements, but also in its stipulated controls over equipment (e.g., heat tracing circuits) with a marginal relation to safety. However, such restrictiveness does not extend to the few direct surveillance requirements that were found to be associated with this Technical Specification.

While General Design Criterion 26 of 10 CFR 50, Appendix A, mandates the design of a reactivity control system with the characteristics of boration capability found in the CVCS design, the need for a Technical Specification to govern such a function is not so clearly based in the regulations. This item represents a surrogate for adequate core reactivity control. Although system-based Technical Specifications can be and are effectively used as surrogate items to properly control design functions, the restrictive provisions, in the case of Technical Specification 3.2.D, may go beyond regulatory intent. As a counterpoint, the Improved Standard Technical Specifications delineate requirements over core reactivity, shutdown margins, and other criticality constraints without the need for specifications governing the relevant CVCS boration flow path components. While boration is a required action for failure to meet the required shutdown margin of the Improved Standard Technical Specifications, controls over the boration flow path and components are not rigidly prescribed. Emergency core cooling system (ECCS) components are not addressed as such in the Surry Technical Specifications. Nevertheless, equipment like the charging pumps are redundantly addressed in both the CVCS and safety injection specifications. Such redundancy is not required by the Improved Standard Technical Specifications.

Given the Surry-specific design features for a two-unit site with common systems, components, and flow paths, the merits of adopting the enhanced flexibility of the Improved Standard Technical Specifications would require additional licensee review. It appears that the benefit of avoiding shutdown requirements if equipment such as heat tracing becomes inoperable may be worth the additional analysis and review effort. However, with the additional flexibility afforded licensees by Generic Letter 91-18 for ensuring the functional capability of a system or component, each licensee may see different advantages, as well as disadvantages, in an item-by-item comparison of the Improved Standard Technical Specifications to plant-specific license conditions.

For Technical Specification 3.2.D, it appears that the Surry licensee would gain not only enhanced flexibility but also a reduction in regulatory burden with the application of Improved Standard Technical Specifications to this item. The impact of such implementation upon the dual-unit system design features and also upon the other technically related areas (e.g., ECCS) warrants further review.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.4.C**

Surry Technical Specification 3.4.C, spray systems, requires that the containment pressure and temperature parameters be maintained within certain limits, given a high refueling water storage tank temperature, in order to maintain the functional capability of the containment spray system. Since spray capability for containment depressurization during accident conditions is dependent upon the containment pressure and temperature, this item relies upon additional conditions delineated in TS 3.8, containment. This item was selected for review because it is representative of safety system controls and also because of the referencing relationship between TS 3.4.C and TS 3.8.

This item has regulatory basis in the reactor containment criteria of 10 CFR 50, Appendix A, with related technical bases in 10 CFR 50, Appendix K, as the containment pressure affects emergency core cooling system performance and in 10 CFR 50.49 relative to the environmental qualification of affected electrical equipment. Safety relevance is established by the need to place the reactor in cold shutdown conditions if the maintenance of containment pressure and temperature within design limits cannot be ensured by the containment spray function. The referencing provisions of this item to TS 3.8 requirements, while technically restrictive, provide a nexus between the containment design limits and the containment spray system capabilities. It appears that such restrictiveness is necessary based upon Surry design basis considerations.

While some flexibility in complying with the provisions of this item is provided by the licensee's control of different parameters (e.g., lower service water temperatures to cool the containment atmosphere allow for higher containment air pressure), there is little inherent flexibility, in general, in this Technical Specification. The containment spray system, as an engineered safety feature, is required to cool and depressurize the Surry containment to subatmospheric pressure following a design basis accident. Given the restriction to maintain refueling water storage tank temperature below a specific temperature in order to enable design functionality of the containment spray system during accident conditions, the prescriptive language of this Technical Specification appears warranted.

While the Surry Technical Specifications could be improved in this area from an organizational standpoint by a better order of the technical provisions of TS 3.4, spray systems, and TS 3.8, containment, the existing requirements appear technically sound and appropriate. Similarly, although there is no surveillance requirement directly correlatable with TS 3.4.C, the spray system test provisions (TS 4.5) appear consistent with ASME

Code, Section XI, requirements that are prescribed by 10 CFR 50.55a(g). However, again from an organizational standpoint, separation of the containment spray surveillance specifications from the limiting conditions for operation does not appear to be the most effective means of communicating the requirements.

The Improved Standard Technical Specifications currently represent the best integrated presentation of limiting conditions, actions, and surveillance provisions with a discussion of the background, safety analysis, and bases for each requirement. Based on the above considerations, while TS 3.4.C may appear prescriptive, it is necessarily so for sound design basis and safety reasons. Therefore, it is concluded that consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive unless accomplished as part of a broader organizational restructuring of the Surry Technical Specifications.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.5.B

Surry Technical Specification 3.5.B, residual heat removal (RHR) system, requires reactor shutdown if specific component operability provisions are not met within a 14-day time period. This item also requires that immediate attention be directed to making repair of the inoperable equipment for the allowed outage time to be applicable. This Technical Specification was selected for review because it is typical of the Surry system-based specifications.

This item has regulatory basis in General Design Criterion (GDC) 34 with pertinence to GDC 35 of Appendix A to 10 CFR 50, as well as in 10 CFR 50.46. Since Surry is an older plant, the emergency core cooling system (ECCS) functions are addressed separately by the safety injection Technical Specifications. Thus, this item specifying RHR component operability requirements governs the functional control of the capability to bring the reactor coolant system (RCS) to cold shutdown conditions under normal shutdown conditions. While the safety relevance of RHR requirements is clearly established, the coherency in the handling of these provisions in the Surry Technical Specifications is not so evident. As an example, the distinction between the need for operable RHR systems is different when the RCS temperature is above 350°F and when the RCS temperature is below this value. However, while TS 3.1.A.1.d.2 requires single RCS loop or RHR loop operation at or below 350°F, Technical Specification 3.5.B fails to differentiate between the different requirements of the different modes of operation.

In effect, the Surry RHR specification allows the licensee to maintain the reactor in hot or intermediate shutdown conditions with RCS temperature greater than 350°F for an indefinite period of time with no apparent constraints on RHR system operability. While this provides flexibility and may in fact be prudent under certain conditions (i.e., if RHR is unavailable, entering conditions where RHR provides the only cooling connection to a heat sink is not advisable), the intent of Technical Specifications is not to cover all operational situations beyond design basis, as is the better defined role of 10 CFR 50.54(x). Additional inherent flexibility of this item derives from the length of the allowable outage time (i.e., 14 days) for an inoperable component rendering one RHR loop out of service.

Therefore, while detailed operability and shutdown provisions are delineated in this Technical Specification, the licensee has flexibility in both the timing and conditional nature of compliance. For the functional requirements of RHR systems in general, risk-based insights may provide the potential for extended allowable outage times to minimize the alternative risk associated with shutdown operations. However, such conclusions must be based upon a detailed, plant-specific Technical Specification analysis. Thus, while the noted inherent flexibility may have technical merit, such a conclusion can only be substantiated by the appropriate application of a risk-based methodology to evaluate this and any similar items.

The Standard and Improved Standard Technical Specification handle RHR requirements with a similar technical approach and both provide a more coherent focus to the cooling functions of the RHR system than is apparent in this Surry item. However, it is noted that Standard Technical Specifications are written to include RHR components in both ECCS and normal shutdown systems, which does not appear fully applicable to the Surry case. Whether the adoption of Improved Standard Technical Specification provisions is consistent with the Surry RHR system design or would be beneficial in providing additional flexibility to the licensee are questions that may merit further review by the licensee. This area is one where performance-based criteria and risk assessment methodology could provide valuable insights from both a safety perspective and an enhanced flexibility potential. While licensee efforts in these areas and their results may not lead to a reduction in regulatory burden, such initiatives may still be worthwhile in order to provide an organizational structure and coherency to the Surry Technical Specifications. Such an approach would also be more consistent with the current regulatory philosophy.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.8.A

Surry Technical Specification 3.8.A, containment integrity and operating pressure, specifies the conditions of the reactor and reactor coolant system under which containment integrity shall not be violated. In addition, this Technical Specification contains a list of containment isolation valves. This item was selected because of its potential for reduced regulatory burden.

The legal bases for the containment integrity requirement is contained in 10 CFR 50, Appendix A, General Design Criteria 16, 50, 51, 54, and 55, which specify the design criteria for the containment and piping systems that penetrate containment. Standard Review Plan Sections 6.2.1 through 6.2.7 provide the guidance related to this requirement.

The requirements in this Technical Specification are important to safety since they provide assurance that containment integrity will be maintained in the event of an accident. This Technical Specification is prescriptive and affords little flexibility with regard to the conditions under which containment integrity shall be maintained prior to startup and during reactor operation. The Surry Technical Specifications still retain a listing of the containment isolation valves that are contained in Table 3.8.1 for Unit 1. Generic Letter 91-08, "Removal of Component Lists from the Technical Specifications," provides guidance for preparing a license amendment to remove such component lists from the Technical Specifications. Removal of the list of containment isolation valves from the Technical Specifications would afford the licensee additional flexibility and does not alter the existing Technical Specification requirement for those components to which they apply. It is noted that, in response to Generic Letter 84-13, the Surry licensee has removed the list of shock suppressors (snubbers) from the Technical Specifications.

Considering that a mechanism for a reduction in regulatory burden is available to the licensee through Generic Letter 91-08, it is concluded that further consideration of this item would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.16.B

Surry Technical Specification 3.16.B, emergency power system, specifies the operational requirements with various offsite and emergency power sources and D.C. battery systems either unavailable or inoperable. This item was selected because it is representative of a Technical Specification limiting condition for operation.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A, General Design Criterion 17, Electrical Power Systems, which states that an onsite and offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. Standard Review Plan Sections 8.2, 8.3.1, and 8.3.2 provide the guidance related to satisfying this requirement.

This requirement is important to safety in that it specifies the minimum sources of onsite, offsite, and D.C. power that must be available to continue operation and the allowed outage times for this equipment during which continued plant operation is permissible.

The Technical Specification provides no inherent flexibility to the licensee since it is very prescriptive with regard to the actions to be taken in the event electrical power sources are not restored to operability within the allocated time period. However, it is noted that the allowed outage times for these electrical power sources is longer than those permitted by both the Standard and Improved Standard Technical Specifications. The surveillance requirements for the emergency diesels and batteries are also less prescriptive than for plants that use the Standard Technical Specifications.

This may be an area where risk assessment methodology could be applied to evaluate the relative risks associated with the outage times specified for various electrical power sources to determine if they are appropriate compared to the risk associated with shutting down and starting up.

Based on the above considerations, it is concluded that this is an area where enhanced flexibility may be possible through the application of risk-based methodology to compare the relative risks associated with the outage times specified relative to the risks associated with shutting down and starting up.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.19

Surry Technical Specification 3.19, main control room bottled air system, specifies that a bottled dry air bank shall be available to pressurize the main control room to a positive differential pressure. The Technical Specification also states that this capability shall be demonstrated by testing in accordance with Technical Specification Section 4.1. This item was selected because it is representative of a technical requirement.

The legal requirement for this Technical Specification is contained in 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, Control Room, which specifies that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions. TMI Action Plan Item III.D.3.4 (NUREG-0737), Control Room Habitability Requirements, provides clarification of guidance contained in GDC 19 and references Standard Review Plan Section 6.4 acceptance criteria and Regulatory Guides 1.78 and 1.95 guidance.

This requirement is relevant to safety in that it is necessary to ensure that control room operators will be adequately protected against the effects of accidental releases of toxic and radioactive gasses so the plant can continue to be safely controlled or shut down under a design bases accident condition. As a result, this requirement has very little flexibility with regard to shutdown if a minimum positive differential pressure in the control room cannot be achieved and maintained.

The only unique feature of this item is the reference to Technical Specification Section 4.1, which contains the testing requirements to demonstrate the capability to maintain a positive differential pressure in the control room. This is one of the few places in the Surry Technical Specifications where a surveillance requirement is cross-referenced to a limiting condition for operation. The limiting conditions for operation and their associated surveillance requirements are contiguous in both the Standard and Improved Standard Technical Specifications. The latter approach not only facilitates the use of the Technical Specifications but also clearly establishes the nexus between the limiting conditions for operation and their associated surveillance requirements.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 4.0.2

Surry Technical Specification 4.0.2, general surveillance requirement, states that the specified time intervals may be adjusted plus or minus 25 percent to accommodate normal test schedules. This item was selected as an example of an item that has inherent flexibility.

This item has regulatory bases in 10 CFR 50.55a and Appendix A to 10 CFR 50. This item is important to safety since it ensures that ASME Code Class 1, 2, and 3 components that are maintained in accordance with the inservice inspection and testing programs for the plant and in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, are tested or inspected within the time intervals specified by the code plus or minus 25 percent to accommodate normal test and outage schedules.

This item has inherent flexibility in that it provides a tolerance for extending the surveillance intervals, which is consistent with guidance contained in Generic Letter 89-14, "Line-Item Improvements in Technical Specifications--Removal of the 3.25 Limit on Extending Surveillance Intervals," and the Improved Technical Specification Program.

Based on the above considerations, it is concluded that, since this item has inherent flexibility consistent with current requirements, further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 5.4.C**

Surry Technical Specification 5.4.C, fuel storage, is a design feature that requires that, when there is spent fuel in the spent fuel pit, the pit shall be filled with borated water at a concentration of not less that 2,300 ppm to match the boron concentration in the reactor cavity and refueling canal during refueling operations. This item was chosen because it is representative of a number of design feature technical requirements.

The regulatory bases for this item are 10 CFR 50.36 and Appendix A to 10 CFR 50. This requirement is important to safety in that it ensures the prevention of criticality in the fuel storage areas and the reactor.

Although the item is prescriptive, it does not appear to be unduly restrictive in view of its safety significance. The item does not appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the item is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of the item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: B   Items: All

The Surry license contains one item in Category B, "Non-Technical License Conditions." The item, OL 4, effective date and expiration condition, specifies the effective and expiration dates of the license. It is required by 10 CFR 50.51.

The item is not directly related to safety. Although the item is prescriptive, it does not appear to be unduly restrictive. The item does not appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the item is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of the item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: C Item: OL 2.C

Surry License Condition 2.C, nuclear materials condition, authorizes the licensee to receive, possess, and use byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation, and radiation monitoring equipment calibration and as fission detectors.

The incorporation of this condition into the operating license is primarily a convenience for both the licensee and the NRC. Prior to the issuance of the operating license, this and similar nuclear materials authorizations were issued to the licensee in the form of separate nuclear materials licenses. These separate licenses were incorporated into the operating license upon its issuance. Authority to combine such licenses is provided by Section 161(h) of the Atomic Energy Act and 10 CFR 50.52.

The item is not directly related to safety. Although the item is prescriptive, it does not appear to be unduly restrictive. The item does not appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the item is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of the item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: C Item: OL 3.I

Surry License Condition 3.I, fire protection condition, requires the licensee to complete certain modifications identified in and in accordance with the schedule prescribed in the NRC's fire protection safety evaluation; to submit certain additional information identified in and in accordance with the schedule prescribed in the NRC's fire protection safety evaluation; and to implement the administrative controls identified in the NRC's fire protection safety evaluation and supplements thereto. This item was selected because it is representative of a number of license conditions that rely on other documents for requirements.

The regulatory bases for the fire protection plan are 10 CFR 50.48, Criterion 3 of Appendix A to 10 CFR 50, and Appendix R to 10 CFR 50. Other than the general authority provided by 10 CFR 50.50 to include in the license any conditions that the Commission deems appropriate and necessary, no explicit regulatory basis exists for the license condition. The item is important to safety in that it ensures that structures, systems, and components important to safety are designed and located to minimize the probability and effects of fires.

Except for completing the specified modifications and submitting the specified information, the license condition only requires the licensee to implement and maintain the administrative controls identified in the NRC's fire protection safety evaluation and supplements thereto. Therefore, except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is needed in order for the licensee to make changes to its fire protection plan. Corresponding license conditions for more recent licenses allow the licensees to make changes to their fire protection plans without prior NRC approval provided the changes would not adversely affect the ability to achieve and maintain shutdown in the event of a fire.

Compared to corresponding license conditions for more recent licenses, the Surry license condition is more flexible in that except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is required for changes to the fire protection plan. More recent licenses generally contain fire protection conditions that require license amendments for fire protection plan changes that adversely affect the ability to achieve and maintain shutdown in the event of a fire. While the need to obtain NRC approval for such changes is evident, it is not clear why the regulatory burden of a license amendment is necessary. Therefore, consideration should be given to eliminating the practice of including fire protection plans and the provisions for making changes thereto as license conditions. In addition, consideration should be given to expanding the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan, and eliminating the inconsistencies in the change requirements for these plans.

SUMMARY OF SURRY ASSESSMENT

Category: D **Item: TS 6.1.C.2**

Surry Technical Specification 6.1.C.2, Management Safety Review Committee (MSRC), requires the MSRC to provide independent review and audit of designated safety-related activities and report to and advise the Senior Vice President-Nuclear on its findings related to those activities. This administrative control implements a continuing monitoring activity that provides independent oversight of safety-related station activities. This item was selected as a representative review activity of a committee required by the Technical Specifications.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.40(b) as it relates to the licensee being technically qualified to engage in licensing activities. The guidance provided by ANSI Standard N18.7 (ANS 3.2), as endorsed by Regulatory Guide 1.33, conveys additional regulatory criteria for the required review activities of an independent review and audit organization. While the MSRC activities have safety relevance in providing oversight of plant operations, the details of exactly what MSRC is responsible to review, document, and report in writing have little basis in the regulations and relate more specifically to Standard Review Plan (NUREG-0800) provisions. The language in this item resembles the wording of the applicable section of the Standard Technical Specifications.

While a certain degree of inherent flexibility exists for the implementation of this item, there is no inherent flexibility in what functions this Technical Specification requires MSRC to accomplish. This prescriptiveness does not appear to be either consistent or commensurate with the intended safety impact because not all of the referenced functions carry the same safety significance. The recording of meeting minutes and reporting requirements also appear somewhat onerous.

It should be noted that the MSRC has only advisory authority in that it recommends and renders determinations; the Senior Vice President-Nuclear has the responsibility for the resolution of any disagreements on overall station operation. Thus the language in this item, which conveys the administrative control of the MSRC requirements, appears to be overly prescriptive and enhanced flexibility could be provided by the use of performance-based criteria or a graded approach to safety-significant review and audit activities. In addition, the use of risk assessment methodology could possibly provide valuable input into the prioritization of MSRC efforts and into the determination of where limited review and audit time could be most effectively directed.

While the safety intent of the MSRC as an independent review and audit authority is soundly based, achieving enhanced flexibility in the administrative control of the MSRC functions would be a worthwhile initiative. The Improved Standard Technical Specifications do not significantly alter the overall MSRC review and audit responsibilities directed by this item. Therefore, it is recommended that further review of this item be conducted to evaluate not only the need for the current prescriptive language of TS 6.1.C.2 but also the prospects for enhanced flexibility by supporting more of a graded safety approach to the MSRC functions.

SUMMARY OF SURRY ASSESSMENT

Category: D   Item: TS 6.4.K

Surry Technical Specification 6.4.K, systems integrity, requires the licensee to implement a program to reduce leakage from systems outside containment that may contain highly radioactive fluids during plant transient or accident conditions. Some general inspection and leak test requirements are specified. This item was selected for review because it is representative of a number of programmatic control requirements listed in the administrative controls section of the Surry Technical Specifications.

While this item has regulatory basis in the radiation dose limits of 10 CFR 100, the origin of this Technical Specification is more directly related to the TMI Action Plan (NUREGs 0660 and 0737) requirements addressing primary coolant sources and other highly radioactive systems outside containment (i.e., action item III.D.1.1). Additionally, since the primary coolant in PWR plants contains boric acid, the program requested in NRC Generic Letter 88-05 for the control of boric acid leakage has an indirect relation to the program mandated by this item. Any leakage reduction program established to comply with Technical Specification 6.4.K appears then to be safety relevant and to have a sound technical foundation and a coherent regulatory basis.

This item has inherent flexibility not only in generalizing the leakage criteria to be "as low as practical levels" but also in allowing the licensee to establish its own preventive maintenance, visual inspection, and integrated leakage test provisions. Further, a licensee may take credit for any inservice inspection (ISI) functional testing, accomplished in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, to fulfill the system integrity requirements of this Technical Specification. While the inspection/test provisions of any leakage reduction program may serve as surrogate to the system structural integrity assumed in the plant design, the use of such surrogate items is both appropriate and technically acceptable, since the functional capability of the containment structure would not preclude a potential leakage of radiation from the systems outside containment that are addressed by this Technical Specification.

Both the Improved and Standard Technical Specifications have language similar to this Surry item for the administrative control of leakage of primary coolant sources outside containment. While the establishment of such a leakage reduction program is delineated as a rigid requirement, the implementation details of the program can be reasonably and flexibly set by the licensee without direct NRC involvement in programmatic or procedural revisions. Based on the above considerations, it is concluded that this Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would be unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: E   Item: TS 3.12.B.7

Surry Technical Specification 3.12.B.7, power distribution limits, specifies what action must be taken if the power tilt ratio exceeds 2 percent for an additional 24 hours after corrective actions have been taken. This item was chosen because it is an example of corrective actions that include notification and reporting requirements.

Standard Review Plan Section 4.3, "Nuclear Design," states that there are no direct or explicit general design criteria for power densities and power distributions allowed during normal reactor operation. The Standard Review Plan also states that the acceptance criteria in the area of power distribution should satisfactorily demonstrate that reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor. This Technical Specification limits the time the allowable quadrant power tilt ratio exceeds 2 percent without the licensee taking corrective actions in the form of a reduction in nuclear overpower and differential overtemperature trip setpoints. The Technical Specification also requires various combinations of differential temperature trip setpoint reduction and notification and reporting requirements depending on whether the design hot channel factors for rated power have not been exceeded or have been exceeded and reactor power is greater than 10 percent or if the hot channel factors have not been determined.

This requirement is important to safety because it provides protection against exceeding fuel design limits by limiting the time the licensee can continue to operate with possible core power distribution asymmetries (while attempting to correct the problem) without the imposition of trip penalties. This aspect of the requirement is very prescriptive and affords no flexibility if the time restrictions are exceeded. However, this requirement is not as prescriptive for plants that use the Standard Technical Specifications. This Technical Specification also requires a special report evaluating the cause of the power tilt to be submitted to the NRC. However, no time limitation for submittal is specified. Similarly, if hot channel factors for rated power are exceeded at power levels greater than 10 percent or if hot channel factors have not been determined, the Technical Specification requires the licensee to notify the NRC. Again, no time limitation for notification is stated. Since this Technical Specification does not contain any time limitations related to reporting and notifications, the safety importance of this requirement appears to be questionable. A check of the Technical Specifications for Seabrook and several other newer plants that use the Standard Technical Specification format indicates that no NRC reporting or notification requirements related to quadrant power tilt ratio is required.

Based on the above considerations, it is concluded that this item warrants further consideration for possible reduction in regulatory burden related to reporting and notification requirements.

SUMMARY OF SURRY ASSESSMENT

Category: E   Item: TS 6.3.A

Surry Technical Specification 6.3.A, action to be taken if a safety limit is exceeded, requires placement of the plant in hot shutdown conditions and the conduct of certain reporting actions in the event that a safety limit is violated. The plant-specific safety limits for the reactor core and the reactor coolant system pressure are delineated in Technical Specifications 2.1 and 2.2. Violation of these safety limits results in operating conditions outside the design constraints of the plant. This item was selected for review because the safety limit action statement is listed as an administrative control and also because the reporting requirements appear to be, in part, redundant to other regulatory requirements.

In addition to 10 CFR 50.36, this item has its regulatory bases in several General Design Criteria of Appendix A to 10 CFR 50, as well as in 10 CFR 100. The safety relevance of the reactor shutdown action is established in 10 CFR 50.36(c)(1) as a general regulatory requirement, as is also the need to notify the NRC. However, the language in the Surry Technical Specification 6.3.A for reports to the NRC is less restrictive than 10 CFR 50.72 for immediate notification and more restrictive than 10 CFR 50.73 for the follow-up written notification. Further, the provision in 10 CFR 50.36(c)(1)--after a safety limit is exceeded, the NRC must authorize the resumption of reactor operation--is not included in this Surry item as it is in Westinghouse Standard Technical Specifications.

As discussed above, the wording of this Technical Specification appears inconsistent with the regulations, thereby resulting in the most restrictive requirement governing the actions to be taken. No inherent flexibility is apparent in this item. For the shutdown provision, such prescriptive language is necessary to comply with the regulations. However, the placement of the action statement in an administrative controls section, rather than with the safety limits section, of the Technical Specifications appears inappropriate. Also, the need to restore compliance with the safety limit that has been exceeded is not specifically addressed in conjunction with the shutdown action, as it is in the Improved Standard Technical Specifications.

For the reporting requirements, the Improved Standard Technical Specifications reference 10 CFR 50.72 and 50.73 requirements for NRC reports and stipulate other provisions for internal licensee reporting and reviews. While these requirements are just as prescriptive as the language in the Surry item, the Improved Standard Technical Specifications provide a more consistent and coherent approach to compliance with the regulations. Further, the Safety Limit Violation Report that is rigidly prescribed in the Surry

Technical Specification could be replaced by a Licensee Event Report, submitted in accordance with 10 CFR 50.73.

Based on the above discussion of the relationship established between the requirements of Technical Specification 6.3 A and 10 CFR 50 and also upon the significance of reactor operation within the bounds of the safety limits, further consideration of this item for enhanced flexibility would be unproductive. However, as a possible reduction in regulatory burden, the Improved Standard Technical Specifications that address safety limits provide a soundly based, clearer, and more consistent approach to the required actions if a safety limit is exceeded. Therefore, the licensee for Surry, as well as other licensees, might consider further evaluation of the benefits of adopting provisions similar to those in the Improved Standard Technical Specifications for handling safety limit violations.

SUMMARY OF SURRY ASSESSMENT

Category: F | Items: All

The Surry operating license contains nine items in Category F, "Unique Plant Features." These items were deemed appropriate to be assessed collectively. They identify the plant and its location, and delineate the plant's major design features. Specifically, the Category F items are as follows:

OL 1	TS 5.1
TS 5.2.A	TS 5.2.B
TS 5.2.C	TS 5.3.A
TS 5.3.B	TS 5.4.A
TS 5.4.D	

These items are basically statements of facts. They generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: G　　　　Items: All

The Surry operating license contains 34 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include license conditions and definitions found in the Technical Specifications. Specifically, the Category G items are as follows:

OL a	OL b	OL c	OL d
OL e	OL f	OL 2.A	TS 1.A
TS 1.B	TS 1.C	TS 1.D	TS 1.E.1
TS 1.E.2	TS 1.F	TS 1.G.1	TS 1.G.2
TS 1.G.3	TS 1.G.4	TS 1.H	TS 1.I
TS 1.J	TS 1.K	TS 1.L	TS 1.M
TS 1.N	TS 1.O	TS 1.P	TS 1.Q
TS 1.R	TS 1.S	TS 1.T	TS 1.U
TS 1.V	TS 4.0.1		

Except for the financial qualification part of License Condition OL d, the license conditions are legal findings that appear to be required by the Atomic Energy Act or the Commission's regulations. The financial qualification finding was required at the time the Surry operating license was issued but is no longer required. The remaining items are definitions found in the "Definitions" and other sections of the Technical Specifications. They are judged necessary for the uniform interpretation of the defined terms.

None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

APPENDIX C

ASSESSMENT OF OPERATING LICENSES

PERRY UNIT 1

PEACH BOTTOM UNIT 2

U. S. Nuclear Regulatory Commission

Regulatory Review Group

April :993

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C.1 ASSESSMENT OF PERRY AND PEACH BOTTOM OPERATING LICENSES

C.1.1 Perry and Peach Bottom Licenses

The Review Group's assessment of the Seabrook license resulted in seven recommendations. The Review Group's assessment of the Surry license largely validated its assessment of the Seabrook license and resulted in only three additional recommendations. Based on the results of the Seabrook and Surry assessments, and its knowledge of and experience with other licenses, the Review Group did not expect to find significant information in its reviews of the Perry and Peach Bottom licenses that would result in a substantial number of additional recommendations. Therefore, the Review Group assessed the Perry and Peach Bottom licenses together. The combined assessment was performed using the same methodology as that used previously for the individual plant assessments.

The Perry Unit 1 operating license was issued on November 13, 1986. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; the environmental protection plan, which is Appendix B to the license; and the antitrust conditions, which are Appendix C to the license. The license as reviewed has been amended through Amendment 43, dated May 28, 1992.

The Peach Bottom Unit 2 operating license was issued on December 14, 1973. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Technical Specifications, which are Appendix B to the license. The license as reviewed has been amended through Amendment 168, dated July 6, 1992.

C.1.2 Assessment of Licenses

The Perry and Peach Bottom operating licenses contain 329 and 275 items, respectively. Each of the items was reviewed and assigned to one of the categories in Table 2 of Section 3.1. The numbers of items in the Perry and Peach Bottom operating licenses by category are shown in Tables C.1A and C.1B, respectively.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on the items' regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 83 items or approximately 25 percent of the total number of items for the Perry license, and 93 items or approximately 34 percent of the total number of items for the Peach Bottom license.

The number of items in the remaining categories that would be assessed individually was determined to be approximately 5 percent or 12 of the 246 remaining items for Perry and 9 of the 182 remaining items for Peach Bottom. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 5 percent, or 7 of the 136 items in Category A would be selected for further assessment for the Perry license. With the items that would be assessed collectively, this meant that 95 or approximately 29 percent of the 329 total items would be assessed either collectively or individually for Perry and 102 or approximately 37 percent of the 275 total items would be assessed either collectively or individually for Peach Bottom.

The items that were to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. All the items that were assessed are listed in Table C.2A for Perry and Table C.2B for Peach Bottom.

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table 3 of Section 3.1. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the Perry and Peach Bottom licenses, certain items that were assessed previously in the Seabrook and Surry licenses were compared to the corresponding items in the Perry and Peach Bottom licenses in order to validate the results of the previous assessments. The items that were selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility and, (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

C.1.3 Results of Assessment

The assessment summaries for each of the items are provided in Attachment A for Perry and Attachment B for Peach Bottom. The summaries are presented in the order of the categories into which each of the items were assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; next, in the Technical Specifications (TS); and, finally, for Perry, in the environmental protection plan (EP) and in Appendix C antitrust conditions (ACs), and for Peach Bottom, in the Environmental Technical Specifications (ES).

The overall findings and recommendations are presented in Section C.2.

Table C.1A

PERRY OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	136
B. Non-Technical License Conditions	7
C. License Conditions That Rely on Other Documents for Requirements	23
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	48
E. Reporting and Recordkeeping Requirements	39
F. Unique Plant Features	11
G. Other	65
Total	329

Table C.1B

PEACH BOTTOM OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	75
B. Non-Technical License Conditions	2
C. License Conditions That Rely on Other Documents for Requirements	19
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	59
E. Reporting and Recordkeeping Requirements	29
F. Unique Plant Features	10
G. Other	81
Total	275

Table C.2A

INDEX OF PERRY OPERATING LICENSE ITEMS ASSESSED

Table C.2A (Continued)

INDEX OF PERRY OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category E (2 of 39)**	
TS 3.3.7.8	Loose parts detection system	C-48
TS 6.9.4	Special reports - fire protection program	C-49
	Category F (11 of 11)***	
OL 2.A	Applicability condition	C-50
TS 5.1.1	Exclusion area, unrestricted area, site boundary	C-50
TS 5.1.2	Low population zone	C-50
TS 5.2.1	Containment - configuration	C-50
TS 5.2.2	Containment - design temperature and pressure	C-50
TS 5.2.3	Containment - secondary containment	C-50
TS 5.3.1	Fuel assemblies	C-50
TS 5.3.2	Control rod assemblies	C-50
TS 5.4.1	RCS - design pressure and temperature	C-50
TS 5.4.2	RCS - volume	C-50
TS 5.5.1	Meteorological tower location	C-50
	Category G (65 of 65)***	
OL 1.A	Finding - application	C-51
OL 1.B	Finding - construction completion	C-51
OL 1.C	Finding - conformance with requirements	C-51
OL 1.D	Finding - reasonable assurance	C-51
OL 1.E	Finding - technical qualification	C-51
OL 1.F	Finding - financial protection	C-51
OL 1.G	Finding - issuance of license	C-51
OL 1.H	Finding - satisfaction of requirements	C-51
OL 1.I	Finding - nuclear material	C-51
OL 2.B.1	Authorization - possess, use and operate	C-51
OL 2.B.2	Authorization - possess	C-51
OL 2.D	Exemptions	C-51

Table C.2A (Continued)

INDEX OF PERRY OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
TS 1.0	Technical Specification definitions (52 items)	C-51
EP 1.0	Objectives	C-51

OL = Operating license condition
TS = Technical Specification
EP = Environmental protection plan condition
AC = Appendix C antitrust conditions

* Page number of assessment summary in Attachment A to this appendix.

** Numbers in parentheses indicate the number of the total number of items in the category that were assessed.

*** Items that were assessed collectively; all others were assessed individually.

Table C.2B

INDEX OF PEACH BOTTOM OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category A (4 of 75)**	
TS 3.6.D	Safety and relief valves	C-53
TS 3.6.F	Recirculation pumps	C-55
TS 3.7.E	Large primary containment purge/vent valves	C-57
TS 3.14.C	Fire detection	C-58
	Category B (2 of 2)***	
OL 3.d	NPDES permit change condition	C-59
OL 4.0	Effective date and expiration condition	C-59
	Category C (1 of 19)**	
TS 3.8.E	Radiological environmental monitoring	C-60
	Category D (3 of 59)**	
TS 6.2.3.1	ISEG - function	C-61
TS 6.5.3.1	Procedure review and approval	C-62
ES 7.1.1.B	Organization	C-63
	Category E (1 of 29)**	
TS 3.8.D	40 CFR 190	C-64
	Category F (10 of 10)***	
OL 2.A	Applicability condition	C-66
TS 5.1	Site features	C-66
TS 5.2.A	Reactor - fuel assemblies	C-66

Table C.2B (Continued)

INDEX OF PEACH BOTTOM OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
TS 5.2.B	Reactor - control rods	C-66
TS 5.3	Reactor vessel	C-66
TS 5.4.A	Primary containment	C-66
TS 5.4.B	Secondary containment	C-66
TS 5.4.C	Containment penetrations	C-66
TS 5.5	Fuel storage	C-66
TS 5.6	Seismic design	C-66
	Category G (81 of 31)***	
OL 1.A	Finding - application	C-67
OL 1.B	Finding - construction completion	C-67
OL 1.C	Finding - conformance with requirements	C-67
OL 1.D	Finding - reasonable assurance	C-67
OL 1.E	Finding - technical and financial qualification	C-67
OL 1.F	Finding - financial protection	C-67
OL 1.G	Finding - issuance of license	C-67
OL 1.H	Finding - satisfaction of requirements	C-67
OL 1.I	Finding - nuclear material	C-67
OL 2.B.1	Authorization - possess, use and operate	C-67
TS 1.0	Technical Specification definitions (56 items)	C-67
ES 1.0	Environmental Specification definitions (15 items)	C-67

OL = Operating license condition
TS = Technical Specification
ES = Environmental Technical Specification

* Page number of assessment summary in Attachment B to this appendix.

** Numbers in parentheses indicate the number of the total number of items in the category that were assessed.

*** Items that were assessed collectively; all others were assessed individually.

C.2 ASSESSMENT FINDINGS AND RECOMMENDATIONS

C.2.1 Introduction

The item assessment summaries were reviewed to determine which of the items appear to exceed the applicable regulatory requirements, given their safety significance and regulatory bases; which of the items should be considered for possible reduction in regulatory burden; which of the items provide at least some inherent flexibility, and why licensees may not be taking full advantage of that flexibility; and which of the items should be considered for enhanced flexibility. The items that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Peach Bottom Item TS 6.2.3.1, ISEG - function, appears in three groups. The item appears to exceed the applicable regulatory requirements, at least in the manner in which it is implemented in the Peach Bottom operating license; it appears to have the potential for possible reduction of regulatory burden for the licensee; and, because options are already available for the elimination of this item, no further consideration of the requirement by the NRC appears to be warranted.

C.2.2 Findings and Recommendations

C.2.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

Findings: Two of the items appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the Perry or Peach Bottom operating licenses. It is recognized that 10 CFR 50.50 authorizes the Commission to include in licenses such conditions as it deems appropriate. The Review Group was not able to review the entire body of underlying regulatory guidance for all these items. Therefore, although all the items appear to prescribe conditions or require actions that exceed applicable regulatory requirements, there may indeed be additional regulatory bases for their presence as license conditions.

The items that appear to exceed the applicable regulatory requirements are as follows:

Perry

None

Peach Bottom

TS 6.2.3.1	ISEG - function
ES 7.1.1.B	Organization

Peach Bottom Technical Specification 6.2.3.1 specifies the function of the Independent Safety Engineering Group (ISEG). An ISEG is not required for plants like Peach Bottom for which operating licenses were issued prior to the imposition of the Three Mile Island action plan requirements. However, it is the Review Group's understanding that the Peach Bottom licensee voluntarily chose to have an ISEG and incorporated this function by amendment into its Technical Specifications.

Peach Bottom Environmental Technical Specification 7.1.1.B appears to exceed the regulatory requirements of 10 CFR 50.36b, which delineate the scope of environmental conditions for an operating license. In prescribing reporting responsibilities and referencing a management organization chart, the intent to establish clear lines of authority and communication is distorted by unnecessary specificity, which also makes the item a regulatory burden.

In its assessment of the Seabrook and Surry licenses, the Review Group found seven items that appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the licenses. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Three of the Seabrook and Surry items--minimum shift crew composition, station staff working hours, and monthly operating reports--also appear to exceed the applicable regulatory requirements for both Perry and Peach Bottom. The four remaining Seabrook and Surry items--isolation of unborated water sources, auxiliary feedwater system, service water system, and D.C. electrical power system--do not apply to, or do not appear to exceed the applicable regulatory requirements for, either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.2.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

Findings: Eight of the items assessed appear to have the potential for possible reduction of regulatory burden. They are as follows:

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.3.7.8	Loose parts detection system

TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration

Peach Bottom

TS 3.8.E	Radiological environmental monitoring
TS 3.14.C	Fire detection
TS 6.2.3.1	ISEG - function
ES 7.1.1.B	Organization

Perry License Condition 2.C.9 contains technical requirements related to Transamerica Delaval, Inc. (TDI) diesel generators. The details pertaining to these requirements are provided in an attachment to the license. Since the license for the most recently licensed plant with TDI diesel generators does not contain these conditions, the regulatory burden could presumably be reduced if the licensee submitted an amendment request to remove this license condition in conjunction with an Updated Final Safety Analysis Report change that incorporates the technical requirements contained in the license condition.

Perry Technical Specification 3.3.7.8 imposes a reporting requirement as a surrogate for corrective action. However, further analysis revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.7.4 contains an augmented inservice inspection program that is similar to but more detailed than that described in Generic Letter 84-13. Further analysis, however, revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.11.1.1 prescribes surveillance requirements in accordance with specific criteria for analysis, frequency, and lower limits of detection set forth in a table that is incorporated in this item. Since changes to the tabular data require a Technical Specification revision, such regulatory burden could be reduced by including the specific provisions of this item in the licensee-controlled Offsite Dose Calculation Manual (ODCM). Relocation of such details to the ODCM may be implemented by licensees in accordance with the Technical Specification Improvement Program and associated NRC guidance.

Peach Bottom Technical Specification 3.8.E contains radiological environmental monitoring requirements, including deviations from sampling schedule, land use census, and analysis to be performed on radioactive materials. Generic Letter 89-01 permits a line-item Technical Specification improvement to be made to place the programmatic controls of the Radiological Effluent Technical Specifications (RETS) in the administrative controls section of the Technical Specifications and relocate the procedural details of the RETS, of which Technical Specification 3.14.E is a part, to the Offsite Dose Calculation Manual or the Process Control Program.

Peach Bottom Technical Specification 3.14.C contains the fire detection instrumentation requirements, including operability, surveillance, and reporting requirements. In accordance with Generic Letter 86-10 and the guidance contained in Generic Letter 88-12, all the requirements related to fire protection systems and fire brigade staffing, including those contained in Section 3.14.C, can be removed from the Technical Specifications and placed in a licensee-controlled technical requirements document.

Peach Bottom Technical Specification 6.2.3.1 specifies the function of the Independent Safety Engineering Group (ISEG). Under the Improved Standard Technical Specifications, the ISEG function may be performed as a staff function under the independent reviews and audits program in the Technical Specifications. Therefore, this item can be pursued by the licensee for possible line-item elimination.

Peach Bottom Environmental Technical Specification 7.1.1.B intends to establish the proper management line of authority for environmental matters. However, in specifying titles and referencing a management organization chart, this item unintentionally creates a burden and a need for amending the Environmental Technical Specifications as a result of any organizational changes affecting this area.

In its assessment of the Seabrook and Surry licenses, the Review Group found nine items that appear to have the potential for possible reduction in regulatory burden. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Seven of the Seabrook and Surry items--physical security condition, seismic instrumentation, meteorological instrumentation, monthly operating reports, fire protection condition, containment integrity and operating pressure, and action to be taken if a safety limit is exceeded--appear to have the potential for possible reduction in regulatory burden for Perry or Peach Bottom or both. The two remaining Seabrook and Surry items--chemical and volume control system and power distribution limits--do not apply to either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.2.2.3 Items That Provide Inherent Flexibility

Findings: Six of the items assessed appear to have at least some inherent flexibility. That an item has at least some inherent flexibility does not preclude it from consideration for enhanced flexibility or reduction in regulatory burden. The items with inherent flexibility are as follows:

Perry

TS 3.6.1.5	Containment structural integrity
TS 6.5.1.2	PORC composition
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.6.F	Recirculation pumps
TS 3.7.E	Large primary containment purge/vent valves
TS 6.5.3.1	Procedure review and approval

Perry Technical Specification 3.6.1.5 allows the licensee to establish the performance criteria against which the technical requirements are measured. It also provides some time for the conduct of repair activities before directing a plant shutdown that would otherwise be required when primary containment integrity is in doubt.

Perry Technical Specification 6.5.1.2 provides inherent flexibility with regard to the composition of the Plant Operations Review Committee (PORC) in that the chairman of this committee can appoint up to two alternates to participate as voting members at any one time. The number of persons required to perform the PORC function is determined by the licensee; however, any changes to this number requires a Technical Specification amendment.

Perry Technical Specification 6.5.3.1 provides for a technical review and control function that relieves the Plant Operations Review Committee of some of its review responsibilities. The shifting of these responsibilities to Technical Review and Control has provided inherent flexibility in the performance of this function.

Peach Bottom Technical Specification 3.6.F provides inherent flexibility by permitting continued plant operation with only one recirculation loop in service. Even with the adjustments required of certain safety limits, limiting safety system settings, and various scram setpoints, the additional flexibility inherent in avoiding a shutdown by allowing plant operation with a reduced power level and tighter setpoint controls is advantageous to the licensee.

Peach Bottom Technical Specification 3.7.E provides inherent flexibility by only limiting the cumulative time that a purge or vent flow path can exist during a calendar year; the licensee has unlimited flexibility within that constraint. However, the Technical

Specification is also prescriptive in this and other aspects.

Peach Bottom Technical Specification 6.5.3.1 contains a technical review and approval function that relieves the Plant Operations Review Committee of some of its responsibilities. The shifting of these responsibilities to the technical review and approval function provides inherent flexibility in the performance of this function.

In its assessment of the Seabrook and Surry licenses, the Review Group found 11 items that appear to have at least some inherent flexibility. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Nine of the Seabrook and Surry items--physical security condition, structural integrity, land use census, minimum shift crew composition, station staff working hours, fire protection condition, residual heat removal system, general surveillance requirement, and systems integrity--appear to have at least some inherent flexibility for Perry, Peach Bottom or both. The two remaining Seabrook and Surry items--containment building penetrations and secondary water chemistry monitoring program--do not apply to, or do not appear to have at least some inherent flexibility for, either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.2.2.4 Items That Should Be Considered for Enhanced Flexibility

Findings: Ten of the items assessed appear to have enhanced flexibility potential. They are as follows:

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.1.3.1	Control rod operability
TS 3.4.1.1	Recirculation loops
TS 3.6.2.1	Drywell integrity
TS 3.7.2	Control room emergency recirculation system
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.7.E	Large primary containment purge/vent valves
TS 3.8.D	40 CFR 190
TS 6.5.3.1	Procedure review and approval

Perry License Condition 2.C.9 contains technical requirements related to Transamerica Delaval, Inc. (TDI) diesel generators. In addition to a reduction in regulatory burden if the license condition was eliminated, as discussed in Section C.2.2.2 of this report, enhanced flexibility would also result since the 10 CFR 50.59 review process could be used to determine changes that can be made without prior NRC approval.

Perry Technical Specification 3.1.3.1 exhibits a potential for enhanced flexibility with certain implementation options available to the licensee. The lead-plant efforts of the LaSalle County Station could be followed in initiating a line-item revision to the provisions governing scram discharge volume component controls. For even greater flexibility, the Improved Standard Technical Specifications, which extend the enhanced flexibility to the control rod requirements as well, could be adopted.

Perry Technical Specification 3.4.1.1 could be made more flexible if the licensee opted to submit an amendment request including a plant-specific analysis justifying single recirculation loop operation. In addition, if the licensee adopted the Improved Standard Technical Specifications, a more consistent approach to the applicable safety limits, setpoints, and direct flow measurements would be provided.

Perry Technical Specification 3.6.2.1 may be unduly prescriptive in that it requires that the plant be shut down within a specified time when actions taken to restore drywell integrity within specified completion times have failed. However, it does not consider the risk of extending the completion times relative to that of shutting down the plant. This is an area where more performance-based methods could be used and where the application of risk assessment methodology could be considered.

Perry Technical Specification 3.7.2 appears to have the potential for enhanced flexibility based upon a plant-specific application of risk assessment methodology. The provisions for control room emergency recirculation system operability at any plant could be tailored to the unique system design and reliability of that plant.

Perry Technical Specification 3.11.1.1 and Peach Bottom Technical Specification 3.8.D, which both address areas covered by a Radiological Environmental Monitoring Program (REMP), can be enhanced by adopting the more flexible provisions of the Improved Standard Technical Specifications. Generic Letter 89-01 provides guidance on the relocation of the REMP to a licensee-controlled document (the Offsite Dose Calculation Manual), the procedural details of which are delineated as program requirements in the administrative controls section of the Technical Specifications. Thus, the handling of these Technical Specifications as administrative controls, which is also endorsed by the Technical Specification Improvement Program, provides a more flexible method of ensuring compliance with 10 CFR 50.36a.

Perry Technical Specification 6.5.3.1 relieves the Plant Operations Review Committee (PORC) of some of its review responsibilities. However, it appears that some of these responsibilities are duplicated between these two functions and that the interfaces between these organizations could be burdensome. The adoption of the staff functional approach, as permitted by the Improved Standard Technical Specifications, would result in enhanced flexibility.

Peach Bottom Technical Specification 3.7.E is prescriptive in that if specified conditions are not met, the penetration must be isolated within 4 hours or the reactor must be shut down. Further, the item requires that the inflatable seals for the large containment isolation valves be replaced at least once every third refueling outage. These are areas in which the use of risk assessment methodology could be considered, e.g., to compare the relative risks of extending the completion times and shutting down the reactor and, under the provisions of the maintenance rule, 10 CFR 50.65, determining a more flexible replacement frequency for the inflatable seals.

Peach Bottom Technical Specification 6.5.3.1, relieves the PORC of some of its review responsibilities. Although there appears to be an effective interface between both organizations, the adoption of a staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in a more flexible requirement.

In its assessment of the Seabrook and Surry licenses, the Review Group found 10 items that appear to have enhanced flexibility potential. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Nine of the Seabrook and Surry items--physical security condition, general limiting condition for operation, containment ventilation system, ISEG composition, SORC responsibilities, temporary changes of procedures, residual heat removal system, emergency power system, and management safety review committee--appear to have enhanced flexibility potential for Perry or Peach Bottom or both. The one remaining Seabrook and Surry item--chemical and volume control system--does not apply to either Perry or Peach Bottom.

An assessment of the findings documented in Section C.2.2.3 of this report reveals that, for many items that provide inherent flexibility, the specifications are less prescriptive and more performance based. Such an approach to the delineation of license requirements allows for the restrictive details to be governed by licensee-controlled documents and also provides an effective means for achieving additional flexibility. For example, Perry Technical Specification 3.6.1.5 allows the licensee to establish specific performance criteria against which the technical requirements for containment structural integrity are measured. Further, several licensee programs were found to have an ample degree of operational latitude, while at the same time providing adequate control over a wide range of technical and administrative areas. This is exemplified in both the Perry and Peach Bottom Technical Specifications TS 6.5.3.1 where certain Plant Operations Review Committee responsibilities have been shifted to a Technical Review and Approval function.

In reviewing the items to be considered for enhanced flexibility, both technical requirements and programmatic areas were identified to have the potential for greater flexibility under licensee control without sacrificing system or program functionality. Additionally, the specification of prescriptive provisions in licensee-controlled documents reduces the regulatory burden on both the licensee and the NRC by allowing changes to be made without having to amend the license. As an example, both the Perry and Peach Bottom Radiological Effluent Technical Specifications could be relocated to the administrative controls section of the Technical Specifications with the implementing details of the applicable programs moved to the Process Control Program or the Offsite Dose Calculation Manual, both licensee-controlled documents.

This approach, allowing licensee-controlled documents to provide the programmatic controls and implementing details for needed technical requirements, could be used to a much greater extent than is currently in practice. For many items, including those where the Perry and Peach Bottom license reviews validate previous assessment findings (e.g., physical security), performance-based guidance could replace prescriptive criteria. The use of such performance-based direction would help clarify the difference between requirements and their implementation details and thus enhance the flexibility of the latter without adversely affecting compliance with the former.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses and, in addition, recommends the following:

- Expand the use of performance-based direction to supplant prescriptive criteria in license conditions and Technical Specifications. In items exhibiting inherent flexibility, the functional requirement is distinguishable from the technical details needed to implement that requirement. As evidenced in the Technical Specification Improvement Program, licensee-controlled programs that govern such implementation details provide both flexibility and the requisite assurance of system functionality.

C.2.2.5 Items Considered or Being Considered in Other Programs

Findings: Eleven of the items assessed have already been or are being considered in other programs. They are as follows:

Perry

TS 3.1.3.1	Control rod operability
TS 3.3.7.8	Loose parts detection system
TS 3.4.1.1	Recirculation loops
TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.6.F	Recirculation pumps
TS 3.8.D	40 CFR 190
TS 3.8.E	Radiological environmentai monitoring
TS 3.14.C	Fire detection
TS 6.5.3.1	Procedure review and approval

Perry Technical Specifications 3.1.3.1 and 3.4.1.1 have more coherent, better organized, and more flexible counterparts in the Improved Standard Technical Specifications, while Perry Technical Specification 3.11.1.1 has been replaced as part of the Technical Specification Improvement Program with an administrative control provision that is more appropriate to the functional requirement.

Perry Technical Specification 3.3.7.8 imposes a reporting requirement as a surrogate for corrective action. Further analysis, however, revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.7.4 and Peach Bottom Technical Specification 3.8.E, in accordance with the provisions of Generic Letters 84-13 and 89-01, respectively, are not included in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry and Peach Bottom Technical Specifications 6.5.3.1, which are similar, could be integrated into a staff functional approach in accordance with the Improved Standard Technical Specifications. This approach simplifies and provides additional flexibility to the entire review and audit function.

Peach Bottom Technical Specification 3.6.F, which is similar to Perry Technical Specification 3.4.1.1, and Peach Bottom Technical Specification 3.8.D, which is similar to Perry Technical Specification 3.11.1.1, have both been replaced with better organized and more appropriate functional items in the Improved Standard Technical Specifications.

Peach Bottom Technical Specification 3.14.C, in accordance with the provisions of Generic Letters 86-10 and 88-12, is not included in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

C.2.2.6 Items for Which No Further Consideration Is Warranted

Findings: One-hundred and ninety-four of the items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is also included here.

The items for which no further consideration is warranted are as follows:

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.1.3.1	Control rod operability
TS 3.3.7.8	Loose parts detection system
TS 3.4.1.1	Recirculation loops
TS 3.6.1.5	Containment structural integrity
TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.1.2	PORC composition
TS 6.5.3.1	Technical review and control
TS 6.9.4	Special reports - fire protection program
Cat B items	Non-technical license conditions (7 items)
Cat F items	Unique plant features (11 items)
Cat G items	Other (65 items)

Peach Bottom

TS 3.6.D	Safety and relief valves
TS 3.6.F	Recirculation pumps
TS 3.8.D	40 CFR 190
TS 3.8.E	Radiological environmental monitoring
TS 3.14.C	Fire detection
TS 6.2.3.1	ISEG - function
TS 6.5.3.1	Procedure review and approval

ES 7.1.1.B	Organization
Cat B items	Non-technical license conditions (2 items)
Cat F items	Unique plant features (10 items)
Cat G items	Other (81 items)

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

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- ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

- Atomic Energy Act of 1954, as amended.

- Code of Federal Regulations, Title 10 - Energy.

- Code of Federal Regulations, 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

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- Facility Operating License No. DPR-44, Peach Bottom Atomic Power Station, Unit No. 2, issued December 14, 1973, through Amendment 168, July 6, 1992.

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- NUREG-1150, "Reactor Risk Reference Document," Volume 1, (Draft) February

1987; and "Severe Accident Risks: Assessment for Five U.S. Nuclear Power Plants - Final Summary Report," December 1990.

- NUREG-1216, "Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc." August 1986.

- NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," Generic Letter 89-01, Supplement No. 1, April 1991.

- NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," September 1992, (referred to in the Assessment Report as "Improved Standard Technical Specifications" for Peach Bottom).

- NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6," September 1992, (referred to in the Assessment Report as "Improved Standard Technical Specifications" for Perry).

- USNRC Generic Letter 84-13, "Technical Specifications for Snubbers," May 3, 1984.

- USNRC Generic Letter 86-10, "Implementation of Fire Protection Requirement," April 24, 1986.

- USNRC Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," August 2, 1988.

- USNRC Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program," January 31, 1989.

- USNRC Letter, dated January 15, 1993, issuing Amendment No. 89 to Facility Operating License No. NPF-11 and Amendment No. 74 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively, with safety evaluation enclosed.

- USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for the Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air

Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, March 1978

- USNRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.
- USNRC Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Nuclear Reactors," May 1981.
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- USNRC Regulatory Review Group, "Draft Assessment of Operating License, Surry Power Station, Unit No. 1," March 1993.
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ATTACHMENT A TO APPENDIX C - PERRY ITEM ASSESSMENT SUMMARIES

SUMMARY OF PERRY ASSESSMENT

Category: A Item: OL 2.C.9

Perry License Condition 2.C.9, Transamerica Delaval, Inc. (TDI) diesel generator reliability, references Attachment 2 to the license, which requires the licensee to perform crankshaft and cylinder block inspections and air roll tests on the diesels. In addition, it is stated that changes to the maintenance and surveillance program approved by the staff are subject to the provisions of 10 CFR 50.59 and, if cracks in the crankshaft are found, this condition should be reported to the NRC. This item was selected because it not only represents a technical requirement incorporated into the operating license, rather than the Technical Specifications, but also contains the details of the requirement in a referenced attachment to the license. Another attachment to the license, related to the detailed control room design review imposes similar types of requirements.

This item has a regulatory basis in General Design Criterion 17 (GDC-17) of Appendix A to 10 CFR 50, which addresses electrical power systems. Concerns regarding the reliability of TDI diesels were first prompted by a crankshaft failure at Shoreham Nuclear Power Station in August 1983. In response to the problem, nuclear utility owners formed the TDI Diesel Generator Owners Group, which developed recommendations related to replacements, modifications, inspections, testing, and maintenance and surveillances. The NRC staff's evaluation, which is contained in NUREG-1216,"Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc.," concluded that implementation of the Owners Group's recommendations, plus additional actions identified in NUREG-1216, established the adequacy of the TDI diesels for nuclear service as required by GDC-17. The recommendations in NUREG-1216 are contained in Attachment 2 to the operating license.

This license condition is important to safety since it ensures the operability of the diesels manufactured by a specific company. However, the need for the incorporation of requirements related to this issue as a license condition and the inclusion of prescriptive technical requirements as an attachment to the license condition appear unnecessary. The Comanche Peak license, which is for the plant most recently licensed with TDI diesels, neither includes this as a license condition nor incorporates it into the Technical Specifications. Presumably, this license condition could be removed if the licensee submitted an amendment request to remove it in conjunction with an Updated Final Safety Analysis Report change that incorporates the requirements contained in the license condition. The regulatory burden would be reduced since a license amendment would no longer be required to change the testing and inspection requirements related to the TDI diesels that were imposed by the NRC. In addition, this would also provide enhanced flexibility since the licensee could use the 10 CFR 50.59 review process to determine changes that could be made without prior NRC approval.

Based on the above considerations, it is concluded that further consideration of this issue would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A   Item: TS 3.1.3.1

Perry Technical Specification 3.1.3.1, control rod operability, requires that all control rods be operable and specifies actions for various conditions of untrippable or otherwise inoperable control rods and scram discharge volume (SDV) valves. Surveillance requirements for scram discharge volume components as well as control rods are delineated. This item was selected for review because it is representative of reactivity control provisions and it references the surveillance requirements of other Technical Specifications for operability determinations.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory bases in Appendix A to 10 CFR 50, as well as in 10 CFR 100. It has clear safety relevance to the reactivity controls required to maintain the reactor with an acceptable shutdown margin and in accordance with minimum critical power ratio limits. Adequate controls over the SDV components also ensure that offsite radiation doses are limited to the levels allowed by the regulations. This item is consistent with other General Electric BWR/6 plant provisions in specifying both control rod and SDV component requirements in the same Technical Specification. The Improved Standard Technical Specifications, however, address control rod and SDV component provisions separately. A coherent application of control rod operability is established in the Perry Technical Specifications by the recognition that this item applies when control rods are declared inoperable as a result of other action items. However, the surveillance requirement of Technical Specification 3.1.3.1, which references the surveillances of four other action items, appears to be redundant and somewhat confusing in that the acceptance criteria of other surveillance tests would be more appropriately applied through their own action item requirements alone.

This item has limited inherent flexibility in the varying level of actions required, dependent upon the number of control rods inoperable and the various causes of inoperability. Additionally, the restriction that not more than eight control rods may be inoperable at power provides some operational latitude while at the same time considers the possibility that a generic problem may exist that requires reactor shutdown for resolution. This provision was retained in the Improved Standard Technical Specifications. However, the Improved Standard Technical Specifications provide other forms of enhanced flexibility in this area, not only in increasing the allowable outage times for inoperable control rods and SDV valves, but also in allowing separate action item entry conditions to apply for each control rod or SDV vent and drain line. This increase in flexibility has already been incorporated, in part, as a Technical Specification amendment to the SDV provisions of the control rod operability requirements (i.e., also TS 3.1.3.1) for the LaSalle Units 1 and 2. Thus, the enhanced flexibility potential for this item at Perry and other similar plants is good, not only because of the lead-plant efforts of LaSalle in the area of a line-item improvement to the SDV limiting conditions for operation, but also because the Improved Standard Technical Specifications offer additional flexibility in the control rod limiting conditions for operation as well.

While the NRC safety evaluation for the LaSalle operating license amendments, NUREG-0803, was issued based on a site-specific analysis and request, the staff review considered elements and criteria of a generic nature related to the SDV system piping. This safety evaluation also documented the consistency between the LaSalle request and the Improved Standard Technical Specifications in the handling of SDV controls. Thus, while extension of such flexibility to control rod operability, in general, requires additional plant-specific analysis for Perry, LaSalle, and other similar plants, the concept of implementing line-item improvements based on the presentation of valid analysis has been demonstrated to be both practical and workable.

As discussed above, while Technical Specification 3.1.3.1 has potential for enhanced flexibility, such improvement has already been considered by the NRC and can be pursued on a plant-specific basis through adoption of the appropriate Improved Standard Technical Specifications. Further review of this item for a reduction of regulatory burden is not deemed to be worth the effort. Individual licensees must determine if Technical Specification revisions at their plants are warranted to take advantage of the enhanced flexibility options available in this area. Therefore, additional consideration of this item by the NRC would be unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.4.1.1

Perry Technical Specification 3.4.1.1, recirculation loops, requires that both reactor coolant system recirculation loops be in operation with added conditions on total core flow and its relationship to thermal power. The action statements require a reactor shutdown if both recirculation loops are not in operation. This item was selected for review because single recirculation loop operation is not permitted for Perry as it is for certain other boiling water reactors.

This Technical Specification has regulatory basis in Appendix A to 10 CFR 50, relative to the maintenance of core operating conditions within fuel design limits and is also related to the fuel cladding integrity criteria delineated in 10 CFR 46 and Appendix K to 10 CFR 50. Such regulatory bases clearly establish the safety relevance of this item. However, even though a special test exception (i.e., TS 3.10.4) allows suspension of the two-loop, matched flow requirements during low-power physics testing, this item is restrictive in both its stipulation of two-loop operation and its constraints on thermal power relative to core flow conditions.

Increased flexibility is available to the licensee by performing a plant-specific analysis that would justify the adequacy of emergency core cooling system (ECCS) performance during single recirculation loop operation. As documented in the Technical Specification bases for this item, operation with one reactor coolant recirculation loop inoperable is prohibited until such an ECCS analysis is performed, evaluated, and determined to be acceptable. Operation with only one recirculation loop in service is authorized at several BWR plants (e.g., Peach Bottom, reference item TS 3.6.F in Attachment B to this report). A licensee can pursue such an option if technically justifiable based upon plant ECCS design and analysis. Thus, although Perry Technical Specification 3.4.1.1 has little inherent flexibility, the licensee has the ability to add flexibility with a technically justified submittal to the NRC for a line-item amendment.

Additionally, the Improved Standard Technical Specifications not only allow single recirculation loop operation if supported by accident analysis, but also increase the allowable outage time for loop flow mismatch beyond that specified in the Perry Technical Specifications. The coherency of the Improved Standard Technical Specifications is evidenced for this item by the focus of the surveillance requirements on flow mismatch rather than the neutron flux noise levels of the Perry Technical Specification. While this allows for a more consistent application of the safety limits and setpoints of the Core Operating Limits Report as it relates to flow requirements, further evaluation of an individual plant's stability region on the core power/flow map would be required prior to the licensee's pursuing adoption of the Improved Standard Technical Specifications. Furthermore, while flow mismatch (i.e., Perry TS 3.4.1.3) is incorporated into this item in the Improved Standard Technical Specifications, the provisions governing the recirculation loop flow control valves have been relocated from this item to a separate Technical Specification.

While enhanced flexibility potential exists for this item, a licensee's pursuit of the above-noted flexibility would probably not result in a reduction in regulatory burden. On a plant-specific basis, each licensee must determine if exercising the available options is worth the effort. It is, therefore, concluded that further consideration of this item would be unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.6.1.5

Perry Technical Specification 3.6.1.5, containment structural integrity, requires that the structural integrity of the containment be maintained during the operational conditions in which primary containment operability is prescribed as applicable, i.e., power operation, startup, and hot shutdown. The action statement dictates placing the reactor in cold shutdown conditions if the containment is found in nonconformance with structural integrity criteria. This item was selected for review because both the limiting condition for operation (LCO) and the surveillance requirements, which include written reports, appear to be redundant to other regulatory requirements.

In addition to the discussion of LCOs and reporting requirements in 10 CFR 50.36, this item has regulatory bases in the reactor containment criteria of Appendix A to 10 CFR 50 and in the containment inspection provisions of Appendix J to 10 CFR 50. The structural integrity of the containment has an important safety significance--not only from a design basis perspective but also with respect to 10 CFR 100 requirements. While this item provides no flexibility in its stipulated compliance provision, some inherent flexibility is evident in allowing the licensee to establish the criteria that determine compliance. Since the required visual inspections are intended to verify no apparent changes in appearance or other abnormal degradation of the containment surfaces and annulus fill concrete, the licensee is permitted to develop its own inspection program and acceptance criteria to fulfill this surveillance requirement.

The stipulation that the inspection be performed prior to the Type A containment leakage rate test not only reiterates the regulatory requirement of Appendix J to 10 CFR 50, but also points out somewhat of an inconsistency in this Perry item. With the surveillance inspections accomplished with the plant already in cold shutdown conditions, directing shutdown actions for failure to conform to the acceptance criteria appears to be a moot requirement. However, it is possible that some structural integrity problems could be identified in the containment during power operations or hot shutdown conditions. In this case then, the licensee is allowed 24 hours for repair before initiating a shutdown that would be otherwise required when primary containment integrity is in doubt.

By comparison, the Improved Standard Technical Specifications allow only a 1 hour completion time for structural integrity repairs but extend the time allotted to achieve cold shutdown conditions. This difference in the allowable outage times might reflect the consideration of repair-time impact upon the various containment designs (e.g., free-standing steel versus prestressed concrete containments). In any case, Perry Technical

Specification 3.6.1.5 appears more flexible than its Improved Standard Technical Specification counterpart. While one of the surveillance provisions of this item specifies a written reporting requirement that is redundant to 10 CFR 50.73, elimination of this surveillance/reporting requirement, although appropriate, will not result in a reduction in regulatory burden. Furthermore, from a practical standpoint, the discovery of problems during surveillance inspections would only prevent a plant startup, not dictate a shutdown, given the plant is already shut down. Such redundancy and inconsistency might justify the elimination of this item as a Technical Specification, but the allowance for a 24-hour repair period for degraded containment structural conditions identified at power provides the licensee additional flexibility that would not be otherwise available.

Therefore, this item is not a candidate for elimination or enhanced flexibility by way of the Technical Specification Improvement Program. The existing requirements already provide some inherent flexibility for regulatory compliance. While the inconsistent action times between the Improved Standard Technical Specifications and Perry TS 3.6.1.5 may warrant additional technical review, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.6.2.1

Perry Technical Specification 3.6.2.1, drywell integrity, requires that the licensee restore drywell integrity within 1 hour or be in hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours (37 hours total) if drywell integrity is lost. This item was chosen because of its potential for enhanced flexibility.

This item has its regulatory basis in the General Design Criteria of 10 CFR 50, Appendix A, Section V, Reactor Containment. This section addresses the integrity of the reactor containment, including the drywell, and the methods of ensuring containment integrity. Drywell integrity is demonstrated by verification of closure of drywell penetrations that do not have automatic isolation and verification that drywell airlock, suppression pool and drywell bypass leakage are all in compliance with corresponding Technical Specification requirements.

This requirement is important to safety to ensure that in the event of an accident, 10 CFR 100 limits are not exceeded. No inherent flexibility is provided by this Technical Specification. However, it is not clear that the Technical Specification could not be made more flexible. Since not all limiting conditions for operation have the same safety significance, the completion times allowed for achieving hot standby, hot shutdown, and cold shutdown could possibly be made more performance oriented, e.g., by considering situation-specific factors. Further, it may not always be safer to change operational modes. For example, if there is reasonable assurance that the situation could be rectified within 1 hour after the completion time for changing modes expires, it might be safer to maintain the reactor in its present mode for that additional period of time than to change modes.

Based on the above considerations, it is concluded that the Technical Specification is appropriate; however, it may be unduly prescriptive. Therefore, it is recommended that further consideration be given to this item for possible enhanced flexibility. This might be an area where risk assessment methodology could be applied to compare the relative risks of extending the completion times and shutting down the plant.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.7.2

Perry Technical Specification 3.7.2, control room emergency recirculation system, requires that two independent control room emergency recirculation subsystems be operable. These provisions ensure that the control room will remain habitable for operations personnel during and following all design basis accident conditions. This item was selected for review because it is representative of a two-train plant safety system for which the application of a risk assessment methodology may be beneficial.

This Technical Specification has a clear regulatory basis in General Design Criterion 19 of Appendix A to 10 CFR 50, which specifies a maximum whole body dose of 5 rem to control room personnel for the duration of an accident. The safety relevance of this item is further established in Regulatory Guide 1.52, which sets forth criteria for the reliable performance of such engineered-safety-feature atmosphere cleanup systems. While less restrictive action requirements are stipulated in the cold shutdown and refueling modes, the action items and surveillance requirements direct generally prescriptive provisions. Particularly in its detail and reference to the position statements of Regulatory Guide 1.52, little inherent flexibility can be found in the scope of the surveillance tests and their quantitative acceptance criteria.

The Improved Standard Technical Specifications provide some increased flexibility, primarily in cold shutdown and refueling conditions where system operability is only required during specific plant activities. Additionally, the surveillance requirements of the Improved Standard Technical Specifications, while also endorsing Regulatory Guide 1.52 positions, appear to be generally less restrictive. However, the shutdown requirements mandated in both Technical Specifications are similarly restrictive under hot operational conditions.

This item appears to have the potential for enhanced flexibility based on a plant-specific application of risk assessment methodology. Different plant control room design features, including proximity of the control room and air intakes to the reactor, have a varying impact upon both the need to enter the recirculation mode of operation and the length of time until shutdown, as dictated by component inoperability. System design features and maintenance practices can contribute significantly to the control room emergency recirculation system reliability. Thus, it is reasonable to expect that the provisions governing the control room emergency recirculation system operability requirements at a plant, if based upon performance and analyzed with a risk perspective, could be tailored to the individual plant, just as the environmental conditions of a design basis accident are calculated and analyzed uniquely for each plant. The current language of this item, as well as that of the Improved Standard Technical Specifications, appears to adopt regulatory guidance and conservative safety margins appropriately. However, the specification of such standard requirements may be unnecessarily restrictive to plants whose design and maintenance practices reduce the exposure risk to control room personnel in other ways.

Based on the above considerations, it is concluded that this Technical Specification is appropriate; however, it may be unduly restrictive based on plant-specific considerations. It is recommended that further consideration be given this item for enhanced flexibility, particularly where risk assessment methodology could be used by a licensee to demonstrate that the accident dose in the control room would remain acceptably low with less prescriptive requirements.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.7.4

Perry Technical Specification 3.7.4, snubbers, requires that all snubbers be operable; that any inoperable snubbers be restored to operable status within 72 hours and an engineering evaluation be performed to determine the cause of the failure; or that the attached system be declared inoperable and the applicable action statement be followed. This item was selected for review because of its potential for reduction in regulatory burden.

The regulatory bases for this item are 10 CFR 50.55a and General Design Criteria 1, 2, 4, 14, and 15 of Appendix A to 10 CFR 50. Regulatory guidance for this item is provided by Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures," and Generic Letter 84-13, "Technical Specifications for Snubbers."

The item is relevant to safety in that the operability of snubbers is necessary to ensure that the structural integrity of the reactor coolant system and other safety systems is maintained during and following a seismic event or other events that initiate dynamic loads.

The Perry snubber Technical Specification was based on similar requirements for previously licensed BWR/6 plants. In accordance with the provisions of Generic Letter 84-13, the list of snubbers is not included in the Technical Specification. However, the associated surveillance requirement contains an augmented inservice inspection program that is similar to but more detailed than that described in Generic Letter 84-13.

The Improved Standard Technical Specifications do not contain detailed snubber Technical Specifications. Only the requirement to include the snubbers in the inservice testing program for ASME Code Class 1, 2, and 3 components remains. The list of snubbers and the details of the associated inservice inspection program have been relegated to licensee-controlled documentation.

Based on the above considerations, it is concluded that possible reduction in regulatory burden can be achieved by the licensee by pursuing the line-item elimination of this item under the Technical Specification Improvement Program. In addition, it is concluded that no further review of this item is warranted.

SUMMARY OF PERRY ASSESSMENT

Category: B   Items: All

The Perry operating license contains seven items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. They deal with sale and leaseback, antitrust, emergency planning, financial protection, and the effective and expiration dates of the license. Specifically, the Category B items are as follows:

OL 2.B.7.a	OL 2.C.3.a
OL 2.C.3.b	OL 2.C.8
OL 2.G	OL 2.H
AC (all)	

The sale and leaseback, antitrust, and emergency planning conditions are not required by either the Atomic Energy Act or the Commission's regulations but are authorized by 10 CFR 50.50, which provides that the license may contain such conditions as the Commission deems appropriate. The financial protection license condition is based on Section 170 of the Atomic Energy Act and 10 CFR 140. The effective and expiration dates license condition is required by Section 103 of the Atomic Energy Act and 10 CFR 50.51.

None of the items is directly related to safety. Although the license conditions are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: C   Item: TS 3.11.1.1

Perry Technical Specification 3.11.1.1, liquid effluents - concentration, requires that the concentration of radioactive material released in liquid effluents to unrestricted areas be limited to a specified concentration for dissolved noble gases and to the concentration delineated in the regulations for all other radionuclides. The limiting condition for operation (LCO) refers to Appendix B to 10 CFR 20 for limits, while the surveillance requirements refer to the Offsite Dose Calculation Manual (ODCM) for calculational methods and sampling analysis. This item was selected for review to determine if redundant requirements are being imposed.

This item has regulatory basis in 10 CFR 50. It limits the "instantaneous" concentration of radioactive materials in liquid effluents to help ensure that the dose objectives of Appendix I to 10 CFR 50 are not exceeded. Additionally, 10 CFR 50.36a requires that operating licenses contain Technical Specifications that require compliance with 10 CFR 20.106, which governs the radioactivity in effluents to unrestricted areas.

Additional flexibility is provided by Generic Letter 89-01, which allows licenses to restructure their Radiological Effluent Technical Specifications (RETS). A license amendment implementing this restructuring moves the requirements of TS 3.11.1.1 to a radioactive effluents controls program in the administrative controls section of the Technical Specifications. The implementing details of the program are moved to the ODCM. The Improved Standard Technical Specifications adopted this restructuring of the RETS. Licensee adoption of the guidance provided in Generic Letter 89-01 and its supplement, NUREG-1302, not only enhances licensee flexibility in implementing the program that governs RETS compliance, but also reduces the regulatory burden by eliminating unnecessary and redundant limiting conditions for operation and action items.

It is noted that licensees may need to amend TS 3.11.1.1 or the radioactive effluent controls program to avoid unnecessarily restrictive limits created by the interface of the Technical Specifications with the newly revised 10 CFR 20. Thus, consideration of this item for enhanced flexibility at this time may be unrealistic given the priority attention needed to establish consistency with the new 10 CFR 20 provisions.

The prescriptiveness of Technical Specification 3.11.1.1 must be viewed in balance with the limited flexibility it exhibits in referencing the ODCM and the need for compliance with 10 CFR 50 and the new 10 CFR 20. As discussed above, Generic Letter 89-01 and the Technical Specification Improvement Program offer both enhanced flexibility and the possibility for a reduction in the regulatory burden associated with this item. Based on these considerations, it is concluded that no further review of this item is warranted since options have been provided under Generic Letter 89-01 and the Technical Specification Improvement Program for revisions to this item.

SUMMARY OF PERRY ASSESSMENT

Category: D Item: TS 6.5.1.2

Perry Technical Specification 6.5.1.2, PORC composition, identifies the composition of the Plant Operating Review Committee (PORC). This item was selected as a representative administrative controls Technical Specification.

The regulatory basis for the plant onsite review function is 10 CFR 50.40(b), Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for this requirement is contained in Standard Review Plan (NUREG-0800), Section 13.4, which specifies that the qualification levels for the PORC members should be at least equivalent to those described in ANSI N18.1, Section 4.4, and the PORC composition should provide for interdisciplinary reviews of the subject matter.

The PORC at Perry, which satisfies the above requirements, has nine permanent members including the chairman, the Director of the Nuclear Engineering Department. The remaining members are managers or staff members that have expertise in various technical disciplines. There appears to be some flexibility in the composition of the PORC with regard to the technical disciplines of the permanent members and the fact that two alternate members can be appointed and participate as voting members of PORC at any one time. In reviewing the composition of PORC or its equivalent for the other three plants assessed by the Review Group, it was found that Seabrook had 10 members, Surry had six members, and Peach Bottom had nine members. There are no specific regulatory requirements regarding the number of members that constitute the onsite review committee, as is evident by the differences among the four plants assessed. In addition, the composition also varies among plants. Surry and Seabrook use only plant management and supervisors while Perry and Peach Bottom include staff members. The blend of technical expertise, however, is fairly consistent among the plants. It appears that each licensee proposed the composition of its onsite review committee and the staff reviewed the acceptability on a plant-specific basis, which has resulted in the variation of the number of members among plants. Changes to the number of members on the onsite review committee may be and have been made by licensees through the Technical Specification amendment process on a plant-specific basis. It is, therefore, up to each licensee to determine the number of members needed for the onsite review committee to perform its function in accordance with the regulations.

Based on the above considerations, it is concluded that since there is no specific regulatory requirement related to the number of persons that constitute the onsite review committee, the licensee has the flexibility to determine its plant-specific needs to satisfy this function. Therefore, further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: D Item: TS 6.5.3.1

Perry Technical Specification 6.5.3.1, technical review and control, requires the review of the procedures/instructions, proposed modifications, and proposed tests and experiments that affect nuclear safety by individuals other than those who prepared the documents. This item was selected because it is an example of a Technical Specification requirement that has inherent flexibility and also the potential for enhanced flexibility.

The regulatory basis for this function is contained in 10 CFR 50.40(b), Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for meeting this requirement is contained in Standard Review Plan (NUREG-0800) Section 13.4, "Operational Review." The licensee has satisfied this regulation through the onsite and offsite committees and the Independent Safety Engineering Group (for plants licensed after the TMI accident). To reduce the workload of the Plant Operations Review Committee (PORC) the licensee submitted a Technical Specification amendment that shifted the review of procedures/instructions, modifications, and tests and experiments that affect nuclear safety from PORC to a newly created technical review and control function. Since the PORC in conjunction with the technical review and control function provide the equivalent of the responsibilities performed previously by PORC, the amendment was approved by the staff in March 1992.

The creation of the technical review and control function, although not required by the NRC, provides the licensee inherent flexibility since it reduces the review responsibilities of the PORC by permitting qualified independent individuals to perform these reviews subject to the approval of the General Manager of the Perry plant. However, it appears that with the creation of this new function, some of the responsibilities (e.g., the review of proposed modifications to plant structures, systems, and components) are redundant to those of PORC.

As a result of this amendment, the Perry Technical Specification requires four organizations (PORC, Nuclear Safety Review Committee, Independent Safety Engineering Group, and technical review and control) to satisfy the plant review and audit functions. Pursuit of the staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in enhanced flexibility for the licensee. For example, the prescriptive Independent Safety Engineering Group could be replaced by a staff function, and the review and audit process could be simplified if a line-item improvement in accordance with the Improved Technical Specifications was pursued by the licensee.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: B Items: All

The Peach Bottom operating license contains two items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. Specifically, the Category B items are OL 3.d, NPDES permit change condition, and OL 4.0, effective date and expiration condition.

The NPDES permit change condition is not required by either the Atomic Energy Act or the Commission's regulations but is authorized by 10 CFR 50.50, which provides that the license may contain such conditions as the Commission deems appropriate. The effective date and expiration condition is required by 10 CFR 50.51.

Neither of the items is directly related to safety. Although the license conditions are prescriptive, they do not appear to be unduly restrictive. Neither of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: C Item: TS 3.8.E

Peach Bottom Technical Specification 3.8.E, radiological environmental monitoring, addresses deviations from the sampling schedule, land use census, and analysis to be performed on radioactive materials. This item was selected because it is representative of a Technical Specification requirement that could be relocated to a licensee-controlled document and, therefore, result in a reduction in regulatory burden.

The regulatory bases for this item are 10 CFR 20.106, which is related to the release of radioactivity in effluents to unrestricted areas; Appendix I to 10 CFR 50, which provides numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion for nuclear power plant effluents; 10 CFR 50.36a, which is related to Technical Specifications on effluents from nuclear power plants; and 10 CFR 50, Appendix A, Section VI, which addresses radioactivity control. The guidance for meeting these requirements is contained in Standard Review Plan (NUREG-0800) Section 11, which addresses radioactive materials, and Regulatory Guide 1.109, which addresses the calculation of annual doses to the public from routine releases of reactor effluents for the purpose of evaluating compliance with Appendix I to 10 CFR 50.

This specification ensures that the doses to the public will be within 10 CFR 20 limits and as low as reasonably achievable in accordance with Appendix I. Peach Bottom Technical Specification 3.8 (which includes Section 3.8.E) contains the procedural details associated with Radiological Effluent Technical Specifications (RETS). Generic Letter 89-01 permits a line-item Technical Specification improvement to be made by permitting the licensee to place the programmatic controls of the RETS in the administrative controls section of the Technical Specifications and relocate the procedural details of the RETS to the Offsite Dose Calculation Manual (ODCM) or the Process Control Program (PCP). Relocation of the details of the RETS into the ODCM or PCP would not result in a reduction in the level of radiological control but would provide a reduction in regulatory burden since a Technical Specification amendment would not be required to change the procedural details of the RETS.

Based on the above considerations, it is concluded that, since Generic Letter 89-01 permits a line-item improvement that would reduce the regulatory burden, further consideration of this item would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D Item: TS 6.2.3.1

Peach Bottom Technical Specification 6.2.3.1, function of the Independent Safety Engineering Group (ISEG), prescribes the functions that are to be performed by this group. This item was selected because it is an example of a Technical Specification that appears to exceed regulatory requirements.

This requirement is based an TMI Action Plan Item I.B.1.2 contained in NUREG-0737. This TMI item was only required of applicants for an operating license. Since Peach Bottom Unit 2 received its operating license in 1973, which was prior to the TMI requirements contained in NUREG-0737, there is no regulatory requirement that the licensee's Technical Specifications include an ISEG function.

Based on discussions with the staff, it is our understanding that the licensee endorsed this concept and voluntarily incorporated the ISEG function into its Technical Specifications. As noted in the Review Group's assessment of the Seabrook license, the Improved Standard Technical Specifications permit the ISEG function to be performed as a staff function under the reviews and audits program in the Technical Specifications.

Since the Peach Bottom licensee voluntarily incorporated the ISEG function into the Technical Specifications, removal of this function through the Technical Specification amendment process could be pursued because an ISEG is not a regulatory requirement for Peach Bottom. Alternatively, the ISEG function could be performed as a staff function in accordance with the Improved Standard Technical Specifications.

Based on the above considerations, it is concluded that further consideration of this item would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D　　　　　　　　　　Item: TS 6.5.3.1

Peach Bottom Technical Specification 6.5.3.1, procedure review and approval, specifies the procedure review responsibilities of the Station Qualified Reviewer, the Plant Operation Review Committee (PORC), the Plant Manager, and the designated superintendent responsible for the procedure. This item was selected because it is an example of a requirement that has inherent flexibility and the potential for enhanced flexibility.

The regulatory basis for this function is 10 CFR 50.40b, Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for meeting this requirement is contained in Standard Review Plan (NUREG-0800) Section 13.4, "Operational Review." The licensee has satisfied this regulation through the onsite and offsite review committees. In addition, the licensee has voluntarily adopted the Independent Safety Engineering Group (only required for plants licensed after the TMI accident). To reduce the workload of the PORC, the licensee's Technical Specifications contain a technical review and approval function, which also is not required by the NRC. The PORC in conjunction with the technical review and approval function provides the equivalent of the responsibilities traditionally performed by PORC alone.

Unlike that of Perry, the Peach Bottom technical review and approval function is limited to the review of procedures and procedure changes. The responsibilities of the PORC and technical review and approval functions are well defined so there does not appear to be any review responsibilities that are duplicated by both functions. Since PORC has the option of reviewing these procedures and procedure changes instead of the Station Qualified Reviewer, inherent flexibility within the organization has been provided.

Similar to Perry, the Peach Bottom Technical Specifications require four organizations (PORC, Nuclear Safety Review Committee, Independent Safety Engineering Group, and technical review and approval) to satisfy the review and audit functions. Pursuing the adoption of the staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in enhanced flexibility for the licensee in the performance of the functions currently handled by the four existing organizations. Therefore, it is concluded that further consideration of this issue would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D Item: ES 7.1.1.B

Peach Bottom Environmental Technical Specification 7.1.1.B, organization, requires the Plant Superintendent to report to and consult with other designated licensee superintendents in all matters pertaining to the operation of the facility or to the Environmental Technical Specifications. The Peach Bottom management organization chart is included in the Environmental Technical Specifications and is referred to by this item. This item was selected for review because of its general, administrative nature and to determine whether it has appropriate basis as license condition.

The issuance of Environmental Technical Specifications as license conditions (i.e., Appendix B to the Operating License) has regulatory basis in 10 CFR 50.36b and is related to 10 CFR 51 in that the required conditions are derived from the environmental assessment conducted as part of the plant licensing process. However, this specific item appears to have no sound regulatory basis as an obligation of the licensee in the environmental area as would be required as a valid license condition in accordance with 10 CFR 50.36b. In fact, the designated chain of command specified by this item conflicts not only with the management organization chart that is referenced and included in the Environmental Technical Specifications, but also with the managerial titles and responsibilities specified in the administrative controls section of the Technical Specifications (i.e., Appendix A to the operating license).

Furthermore, while the organizational requirements of the Technical Specifications generally discuss lines of authority, responsibility, and communication, this item is prescriptive in its detail, making it prone to error unless revised with every licensee management organization change. The above-noted conflicts more than likely reflect the fact that this item has not been updated when changes affecting the license conditions were implemented.

The requirement to update such a license condition represents an unnecessary regulatory burden upon the licensee. It is concluded that this item should be considered for possible elimination or, at a minimum, revision to reflect general consistency with the existing organizational requirements of the administrative controls section of the Technical Specifications. Other similar line items in the Environmental Technical Specifications appear to warrant the same attention. However, this problem appears to be unique to the Peach Bottom Environmental Technical Specifications, which have undergone major item revisions or deletions over time. Thus, additional consideration of this requirement for enhancement by the NRC would prove unproductive and no further review of this area is warranted.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: E Item: TS 3.8.D

Peach Bottom Technical Specification 3.8.D, 40 CFR 190, requires compliance with the provisions of 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." Using the doses calculated from radioactive effluent releases that are governed by other Technical Specifications in combination with doses associated with direct radiation (turbine shine, storage tanks, etc.), a total dose is calculated. If the limits of Technical Specification 3.8.D are exceeded, a special report is required to be submitted to the NRC. This item was selected for review because its provisions appear to be redundant to existing regulations and the only stipulated action is a reporting requirement.

This item has regulatory bases in 10 CFR 20 and Appendix I to 10 CFR 50. Since 10 CFR 20 specifically requires compliance with 40 CFR 190 and also specifies reporting requirements for releases of radioactive material in excess of the limits of 40 CFR 190, this Technical Specification appears to be redundant to the regulations. However, 10 CFR 20 and 40 CFR 190 merely provide an overall, broadly defined limit. Technical Specification 3.8.D provides the details of how to meet the regulations as well as actions to be taken when the limits are exceeded. Since the 40 CFR 190 dose levels are generally the more limiting requirements for permissible levels of radiation in unrestricted areas governed by 10 CFR 20, this item is safety relevant; it also includes the practice of ALARA principles.

While 10 CFR 50.36a requires Radiological Effluent Technical Specifications (RETS), Generic Letter 89-01 allows licensees to relocate the procedural details of the RETS to the Offsite Dose Calculation Manual (ODCM). Further, the Technical Specification Improvement Program governs the ODCM, Radiological Environmental Monitoring Program, and Radiological Effluent Control Program requirements as administrative controls in the Improved Standard Technical Specifications. One of the Radiological Effluent Control Program administrative provisions is the requirement that it include limitations to annual doses in accordance with 40 CFR 190. Thus, while the technical requirements that represent the origin of this item are retained in the Improved Standard Technical Specifications, they are handled there more coherently and flexibly as program requirements.

The recent revision to 10 CFR 20 impacts this area. The manner in which Technical Specification revisions relative to 10 CFR 20 changes will be handled is still under review by the NRC to effect a coherent and consistent, yet technically correct, process for compliance with the regulations.

Based on the above considerations, this item has the potential for enhanced flexibility if a change is pursued in accordance with the Technical Specification Improvement Program. However, given the recent revision to 10 CFR 20, further evaluation of the most effective way to implement regulatory compliance in this regard is ongoing by the NRC staff. Therefore, in light of the Technical Specification Improvement Program and the ongoing reviews, it is concluded that any additional consideration of this item for a further reduction in regulatory burden would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: F　　　　　　　　Items: All

The Peach Bottom operating license contains 10 items in Category F, "Unique Plant Features." These items were deemed appropriate to be assessed collectively. They identify the plant and its location and delineate the plant's major design features. Specifically, the Category F items are as follows:

OL 2.A	TS 5.1
TS 5.2.A	TS 5.2.B
TS 5.3	TS 5.4.A
TS 5.4.B	TS 5.4.C
TS 5.5	TS 5.6

These items are basically statements of facts. They generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: G Items: All

The Peach Bottom operating license contains 81 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include license conditions, Technical Specification definitions, and Environmental Technical Specification definitions. Specifically, the Category G items are as follows:

OL 1.A	OL 1.B
OL 1.C	OL 1.D
OL 1.E	OL 1.F
OL 1.G	OL 1.H
OL 1.I	OL 2.B.1
TS 1.O (56 items)	ES 1.0 (15 items)

Except for the financial qualification part of License Condition OL 1.E, the license conditions are legal findings that appear to be required by the Atomic Energy Act or the Commission's regulations. The financial qualification finding was required at the time the Peach Bottom operating license was issued but is no longer required. The Technical Specification definitions and Environmental Technical Specification definitions are judged necessary for the uniform interpretation of the defined terms.

None of the items is directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appears to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

Volume Four

Regulatory Review Group

Risk Technology Application

U.S. Nuclear Regulatory Commission
Office of the Executive Director For Operations

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4.1 INTRODUCTION

In 1975, the U.S. Nuclear Regulatory Commission (NRC) completed the first quantitative study of the probabilities and consequences of severe reactor accidents in commercial nuclear power plants — the Reactor Safety Study, published as WASH-1400 *[Ref. 4-1]*. This work for the first time used the techniques of probabilistic risk analysis (PRA) for the study of severe core damage accidents in two commercial nuclear power reactors. The product of probability and consequence, a measure of the risk associated with severe accidents, was estimated to be low relative to other man-made and naturally occurring risks for the two plants analyzed.

Following the completion of WASH-1400, and similar efforts conducted in other countries (most notably, Phase A of the German Risk Study *[Ref. 4-2]*), research efforts were initiated to develop advanced methods for assessing accident frequencies, improved means for collecting and analyzing operational plant data were put in place, methods were initiated to improve the ability to quantify the effects of human errors, and studies to better predict the nature and effect of common cause failures were begun. Further, limited research was begun on those key severe accident physical processes identified in the Reactor Safety Study.

The 1979 accident at Three Mile Island (TMI) substantially changed the character of the analysis of severe accidents worldwide. Based, at least in part, on the comments and recommendations of the major investigations of that accident, a substantial research program on severe accident phenomenology was planned and initiated with international sponsorship *[Ref. 4-3]*. This program has been the subject of many reviews and comments and included both experimental and analytical studies. It was also recommended in the various TMI investigation reports *[Ref. 4-4]* that PRA techniques be used to complement the traditional non-probabilistic methods of analyzing nuclear plant safety.

A large number of nuclear power plants have been or are being analyzed using probabilistic techniques throughout the world. Individual plant examinations (IPEs) are being or have been performed on all U.S. plants, most of which are using PRA. At the present time, most nuclear power plants have been or are being analyzed to identify potential vulnerabilities and to determine the frequency of severe accidents. Important insights are being gained relative to the actions that might be taken to maintain or improve the plant safety envelope while providing increased flexibility to the plant operator.

In 1984, a study was performed by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the state of the art in risk analysis techniques, and a summary of PRA perspectives was published (NUREG-1050 *[Ref. 4-5]*). Before commenting on the proper usage of PRA analyses at present, the general conclusions of that document relative to the current state of the art, recognizing both the strengths and weaknesses in the technology at present, needs to be revisited.

In the area of systems modeling, much of the basic methodology remains unchanged from that of the Reactor Safety Study. However, there is now a wealth of experience in applying these methods, and improved computer codes now permit the efficient handling of the more complex models required to analyze the effects of fires and external events such as earthquakes. Much, if not all, of the analysis of internal events (and external events) can now be performed on personal computers, substantially reducing the cost and improving the efficiency of studies performed today. Techniques are available to calculate importance measures of plant systems and components from a variety of viewpoints, in a form amenable for use in determining the relative importance of systems and components to plant safety. The decision of the detail to which systems are modeled, however, is generally left to the judgment of the analyst, usually based on a perception of what may be important relative to other components or subsystems. Little guidance is available in the literature in this regard. Thus, before the results can be used in a regulatory application, the boundary conditions and assumptions used in the analysis must be examined to ensure they are appropriate to the specific usage envisioned.

Considerable data have been acquired on initiating event frequencies and component reliability, although this data may vary somewhat from plant to plant. Thus, while a comprehensive plant-specific data analysis is within the current capabilities, it sometimes is not performed because of the costs and resource allocations required. Thus, before a current probabilistic analysis is relied upon to support plant-specific regulatory initiatives, the degree to which the PRA analysis is also plant-specific may need to be ascertained. As discussed in the sections that follow, generic data may well suffice when using the PRA as a coarse screening device to separate the important from the unimportant, but plant-specific data will be needed for more complex usages.

Detailed methods have been developed for evaluating the significance of dependent failures that address both the quantitative aspects of the analysis and the qualitative knowledge gained that can help prevent their occurrence. At the present time, the lack of readily accessible root cause data on dependent failures from operating and maintenance logs is the more limiting factor, rather than the methods for analyzing the data. (The raw data is generally available to the plant owner/operator, but in many cases it may not be in readily usable form to the PRA analyst or to the regulator.) Guidance on acceptable ways of analyzing the raw data for dependent failures has been developed jointly by Electric Power Research Institute (EPRI) and NRC.

It should be noted that methods for evaluating the reliability of solid-state control and protection devices are not yet available for routine application, particularly with respect to the adequacy of the software associated with the solid-state device. Information is available from the aerospace and defense industries in this regard and this information, when coupled with research efforts currently under way, should do much to improve the situation. Therefore, at the present time, when software-driven solid state devices are analyzed, quantitative results should be viewed with considerable caution and care should be given to examining the adequacy of the methods employed.

In the area of human interactions, improved methods are available and additional data have been acquired that permit a more detailed analysis of the likelihood of failing to follow procedures for a number of situations. The state of the art is still relatively weak in the ability to address cognitive and comprehension errors, or to consider the pervasive effect of a poor safety attitude at a plant. Substantial work is under way in these areas in many countries, and some improvements are expected in the future. However, at the present time, the use of PRA information in a regulatory framework will be enhanced if such applications are structured such that they minimize the influence of the uncertainties inherent in the human error probabilities. Even when human errors are treated in a relative manner, however, care must be taken to ensure that dependencies and boundary condition changes are properly considered.

In the area of accident progression and consequence analysis, models have been substantially improved, and many sensitivity analyses are now available. However, comprehensive uncertainty analyses of the models are only now being performed. As identified above, a detailed and comprehensive research program is directed to those elements necessary to reach regulatory closure on severe accident issues. The most recent assessment of the uncertainties in these portions of the analyses was contained in the NRC-sponsored NUREG-1150, "Severe Accident Risks, An Assessment for Five U.S. Nuclear Power Plants" *[Ref. 4-6]*, which considered uncertainties associated with both input parameters and modeling. While, in general, the central estimates (means, medians) of the distributions associated with the releases of the various radionuclides to the environment in NUREG-1150 are lower in magnitude than those predicted in earlier studies such as WASH-1400, the uncertainty ranges remain large.

The ability to perform comprehensive uncertainty analyses, including consideration of both modeling uncertainties as well as those associated with input parameters, has improved greatly. The most detailed study of this type is included in NUREG-1150. However, that method relies heavily on expert elicitation and is extremely resource intensive and time consuming. Improved, more efficient methods are needed if such analyses are to be routinely used in regulatory decisionmaking. Alternately, means should be devised to use risk insights in a manner consistent with a somewhat limited overall assessment of uncertainties.

To the extent possible, the use of probabilistic information in developing performance-based criteria may be more appropriate and robust when applied to the potential for severe core damage or to system availability under given conditions rather than to public risk. The inherent uncertainties in assessments of individual or societal risk make analyses of such parameters more amenable to comparisons with goals rather than determination of compliance with criteria.

The ability to analyze the effect of fires, floods, and other external events has improved substantially. Major limitations still exist relative to the ability to estimate recurrence frequency for very rare catastrophic events (such as great earthquakes) and it does not appear that the uncertainties associated with such estimations will be narrowed substantially in the near future. Similarly, some of the subtle effects associated with certain other external events will require more study before they can be quantified without considerable uncertainty (e.g., effects of smoke and soot during fires). These factors may limit the use of probabilistic-type approaches in these areas of regulation unless consideration is given to the impact of the uncertainties involved on the regulatory decisionmaking process.

Given these strengths and weaknesses, how can probabilistic results be used? A comprehensive discussion appears in "Probabilistic Safety Assessment in Nuclear Power Plant Management," edited by N. J. Holloway and sponsored and published by Principal Working Group 5 (Risk Assessment), Organization for Economic Cooperation and Development/Nuclear Energy Agency *[Ref. 4-7]*. It evaluates the value of PRA as an increasingly valuable complement to general engineering analysis for assessing and managing the safety-related operations of a nuclear power plant. The report draws the following conclusions:

- The application of PRA provides plant management with a general systems engineering tool that generates insights not readily available from the traditional deterministic safety and licensing analyses. While some of these insights derive from probabilistic evaluation, the majority do not, but simply arise from the systematic yet unprejudiced nature of the PRA procedures. Some of the most important new insights have been derived from the integrated model of plant system behavior and operator actions that PRA can create.

- The existence of a PRA capability within a plant operator's organization provides for a logical framework of regulatory discussion and negotiation to be created. Furthermore, this framework is plant-specific and can thus be used for plant-specific evaluation and more logical resolution of generic safety issues.

- The benefits derived by plant operators are generally greatest when there is a full commitment to development and maintenance of an internal PRA capability, with minimal dependence on outside experts except for an initial technology transfer phase. Although such commitments are quite expensive, those who have undertaken them are generally of the opinion that the benefits more than compensate.

- The application of PRA to an existing plant has always resulted in the identification of effective ways of achieving plant safety, and has thus contributed to the overall effectiveness of plant operation.

Therefore, the report comes to the conclusion that the implementation of PRA as an aid to nuclear power plant safety management is directly beneficial to those implementing it in support of their plant designs or operations and to all those concerned with ensuring nuclear plant safety. It is in this vein that the NRC has initiated the IPE process, in which each licensee is requested to conduct plant-specific risk-based searches for vulnerabilities.

Probabilistic analysis techniques also are of interest to the regulator in a variety of ways, and most of the comments addressing utility use in the OECD/NEA report referenced above are applicable in this venue as well. These techniques provide a unique perspective that permits an independent consideration of the body of regulatory requirements to ensure that potentially risk-significant factors are properly considered and that regulatory resources are not needlessly expended on unimportant matters by either the regulated or the regulator. They can be used to identify those systems, trains, and components that are important in maintaining a low likelihood of core damage, and, conversely, can also identify those items that have little influence on the likelihood of an accident. However, such analyses must be done with a clear appreciation for the strengths and weaknesses discussed above.

The results of PRA studies, including detailed uncertainty analyses, provide information useful in prioritizing the expenditure of resources for plant evaluations. The models generated in a probabilistic study are useful in evaluating the significance of both plant-specific and generic issues. They are also useful when developing strategies to react to or manage a severe accident as it occurs. As before, this must be done with an appreciation of the boundary conditions and assumptions used in the original analyses. While items found risk-significant might warrant further analysis or regulatory attention, this will depend on the specifics of the situation, the degree to which existing regulatory instruments are met, and the potential for approaching or exceeding any safety goals that might be established. Similarly, items cannot be dismissed on the basis of low risk until it is clear the analysis is sufficiently robust in the area of interest and that it adequately supports the decision.

In summary, the strongest insights gained from a probabilistic analysis are derived from (1) the integrated and comprehensive examination that analyses of these types entail, (2) the attention devoted to interactions between systems, the operating staff, and the plant systems, and (3) the structured examination of operating experience. In general, the insights and importance rankings developed from the analysis of a system, or from analyses of groups of systems, to assess the frequency of severe core damage are more robust than those that require an evaluation of overall risk; this determination is because the analyses in the former case are simpler and the uncertainties involved are not as broad as in the latter situation. The weakest insights are those that are derived primarily from the quantitative rankings alone, without considering the meaning of the results in an engineering context. While the quantitative results are important, they should be considered as most useful for a screening of the results to identify important accident sequences and plant features at the present time and to give indication of areas with relatively little or relatively high importance in a probabilistic context.

Probabilistic analysis presents an additional tool, an additional source of information that can be used to focus regulatory decisionmaking in many areas, identifying features most important to plant safety. Used properly, with recognition of the limitations and proper attention to the scope, boundary conditions, and assumptions of the analysis, it can be used to exploit the flexibility presently existing within the regulatory environment to improve plant safety while reducing undue regulatory burden. It can also be used to suggest areas where performance-based regulatory practices can be employed in the future. Techniques are now being developed and employed to improve plant configuration control and to optimize the required plant response to equipment outages or mode changes.

Recognizing these strengths and weaknesses, a set of general guidelines have been developed regarding the constraints that are needed on the boundary conditions and assumptions of a probabilistic analysis used to support various types of regulatory initiatives. The qualifications are discussed in detail in Section 4.2. A proposed approach to PRA application in the regulatory process is provided in Section 4.3. Detailed discussions of these applications are presented in Sections 4.4 through 4.6. How PRA can be used to provide a relative ranking and importance of rules and regulations is provided in Section 4.7. Perspectives from non-NRC organizations regarding the use of PRA is provided in Section 4.8. A summary of NRC programs, particularly how they can support the recommended applications, is provided in Section 4.9. Conclusions are provided in Section 4.10. A list of acronyms, abbreviations, and references is provided in Section 4.11.

4.2 PRA SUMMARY

In using a PRA-type analysis to provide additional flexibility in the regulations and their implementation, it is necessary to understand the purpose, boundary conditions, and type of results associated with this type of analysis. In addition, it is important to note that PRA terminology has been used by the NRC and throughout the industry with a variety of meanings. The discussions in this section are, therefore, providing the definitions of the PRA terminology as used in this report. These definitions are based on the NRC draft PRA Working Group report *[Ref. 4-8]*, one part of which provides guidance on NRC PRA term definitions.

A PRA of a nuclear power plant is an analytical process that quantifies the potential danger of the design, operation, and maintenance of the plant to the health and safety of the public. The danger or hazard that has been identified as posing the greatest risk to the public is the consequences associated with possible reactor core melt accidents. Therefore, in the calculation of the risk, those events that could potentially lead to a reactor core melt and a release of radionuclides from the reactor are identified and their frequency quantified.

A PRA can be performed to different levels. The first phase of a PRA, called a Level 1 PRA, involves the calculation of the potential core damage frequency. The second phase, a Level 2 PRA, calculates the frequency of the core damage progressing to a core melt and the release of radionuclides to the environment. The last phase, a Level 3 PRA, calculates the consequences of the fission product releases to the environment.

A PRA can also be performed based on either internal or external events or both. Internal events only consider equipment failure internal to the component (or to systems supporting the component) when examining the potential failure of SSCs. Internal flooding is, however, considered part of the internal events analysis for the purpose of this discussion. External event analysis considers equipment failure external to the component, and therefore, involves the examination of the effects of fire, earthquakes, high winds, flooding, etc. The term "Level 1 PRA," however, generally refers only to internal events and is used as such in this report.

Each PRA level consists of numerous elements of which several are critical when considering various applications of the PRA. That is, the attributes of each element in the PRA will dictate the ability of the PRA to be used beyond its original purpose (for example, the original purpose might be an IPE). Primarily, only those attributes associated with a Level 1 PRA are discussed since the applications under consideration generally involve the Level 1 portion of the PRA.

In this report, the various potential applications of PRA in providing additional flexibility in the implementation of the regulations will focus on those aspects that address core damage prevention based on internal events and not core damage based on external events nor mitigation of the effects of core damage (e.g., containment performance, source term releases). Ultimately, some expansion will be needed to consider external events and engineered safety features with mitigative functions. This expansion will be done in conjunction with any pilot programs in this regard proposed by the industry.

One objective of the Regulatory Review Group (hereinafter referred to as the Review Group) is to determine how an integral analysis can be used to provide more flexibility in the regulations and the implementation of the regulations. Therefore, in providing a general set of principles or guidelines, the various methods that are generally used by licensees — level of detail, scope, and assumptions — needs to be understood. The following discussion is written from the perspective of the content of licensees' PRAs.

4.2.1 PRA Elements

A Level 1 PRA is comprised of three essential elements as follows:

- The delineation of those events that, if not prevented, could result in a core damage state and the potential release of radionuclides.
- The development of the models representing the core damage events.
- The quantification of the models in the estimation of the core damage frequency.

The first element of a Level 1 PRA delineates those events that, if not prevented, could result in a core damage state and the potential release of radionuclides. This process, generally referred to as the Accident Sequence Analysis, is typically divided into two parts: identification of the initiating events and development of the potential core damage accident sequences associated with the initiating events.

The initiating events generally modeled in current PRAs include loss-of-coolant accidents (LOCAs), general plant transients, and plant support system transients.[1] Event trees are developed for each of these initiators that delineate the core damage accident sequences that could potentially occur. The accident sequences are comprised of those sequences of events (i.e., success and failure of the functions and systems) that, if they occur, will result in core damage. The initiating events and accident sequences, therefore, identify the various systems for which a mathematical (i.e., Boolean algebra) model is required.

[1] In recent PRAs, internal flooding has been defined as an internal event. The IPE, which is an examination of internal events, includes the consideration of internal flooding.

The plant models are developed in the second element of a Level 1 PRA. These models depict the different failure paths associated with each system in determining the system's unavailability and unreliability.

Two different types of fault trees are generally used to model a system's potential performance. The *"large fault tree"* concept involves developing a single fault tree that models each of the different failure configurations of a system. Special events (called "house" events) are modeled in the fault trees that are used to activate each configuration. The *"support state fault tree"* concept involves developing a separate fault tree for each different failure configuration (or support). Each support state fault tree is, therefore, comprised of independent events.

The third element of a Level 1 PRA estimates the plant's core damage frequency and the associated statistical uncertainty. This estimation is performed by first quantifying the failure probabilities and unavailabilities of the various structures, systems, and components (SSCs), quantifying the initiating event frequencies, and quantifying the human error probabilities (HEPs) associated with the various operator actions. The frequency for each event tree core damage accident sequence is then quantified by integrating the failure probabilities (i.e., event data) of the SSCs and the HEPs with the initiating event frequencies into the Boolean models. These frequencies are summed to yield the overall mean core damage frequency of the plant. This value represents the average annual core damage frequency associated with the design, operation, and maintenance of the analyzed plant.

Part of the core damage frequency estimation is the quantification of its associated statistical uncertainty. This uncertainty reflects the lack of precision in the data or a lack of detailed understanding of the modeled physical phenomena.

4.2.2 PRA Scope and Level of Detail

PRAs examine the consequences of events that involve a reactor scram[2] or forced shutdown with the need for subsequent core heat removal. These events can occur at different reactor operating states from full to low power and various shutdown modes.

Initiating Event Analysis —

The initiating events are generally incorporated in the current PRA models by a single event that represents the average annual frequency of the event. Most initiating frequencies are developed from operating data, and although a logic model explicitly

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[2]The resulting reactor scram is an "immediate" occurrence. That is, inoperability of a system that requires the plant to go to shutdown conditions after, for example, 8 hours, may not be considered an initiator.

depicting the various systems and components contributing to the initiator occurrence may be developed and quantified, it is generally not incorporated into the PRA model.

The initiating events generally modeled in current PRAs include LOCAs (e.g., pipe breaks, stuck open relief valves, steam generator tube rupture), general plant transients associated with BOP systems such as loss of feedwater, and support system transients associated with non-BOP systems such as loss of a vital AC bus.

Event Tree (Accident Sequence) Analysis —

The accident sequences are generally depicted at the functional or systemic level of detail. The selected functions or systems are dependent on the scope of the success criteria analysis. The success criteria determines those functions or systems, or combination of functions or systems, which if performing to defined conditions, will maintain the core in a safe condition (i.e., prevent the occurrence of a core damage state). Conversely, the success criteria identifies those combinations of functions or systems, which if not performing to specified conditions, will result in an unsafe condition (i.e., core damage). Generally, in most PRAs, the core is assumed to be in a safe condition when the consequences of the radionuclide releases from the damaged fuel would be negligible. Typically, this state is assumed to be prevented if reactor water level is not allowed to decrease below 2 feet above the bottom of the active fuel for BWRs and below the top of the active fuel for PWRs.[3]

The requirements of a defined core damage state is determined from detailed engineering analysis of both core and plant behavior under different accident conditions (e.g., large LOCA versus normal plant transient). The results, therefore, are subject to the codes, modeling assumptions, etc. that are used on the analysis.

As noted, the defined success criteria and extent of supporting engineering analysis determines those plant-specific functions and systems that are identified as capable of preventing a core damage state. There are, however, numerous plant systems that either have no relationship to the needed function or do not meet the necessary success criteria. These systems are not evaluated (e.g., modeled) in the PRA. An example of the number of plant systems as compared to those modeled in a PRA is shown in Table 4.2-1. It is easily seen from this table that a PRA, while successfully integrating the impact of design, operational, and maintenance faults on the plant from a core damage prevention perspective, is limited to a narrow set of systems.

[3]The level is much higher for PWRs since two-phase cooling is not inherently part of its design.

Table 4.2-1
Example of BWR Plant Systems Versus PRA Modeled Systems

SYSTEMS	PRA	SYSTEMS	PRA
Nuclear Boiler System	•	Auxiliary Steam System	•
Recirculation System	•	Condensate System	◇
CRD Hydraulic System	◇	Feedwater System	◇
Redundant Reactivity Control	•	Condensate Cleanup System	•
Feedwater Control	•	Heater, Vents Drains System	•
Standby Liquid Control System	★	Turbine Systems	□
Neutron Monitoring System	•	Generator Systems	□
Remote Shutdown System	•	Condenser Systems	□
Reactor Protection System	□	Off Gas Systems	•
Plant Annunciator System	•	Circulating Water System	□
Fire Protection System	◇	Chlorination System	•
Meteorological Monitoring	•	Water Storage and Transfer	•
Seismic-Instrumentation System	•	Emergency Service Water	★
Vibration Monitoring System	•	Component Cooling Water	△
Loose Parts Monitoring System	•	Turbine Bldg Cooling Water	△
Transient Test System	•	Normal Service Water	△
Drywell Monitoring System	•	Plant Air System	◇
Residual Heat Removal System	★	Instrument Air System	★
Low Pressure Core Spray	★	Plant Chilled Water System	△
High Pressure Coolant Injection	★	Drywell Chilled Water	△
Leak Detection System	•	Diesel Generator Systems	★
MSIV Leakage Control System	•	Transformer Systems	•
Feedwater Leakage Control	•	Switchgear Systems	•
RCIC System	★	Auxiliary Bldg Vent System	△
Liquid Radwaste System	•	Radwater Bldg Vent System	•
Reactor Water Clean-up System	△	Turbine Bldg Vent System	△
125V & 24V Batteries	★	Drywell Vent System	★
125V DC Power Supplies	□	Wetwell Vent System	★
125V Battery Chargers	★	Emer Swgr and Batt Rm Vent	△
Static Inverters	★	Other Bldg Vent Systems	△
Cont./Drywell All Monitoring	•	Control Bldg HVAC System	△
Drywell Cooling System	△	Control Room HVAC System	△
Main and Reheat Steam System	◇	Load Seq and Shedding	△

★ Component level of resolution model
◇ Failure mode level of resolution model
□ Event level of resolution model
△ Not explicitly modeled
• Not evaluated

Systems Analysis —

The fault trees constructed for the various systems can be developed to different levels of resolution as follows:

- *Component Resolution* — The individual components comprising the function or system and the possible failure modes of each component are explicitly depicted in the fault tree model as unique basic events. It should be noted, however, that not every system component and failure mode is modeled. Generally, only those components whose failure mode results in the loss of system function with a relatively significant probability (e.g., ≥ 1E-6) are modeled.

  A component in a PRA is generally the major piece of equipment that is essential to the function of the system such as pumps, valves, heat exchangers, diesel generators, etc. Parts that are essential to the component's function (e.g., valve disk) are not explicitly modeled as unique basic events but are included within the boundary of the component (e.g., valve). Only those failure modes that prevent system function are usually modeled.

- *Failure Mode (Train) Resolution* — The individual components comprising the system or function, and their different failure modes, are not explicitly depicted in the fault tree model as unique basic events. Only the failure modes of each train (or system) are modeled as unique basic events (e.g., train hardware fault, train out for maintenance, loss of power).

- *Event (System) Resolution* — The function or system is represented by a single event; that is, a Boolean model explicitly depicting the components (or trains) and the failure modes as unique basic events is not constructed in computing the system failure probability. This level of resolution can be referred to as a "black box" model.

Data Analysis —

The data analysis basically involves the quantification of the different failure mode probabilities associated with the SSCs modeled in the system fault trees. The failure modes considered in current PRAs generally include the following:

- Hardware faults — This failure mode examines the potential for demand and time-related type failures associated with random hardware faults caused by such items as crud buildup on valve disk.

- Test and maintenance faults — This failure mode examines the potential for a component, train, or system to be unavailable when demanded because it is out-of-service for a test or maintenance activity.

- Common cause faults — This failure mode examines the potential for several components to dependently fail from the same specific cause such as replacing the same part in several components where each replacement part is defective.

The data analysis also involves the quantification of initiating event frequencies. This quantification examines those failure modes (e.g., hardware faults, human performance faults) that can cause the occurrence of the initiating event (e.g., main steam isolation valve closure).

The identified events and the defined failure modes dictate what plant information is required to quantify the failure rates and unavailabilities and initiating event frequencies. The estimation of the probabilities and frequencies is dependent on the supporting plant information that provides the necessary information on plant history. If adequate plant information exists, then plant-specific equipment failure rates, unavailabilities, and initiating event frequencies are computed; however, if inadequate plant information exists (e.g., failures have not occurred), "generic"[4] data must be used, which places a limitation on the PRA application.

The period of time of the plant's history that is used to compute equipment failure rates, unavailabilities, and initiating event frequencies must be considered. A plant's historical performance changes over time; design, operational, and maintenance changes are occurring, which affects the reliability and unavailability of systems and components. It is important that the data reflect, as much as possible, the current performance of the plant.

Human Reliability Analysis (HRA) —

The estimation of event probabilities also involves the quantification of human performance events. This task is very diversified, and standardization among PRAs does not exist. This task, however, has the ability to change the dominant accident sequences; that is, change the results of the PRA. The HRA, therefore, not only impacts the estimated core damage frequency but what are identified as the most likely contributors to realizing a core damage state.

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[4]Generic data are based on compilation of data of the operating history of components taken from the nuclear industry.

The human events include those operator actions conducted during normal plant operation that result in inoperable equipment without causing an initiating event (generally referred to as pre-initiator human actions). Also evaluated are those operator activities that are required to achieve a safe plant shutdown (generally referred to as post-initiator human actions). Post-initiator human actions include response type actions and recovery type actions.

Response-type actions are those human actions performed in response to the first level directive of the EOPs. For example, suppose the EOP directive instructs the operator to determine reactor water level status, and another directive instructs the operator to maintain reactor water level with system x. These actions — reading instrumentation to determine level and actuating system x to maintain level — are response-type actions.

Recovery-type actions are those human actions performed to recover a specific failure or fault. For example, suppose system x failed to function and the operator attempts to recover it. This action — diagnosing the failure and then deciding on a course of action to "recover" the failed system — is a recovery-type action.

Quantification —

Using the event data and HEPs, the quantification of the core damage frequency is performed by integrating the initiating event models[5] with the system models as depicted by the event trees. This computation is typically performed on a sequence basis, with the core damage frequency equal to the Boolean summation of the core damage frequencies of the individual sequences.

The core damage frequency is generally based on the summation of only the *dominant* accident sequences, and not every defined accident sequence. Those accident sequences whose calculated core damage frequency is typically less than 1E-8 to 1E-10 are generally truncated; they are not integrated into the overall PRA model. If the PRA contains quantified conclusions, i.e., importance measures, these conclusions are generally based on the dominant accident sequences alone.

Uncertainty Analysis —

The uncertainties associated with the parameter values are defined by assigning a probability distribution to the component failure rates and unavailabilities, initiating event frequencies, and human error event probabilities. The uncertainties associated with physical phenomena are estimated by assigning probability distributions to the different

[5]These models, as mentioned previously, are generally a single event.

modeling hypotheses. The data and modeling distributions are then propagated in the core damage frequency quantification generally using a type of Monte Carlo technique.[6]

4.2.3 PRA Boundary Conditions

In reviewing the scope and level of detail of a Level 1 PRA, certain boundary conditions can be identified that have the potential to impact the application of a PRA. These boundary conditions need to be addressed when considering using a PRA in the regulatory process.

The boundary conditions include the following:

- *Scope* — The PRA must involve a Level 1 analysis and must address at least internal events (including internal flooding). It needs to be noted that the importance of external events and core damage mitigation cannot be ignored. That is, depending on the PRA application, consideration of these two issues will need to be addressed so that safety significant SSCs due to either external events or core damage mitigation (but not important to core damage prevention) are not overlooked.

- *Structure, System, and Components* — The application of the PRA is limited to those SSCs that are part of the PRA. If the SSCs are not modeled in the PRA, it does not mean they are unimportant to core damage prevention, but that, from a probabilistic perspective, they do not contribute significantly to the core damage frequency. Therefore, for SSCs not modeled (e.g., evaluated) in a PRA, it is difficult to use the PRA for insights relative to the impact of potential changes associated these SSCs.

- *Level of Resolution* — The usefulness of a PRA is dependent on the level of resolution of its SSCs. If a PRA is performed at a system level, the insights of the PRA are at a system level. Conversely, a component level of resolution provides insights at the component level.

- *Failure Modes* — Although a PRA may be performed to a component level, the application will be restricted to those failure modes (i.e., hardware, test and maintenance and common cause) modeled for the component. A component level of resolution does not mean that each failure mode is modeled in the PRA.

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[6] An uncertainty analysis of the core damage frequency estimation is not required by Generic Letter 88-20, although some licensees have elected to perform an uncertainty analysis in their IPE efforts of the Level 1 portion.

- *Data* — The degree of plant-specific data that is used in the quantification of component failure rates, unavailabilities, and initiating event frequencies provides the degree of actual plant-specific representation. Therefore, whether generic data or plant-specific data are used will determine the extent of the use of the PRA in the regulatory process.

- *HRA* — The incorporation of human activities into the PRA model has the ability to determine the dominant accident sequences and the dominant contributors to core damage. Insights from a PRA can, therefore, be misleading dependent on the type of human activities that were modeled. There are considerable uncertainties in the current ability to model human actions, and different assumptions can lead to significant changes in results.

- *Truncation* — In quantifying the core damage frequency, truncation of low probability events and sequences is generally performed. Although this truncation is normally preformed such that ~95 percent of the core damage frequency remains after truncation, the insights (e.g., importance measures, sensitivities) do not generally include the impact on the truncated events and sequences.

- *Uncertainty* — The quantification of the data and modeling uncertainties indicates the possible range of occurrence of a core damage state; that is, due to lack of knowledge, the uncertainty estimates the upper and lower bounds for which a core damage accident sequence could actually occur.

These boundary conditions are discussed in more detail for the individual applications in Sections 4.3 through 4.6.

4.2.4 PRA Results

The form of the results will dictate, in a sense, the usefulness of the PRA. Besides the calculated core damage frequency, there are numerous other types of results that are quantified in a PRA. A PRA model is comprised of hundreds of accident sequences that can potentially result in a core damage state. The results of a Level 1 PRA indicate the ***dominant*** accident sequences which generally comprise at least 95 percent of the total core damage frequency and generally include less than two dozen sequences. These dominant sequences are those potential accident sequences that are the most likely to occur. As part of these results, the individual events that are dominating these accident sequences are also identified. These events are, therefore, the ones that are the most likely to occur that could result in a core damage state.

The most meaningful results of a Level 1 PRA, perhaps, when considering the use of PRA in the regulatory process are the importance measures. These measures show different types of insights to the core damage frequency if changes regarding the availability and reliability were made to a SSC.

The importance measures generally seen in PRAs include one or all of the following:

- *Reduction Importance Measure* — provides a ranking of the events (e.g., components) by those most crucial for safety improvement. The importance value for each event indicates the potential reduction to the core damage frequency if the event's (e.g., component's) probability was quantified as 0.0, or, for example, the component was assumed to be perfectly reliable. This measure, therefore, indicates how much the core damage frequency can be improved (i.e., reduced) if it can be assured that a SSC will always function as required when demanded.

- *Increase Importance Measure* — provides a ranking of the events (e.g., components) by those most crucial to maintaining safety at the current estimated level. The importance value for each event indicates the potential increase to the core damage frequency if the event's probability was quantified as 1.0, or for example the component was assumed to be always unavailable. This measure, therefore, indicates how much the core damage frequency can be hurt (i.e., increased) if failure of the SSC was always certain.

- *Fussell-Vesely Importance Measure* — provides a ranking of the events (e.g., components) by contribution to the core damage frequency by computing their potential to change the core damage frequency. The importance value for each event is the summation of core damage frequencies of the cut sets[7] (or sequences) containing the event under consideration divided by the total core damage frequency.

These importance measures are significant because they can indicate the relative safety importance of an issue without requiring further manipulation of the PRA model. That is, safety insights can be gained from these measures. For example, the Reduction Importance Measure shows both those events (e.g., components) that are most likely to cause core damage and those events that have little-to-no impact on core damage. The Increase Importance Measure, on the other hand, indicates those events (e.g., components) that are critical to maintaining the current level of safety. That is, if their reliability and availability were to decrease, they would have the most significant impact on the core damage frequency. These measures can then be used to define generic categories to provide safety insights in the regulatory process.

[7]The minimum, unique combination of events that will result in the defined end state, e.g., core damage.

4.3 PRA APPLICATION DEFINITION

4.3.1 Risk-Based Reactor Regulation

Risk (or probabilistic risk) can be defined as the frequency of the consequences associated with an identified hazard that poses a potential danger to the health and safety of the public. Risk-based regulation involves the use of PRA regarding these identified hazards in the development and implementation of the regulations. PRAs of current facilities, however, address the frequency of the consequences of radionuclide releases from the reactor.[8] In this context, risk-based regulation is then defined as risk-based ***reactor*** regulation.

The staff has defined risk-based regulation as the use of PRA insights to focus licensee and regulatory attention on design and operational issues commensurate with their impact on risk to the public *[Ref. 4-9]*.

As used in this report, risk-based regulation refers to the panoply of possible current and future uses of probabilistic analyses to support regulatory actions. These applications include present uses such as generic issue prioritization and resolution, backfit decisions under 10 CFR 50.109, regulatory analysis in support of rulemaking, prioritization of licensee activities in response to regulatory requests, and justifications for continued operation. Other possible uses (described below) are included such as development of graded approaches to the maintenance rule and to quality assurance requirements, optimization of Technical Specification requirements for allowed outage times (AOTs) and surveillance test intervals, Technical Specification schemes that are based on risk-based configuration control, and ultimately, a set of regulatory requirements almost totally dependent on the risk analysis of the facility.

4.3.2 PRA Utilization in Regulatory Process

The Review Group charter regarding the assessment of risk technology directs the group to "*examine how an integral analysis (PRA) can be used to provide more flexibility in the regulations and the implementation of the regulations. Determine what types of general ground rules or restrictions would be necessary to confidently sustain broad PRA usage as an accepted, credible tool for optimizing operations while maintaining the current level of safety. This will include addressing uncertainties and limitations of analytical tools and restrictions that should be placed on their use, identifying ways of accommodating limitations and specify conditions under which NRC could support broad application of risk technology to optimize licensee flexibility.*" *[Ref. 4-10]*

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[8]The danger or hazard that has been identified as posing the greatest risk to the public is the consequences associated with possible reactor core melt accidents.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

In response to the above charter, the use of PRA to provide additional flexibility in the implementation of the regulations requires that general sets of PRA principles be defined. These principles need to define a set of general rules or guidelines that will establish major boundary conditions and assumptions; however, it needs to be recognized that this set will change as one changes application. It will be most useful, therefore, to construct these principles in terms of requirements as the application progresses from the generic to the plant specific.

A possible structure from the more generic application to the plant specific is illustrated below in Figure 4.3-1.

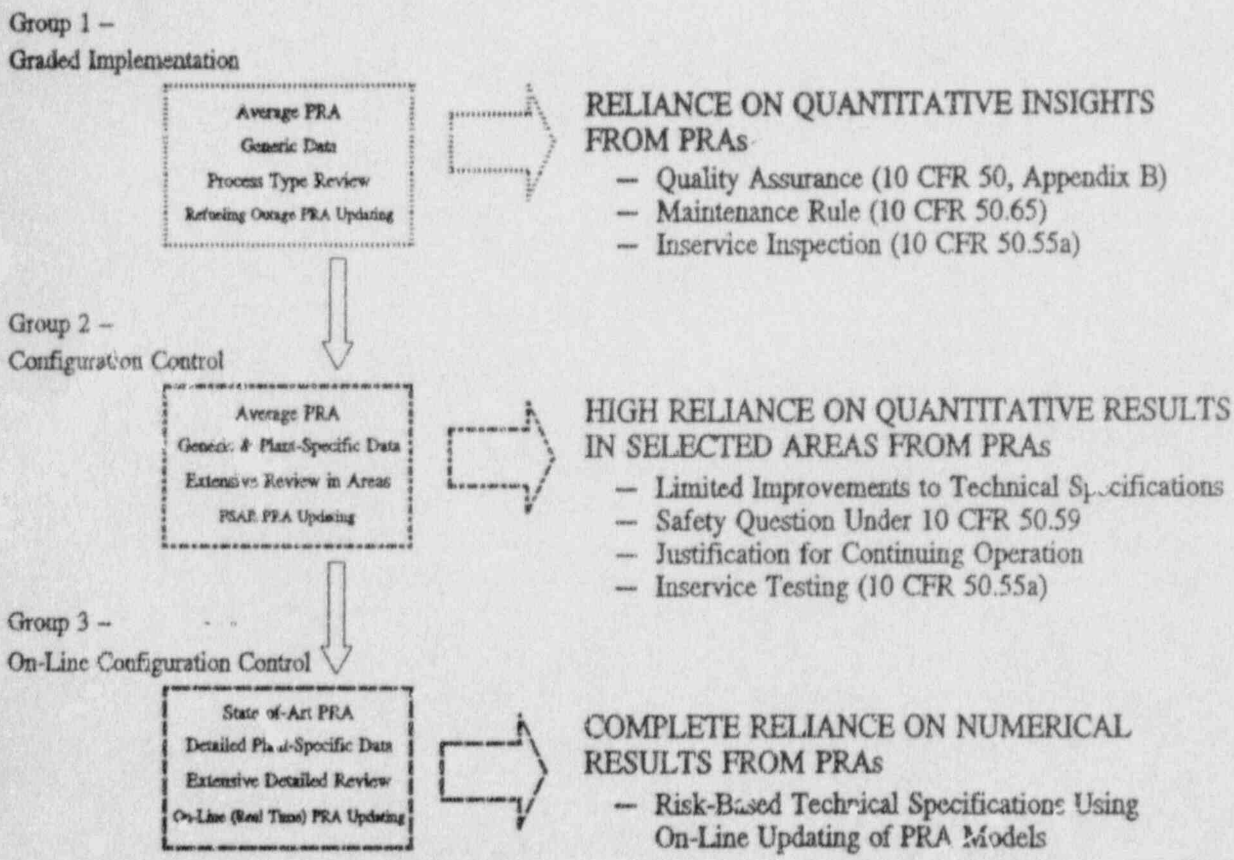

Figure 4.3-1. PRA Applications.

For the first group, emphasis would be placed on those areas of PRA that would identify general categories of plant SSCs in terms of their safety significance. Use of the PRA in this manner would not require great precision in the PRA. Conversely, the regulator does not require a high degree of precision in the PRA, and therefore, it would not be necessary to conduct a thorough *de novo* review of the PRA.

For this type of utilization, generic failure rate data would suffice, supplemented with plant-specific data only where a qualitative examination of operating experience might indicate some anomalous behavior relative to the overall generic data base for such components. Because the real purpose underlying Group 1 type uses is the separation of the important from the unimportant, and, only secondarily, the development of rank ordered groups of the "important," frequent updates of the PRA would not be required. Rather, they would need to be done only when there was a major redesign of one of the plant systems or a major modification in the basic operational principles; update of the PRA at each refueling outage would suffice.

For Group 1, based on the principles stated above, PRA applications would involve a *"graded implementation"* of regulatory rules. This application would differentiate the safety significant SSCs from the insignificant where the rule implementation for these SSCs would involve a graded approach commensurate with their safety significance. Three possible examples could involve implementation of 10 CFR 50, Appendix B, Quality Assurance, 10 CFR 50.65, Maintenance Rule, and 10 CFR 50.55a, Inservice Inspections.

For the second group, emphasis would be placed on those areas of the PRA that would be used directly in developing a probabilistic-based strategy to implement or modify a given regulatory practice. Use of the PRA in this manner would require a higher degree of precision in the PRA since emphasis is being focused on numerical results of a specific PRA; a higher degree of confidence in plant-specific representation is needed.

For this type of use, a detailed analysis of plant-specific data would be needed for those areas of the PRA that would be used in support of the regulatory change; that is, generic data could suffice in the unaffected areas except where (as in Group 1), a qualitative examination of operating experience might indicate some anomalous behavior relative to the overall generic data base for such components. Since a higher degree of precision would be required by the regulator, a more detailed and comprehensive review would need to be performed. This review, however, would be focused on the specific application, and therefore, on those areas of the PRA supporting the application. Because Group 2 uses are relying on results from a specific PRA, updates would be required similarly to Group 1, when there is a major redesign of one of the plant systems or a major modification in the basic operational principles. Update of the PRA at each Final Safety Analysis Report update, however, would be needed if the change affects that part of the PRA.

For Group 2, based on the principles stated above, PRA applications would involve a *"configuration analysis"* in the implementation of regulatory rules. This application would propose a change or modification to specific requirement based on the results of the plant configuration modeled in the PRA. Possible examples could involve improvements to Technical Specifications, improvements to ASME Boiler and Pressure Vessel code requirements for Inservice Testing, justification for continuing operation, and safety questions under 10 CFR 50.59.

For the third group, emphasis would be completely placed on the numerical results of plant-specific PRAs. The PRA would be an integral part of the regulatory structure and plant safety decisions would, therefore, be based on the plant PRA. For this type of use, a PRA of high calibre with a high degree of precision would be required. Detailed models would need to be developed and standardization on level of detail, resolution and scope would be required. The uncertainties surrounding modeling assumptions, physical phenomena and data would require a higher degree of resolution. Since the PRA would be an integral part of the regulatory structure, the PRA would require a comprehensive review by the staff. Perhaps of most significance, a comprehensive analysis of plant data would be required, since many of the methods currently available to optimize regulatory practices have imbedded assumptions regarding the characteristics of the failure data of the various components. Updating of the system status would be needed on a frequent basis, perhaps even in real time.

For Group 3, PRA applications would involve an *"on-line configuration control"* in the implementation of regulatory rules. This application would use a *"real time living-PRA"* to make daily regulatory and safety decisions, for example, in support of Technical Specifications.

The inherent difficulties in progressing beyond Group 1 type applications suggest that pilot programs be organized between the NRC and the regulated industry to test the viability of the more complex applications before they are offered to the industry as a whole.

A discussion of an application from each group is provided in the following sections. These discussions focus on the general sets of rules for the PRA relative to its application.

4.4 PRA APPLICATION FOR GRADED IMPLEMENTATION (Group 1)

The use of a PRA to support a graded implementation of the regulation means that PRA results and insights are used to identify different safety categories of SSCs (from the insignificant to the significant), and the implementation of the rule for each SSC category is commensurate with their safety significance. That is, as the importance of an SSC becomes less safety significant, the implementation of the rule becomes less stringent (i.e., less detailed, comprehensive and prescriptive). Conversely, as the importance of an SSC increases, the implementation becomes more restrictive.

To use PRA in support of a graded approach in rule implementation, PRA criteria associated with the application need to be defined. Criteria determining the definition of importance, criteria used to identify the important SSCs, and criteria establishing the basic conditions of the PRA in identifying importance are each necessary. In addition, these criteria need to ensure that the PRA application does not negatively affect the current level of safety associated with the design, operation, and maintenance of the plant.

4.4.1 Importance Definition

The major element of the graded application is identifying those systems, trains, and components that are important and then determining their relative importance. It is, therefore, necessary to define what is meant by importance and define the criteria for relative importance. These definitions are both based on insights from PRA.

In defining what is meant by importance, this definition can be based on either "deterministic" or "probabilistic" perspectives. It is, therefore, important to qualify what is meant by these terms. A deterministically important SSC is one whose failure would result in, for example, core damage but for reasons of design, operation or maintenance is judged to have very low failure probability. These SSCs may not be included in the PRA explicitly, but the boundary conditions under which the PRA is formulated make it clear that they are assumed not to fail. A probabilistically important SSC is one that in the context of the integrated plant system and the relative frequency of the challenges it faces, has a relative higher failure probability. These SSCs are, therefore, explicitly modeled in the PRA.

In this context, importance is initially defined as those SSCs that are necessary to maintain the current level of safety that is further defined by core damage prevention. Those SSCs necessary to core damage prevention are, therefore, defined as important; that is, those SSCs with the potential to impact the core damage frequency are identified as important.

It must be noted, however, that core damage mitigation is also an integral part of safety. That is, an SSC that is important to mitigation could be unimportant to core damage prevention. It would, therefore, be incorrect to classify an SSC as safety insignificant if it is important to core damage mitigation although unimportant to core damage prevention.

The relative importance of the SSCs necessary to core damage prevention can be determined from PRAs. The Increase Importance Measure is an excellent measure to use in defining different importance categories of SSCs. It provides a ranking of the events (i.e., SSCs) that are critical in maintaining the core damage frequency at its current estimation (i.e., maintaining safety at the current estimated level).[9] Therefore, those SSCs whose reliability and availability need to be closely maintained are identified by this measure. For example, in a graded quality assurance (QA) application, controls need to be assured for the relatively important SSCs so that their reliability and availability is not adversely impacted.

One definition for identifying the relative important SSCs can be based on the Increase Importance Measure. The relatively important SSCs can be defined as those whose Increase Importance Measure impact on the core damage frequency is greater than or equal to a factor of 10-to-100 (depending on the application) as shown by the following equation[10]:

$$\frac{\text{Increase Importance Measure}_{SSC}}{\text{Core Damage Frequency}_{plant}} \geq 10\text{-}100 \qquad \textit{Equation 4-1}$$

where

Increase Importance Measure$_{SSC}$ = Interval value estimation where core damage frequency is calculated with SSC value(s) set equal to 1.0.

This equation defines a relatively important SSC as one where, if its failure were certain, the core damage frequency would increase by more than a factor of 10-to-100. A relatively non-important SSC is, therefore, one where the core damage frequency would increase by less than a factor of 10 if its failure was always certain.

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[9]This measure provides the impact on the core damage frequency if an event's failure probability is 1.0; that is, it identifies how badly the core damage frequency is impacted if the availability and reliability of an SSC is degraded to the point where failure is certain.

[10]Equation 4-1 is the ratio value estimation of the Increase Importance Measure.

This definition is one suggestion for defining the relative importance. Whatever is used, the definition should include *all* the applicable failure modes of the SSC that are addressed by the application and not just, necessarily, only one of the basic events of the SSCs. In addition, the identification should not exclude an important core damage mitigative SSC.

For example, if an SSC's estimated Increase Importance Measure is 4E-4, its unavailability impact on a core damage frequency of 1E-5 is a factor of 40. For this example, the SSC would be classified as "relatively important" to core damage prevention. If its estimated importance measure is, however, 5E-5, its unavailability impact to the core damage frequency of 1E-5 is only a factor of 4. In this case, the SSC would be classified as "relatively non-important" to core damage prevention.

4.4.2 Importance Classification

The objective of the graded application is to define different categories of rule implementation based on the relative importance of an SSC. Based on the results of PRA, the plant's SSCs of concern can be identified and classified into different importance categories. These categories can be determined from either a "generic" perspective or from a "plant-specific" perspective.

Generic Importance Classification --

A generic classification of SSCs would be formed when, for example, plant-specific analyses are not available (e.g., generic data has been used in the PRA). It is also formed to provide a sense of completeness and account for the uncertainties and inherent subjectivity that is imbedded in a PRA. These SSC categories are determined based on PRAs of plants of similar design. Therefore, for each class of similar plants, relatively important and relatively non-important SSCs are identified.

Initially, similar designs could be considered one of the following:

- BWRs 1-4.
- BWRs 5&6.
- PWR Westinghouse.
- PWR CE.
- PWR B&W.

In using a graded approach for rule implementation, the plant's SSCs of concern would be ranked according to their importance to core damage prevention based on PRA information. Therefore, for each class of plants, the relatively important SSCs are those SSCs that have been found to be relatively important in <u>*any*</u> of these PRAs. For each of these SSCs, the ratio of their Increase Importance Measure to the core damage frequency is greater than or equal to a factor of 10-to-100 for at least one of the plants in that class. These SSCs, because of their relative importance to core damage prevention, would be subject to the current implementation in meeting the regulatory requirements. The remaining list of SSCs are then classified as relatively non-important since *no* PRA (of a similarly designed plant) identified any of these SSCs as relatively important.

This process can be illustrated by applying this graded approach, for example, to the implementation of 10 CFR 50, Appendix B, QA. In a graded QA implementation, only the SSCs identified as relatively important from a plant's Q list would be subject to the current implementation in meeting the QA regulatory requirements. The SSCs, however, identified as relatively non-important would be subject to a graded implementation in meeting the QA regulatory requirements involving some relaxation in what is now common for Q list items. This process is illustrated below in Figure 4.4-1.

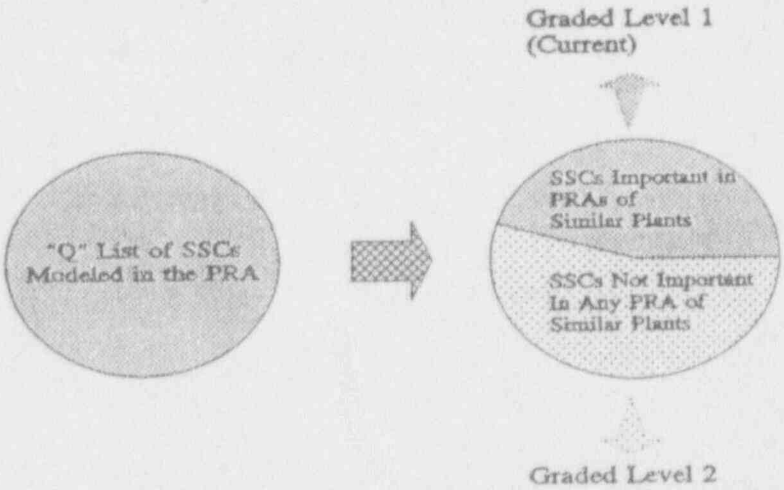

Figure 4.4-1. Generic Classification of SSCs for Graded QA.

For each class of plants, those SSCs that have been found to be relatively important in *any* of these PRAs would be subject to the current implementation in meeting the QA regulatory requirements. The remaining Q list SSCs, classified as relatively non-important since *no* PRA (of a similarly designed plant) identified any of these SSCs as relatively important, would be subject to a graded implementation of the QA regulatory requirements. Although these SSCs have been determined to be probabilistically unimportant to core damage prevention, they have been identified as deterministically important to core damage prevention; therefore, completely removing all quality assurance requirements would be inappropriate. This graded approach might focus on pre-operational functional testing, installation inspection, and compliance with recognized industrial procurement practices.

The initial identification of the relatively important and relatively non-important SSCs is performed based on probabilistic criteria. There are, however, SSCs that have been identified as deterministically important (i.e., identified as part of the "Q" list) but have not been modeled in the PRA. They have been determined to be probabilistically unimportant; that is, their failure probability is estimated to be negligible as compared to other SSCs. However, because of their deterministic importance (or because, if modeled they would have a large Increase Importance Measure) and unless appropriate justification is provided, they would still be classified as relatively important to core damage prevention and would be subject to the current QA implementation requirements. An example of a component in this category would be the reactor pressure vessel, which is usually not directly included in the PRA model after truncation.

Those SSCs of a plant that are not included on the Q list have been determined to be deterministically unimportant, and therefore, are currently not required to be subject to the current QA requirements. If one of these SSCs was identified as probabilistically important (that is, modeled in the PRA and determined to be relatively important), this SSC should be subject to QA requirements and subject to a graded implementation of the QA regulatory requirements in accordance with its importance

A plant's SSCs of concern have now been divided into two categories of SSCs. One category of relatively important SSCs where the current regulatory implementation is maintained. The second category of relatively non-important SSCs, however, will now be subject to a graded regulatory implementation.

Plant-Specific Importance Classification —

The initial classification of the relatively important SSCs is based on the results of PRA of plants of similar design and is a generic classification. There could, however, be plant-specific SSCs that are relatively non-important based on their plant-specific PRA. These differences could be due either to PRA reasons (e.g., boundary conditions or assumptions) or plant-specific design differences (e.g., core spray pump net positive suction head (NPSH) requirements differ). Plant-specific classification of SSCs can, therefore, also be performed.

Three categories of importance would be defined in the plant-specific classification. The first category would include those SSCs found to be relatively important by the plant-specific PRA. The second category would include those SSCs found to be relatively non-important by the plant-specific PRA, but a PRA of a similarly designed plant found them to be relatively important. This difference is due to PRA considerations (e.g., different assumptions); that is, the difference is ***not*** due to design differences between the plants. The third category would include, in this category, those SSCs found to be relatively non-important in plant-specific PRA but relatively important in PRA of a similar plant. The difference here, however, is clearly due to plant design differences. Note that "design differences" includes not only physical differences (e.g., pump NPSH requirements), but also technical and analytical differences. For example, Pump 'x' is found to be relatively important in Plant 'A' PRA while this same pump is found to be relatively non-important in Plant 'B' PRA. This difference is because Plant 'A' assumed a 4 hour room cooling dependence while Plant 'B' performed a technical analysis that confirmed a 24 hour dependence. In this type of situation, a design difference exists, and therefore, the third category includes those SSCs found to be relatively non-important in plant-specific PRA and in PRAs of similar plants.

In the second category of SSCs, these SSCs would not be subject to the current rule implementation but to a graded rule implementation. The implementation for these SSCs, however, would be more stringent than the implementation for those SSCs identified as relatively non-important (i.e., third category). As a possible example, when considering graded QA, these components might be subjected to most elements of the present QA program, but the need to maintain the "pedigree" of the component could be eliminated. Further requirement reductions might be obtained if it could be shown that commercially available equipment of this type met the expected reliability characteristics of the PRA.

The process of defining plant-specific SSC categories is illustrated in Figure 4.4-2.

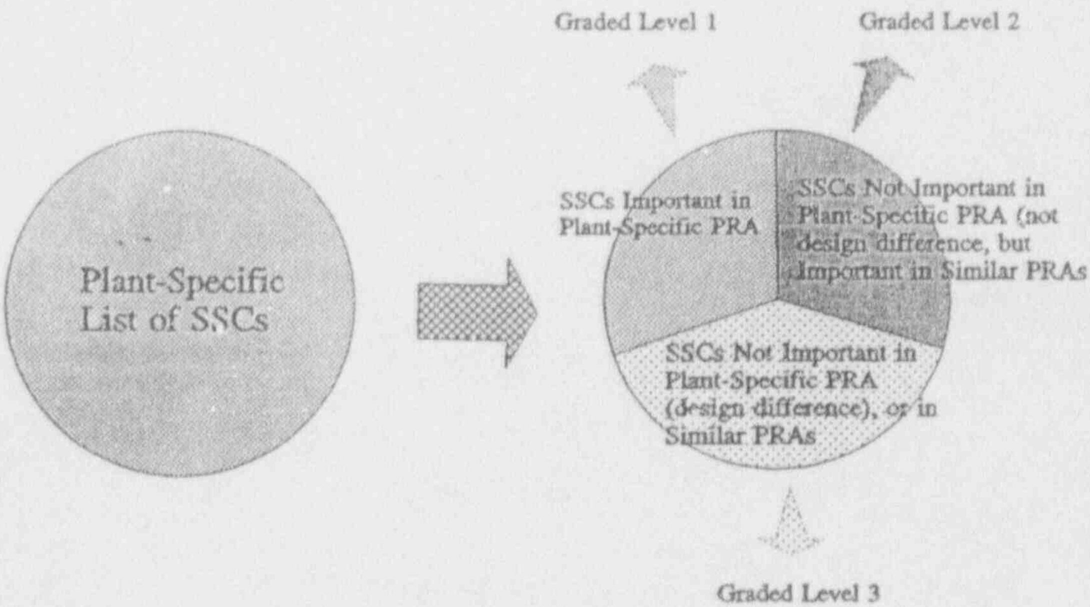

Figure 4.4-2. Plant-Specific Classification of SSCs for Graded Implementation.

4.4.3 Component Definition

In considering a graded approach for rule implementation, it must be remembered that the PRA definition of a "component" is different from, for example, a Q list's definition. Many of the items on a Q list either are not modeled in a PRA, or if modeled are not explicitly depicted in the PRA model. These items are ref^rred in the PRA as "parts."

In a PRA, if a component part is essential to the component function (as defined by the PRA), then the part is included in the component boundary. There may be parts, however, that are not essential for the component function even though the component has been identified as relatively important in the PRA. These parts would not be classified as relatively important and subject to the current implementation of the regulatory requirements. They would be classified as relatively non-important and subject to a graded implementation in meeting the regulatory requirements. The determination that a part is not critical to component function would be based on an engineering evaluation of the need for the piece part, considering the failure modes involved. For example, when considering graded QA, if an O-ring failure led to minimum leakage, but did not prevent functional performance, it could receive reduced QA coverage. This classification is illustrated below in Figure 4.4-3.

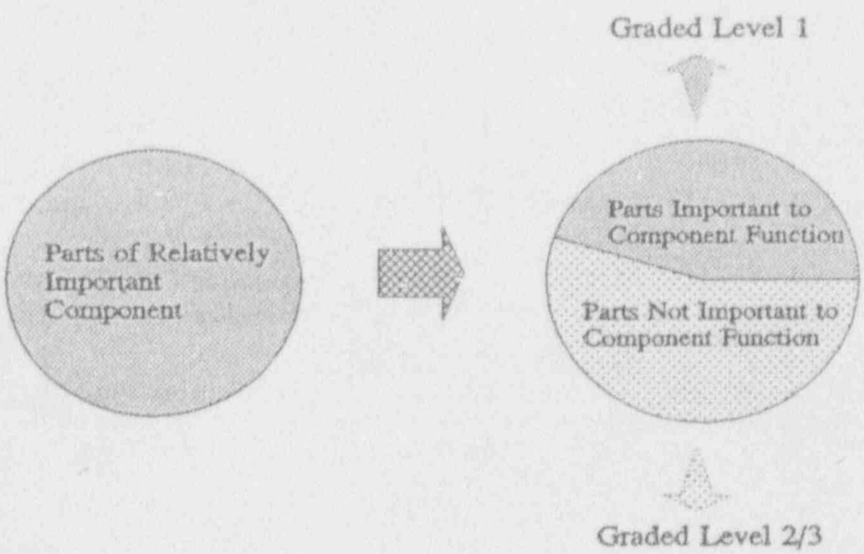

Figure 4.4-3. Classification of Component Parts.

4.4.4 Graded Implementation Requirements

Utilization of PRA for graded rule application potentially results in different levels or categories of importance of the SSCs depending whether a generic or plant-specific classification is used:

Generic Importance Classification —

- *Generic Category A* — Those SSCs that have been found to be relatively important to core damage prevention in a PRA of plants of similar design. Also included in this group are those SSCs that have been found to be deterministically important, but not probabilistically important; and those SSCs that have been found to be probabilistically important but not deterministically important.

- *Generic Category B* — Those SSCs that have not been found to be relatively important to core damage prevention in *any* PRA of plants of similar design.

Plant-Specific Importance Classification —

- *Plant-Specific Category A* — Those SSCs that have been found to be relatively important to core damage prevention in the plant-specific PRA. Also included in this group are those SSCs that have been found to be deterministically important, but not probabilistically important; and those SSCs that have been found to be probabilistically important but not deterministically important.

- *Plant-Specific Category B* — Those SSCs that have not been found to be relatively important in the plant-specific PRA, but have been found to be relatively important to core damage prevention in a PRA of plants of similar design. This difference is *not* due to design differences.

- *Plant-Specific Category C* — Those SSCs that have not been found to be relatively important to core damage prevention in *any* PRA (plant-specific and similar plants). Also included are those SSCs that have not been found relatively important in the plant-specific PRA, but have been found to be relatively important in a PRA of plants of similar design, but clearly due to design difference.

For each of these categories, the actual implementation of the rule needs to be defined. Detailed development may require a pilot study to explore the most efficient implementation strategy.

4.4.5 PRA Criteria

In using a PRA to identify relatively important SSCs and subsequently define different categories of relatively important SSCs, the PRA must be performed to certain standards. These standards, or criteria, address those boundary conditions associated with a PRA (discussed in Section 4.2).

In addition to the boundary conditions listed in Section 4.2 that require established criteria, there are several others that must also be addressed when considering the use of PRA in the regulatory process. These criteria address the updating of the PRA and the level of review of the PRA.

In performing a PRA, the time period involved is generally 2 to 3 years. The models developed as part of the PRA reflect the design, operation and maintenance of the plant typically at the start of the PRA. As the PRA is used, the potential, therefore, exists for the PRA to be outdated and not reflect the current core damage frequency estimation (i.e., current level of safety) of the plant since the design, operation, and maintenance of the plant does change. How often the PRA needs to be updated must be addressed when considering the PRA application. This issue can be divided into three different criteria as follows:

- *Outage Driven* — The PRA is updated at each plant refueling outage considering the plant design, operational, and maintenance changes.

- *PRA Driven* — The PRA is updated at the time of the plant design, operational, or maintenance change if the change has the potential to affect the PRA.

- *Real-time Driven* — The PRA is made "living" such that it continually reflects the status of the plant in real-time.

From a regulatory perspective, in considering the use of the PRA, the adequacy of the PRA for the identified use must be addressed. This determination will be based on the type and level of review that is performed by the NRC. The different levels and types of review that can be performed include the following:

- *Process* — The review primarily focuses on the methods, boundary conditions and assumptions of the PRA such that it can be determined that the SSCs important to core damage prevention are adequately addressed and identified in the PRA. Guidelines on the specific review criteria should be developed by the staff as part of any pilot study.

- *Detailed* — The review focuses on the accuracy of the core damage frequency estimation. The methods, boundary conditions, assumptions, scope, level of detail, models, and data of the PRA are reviewed. Guidelines on the specific review criteria should be developed by the staff as part of any pilot study where a detailed review is required.

PRA Criteria for Generic Importance Classification —

This categorization is basically determining relative importance of a plant's SSCs based on generic insights. Since only those SSCs that have never been shown to be relatively important in *any* PRA (of similar designed plants) receive graded implementation, generic types of criteria are adequate. The following is a list of suggested standards that are recommended if the Group 1 (Graded Implementation) generic PRA application is used.

An NRC review needs to have been performed of the plant-specific PRA of the licensee using the application. A review of the PRAs of the similar plants also needs to be performed. However, only a process-type review of these PRAs similar to that afforded to IPE submittals is needed for this generic application.

The PRAs need to have addressed at least internal events, including internal flooding. External events need to be addressed to the extent that a non-important SSC (as defined by Equation 4-1) combined with an external event remains non-important. It has been assumed in this report that the likelihood of occurrence of this combination is insignificant (i.e., < 1E-8); however, the PRA needs to address this issue by confirming this assumption. If the assumption cannot be confirmed, then the SSC initially identified as relatively non-important needs to be reclassified as relatively important.

The classification of relatively non-important SSCs is bounded by the level of detail of the PRAs. For an SSC of a plant's Q list to potentially be considered as relatively non-important, the PRAs need to have addressed these SSCs. Therefore, the SSCs not addressed by any PRA (or parts of any component modeled), but on the plant's Q list, are classified as relatively important until appropriate justification is provided to remove it.

The PRAs only need to have addressed the probabilistically significant failure modes for each classified SSC (as either relatively important or non-important). Hardware, test and maintenance, and common cause faults need to be evaluated as potential failure modes. Probabilistically significant can be defined as an unavailability or failure probability greater than or equal to 1E-5 at the component level.

The level of model resolution determines the degree of application of a plant's SSCs. To determine that an SSC may potentially be classified relatively non-important, then that

SSC needs to be explicitly represented in the model. For example, if an SSC is modeled in the PRA, but not explicitly represented, it should be classified as a relatively important SSC.

The use of generic data for the quantification of events failure rates and unavailabilities is adequate for this generic application.

HRA has the ability to impact the identification of the dominant sequences. Inadequate HRA could, therefore, erroneously result in identifying relatively important SSCs as relatively non-important. To preclude this possibility, the classification of the SSCs is performed with the HEPs for the various operator activities as follows:

- A screening value of at least 3E-2 must be used for pre-initiator human events (unless technical justification is provided for lower values).

- A screening value of at least 0.1 must be used for all response type post-initiator human events and a screening value of 0.5 for all recovery type post-initiator human events, with a bottom threshold value of 1E-4 for all post-initiator human events per accident sequence (unless technical justification is provided for lower values).[11,12]

The PRA quantification process may take advantage of truncation of low probability events, cut sets, or sequences. The truncation value must ensure, however, that at least 95 percent of the core damage frequency is captured. This truncation value may need to be reconsidered if the core damage frequency is dominated by a single SSC. Since a generic grouping is being performed, an uncertainty analysis is not required. Data and modeling uncertainties (particularly modeling assumptions) are offset by the grouping of results from similar plants.

The PRA needs to be current in regards to the design, operation and maintenance of the plant at the time of its application. Generally, updating the PRA at every refueling outage will provide this currency.

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[11] As used here, recovery actions refer to all post-initiator human actions outside the Emergency Operating Procedures for the plant (see Section 4.2.2 for definition of response versus recovery type actions). The lower threshold value should be applied in the Boolean combination of all human errors in a given accident sequence.

[12] The screening values are employed for initial screening of important SSCs from the unimportant. Further analysis is encouraged based on a thorough, documented HRA analysis.

PRA Criteria for Plant-Specific Importance Classification —

This categorization credits plant-specific differences for the relatively important SSCs. Those plant-specific SSCs that are determined from the plant-specific PRA to be relatively non-important are differentiated from the generic list of relatively important SSCs.

For a plant-specific SSC (that has been found relatively important in a PRA of a similar plant) to be classified in the generic category of relatively non-important, the reason for the difference must be because of plant design differences and not PRA differences (e.g., different assumptions).

The category of SSCs found non-important in a plant-specific study, not because of plant design differences, would be subjected to a graded implementation. The implementation, however, would be more stringent than for those SSCs that have been found to be relatively non-important in another PRA of a similar plant if some PRA of a similar plant has found this SSC to be relatively important and the difference is not due to plant design differences.

To categorize SSCs on plant-specific information, the criteria imposed on the plant-specific PRA is also more stringent. This stringency is applied to the data and truncation criteria. The other criteria are the same as for the generic application.

The data for those SSCs under consideration need to be based on plant-specific information. For example, if a specific SSC is determined as relatively important from a PRA of a similar plant, but this SSC is determined relatively non-important from its plant-specific PRA, the data used to estimate the plant-specific SSC's reliability and availability need to be based on plant-specific information.

In quantifying a PRA, it is natural to truncate low probability events, cut sets, or sequences. When this truncation is performed, the importance measures are only computed for those SSCs that are not truncated and do not consider the effect on the truncated portion. For a plant-specific SSC determined to be relatively non-important from its plant-specific PRA (although some PRA of a similar plant found it to be relatively important), the quantification of this SSC's importance measure needs to consider the effect of the SSC's unavailability and unreliability on the entire PRA model. In addition, any truncation value may need to be reconsidered if the core damage frequency is dominated by a single SSC.

4.4.6 Graded Type Applications

Appendix B to 10 CFR 50, states that *"the Quality Assurance program shall provide control over activities affecting the quality of the identified structure, system and components, to an extent consistent with their importance to safety."* A PRA provides a tool that can categorize the SSCs according to their relative importance to safety and, therefore, define different categories of QA implementation.

The graded QA implementation approach outlined above is but one example of fulfilling a regulatory request, a generic letter, etc. This type of approach — defining different categories of implementation for the SSCs commensurate with their relative importance — is not unique. For those regulations, generic letters, etc. where a ranking approach is appropriate to provide either gradations in the degree of response, or to prioritize the timing of the response, a similar process would be followed as illustrated below in Figure 4.4-4. Other examples for a graded approach of rule implementation could include the Maintenance Rule (10 CFR 50.65), or Inservice Inspection requirements of the ASME Boiler and Pressure Vessel code (10 CFR 50.55a).

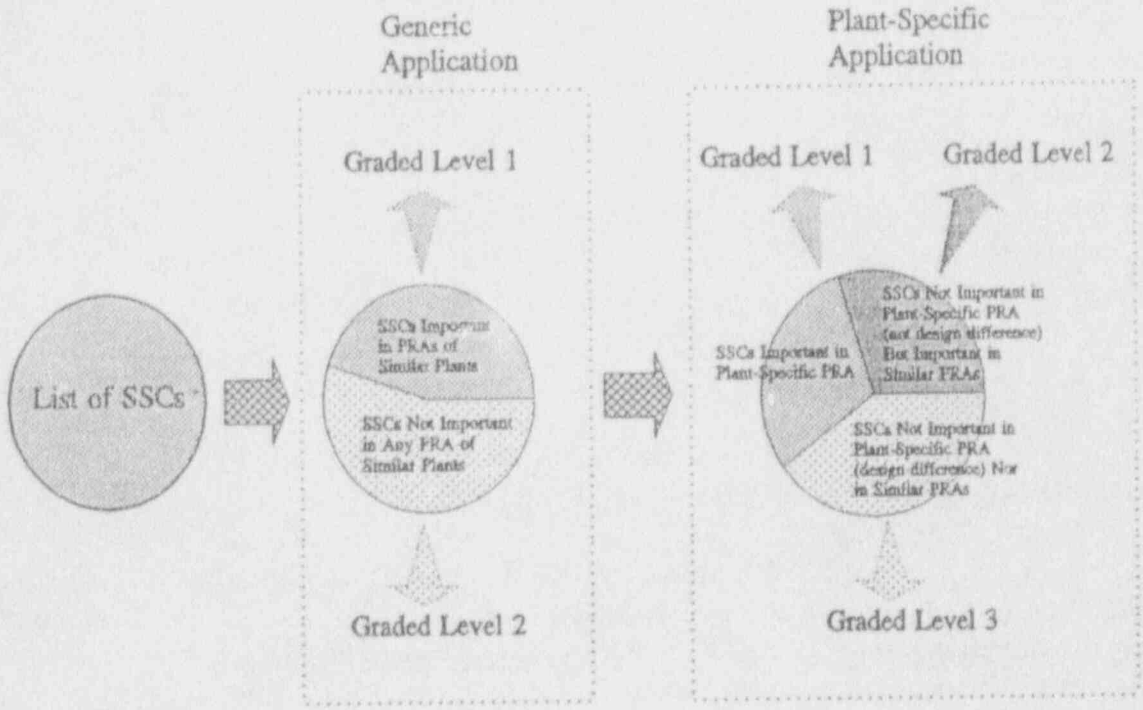

Figure 4.4-4. Graded Approach Process.

The criteria used in classifying relatively important and relatively non-important SSCs would be the same. In addition, the criteria established for the PRA would be the same.

The requirements for the various categories would need to be defined. These requirements should be commensurate with their relative importance.

4.5 PRA APPLICATION FOR CONFIGURATION ANALYSIS (Group 2)

The use of PRA in the regulatory process can involve plant-specific applications where reliance is now focused more on the numerical results of the PRA and manipulation of the PRA models. In addition, the application involves changes or modifications to regulations that are focused more to specific requirements. The plant configuration, the boundary conditions and assumptions modeled in the PRA, therefore, play a greater role in the application. One example of the type of application that would be classified in this group are changes to Technical Specifications.

A discussion of the types of applications and their associated criteria are provided below.

4.5.1 Configuration Application Supporting Technical Specifications

One aspect of the regulatory process involves Technical Specifications that, in a sense, control the configuration of a plant. The configurations are established by the AOTs associated with the limiting conditions of operation (LCO) in the Technical Specifications. The AOT defines that period of time that an SSC is allowed to be out-of-service before a plant shutdown is required. An "optimal" AOT can be estimated by considering the safety of continued operation versus manual shutdown or by also crediting the as-licensed plant conditions (i.e., design, operation, and maintenance). Both of these approaches are discussed.

The Technical Specifications also provide the surveillance test intervals (STIs) required for various plant SSCs. The surveillance test is performed to verify that important standby systems will function as demanded when required.

The AOTs and STIs are modeled in a PRA as they affect the SSCs availabilities. The AOTs control SSC unavailability due to maintenance by the specified AOT time. The STIs control SSC unavailability due to failures by limiting the fault exposure time. A PRA can, therefore, be used to optimize these conditions.

Configuration Analysis of AOTs Based on Simple Comparison of Continued Operation Versus Manual Shutdown —

Currently, Technical Specifications usually require a plant to shut down or take appropriate action when an AOT is exceeded. This requirement may, however, pose a challenge if the AOT applies to a system needed for shutting down or continued shutdown. Therefore, the required shutdown of the plant may present a greater safety concern than remaining at power for an additional amount of time. The core damage probability should then be evaluated for remaining at power versus shutting down when an LCO occurs to determine whether it is best to repair the SSC with the plant at power or in shutdown.

The probability of core damage occurring from remaining at power when an AOT is exceeded is computed by analyzing the specific configuration existing at the time using the PRA model. The PRA, however, estimates the core damage *frequency* for an average configuration and any possible event. The PRA model must then be modified to account for the specific configuration and quantified for the core damage ***probability*** of that specific configuration of plant components.

The likelihood of core damage occurring from continued operation is compared to the probability of core damage occurring from shutting down. This latter state is comprised of three phases. A core damage state could potentially occur during the period of shutting down, during the shutdown period, or during the period of starting up. Each of these phases should be evaluated and compared to the likelihood of a core damage state from remaining at power.

It can be assumed that the probability for a core damage state occurring during the period of shutting down is comparable to one of a manual shutdown with the specified equipment out-of-service. The potential accident sequences associated with a manual shutdown, or normal transient, are delineated in a PRA. Therefore, the core damage probability associated with a normal transient is computed considering an initiating event probability of 1.0.

For a temporary Technical Specifications relief on an AOT change, for a single line item AOT change, or for a group of several line item changes, the ratio of the core damage probabilities of continued operation to manual shutdown should be less than or equal to 1.0. This criterion is based on the requirement of the core damage probability of continued operation being less than the core damage probability of manual shutdown. In addition, an optimum AOT can be estimated. This AOT is optimized when the core damage probability of manual shutdown is equal to the core damage probability of continued operation. These concepts are illustrated by the following equations and in Figure 4.5-1:

$$CDP_{CO}/\text{day} \leq CDP_{MS} \qquad \textit{Equation 4-2}$$

$$\frac{CDP_{CO}/\text{day}}{CDP_{MS}} \leq 1 \qquad \textit{Equation 4-3}$$

$$AOT_{opt} = \frac{CDP_{MS}}{CDP_{CO}/\text{day}} \qquad \textit{Equation 4-4}$$

where CDP_{MS} = core damage probability of manual shutdown
CDP_{CO} = core damage probability of continued operation in specified configuration

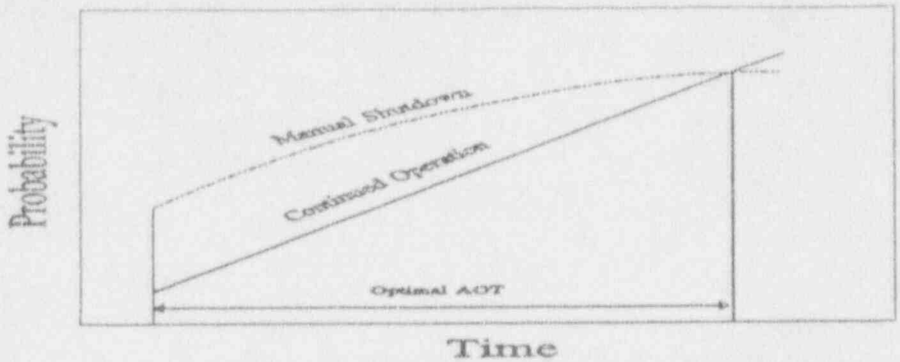

Figure 4.5-1. Comparison of Core Damage Probability of Continued Operation versus Shutdown.

It must be noted that in the optimization of an AOT, the lowest functional capability or performance levels of equipment required for safe operation of the facility must be maintained, 10 CFR 50.36(c)(2). Therefore, certain limits may be imposed in optimizing an AOT to account for the uncertainties associated with the data and physical phenomena imbedded in a PRA.

Configuration Analysis of AOTs Crediting As-Licensed Plant Conditions —

For a case involving a temporary Technical Specifications relief for an AOT change, or for pre-planning purposes regarding a single line item change or a group of several line item changes, the optimization of AOTs can also be estimated by crediting the as-licensed conditions of the plant by following a similar process. In this approach, consideration of the average annual core damage frequency is included. This frequency is the "accepted" core damage frequency associated with the plant based on its current design, operation and maintenance.

In this type of approach, the optimized AOTs for each SSC are still evaluated for different configurations. There could be a specific configuration, however, where the estimated core damage frequency exceeds the average core damage frequency. This increase would imply there is a temporary degradation in the current estimated level of safety, but this degradation can be controlled by managing the configuration. Management controls can involve imposing a strict limitation of the time the adverse configuration is permitted to exist, ensuring certain other SSCs are available, etc. It is, therefore, important to ensure that any individual AOT extension or set of extensions does not cause the current estimated average level of safety to be decreased. The probability per day of a core damage state from any extended AOT must then be equal to or less than the average core damage frequency.

Using this ground rule, the optimal AOT (i.e., maximum pre-determined AOT extension) for any single SSC can also be computed as follows:

$$AOT_{opt} = \frac{CDP_{MS}}{CDP_{CO}/day - CDF_{avg}} \qquad \textit{[Ref. 4-11]} \qquad \textit{Equation 4-5}$$

where CDF_{avg} = average core damage frequency

Note that if a train (not affected by the AOT) were successfully tested, showing evidence of continued operability, the core damage probability for continued operation would decrease, thus extending the AOT. In this manner, a family of AOTs could be calculated for a variety of train operability configurations. Recognition of these types of activities are an essential element of successful configuration management that is aimed at maximizing safety.

Figure 4.5-2 illustrates this concept of continued operation versus shutdown considering the average annual acceptable core damage frequency. In this approach, the optimum AOT would be greater than that estimated in the approach based on a simple comparison between manual shutdown and continued operation. Each plant with this approach would, in effect, be credited for the impact of their configuration, maintenance and surveillance practices.

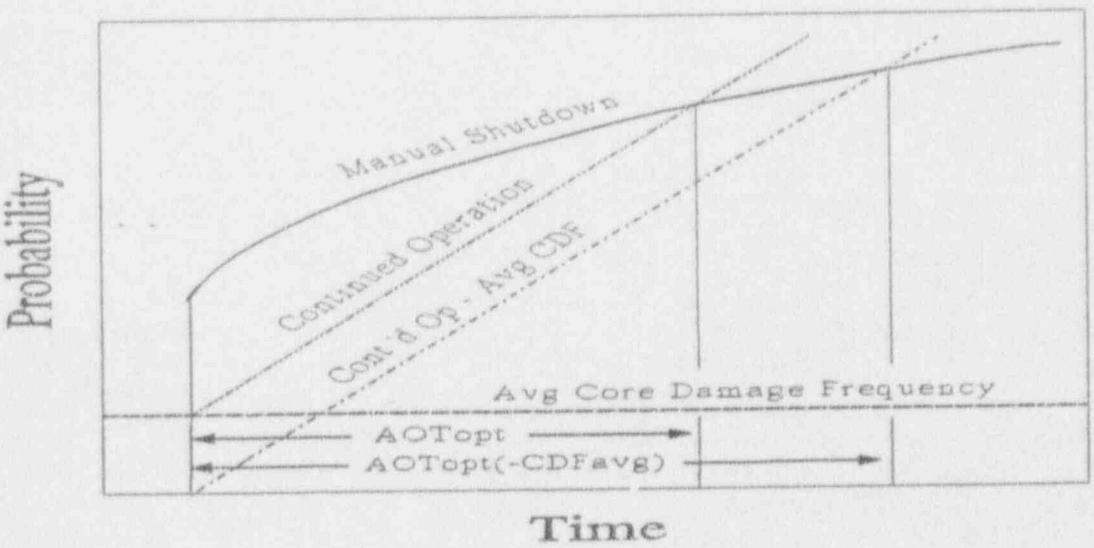

Figure 4.5-2. Comparison of Core Damage Probability of Continued Operation versus Shutdown Considering Acceptable Current Core Damage Frequency.

In either approach, the PRA is used to optimize the AOTs in the Technical Specifications. That is, for either a single line item or a group of several line items, the current Technical Specifications AOTs for SSCs are evaluated and probabilistically-based AOTs, considering plant configuration control, are added to the Technical Specifications for these SSCs. These new AOTs are specified for certain predetermined plant configurations that must be maintained; that is, configuration management that is aimed at maintaining the current level of safety is used to allow probabilistically-based AOTs for the current AOTs in the Technical Specifications. (See Section 4.8 for a discussion of the Torness Technical Specifications.)

As discussed in the first approach (i.e., AOTs based on simple comparison of continued operation versus manual shutdown), it must be reemphasized that in the estimation of these optimal AOTs, the lowest functional capability or performance levels of equipment required for safe operation of the facility must be maintained, 10 CFR 50.36(c)(2). Therefore, certain limits may be imposed in optimizing an AOT to account for the uncertainties associated with the data and physical phenomena imbedded in a PRA.

Configuration Analysis of STIs —

The Technical Specifications state the frequency at which standby components need to be tested. These requirements, however, have been argued to pose adverse effects on safety by causing plant transients or causing undue wearing of SSCs. A PRA can be used to optimize the STIs without affecting the current level of safety.

In considering an STI change, the core damage frequency based on the new STI needs to be compared to the core damage frequency based on the current STI.

As noted above, the STIs control SSC unavailability by limiting the fault exposure time. In a PRA, this unavailability has the following relationship:

$$Q = f(q_d, q_{t/m}, \lambda, T) \qquad \textit{Equation 4-6}$$

where

Q = component unavailability probability
q_d = component unavailability due to demand failures
$Q_{t/m}$ = component unavailability from test and maintenance (i.e., out of service)
λ = component failure rate
T = component STI

Based on the above equation, if the STI for a component were increased, it can easily be seen that the probability of the unavailability of the component, not the failure rate, is increased. Conversely, the opposite is true. If the STI were decreased, the unavailability is decreased. Therefore, if the availability of a two train system, for example, were to remain "safety neutral," and if one train is failed, the availability of the second train needs to be improved to keep the system availability "constant." One way to achieve this neutrality is by increased surveillance testing of the second train.

Using the average core damage frequency as the upper limit, one approach to optimizing STIs is to investigate functional availabilities rather than the availability of a single component. In this approach, the functional availability is held constant while manipulating the availabilities of the systems comprising the function. That is, the STIs for some of the SSCs could be increased but decreased for others with the availability of the function remaining constant. The same would apply for either a system or train. The system or train availability would, therefore, be held constant while manipulating the availabilities of the trains and components comprising the system or train. These applications would require specifications at the functional, systemic, or train level to remain constant.

In this type of application, the availability for either the function, system, or train would be established from the average core damage frequency based on the current STIs. Criteria similar to that defined for graded implementation (see Section 4.4.1) can also be used here to identify relatively important and non-important components; and therefore, identify the candidate components for increasing STIs, and identify the components where the STIs should not be changed. For example, the STIs for the relatively non-important SSCs could be increased, since the limits on the relatively important SSCs would control, and the current estimated safety level would not be impacted.

In the above application, the safety level is not impacted because either the overall function, system, or train availability is not changed. The STIs can also be manipulated with the safety envelope unchanged without maintaining the function, system, or train availability constant. Other compensatory measures could be proposed to offset any increased STI that would maintain the current level of safety.

It is noted that the situation is more complex than addressed here. The unavailability is a function of not only the time-dependent failure rate but also of the demand stresses placed on the system. In evaluating STIs, attention should be given to the root cause analyses of plant-specific failure to properly evaluate the effect of STIs.

4.5.2 PRA Criteria

In using a PRA to optimize AOTs and STIs, the PRA must be performed to certain standards. These standards (or criteria) address those boundary conditions associated with a PRA (discussed in Section 4.2). The following is a list of suggested standards that are recommended if Group 2 (Configuration Analysis) PRA application is used.

The PRA needs to have addressed at least internal events, including internal flooding. External events need to be addressed to ascertain the impact of an external event on the SSC under evaluation. That is, the occurrence of an external event combined with the failure probability or unavailability (as appropriate) on the core damage frequency needs to be considered. It has been assumed in this report that the likelihood of occurrence of this type of combination is insignificant (i.e., <1E-8); however, the application needs to address this issue to the extent of confirming this assumption. If the assumption cannot be confirmed, then the application needs to address the potential core damage frequency change accounting for the external event.

Proposed AOT and STI changes for SSCs are bounded by the level of detail of the PRAs. For a change to be considered, the PRA needs to have addressed these SSCs.

The failure modes that characterize the AOT and STI for the SSCs under consideration need to be included in the PRA model.

The level of model resolution determines the degree of application of a plant's SSCs. To determine the impact of changing an AOT or STI of an SSC, that SSC needs to be explicitly represented in the model.

The use of generic data for the quantification of events failure rates and unavailabilities is adequate for most of the SSCs. For the SSCs involving single line item type reliefs, plant-specific data are required.

HRA has the ability to impact the identification of the dominant sequences. Inadequate HRA could, therefore, erroneously result in identifying relatively important SSCs as relatively non-important. To preclude this possibility, the classification of the SSCs is performed with the HEPs for the various operator activities as follows:

- A screening value of at least 3E-2 must be used for pre-initiator human events (unless technical justification is provided for lower values).

- A screening value of at least 0.1 must be used for all response type post-initiator human events and a screening value of 0.5 for all recovery type post-initiator human events, with a bottom threshold value of 1E-4 for all post-initiator human events per accident sequence (unless technical justification is provided for lower values).[13,14]

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[13]As used here, recovery actions refer to all post-initiator human actions outside the Emergency Operating Procedures for the plant. The lower threshold value should be applied in the Boolean combination of all human errors in a given accident sequence.

The PRA quantification process may take advantage of truncation of low probability events, cut sets, or sequences. The truncation value must ensure, however, that at least 95 percent of the core damage frequency is captured. If truncation is performed, the quantification of importance measures, for the single line SSCs needs to consider the effect of the SSC's unavailability and unreliability on the entire PRA model. In addition, the truncation value may need to be reconsidered if the core damage frequency is dominated by a single SSC. Since this type of application relies on the numerical results of the PRA, the uncertainties associated with the data need to be addressed. The mean value, therefore, needs to be estimated for the core damage frequency.[15]

The PRA needs to be current in regards to the design, operation and maintenance of the plant at the time of its application. Generally, updating the PRA at every refueling outage will provide this currency.

An NRC review needs to have been performed of the plant-specific PRA of the licensee using the application. A process type review for the majority of the PRA is adequate for this type of application; however, focus needs to be particularly emphasized in the data area regarding the computation of the core damage frequency value. Guidelines on the specific review criteria should be developed as part of any pilot study.

4.5.3 Other Configuration Analysis Applications

The approach outlined above for Technical Specifications is but one example of using a PRA where the model and numerical results are used to change or modify specific regulatory requirements. This type of approach is not unique.

10 CFR 50.55a states that *"Structures, systems, and components must be .... tested .... to quality standards commensurate with the importance of the safety function to be performed. Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code...."* In implementing, for example the inservice testing requirements of the ASME Boiler and Pressure Vessel Code, this type of application can be applied. For those regulations where a configuration analysis is appropriate, a similar process would be followed.

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[14](...continued)

[14]The screening values are employed for initial screening of important SSCs from the unimportant. Further analysis is encouraged based on a thorough, documented HRA analysis.

[15]The mean value is not estimated by using mean values for the basic events and then treating them as point estimates in calculating the core damage frequency. The distribution of each basic event is propagated in the quantification process to estimate a true mean value for the core damage frequency.

The criteria used in classifying relatively important and relatively non-important SSCs would be the same. In addition, the criteria established for the PRA, in a configuration nalysis type approach, would be the same. The lowest functional capability or performance levels of equipment required for safe operation of the facility must still be maintained in any process or approach used. Therefore, certain limits may still need to be imposed to account for the uncertainties associated with the data and physical phenomena imbedded in a PRA.

4.6 PRA APPLICATION FOR ON-LINE CONFIGURATION CONTROL (Group 3)

PRA, at its most optimum, can be used in a living manner. In this type of application, the probability of a core damage state is computed in real-time and plant decisions — operational, maintenance, etc. — are made based on the core damage probability. This applications would essentially replace the current concept of LCOs in 10 CFR 50.36.

A real-time computation of the core damage probability would mean that the PRA model and the entire PRA process is computerized such that the PRA inputs can be manually or automatically fed into the PRA model for the plant. Therefore, at any given time, a core damage probability for the plant is known. A system would then need to be designed and implemented that could perform this task.

In this application, some plant safety decisions would be made based on a calculated core damage probability. A baseline core damage probability (or upper limit) would be established, and the plant would be designed, operated, and maintained within this baseline. Therefore, the absolute value of the core damage probability becomes critical. A standardization for PRA regarding such items as boundary conditions, assumptions, scope, level of detail, etc., would need to be established and uncertainties resolved.

Since the PRA would be used to regulate the plant, assurance would need to be provided of both the adequacy and accuracy of the PRA and the PRA supporting software and hardware. Also, the real-time input of plant conditions to the PRA computer model would have to meet standards of acceptance. This provision would need to occur at all levels; therefore, requirements, audits, and inspections would more than likely be required at each level.

Development of systems such as this are within the state of the art. Efforts are well under way to implement the capability to evaluate the instantaneous risk level on close to a real-time base in the U.S. and in other countries, and operational systems have been functioning for several years in the United Kingdom. These systems can be an excellent aid to the plant operators and provide the analytic capability to make those plant analyses suggested by the Group 1 and Group 2 type applications relatively easy. However, although PRA can provide valuable insights, and a living-PRA can be a tremendous asset to the internal operations of a licensee, it is felt that the state of the art of PRA will ***not*** currently support this type of ***regulatory*** application outlined for on-line configuration control, noted as Group 3 in this report.

4.7 RELATIVE IMPORTANCE OF REGULATIONS

The objective of this effort is to assess the consistency of regulations with the safety goals by determining the feasibility of estimating the relative "safety importance" of regulations considering their importance to public safety and health.[16] This section is a brief summary of the effort. Appendix A to this volume provides a detailed discussion of the effort and results.

4.7.1 Background

The Regulatory Review Group was tasked with conducting a detailed review with special attention *"placed on the feasibility of substituting unnecessarily prescriptive requirements and guidance with performance-based requirements and guidance founded on risk insights."* PRA is an excellent tool to identify those rules and regulations that need to be prescriptive in nature versus those that should be performance-based. This identification can be performed by determining the relative impact of a rule to safety. For example, as a rule becomes more safety important, the rule would be more prescriptive; therefore, the rule would become more detailed and comprehensive in its specific requirements. Conversely, as a rule has less safety importance, it would be more performance oriented; that is, the rule would be less detailed and less comprehensive in its specific requirements. This philosophy is illustrated in Figure 4.7-1.

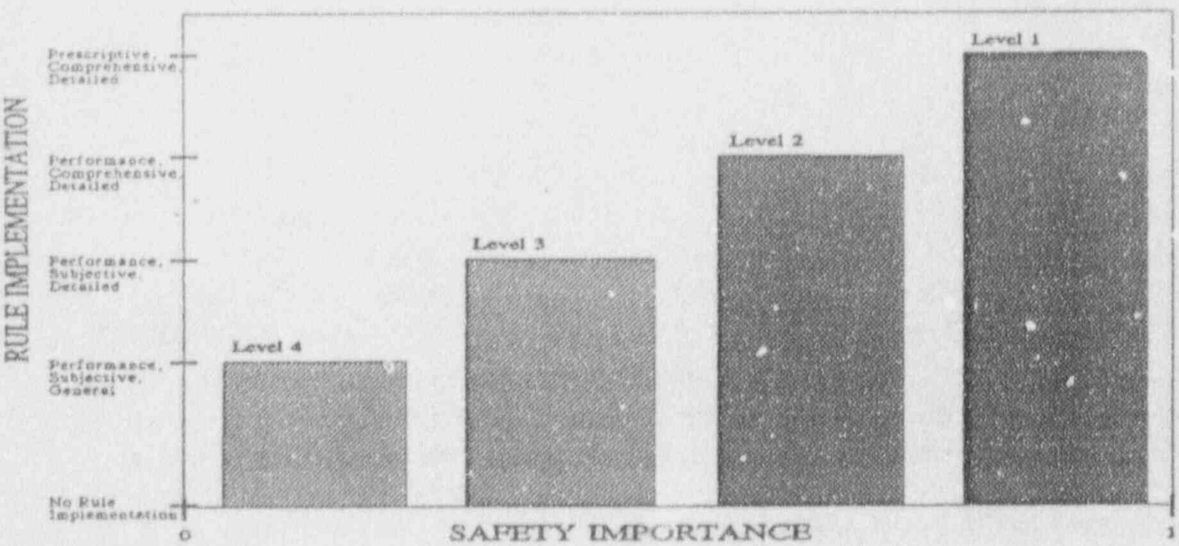

Figure 4.7-1. Example of Prescriptive-Based versus Performance-Based Rule Relative to Safety.

[16] This effort was performed with the assistance of Idaho National Engineering Laboratory.

To determine when a rule should be prescriptive versus performance-based, an approach that can *quantify* a rule's impact on safety needs to be developed. The effort discussed in this section describes the feasibility study that was performed to determine an approach that quantifies (in a subjective and approximate manner) a rule's impact on the design, maintenance and operation of a plant using PRA techniques. That is, the degree to which a rule could potentially impact the reliability and availability of those SSCs and human actions important to safety was examined. This approach, therefore, would both determine the overall impact to safety by a rule and identify which SSCs and human actions are affected by the rule. These results would then serve as an aid in determining whether a rule should be prescriptive or performance based, the degree to which it should, and provide insights towards the affected or non-affected SSCs and human actions.

4.7.2 Technical Approach

To determine the potential impact by a rule on the safety of a plant, an approach was developed that examined the potential effect of a rule on a plant's core damage frequency. The Surry and Peach Bottom NUREG-1150 models were used to determine this impact. The development of the approach involved the following basic tasks:

- Sample regulations were selected for use as "test cases."

- Specific changes to the selected regulations were hypothesized to determine their potential safety impact.

- A framework was developed to systematically relate regulations to PRA model changes.

- The impact of the hypothetical regulation changes on the plant design, operation and maintenance was determined using expert elicitation.

- The results of the expert elicitation was propagated through the NUREG-1150 PRA models to determine the impact on the core damage frequency.

As a feasibility study, a limited number of hypothetical regulatory changes were investigated, and an analysis of the core damage frequency impact was performed for each change.

A regulatory change could lead to changes in the input parameters (e.g., component availability and reliability) of the PRA or changes to the PRA model (e.g., system fault tree) itself to account for new failure paths or the elimination of safety barriers. For this feasibility study, it was necessary to focus on the most core damage significant components, systems, initiating events and human actions in the PRA; structural changes were not investigated. The impact of changes in the characteristic parameters for these components, systems, initiating events and human actions were then quantified using the IRRAS 4.0 code.

Selection of the Sample Regulations —

A classification scheme was used to group and evaluate the 10 CFR 50 regulations to generate a sample of regulations for use in this feasibility study. This classification scheme employed a number of variables that characterized the impact a given regulation can have on a PRA model. In particular, they indicate the mechanism by which the regulation can impact the PRA model and the scope (extent) of this impact. The variables are binary and are defined as follows (the possible values are indicated in the brackets):

<u>Mechanism of Impact</u>

X_1:	Directness of impact mechanism	[Indirect/Direct]
X_2:	Potentially affects numerical values of PRA model parameters	[Yes/No]
X_3:	Potentially adds new parameters to PRA model	[Yes/No]
X_4:	Potentially removes parameters from PRA model	[Yes/No]

<u>Scope of Impact</u>

X_5:	Impact extent	[Localized/Pervasive]
X_6:	Potentially affects PRA dominant sequences	[Yes/No]
X_7:	Potentially affects non-dominant sequences in PRA	[Yes/No]
X_8:	Potentially affects systems/components/failure modes not in PRA	[Yes/No]

By rating the 10 CFR 50 regulations with these variables, 14 natural groupings of rules were identified. A representative sample set of four regulations were chosen for this study:

- 10 CFR 50.62 (The ATWS Rule).
- 10 CFR 50 Appendix B (The Quality Assurance Rule).
- 10 CFR 50.120 (The Training Rule).
- 10 CFR 50.65 (The Maintenance Rule).

The impact of the ATWS Rule (10 CFR 50.62) is in a grouping that has a direct impact on the PRA model, potentially affects model inputs, and may remove parameters in the PRA. The scope of the rule is considered local and changes would impact both dominant and non-dominant sequences. As an additional point the rule was chosen because its core damage significance is expected to differ for BWRs and PWRs. It also provides a case where a number of detailed PRA studies have been done to analyze alternative strategies for compliance.

The Quality Assurance Rule (10 CFR 50 Appendix B) and the Maintenance Rule (10 CFR 50.65) are in a grouping that has a direc impact on the PRA model and potentially affects model inputs. They have a pervasive effect on numerous components, systems, and human actions in the plant. Appendix B has been in effect since 1970 while the Maintenance Rule has only recently been adopted and has not yet been completely implemented.

The Training Rule (10 CFR 50.120) is still in draft form. It will be in a grouping that has a direct impact on the PRA model, potentially affects model inputs, and may add parameters to the PRA. It will have a pervasive impact on numerous components, systems, and structures in the plant. Changes in this regulation will impact the numerical input values of both the dominant and non-dominant sequences in the PRA.

Identification of Regulation Changes —

The regulation changes were developed in a manner to illustrate typical ("average") variations in safety significance variations between the regulations. The identified changes to the regulations to determine a rule's impact were as follows:

- 10 CFR 50.62 (The ATWS Rule) — Determine the impact of implementing the ATWS Rule as if the regulation had never existed. The average plant response was considered different for BWRs and PWRs.

- 10 CFR 50 Appendix B (The Quality Assurance Rule) — Determine the impact of the rule on general component reliability, first by examining the effect if the regulation was eliminated and second examining the effect if the regulation had not existed.

- 10 CFR 50.120 (The Training Rule) — First, determine the impact of total implementation of the Training Rule. Secondly, determine the impact of the industry implementing training that met the intent of the training regulation but without having a formal regulation.

- 10 CFR 50.65 (The Maintenance Rule) — Determine the impact of total implementation of the Maintenance Rule.

Modeling Framework —

The largest technical difficulty i :e study was associated with developing credible linkages between a given regula..on and the PRA model itself. The framework adopted for identifying these linkages is shown in Figure 4.7-2. Basically there are only four classes of model parameters that impact the results of a risk assessment model: initiating event frequency changes, component availability and reliability changes, human action probability changes, and changes to PRA model structure. The changes in these parameters associated with a given rule change were determined using an expert elicitation process. This process was designed to make the experts consider multiple mechanisms by which a particular parameter might be affected.

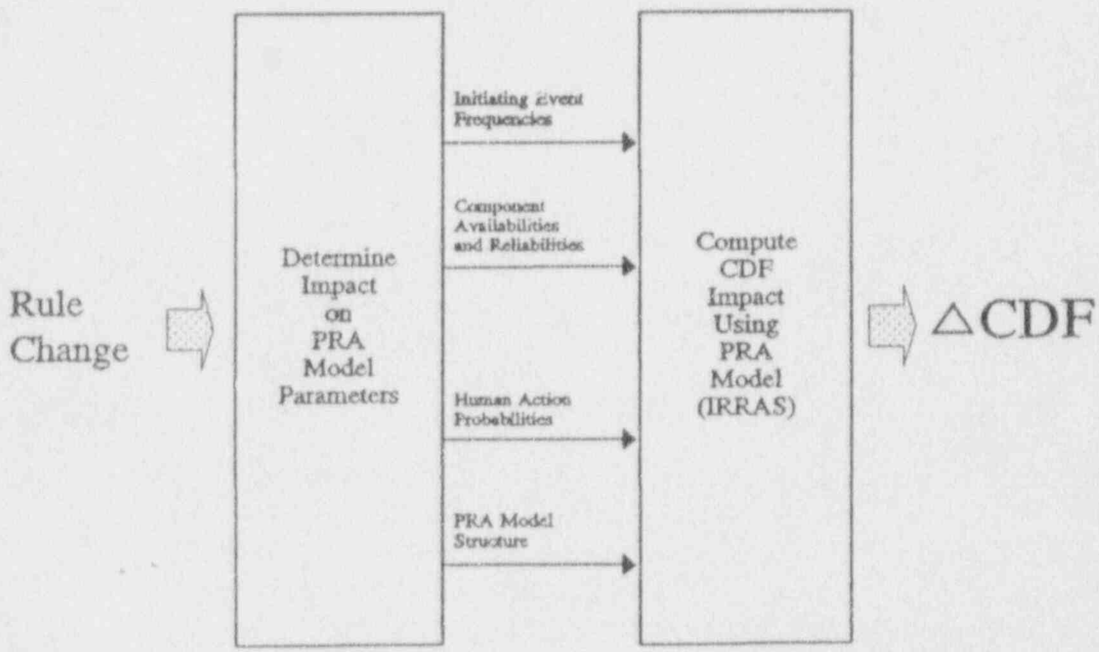

Figure 4.7-2. Modeling Framework.

To reduce the number of parameters elicited, importance analysis was used to focus on those parameters which had the greatest impact on the core damage frequency. These parameters are summarized in Table 4.7-1.

Table 4.7-1
Example Summary of Parameters Elicited

- Availability of scram system (e.g., RPS and ARI for BWR).
- Ability to achieve reactor subcriticality (for BWRs).
- Operator likelihood to initiate emergency actions (e.g., SLC for BWRs).
- Frequency of general plant transients and support system plant transients.
- Availability and reliability of active valves, diesel generators, turbine driven pumps, motor driven pumps, and batteries to perform as needed.
- Likelihood of proper actions by non-licensed operators.
- Likelihood of latent errors by maintenance personnel.

Importance measures were systematically developed for all PRA model parameters using the IRRAS 4.0 code. These measures are useful for predicting the effect of a change in a single PRA model parameter (e.g., a failure rate) or of small changes in a number of parameters. However, in this project, the simultaneous impact of a single regulation change on multiple basic events must be evaluated. Moreover, if the risk impact is significant, it can be expected that the magnitude of change in a given parameter may be one or more orders of magnitude greater than the original parameter value. Although importance measures for multiple basic event variations can be developed, it is more direct and convenient to simply recompute the PRA model (i.e., to requantify the core damage frequency) using available computer software.

Quantification of the Impact of the Regulation Changes --

For each regulation change discussed above, the following steps were performed.

1. The specific scope of impact (i.e., which components, human actions, initiators, and basic events are affected by the regulation change) was determined and the qualitative impact documented in a summary discussion for the expert elicitation.

Once the impacts were identified, a walk-through example was presented by a normative expert for each elicitation that illustrated how the regulatory change affects safety. For example, in the case of quality assurance, the discussion could cover the different ways that QA impacts the procurement, installation, and testing of equipment.

2. The mechanism of the impact (i.e., what PRA parameters might be modified) was determined for the rule change, and elicitation questions were developed.

   For each regulatory change, selected PRA parameters in the dominant sequences were grouped. The elicitation questions for each regulation were designed to directly address these PRA dominant sequence groups.

3. Qualitative measures of the expected change in groups of PRA parameters as a result of the benefit or elimination of selected regulations were elicited.

   Both the direction and magnitude of change were elicited. The magnitude elicitation was limited to a descriptive scale (obvious, noticeable, subtle).

   Direction of the change was indicated by an increase, decrease, or no change.

   No change. This implies there is absolutely no relationship or correlation between the regulation and the PRA parameters. If no change is anticipated then the magnitude is not questioned.

   Magnitude of the increased or decreased change was elicited regarding whether the increase or a decrease in the parameter (e.g., component availability) was subtle, noticeable or obvious.

   Subtle. A subtle change is anticipated that implies a long period of time is required to discern any change (e.g., the trend if it exists is within the range of random fluctuations in data).

   Noticeable. A noticeable change is anticipated that implies any change will be detected over time.

   Obvious. An obvious change is anticipated that implies any change will be immediately observable (i.e., marked and dramatic).

4. Quantification was accomplished by transforming the descriptive scale (obvious, noticeable, subtle) into numerical values. In this case, the descriptions were transformed into values based on the judgment of PRA experts.

5. The transformed numerical values were combined into a single factor (estimator) for each question; this factor was used to modify the dominant sequence groups, and then the PRA was requantified.

One of the weaknesses in this work concerns the connection between the qualitative data and the development of quantitative values. This is because each expert may have a differing opinion of the numerical values associated with their qualitative responses. Despite this weakness, it is expected the output of the estimation process is useful (in this feasibility study) to indicate the safety significance of these regulations.

4.7.3 Expert Elicitation Technique

To evaluate the magnitude of PRA model parameter changes used as inputs to the re-computation effort an expert elicitation was used. An expert panel of individuals knowledgeable of the regulations and their implementation at a plant relative to the design, operation and maintenance (i.e., their impact on system, component and human performance) was essential. This panel of experts was assembled consisting of two senior level nuclear utility managers (with expertise in operational safety assessment), an NRC Senior Resident Inspector, an NRC Senior Licensing Project Manager, and a Senior Manager from the NRC Office of Research. A formal training and normative session were held, and the experts were formally polled on the direction and magnitude of regulatory driven changes in key parameters impacting the PRA models.

The Nominal Group Technique was used for the expert elicitation in this feasibility study. The selection of this technique was based on considerations of available resources to demonstrate feasibility, expert estimation theory, and past experience with other methods.

The knowledge and expertise of experts was captured in pre-meeting preparation and information gathering and in the problem definition portion of the elicitation session itself. Problem decomposition into components was used to assist the development of each expert's final estimate.

The Nominal Group method employed no effort to obtain consensus judgments between experts. This structure, along with the facilitator's direction of the discussion, was designed to minimize bias due to domination of the group's thinking by any individual.

The expert elicitation session began with a brief introductory period in which participants were introduced, roles explained, and agenda reviewed. This was followed by an elicitation training session whose major focus was to introduce participants to the elicitation processes and to show them the importance of remaining open to new information as it becomes available. It also was used to introduce the scales to be employed.

The elicitation training was followed by a normative session in which general issue statements for each rule were introduced, and an overview of the linkages between plant functions and reliability was explained. Each issue statement summarized the objectives of elicitation, note background information provided, and defined the suggested baseline plant to be used. Issue statements also defined direction and magnitude of the changes to be elicited, explained the attached elicitation tables, and defined the suggested primary impacts of each rule.

For each rule change elicited, experts were asked to judge both the direction and the magnitude of the impact (i.e., would the rule change increase, decrease, or not change reliability and would the impact be subtle, noticeable, or obvious). Specific components, systems, and functions to be considered were specifically named on the elicitation tables provided.

The experts were privately asked first to write their own estimates of the impacts of rule changes and the basis for those impacts on the forms provided. After all experts had completed making their estimates, they were asked in random sequence to disclose and explain their initial estimates to the rest of the group. The experts were then given the opportunity privately to change their estimates and to provide any additional reasoning on the provided forms. For each rule, a brief closing discussion was held.

For the expert elicitation, the qualitative results were summarized for each regulation as shown in the sample expert elicitation table shown in Table 4.7-2. There was excellent agreement among the experts concerning the direction of the regulation change impacts, and reasonably good agreement concerning the magnitude of impacts.

Table 4.7-2
Summary of Example of Expert Elicitation

	ELICITATION QUESTION/EXPERT RESPONSE	EXPERT A	EXPERT B	EXPERT C	EXPERT D	EXPERT E
	10 CFR 50.62 BWR ELICITATION					
1	Has the availability of the automatic scram system (RPS, ARI) increased or decreased?	Subtle Increase	Noticeable Increase	Subtle Increase	Subtle Increase	Expert not elicited
2	Has ability to achieve reactor subcriticality (via boron injection and RPT) increased or decreased?	Noticeable Increase	Subtle Increase	Noticeable Increase	Subtle Increase	Expert not elicited
3	Has the likelihood of the operator to initiate SLC (given an ATWS event) increased or decreased?	Noticeable Increase	Noticeable Increase	Obvious Increase	Noticeable Increase	Expert not elicited

4.7.4 Results

As described in Section 4.7.2, this feasibility study investigated four hypothetical regulatory changes and analyzed the risk impact from each change. The method used to quantify the impact of regulation changes was based on identifying appropriate changes to the PRA model, eliciting qualitative measures, and then quantifying the impact of these changes as described above.

After the expert elicitation, the qualitative results were summarized as shown in the sample expert elicitation table shown in Table 4.7-2. The qualitative results were then transformed into quantitative values and these numerical values were combined into a single factor (estimator) for each question. To transform the qualitative results, the descriptive scale for both availability and frequency of an event was given a quantitative value as follows:

	Frequency	Availability or Likelihood
No Change	1.0	1.0
Subtle Change	1.2	2.0
Noticeable Change	2.0	5.0
Obvious Change	3.0	10.0

The PRA experts based these values on the knowledge of what constitutes a magnitude of change in the PRA. Each factor was then used to modify the dominant sequence groups of events from the dominant sequences, and the PRA was requantified to yield a new core damage frequency. The results (as shown in Figures 4.7-2 and 4.7-3 and given in Table 4.7-3) demonstrate that the methodology can distinguish between core damage impacts of different regulations. The experts elicited determined that the implementation of each regulations had some positive effect in reducing core damage frequency. It is important to note that the experts were careful to indicate that it is difficult to determine all the combined influences on core damage from other policies, regulations, and general knowledge. It was beyond the scope of this elicitation to investigate the interaction of combinations of regulations and other policies. The short time frame only allowed for preparation of both the elicitation questions and background on each regulation at a gross level of detail. It is felt, however, that an expanded elicitation could provide the experts guidelines as to how to determine the contribution from each specific regulation, or in some instances, combinations of regulations and other policies as appropriate.

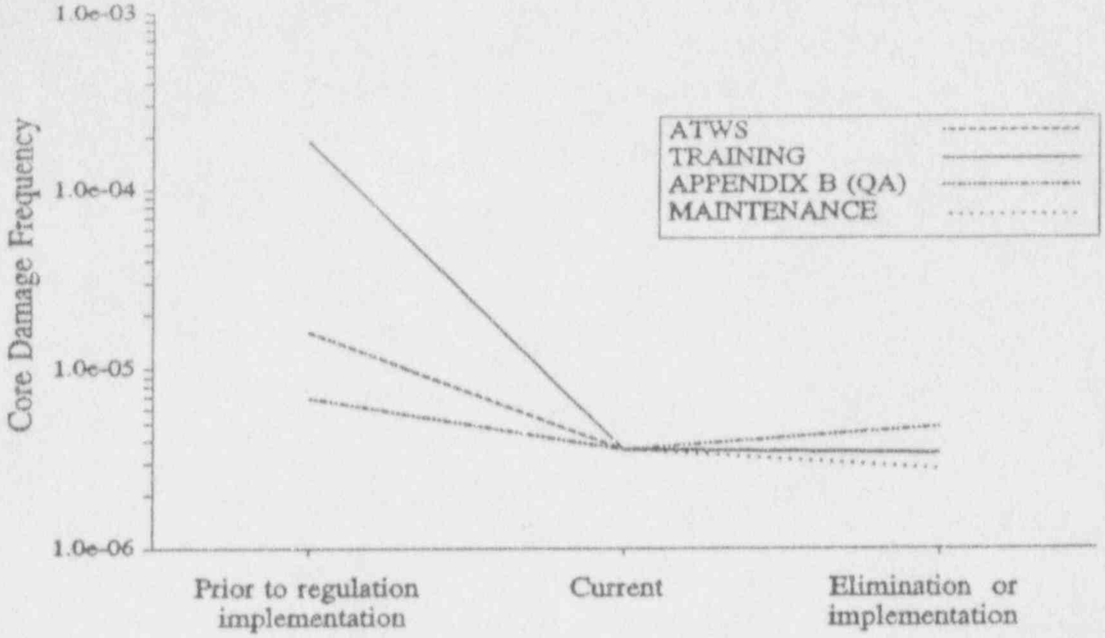

Figure 4.7-2. Resultant BWR Core Damage Frequencies Due to Changes in the Regulation.

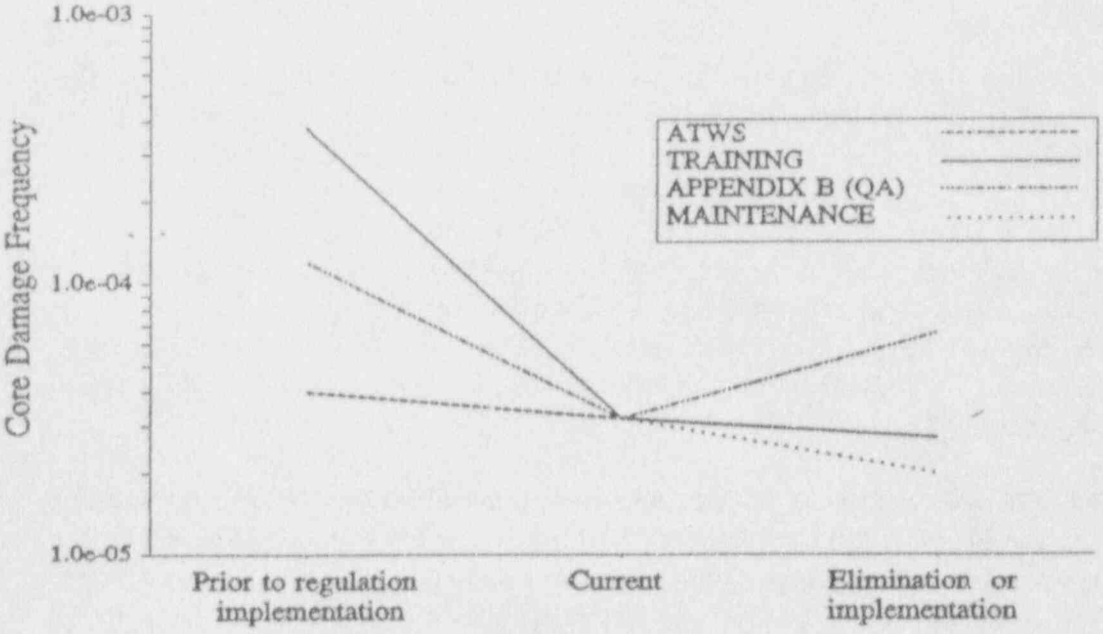

Figure 4.7-3. Resultant PWR Core Damage Frequency Due to Changes in the Regulation.

Table 4.7-3
Summary of Core Damage Frequency Impacts

	BWR BASE CASE CDF = 3.6E-6			PWR BASE CASE CDF = 3.2E-5		
REGULATION CHANGE	CDF AFTER	DELTA CDF	FACTOR	CDF AFTER	DELTA CDF	FACTOR
ATWS (impact of the rule)	1.6E-5	-1.3E-5	4	4.0E-5	-8.2E-6	>1
APPENDIX B (impact if the rule had never existed)	6.9E-6	-3.3E-6	2	1.2E-4	-9.0E-5	4
APPENDIX B (impact if the rule were eliminated)	4.8E-6	-1.2E-6	>1	6.6E-5	-3.4E-5	2
TRAINING (impact if the rule were implemented)	3.4E-6	2.4E-7	0.9	2.7E-5	4.5E-6	0.8
TRAINING (impact of the intent of the rule)	1.9E-5	-1.6E-5	5	3.8E-4	-3.5E-4	12
MAINTENANCE (impact if the rule were implemented)	2.8E-6	8.2E-7	0.8	2.0E-5	1.2E-5	0.6

The Training Rule had the greatest impact of any of the regulation changes on the dominant sequences in both the BWRs and PWRs. The Training Rule was elicited for two impacts: (1) the impact of total implementation of the rule from the current 1993 situation and (2) the impact of the intent of the regulation prior to the 1982 timeframe. As is shown in Figures 4.7-2 and 4.7-3, the plant core damage frequencies, prior to implementation of the regulation, currently, and after total implementation are presented from left to right. The experts commented that the implementation of a systematic approach to training (this was initiated in 1983) greatly reduced risk. They felt it was difficult to tell how much of this change was actually due to the regulation and how much is related to other influences.

The ATWS Rule was elicited only on its impact prior to implementation. For the BWR it provided a greater factor of reduction in risk (4.4) than in the PWR (1.3). The experts felt that its greatest contribution was in increasing the knowledge base concerning reactor behavior and as a result the operators have an increased likelihood to initiate borate injection given an ATWS.

The Quality Assurance Rule, 10 CFR 50 Appendix B, was also elicited for two impacts: (1) the impact of total elimination of the rule from the current 1993 situation and (2) the impact of the intent of the regulation prior to the current timeframe. The results of the elicitation on Appendix B demonstrate that some benefit (a factor of 2 in BWRs and 4 in PWRs) was obtained from the original institution of this event. The benefit appears to produce a slightly better reduction in risk for the PWRs. The experts discussed that, with the elimination of this regulation, some benefit would be lost (a factor slightly greater than 1 in BWRs and 2 in PWRs) but that the average plant would maintain a level of QA that would maintain risk at much the current level. They also stated that information added to the knowledge base would not be lost with subsequent elimination of the regulation and this would also contribute to the maintenance of the current level of risk.

Information on the Maintenance Rule was elicited only on its impact after full implementation in the future. All the experts determined that the Maintenance Rule contributed the least to reduction of core damage frequency. For the BWR, it provided a factor of reduction in risk (4) than in the PWR (slightly greater than 1). The elicitation brought out that the Maintenance Rule's greatest significance was an increased focus on the safety significance of support systems. The reduction in risk will be subtle and largely due to the increase in the knowledge base.

4.7.5 Conclusions and Recommendations

Based on the results of this limited scope feasibility study, the following conclusions can be made:

- While some biases may exist in the interpretation of the impacts of regulations on PRA model parameters (e.g., event frequencies, component reliabilities), the relative directions and relative magnitudes of changes are not generally disputed by either the utility or NRC experts. This result was unexpected.

- Using the adjusted PRA model parameters obtained by the expert elicitation, it is possible to quantify and differentiate the core damage impacts of specific NRC regulations. This can be done by identifying in a systematic fashion how regulations impact the frequency of potential initiating events, component reliability and availability, human performance probability, and PRA model structure (number of barriers available).

In the course of performing the study, several issues were identified that warrant further consideration in interpreting the absolute values obtained. The results obtained are a reflection of the specific modeling approaches taken in the Surry and Peach Bottom NUREG-1150 PRA studies. As an example, differences in the approaches taken for modeling the reactor protection system unavailability has an effect on the magnitude of the ATWS Rule risk impacts between PWRs and BWRs. Consideration of how to deal with these subtle differences could be the subject of future work. As an additional issue, the use of expert elicitation to estimate the likely changes to PRA model inputs is potentially biased by the inability to separate out the effects of numerous rule changes and industry initiatives that have been under way for the last decade. Each of the experts commented on this problem, and this concern should be given further thought.

To improve the ability to differentiate the risk impacts of a wider body of regulations, it will be necessary to eventually consider the impacts of PRA "back-end" parameters (e.g. those parameters that impact containment performance, source terms and public exposure). This expansion is a recommendation for future work that would lead to the ability to evaluate rules like the Combustible Gas Control Rule (10 CFR 50.44). The evaluation of these inputs can be done as an extension of the basic methodology put together for this feasibility study.

4.8 OTHER NON-NRC PERSPECTIVES

As part of the evaluation of the use of risk-based techniques in regulation, public comments were solicited to gain their perspectives. A public meeting was held (May 6, 1993) with over 75 attendees. Representatives from over 25 licensees, several nuclear vendors and architectural engineering firms, numerous consultants and national laboratories, EPRI, NUMARC attended. Verbal and written comments were provided, reviewed and integrated into the report.

In addition, discussions were held with the regulatory authorities in Mexico, Sweden, Germany, and the United Kingdom, and with members of a working group of the OECD Committee on the Safety of Nuclear Installations Principal Working Group 5 that is exploring the state of the art of risk-based configuration control. Visits were also made to the Laguna Verde plant in Mexico and the Torness Power Station in Scotland to gain first-hand knowledge of their experiences.

The implementation of risk-based regulation in Mexico is comparable to that seen in the United States. PRA is used for issue prioritization and resolution, support in rulemaking, justification for continued operation, etc. A major PRA effort of the Laguna Verde plant is currently under way with the plant using the model in its operation and maintenance decisions. The plant is starting to explore the use of PRA in regulation in more detail with the encouragement of the Comisión Nacional de Seguridad Nuclear y Salvaguardias.

The joint Nordic study *[Ref. 4-11]* and *[Ref. 4-12]* is still in progress, but they are exploring the use of PRA in regulation in considerable detail, particularly in the regimes of AOT and STI determination. Results of this study should be available to assist the development of pilot studies in this area.

The Torness Power Station in Scotland has developed a technique for management of plant configuration control in a manner that appears to be consistent in many ways with the regulatory system employed in the United States and that may offer significant insights to those developing pilot applications relative to Technical Specifications or configuration control in this country.

The Torness Power Station employs a Mark II Advanced Gas Reactor. It uses a highly redundant and diverse combination of systems to provide essential post-trip cooling services. A quadrant approach is utilized in the design of the safety systems that provides substantial physical separation. A PRA was performed for the essential post-trip cooling function.

In setting the requirements for allowed time for component maintenance and repair, the plant examined a variety of possible configurations that might obtain from outages of selected equipment. Their approach to what are essentially AOTs for the various configurations was to permit the instantaneous core damage frequency to increase by a factor of up to 10 over that of the baseline PRA for a period of time not to exceed 30 days (approximately 1/10 year), and to allow the instantaneous core damage frequency to increase by a factor of 10-to-100 for a period not to exceed 3 days (approximately 1/100 year). In addition, they have overlaid requirements to preserve the "single failure criterion" and to limit the overall amount the integrated instantaneous risk may exceed the baseline PRA.

Because the systems involved are complex and highly redundant, there are a large number of possible configurations. They have calculated the risk increase for a large number of possible configurations and incorporated the results into a series of approximately 200 rather complex tables that define the permissible outage times associated with the various configurations. These tables are incorporated into the plant's Identified Operating Instructions, which are roughly akin to the Technical Specifications of U.S. plants. A computer is available in the control room to search the outage tables and determine which may be satisfied given the actual status of the plant. Hard-copy versions of the table are also available to verify the computer search or to determine appropriate limits if the computer becomes unavailable. This type of operation, with pre-calculated and verified tables presenting AOTs for a wide variety of system configurations, is an excellent example of what might be accomplished under the Group 2 type of application of PRA methods, discussed above.

The Torness system also has other features that increase its utility operationally. The computer maintains an accurate log on the number and outage times associated with equipment outages, permitting an easy evaluation in trends in component reliability. The system can be used in a prospective mode and is routinely used to plan outages of equipment to minimize the risk impact. It can also provide a prioritized list of what repairs would have the greatest risk reduction potential if an undesirable configuration were to occur.

A more detailed description of the Torness approach can be found in the "Operational Experience of a Reliability Based Maintenance Strategy for the Control of Essential Post-Trip Cooling Plant in a Nuclear Power Station" by W. B. Waddell *[Ref. 4-13]*.

4.9 EXISTING NRC EFFORTS

PRA applications having the potential to provide more flexibility in the regulations and in the implementation of the regulations while maintaining safety are those that primarily address configuration control and QA issues. Configuration control applications generally involve the utilization of PRA methods to optimize STIs and AOTs. QA applications generally involve the utilization of PRA to support "graded" QA; that is, optimizing QA for those structures, systems, or components that are safety significant based on PRA insights. Current NRC-sponsored programs were examined to identify those efforts that are using PRA that could provide potential insights in these areas.

The use of PRA by the NRC has been both broad and narrow. The broad application is seen in the many various and diverse activities that have increased over time, particularly since the TMI accident. The utilization of PRA, however, has been narrow in that it has been limited to a small set of applications. These activities have been defined and summarized into several categories (as reported in the draft NRC PRA Working Group Report) as follows:

- *Licensing of reactors* that involves using PRA in the review of analyses submitted as part of advanced reactor design certification applications, and plant-specific licensing actions such as Technical Specification modifications, justifications for continued operations, etc.

- *Regulation of reactors* that involves using PRA in monitoring of operations (with risk-based inspections); screening of events for significance (including operational event screenings, generic safety issue screenings, and facility screening risk analyses); analyses of events and issues (including operational events analyses, component and system failure data analyses and trends, reliability monitoring now developing as a result of the maintenance rule, generic safety issue analyses, and severe accident research studies); facility analyses (both those performed by the staff such as NUREG-1150 and those performed by licensees in the individual plant examination process); and regulatory analyses supporting regulatory actions such as backfits.

- *Licensing of fuel cycle and materials* that involves using methods similar to risk analyses (called performance assessment methods) that are being used as part of the licensing of proposed high-level-waste repository.

These activities are summarized below in Table 4.9-1.

Table 4.9-1
Summary of Staff PRA Uses

CATEGORY	APPLICATION
Licensing of Reactors	• Reviews of advanced reactors. • Reviews of plant-specific licensing actions.
Regulations of Reactors	• Monitoring operations by inspection. • Issue screening of operational events, generic safety issues, and facility screening risk analyses. • Issue analyses of operational events analyses, operational data and trending analyses, maintenance rule regulatory guide, generic safety issues, and severe accident issues. • Facility analyses involving staff studies and individual plant examinations. • Regulatory actions including regulatory analyses.
Licensing of Fuel Cycle and Materials	• Reviews involving high level waste facilities.

As can be seen, these PRA efforts are relatively diverse; and although each NRC office is involved in programs using PRA, current utilization of this type of integral analysis by the NRC is rather limited when focused on attempts to reduce regulatory burden or provide additional flexibility with the regulations and licenses. Current NRC-sponsored programs that can provide insights in support of this area primarily involve configuration control regarding Technical Specification optimization. No NRC-sponsored programs supporting graded QA based on PRA were identified.

These specific types of activities are summarized below for each NRC office.

4.9.1 AEOD-Sponsored Programs

The Office for Analysis and Evaluation of Operational Data (AEOD) utilizes PRA techniques and insights in the accomplishment of its mission. Although their ongoing PRA-related programs are not focused on determining ways to reduce regulatory burden and provide flexibility in licensing and regulatory actions, the Trends and Patterns Analysis and the Reactor Operations Analysis Branches within the Division of Safety Programs are involved in efforts that can ultimately assist in providing the data requirements and insights for PRA-based programs supporting configuration control and graded QA (from a regulatory perspective).

The Trends and Patterns Analysis Branch has ongoing programs that analyze operational data to identify and provide a quantitative content for new safety issues; evaluates the effectiveness of current regulations, regulatory actions, and initiatives taken by licensees to resolve safety issues concerns; and helps guide and focus engineering evaluations. These programs support four major activities as follows:

- Hardware performance studies of risk-important components, systems, initiating events, and accident sequences.
- Safety and regulatory studies of trend performance for selected regulatory issues through an appropriate parameter related to the specific issue to determine effectiveness of implementation.
- Data base studies involving common cause failure event data and a human performance data base that trends human actions important to plant safety and risk.
- Risk assessment studies evaluating the risk implications of trending results from the hardware, safety issues, and special data analyses.

The Reactor Operations Analysis Branch's ongoing Accident Sequence Precursor (ASP) Program also provides needed support for the PRA utilization in configuration control and graded QA optimization. The ASP program provides a safety significance perspective of nuclear plant operational experience. The program uses PRA techniques to provide estimates of operating event significance in terms of the potential for core damage; that is, accident sequence precursors are events that are important elements in core damage accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one of more postulated events, could result in a plant condition leading to severe core damage. The precursors are selected and evaluated using an evaluation process and significance quantification methodology. The types of events evaluated include initiators, degradations of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences.

4.9.2 NRR-Sponsored Programs

The Office of Nuclear Reactor Regulation (NRR) has current PRA efforts directly supporting licensing and regulatory activities that can provide regulatory burden reduction and flexibility in the implementation of the regulations. These efforts are being performed in the Operational Reactor Support and Systems Safety Analysis Divisions by the Technical Specifications and Probabilistic Safety Assessment Branches, respectively.

In 1987, the Commission issued its interim "Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" encouraging licensees to voluntarily implement a Technical Specification Improvement Program. As a result of this policy statement, five sets of improved STS were developed; one for each Nuclear Steam Supply System (NSSS) vendor (i.e., Westinghouse, Babcock and Wilcox, Combustion Engineering, General Electric BWR 4, and General Electric BWR 6). PRA was utilized in the development of these STS as follows:

- A number of completion times (i.e., AOTs) and STIs were relaxed based on NRC staff-approved topical reports and on draft NUREG-1366 *[Ref. 4-14]*. In their topical reports justifying the relaxations, the NSSS vendors based their conclusions on PRA insights. NUREG-1366 used qualitative rather than PRA insights to support such relaxations.

- Using the Grand Gulf and Surry PRAs from NUREG-1150, the core damage frequencies were recalculated with the new STS changes to identify any potential concerns. No significant increase in core damage frequency was observed as a result of these changes.

A "lead" plant for each NSSS STS has been identified by industry.

As the implementation of the improved STS and development of line-item improvements proceeds, the staff's intends to utilize PRA along with deterministic bases to support its decisions. This utilization will primarily be based on evaluations of industry's proposals. The information from the programs currently in progress in the Office of Nuclear Regulatory Research (RES) will be used to support or validate, as appropriate, industry's risk-based proposals.

Currently the staff is evaluating risk-based changes to Technical Specifications proposed by the South Texas Nuclear Project. This effort is currently in progress in RES.

The Probabilistic Safety Assessment Branch activities that directly involve PRA efforts to improve plant operations and maintenance primarily include providing risk assessment of potentially safety significant issues and reviewing applications submitted by the licensees. The issues reviewed for their risk impact are a result of identified safety concerns. Recent examples include:

- Intersystem LOCA.

- Shutdown Risk.
- Alternative Tube Plugging Criteria.

The applications submitted by the licensees are generally requests for exemptions (or waivers) from regulatory requirements. The justification for requesting and granting the exemption includes PRA insights. Recent examples include:

- Waiver to allow refurbishment of service water system.

- Minor actions involving man-made hazards, tornado protection, containment penetrations, and toxic gas detectors.

4.9.3 RES-Sponsored Programs

RES has several ongoing PRA efforts directly supporting licensing and regulatory activities. These programs are being performed in the System Research, Safety Issue Resolution and Engineering Divisions by the Human Factors Branch, the Severe Accident Issues and Probabilistic Risk Assessment Branches, and the Electrical and Mechanical Engineering Branch, respectively.

The PRA programs in the Human Factors Branch are currently those that have the greatest potential in assisting in the assessment of risk technology for providing regulatory burden reduction and flexibility while maintaining safety. These efforts are primarily focused on developing methods in direct support of Technical Specification improvements as follows:

- Risk impact in varying AOTs and STIs at power and during shutdown and considering the effects of test errors on optimum test intervals.

- Risk impact from action statements requiring shutdown if equipment needed during shutdown (e.g., residual heat removal) fails.

- Risk implications of taking equipment out-of-service for maintenance looking at rolling maintenance schedules, optimizing the frequency of schedule maintenance, and integrating surveillance with preventive maintenance.

- Dependent failures examining improved methods for recognizing and preventing dependent failures.

- Configuration management considering a conceptual framework for risk-based configuration management.

The methods that are being developed are reliability-engineering tools that analyze Technical Specification requirements within the framework of a PRA and that can estimate the risk impact of changing the level of a particular requirement in Technical Specifications; and therefore, they can provide a risk perspective on the bases for these Technical Specification requirements and for related maintenance guidelines.

These applications share the strengths and weaknesses of PRA. They are useful to integrate and prioritize only those considerations that can be quantified in terms of reliability and availability; therefore, they are applicable to only a fraction of the requirements in Technical Specifications. In general, these methods are directly applicable to evaluating AOTs and STIs for active, front-line systems, and support systems. The methods are only marginally applicable to instrumentation, and are not applicable to concerns not modeled in PRA, such as security and occupational health. In general, these methods are not yet sufficiently refined to treat uncertainties in detail. It is expected that consideration of uncertainties will be incorporated with the use of these methods.

There are currently six ongoing programs that are developing these methods as described below.

1 — Procedures for Evaluating Technical Specifications

In 1983, a task force established by the Executive Director for Operations (EDO) provided recommendations to improve surveillance testing requirements in Technical Specifications. The resulting actions formed the Technical Specification Improvement Program. In 1987, a Commission Interim Policy Statement on Technical Specifications Improvement encouraged licensees to voluntarily implement a Technical Specification Improvement Program that included applying risk analysis methods and human factors principles to improve Technical Specifications. In support of this program, research began to develop methods for evaluating the risk impact of requirements in Technical Specifications, to explore alternative approaches, and to provide a technical basis for improvements.

This research, which is largely completed, has published methods to evaluate the risk impact of AOTs and STIs (including the impact of test errors). The work also outlined a conceptual approach for operational configuration control. The remaining work on this project, which is being completed in 1993, will provide a method to evaluate the risk impact of scheduled maintenance intervals. The approach analyzes the balance between beneficial and adverse effects of maintenance, and models three states: operable, degraded (i.e., ready for preventive maintenance), and failed. The method can use NPRDS data for incipient, degraded, and complete failures. The results of this research will allow analysis of the risk impact of issues such as not permitting certain preventive maintenances during power operation and instead requiring that AOTs during power operation be used only for corrective maintenance.

One of the new STS's will be used as a testbed for a limited pilot application of the methods described in this report for evaluating requirements in Technical Specifications. This pilot application involves developing a strategy and criteria that will result in clear, simple statements of requirements that integrate risk and practical considerations to control risk efficiently. These criteria are intended to address:

- The scope and frequency of updating of the PRA and data base that form the basis for the licensee's risk analysis.

- What risks must be assessed to support Technical Specification changes and acceptable ways to model them (e.g., test intervals, test effectiveness, test errors, and aging effects).

- Prioritizing risk contributors in Technical Specifications.

- Acceptable changes in risk.

- Experience feedback, if appropriate, in updating Technical Specification requirements.

2 — Technical Specification Requirements During Shutdown

NRC is reevaluating regulatory requirements for nuclear power plants during shutdown. One aspect of this reevaluation is to consider how effectively Technical Specifications control risk during shutdown.

In support of this endeavor, this project was established to develop methods for evaluating the risk impact of plant configurations permitted and surveillance required by Technical Specifications during shutdown; to explore alternative approaches; and to provide a technical basis for improvements. These analysis methods use as a framework the low-power-and-shutdown PRAs (described elsewhere in this report).

These models and trial applications to a pressurized water reactor (PWR) and a boiling water reactor (BWR) will be completed in late 1993.

3 — Action Statements That Require Shutdown

As part of the program to improve Technical Specifications, action statements that require plant shutdown if an AOT time is exceeded are being developed.

The issue concerns a few systems, such as residual heat removal (RHR), standby service water (SSW), and auxiliary feedwater, that may be required to cool the plant during shutdown. Currently, action statements in Technical Specifications typically require that plants shut down when an AOT is exceeded, even though shutdown may require use of the system that is out-of-service for maintenance. The work has developed a decision-analysis method for comparing the risk impact of transferring the plant to shutdown versus the risk impact of continued power operation.

The method and trial application to RHR and SSW at a BWR-6 are being published this Spring. An equivalent method and trial application to a PWR will be completed in early 1994.

4 — Technical Specification Defenses Against Dependent Failures

Technical Specifications set surveillance requirements and AOTs in order to ensure the availability of a plant's safety systems. These safety systems are designed to achieve high availability through redundancy. Redundancy, however, can be defeated by dependent (e.g., common cause) failures. For example, the Davis-Besse loss of all feedwater in 1985 involved several valves stuck shut (dependent failures). Despite the importance of dependent failures, most Technical Specification requirements do not explicitly address and protect against dependent failures.

In support of this concern, a method and criteria are being developed for explicitly addressing dependent failures in setting STIs and AOTs. This method uses a NUREG-1150 PRA as the framework within which to model and evaluate the risk impact of postulated Technical Specification improvements. A recent AEOD analysis of industry-wide experience with dependent-failure events is used as a reality check to supplement the PRA. Possible improvements in Technical Specifications that might better defend against such dependent failures are being postulated.

The purpose is to determine whether simple changes in surveillance requirements and AOTs would substantially reduce the risk of operating reactors. The result will be an assessment of the effectiveness of this approach.

5 — Method for Monitoring Dependent Failures

This effort is a related project that supports AEOD trends and analysis of operational data. This project has developed a method for analyzing failure data to estimate the fraction of failures that are dependent failures. The method compares the distribution of observed times-between-failures with the distribution expected if the failures were independent. The difference reflects dependent failures. The method estimates the fraction of dependent failures (e.g., a beta factor) and the actual safety system unavailability with this degree of dependency.

The methods development has been completed, and the report will be published in mid-1993. AEOD and RES are discussing whether additional work is warranted to make the software directly applicable to AEOD screening of data to help recognize dependent-failure events.

6 — Handbook

This task is developing a handbook of methods for evaluating the risk impact of Technical Specification requirements. The handbook will facilitate staff evaluation of licensee proposals for changes to Technical Specifications and for scheduling of AOTs for preventive maintenance. This handbook will also transfer research results to support NRR's Technical Specifications Branch.

The scope of the handbook includes reliability and risk based methods for evaluating: AOTs, use of AOTs for preventive maintenance, action statements requiring shutdown, STIs, defenses against common cause failures, and managing plant configurations. For each of these topics, the handbook will summarize useful analysis methods and data needs, will outline in common-sense terms the insights to be gained from a risk perspective, and will list a few references for more detailed information and alternative methods. Writing of the handbook is starting in March 1993. A draft will be circulated for staff review and comment in October 1993. The completed handbook will be available early in 1994.

These six programs are focused on developing methods for Technical Specification optimization. The methods developed, given that the limitations, boundary conditions, assumptions, uncertainties, data, and human performance issues associated with PRA are properly addressed, can provide assistance in determining the ground rules or restrictions that would be necessary to maintain the current level of safety while providing additional flexibility in the implementation of the regulations. In addition, there are other ongoing programs within RES that also utilize PRA, will provide necessary insights, and will provide assistance in addressing the above-mentioned concerns.

Technical Analysis of Proposed Changes to the South Texas Technical Specifications

Houston Lighting and Power, the licensee for the South Texas Nuclear Project (STNP), submitted a proposed amendment to its operating license. The Probabilistic Risk Analysis Branch is developing a framework for analysis and a technical basis for evaluating the proposed changes to AOTs and STIs for the STNP. The evaluation involves reviewing the system failure models and sequence level cut sets of the STNP PSA, establishing a systematic risk profile for the base case three-train configuration of the STNP, obtaining the overall risk impact of the proposed changes in AOTs and STIs, and developing a framework that will support the bases for approval of the proposed changes in AOTs and STIs based on risk arguments.

Although this effort is not a formal program to develop "generic" methods for evaluating proposed Technical Specification changes, insights can be used for generic applications.

Individual Plant Examination Data Base

On November 23, 1988, Generic Letter 88-20 was issued requesting licensees to perform an Individual Plant Examination (IPE) with the general purpose of each licensee "to develop an appreciation of severe accident behavior, to understand the most likely severe accident sequences that could occur at its plant, to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (if necessary) to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents" *[Ref. 4-15]*.

In support of this effort, an IPE data base has been developed, which catalogs the information provided in each licensee's IPE submittal. The type of information being input to the data base for each IPE includes the following:

- Plant information (e.g., reactor and containment type).
- Initiating event information (e.g., initiating event and its associated frequency).
- Accident sequence information (e.g., accident sequence description and associated frequency).
- System and component dependency information.
- Core damage frequency information.
- Plant damage state information.

The data base will allow users to gather information both by plant and across plants. For example, the data base will identify those plants where a certain issue such as loss of offsite power is a concern; will identify concerns for a group of plants such as identifying the dominant contributors for 3-loop westinghouse plants; will identify those plants where a system concern may exist such as identifying plants where diesel generators are dependent on instrument air. These are a few examples of the IPE data base.

The information currently being entered into the data base only includes IPE data. As part of the IPE effort, licensees were only required to examine internal initiators and internal flooding. NUREG-1407 *[Ref. 4-16]* provides the guidelines for the IPE of external events. The data base will be expanded to include this information for each licensee.

Low Power and Shutdown PRA

PRAs have traditionally examined severe accidents only occurring at full-power operation. Analyses have indicated that severe accident occurring at low power and shutdown could be significant. A major program has been in progress to assess the frequencies and risks of accidents initiated during low-power and shutdown modes of operation for two nuclear power plants by performing detailed PRAs for the various operational modes. This effort also involves the development of new methods and will compare the assessed risk with those of an accident initiated during full-power operation.

The work involves examining the accidents initiated by internal events (including flooding and fire) as well as external events (e.g., earthquakes). Ultimately a full PRA (core damage frequency, fission product releases and consequences) will be completed.

PRA Working Group

In 1991, the EDO formed a working group of staff management (i.e., PRA Working Group) to "consider what improvements in methods and data analysis are possible and needed, the role of uncertainty analysis in different staff uses of PRA, if improvements are needed in the allocation of existing PRA staff, and the need for recruitment of more staff (or for identifying other means for supplementing staff resources)." *[Ref. 4-17]*

The objectives of the PRA Working Group are to develop guidance on consistent and appropriate uses of PRA within the NRC; to identify skills and experience necessary for each category of staff use; and to identify improvements in PRA methods and associated data necessary for each category of staff use. In support of these objectives, the Group has defined the scope of its work as follows:

- Ascertain present uses of PRA by the staff; future PRA uses that are not now well defined (e.g., possible transition to risk-based reactor regulation) are not included in the Group's scope of work.

- Review available or developing risk analysis documents and guides, and develop recommendations for improvement. Such improvements are the responsibility of the user organization, with oversight by the Working Group. It is not within the Group's scope to update or replace such guides although the group may make recommendations to update them.

- Assess staff skills and experience needed to appropriately apply PRA, including staff organizational considerations, if appropriate. While the skills and experience assessment is within the scope of the Group's work, the development and implementation of plans to change staffing levels, staff training, or organizational arrangements are the principal responsibility of the Office of Personnel and the affected offices, as part of the overall development and implementation of the agency's Human Resources Strategic Plan.

- Assess needed improvements in PRA techniques and data to support appropriate staff use of risk analysis. This assessment focuses on improvements needed for particular uses, rather than a broad assessment of needed improvements in risk analysis methods, and uses state-of-the-art risk studies such as NUREG-1150 as reference and resource material. The performance of any such improvements is the responsibility of the appropriate staff organization, not the Working Group.

It must be ensured that the current level of safety is maintained when using an integral analysis, such as PRA, to provide more flexibility in the regulations and in the implementation of the regulations. NRC-sponsored programs were inventoried in a first step to determine what types of general rules and restrictions would need to be imposed so that PRA can be used while maintaining the current level of safety. A summary of these PRA programs that could provide insights are provided in Table 4.9-2 below.

Table 4.9-2
Summary of NRC-Sponsored PRA Programs

RESPONSIBILITY	PROGRAMS	APPLICATION
AEOD/DSP/TPAB	Analysis of operational data to identify and provide quantitative content for safety issues	Data support to Technical Specification and graded QA optimization
AEOD/DSP/ROAB	Accident Sequence Precursor Program	Data support to Technical Specification and graded QA optimization
NRR/DORS/TSB	Technical Specification Improvement Program	Utilization of Technical Specification optimization
NRR/DSSA/PSAB	• Risk Evaluation of Safety Issues • Review of Licensee Requests for Exemption	Information support to Technical Specification and graded QA optimization
RES/DSR/HFB	• Procedures for Evaluating Technical Specifications • Technical Specification Requirements During Shutdown • Actions Statements That Require Shutdown • Technical Specifications Defenses Against Dependent Failures • Method for Monitoring Dependent Failures • Handbook of Methods for Evaluating the Risk Impact	Development of Technical Specification optimization methods
RES/DSIR/PRAB	Technical Analysis of Proposed Changes to the South Texas Technical Specification	Information support to Technical Specification and graded QA optimization
RES/DSIR/SAIB	Individual Plant Examination Data Base	Information support to Technical Specification and graded QA optimization
RES/DSIR/PRAB	Low Power and Shutdown PRA	Information support to Technical Specification and graded QA optimization
RES/DSIR/PRAB	PRA Working Group	Information support to Technical Specification and graded QA optimization

4.10 CONCLUSIONS

The current state of the art in PRA technology was examined to determine under what circumstances information, either qualitative or quantitative, gleaned from PRA methods could be used in the regulatory process. It was determined that PRA methods provide an integral tool that can be used to help ensure coherence and consistency in the regulatory process and provide a means of converting diverse deterministic requirements to performance based requirements. This provision can occur with equivalent protection to public health and safety, while offering increased flexibility to licensees, provided the risk-based criteria are met. To this end, the current state of the art in PRA methods were assessed considering how the many strengths of these methods could be exploited, while minimizing the significance of those weaknesses that still remain in the application of risk-based methods in regulation.

Work in progress under the NRC sponsorship devoted to the research, development, and application of risk-based methods to aid the regulatory process was surveyed. NRC has had an active program investigating the use of PRA methods in regulatory practices since 1983, and much (but not all) of the work done by others in this area in the U.S. draws heavily from this research. (The more important elements relative to use of risk-based techniques in regulation are described in Section 4.9.1.)

A number of papers sponsored by the regulated industry that address the potential use of risk-based techniques in regulation have been published in the literature. Informal discussions were held with several of the authors. Broad-based industry research on use of PRA methods for regulatory purposes, as reflected in the literature, is fairly recent, but several utilities have had long-term ongoing programs on use of risk methods to improve operations. These programs could be extended to the regulatory environment.

Further, international literature addressing the potential for risk-based regulations were reviewed, particularly the information contained in reports and workshops sponsored by the Organization for Economic Cooperation and Development/Committee on the Safety of Nuclear Installations Principal Working Group 5*[Ref. 4-18]*. Because of their current substantial efforts in this regard, detailed discussions were also held with utility and regulatory authorities in Mexico and the United Kingdom to gain the benefit of their experience.

4.10.1 Recommendations

Based on the above, it is recommended that the utilization of PRA-based techniques in the regulatory process be characterized by three general groups, each having similar requirements in terms of the boundary conditions and assumptions used in the analysis, as well as similar requirements in terms of the depth and breath of the review that would be required by the NRC staff.

- *Reliance on Quantitative Insights From PRAs* — This category of risk-related regulatory actions would utilize the risk analyses to separate the potentially important components and systems from the unimportant. This relative importance would be from a PRA perspective based on core damage prevention, relying on both plant-specific studies as well as on compilations of the results of risk-based studies on similar plants.

  This type of usage could be based on the type of PRA modeling effort that is common in responses to the IPE Generic Letter 88-20 and the type of review currently being applied to IPE reviews by the NRC staff would likely suffice. Generic failure rate data could generally be employed and frequent updates of the PRA studies would not generally be required.

  Performance-based responses to the Maintenance Rule, risk-based approaches to graded quality assurance and inservice inspection are possible examples of potential usage.

- *Heavy Reliance on Quantitative Results From PRAs in Selected Areas* — Efforts of this type would require careful attention to the PRA methods and analyses in selected areas but would not involve close scrutiny of the entire plant risk analyses. It could be used to improve regulatory flexibility for a given component, or applied broadly to selected portions of the plant at the train level, without examining the detailed modeling at lower levels in the analytical trees.

  This type of application would also generally require average PRA modeling. Generic failure data would be sufficient in most instances, but it would need to be augmented with plant-specific data in those selected areas where heavy reliance was placed on the plant-specific results. For greater than one-time use, the PRA would have to be modified as necessary to reflect any changes in the current plant design and operational practices. This would likely require updating at least each refueling outage.

  Examples of this category would include optimization of selected Technical Specifications, evaluations of "unreviewed safety question" under 10 CFR 50.59, use of pre-calculated configuration management analyses to support extension of AOTs under certain circumstances, justification for continued operation and inservice testing evaluations under 10 CFR 50.55a.

- *Complete Reliance on Numerical Results from PRAs* — In this category, regulatory decisions would be based almost exclusively on the numerical PRA results. It would require a very comprehensive analytical effort since, in this type of application, apparently minor changes in assumptions or boundary conditions may significantly affect regulatory decisions.

  This type of application would require a level of detail that either stretches or exceeds the current state of the art. It would require a comprehensive plant-specific data analysis, and would require that the PRA be reviewed at a depth equivalent of that afforded to a final safety analysis report in the course of a Part 50 operating license review.

  An example of this type of usage would be the development of risk-based Technical Specifications requiring on-line updating of PRA models.

Tentative requirements have been developed for the boundary conditions and assumptions used in the analyses for each of the above classes in the preceding sections. These requirements should be regarded as tentative and can serve as a jumping off point for detailed discussions with the public and the regulated industry.

Beyond the technical recommendations, more specific recommendations regarding the nature of the regulatory environment needed to introduce the use of risk-based analyses in a broad fashion are offered.

- In effect, the NRC currently uses PRA insights to primarily add requirements to the industry. This utilization of PRA needs to be changed to allow PRA-based insights to reduce regulatory burden when it is shown that such a reduction does not reduce the safety envelope of the plant. Thresholds (e.g., NRC guidelines on content of submittals, acceptable PRA methods, and decision criteria) must, therefore, be established by the NRC for each PRA usage class (as described above) in concert with any industry-proposed pilot applications of these potential uses.

- The current state of development and utilization of probabilistic techniques in the industry can support use of risk-based regulatory approaches at the present time. Several utilities have ongoing programs using risk methods and "living" probabilistic analyses to improve operations and maintain plant safety and efficiency that could be extended to the regulatory environment and provide increased licensee flexibility while maintaining or improving the safety envelope. It is recommended that the Commission elicit licensee proposals in this regard to support such an effort.

- The development by NRC of methods for optimizing Technical Specifications using risk-based techniques is nearing completion and, with publication of a handbook early in calendar year 1994, will provide a technical basis for judging the acceptability of risk-based approaches proposed by licensees. In addition, this handbook could serve as the point of departure for discussions between the NRC staff and the industry leading to industry-proposed guidance, suitably endorsed by NRC. It is recommended that this handbook be published as a regulatory document, perhaps as a regulatory guide. This handbook can provide guidelines for methods or similar techniques that could be used in a pilot program in the near future, if there is industry interest in such an application.

- NRC programs and interests on the development and implementation of risk-based methods in regulation currently span multiple offices and organizations. An integral agency plan covering the research, development, implementation, and use of risk-based techniques in regulation is needed in maintaining a consistency of approach throughout the agency and in allocating scarce resources. This plan would also assist in the efficient use of the limited number of NRC staff with expertise in quantitative risk assessment.

- Possible risk-based regulatory approaches span a continuum from modest applications of conventional probabilistic methods to techniques for risk-based configuration control on a real-time basis. They represent an increasingly valuable complement to the present regulatory structure. The required resource commitments for both the licensee and NRC are likely to increase in this area as more complex approaches are investigated; however, these more comprehensive approaches will also offer the most flexibility. and therefore, resource savings to the licensee while maintaining the safety envelope.

- An incremental approach is recommended for the evolution to a more risk-based approach, testing benefits gained versus costs of implementing in pilot programs before proceeding to complete implementation industrywide. As indicated above, certain risk-based approaches can be implemented now, while others will be suitable for trial investigation in the near future. An investigation of the usages that are compatible with the current strengths and limitations of risk methods needs to be pursued in supporting the evolution to PRA-based regulation.

4.10.2 Analysis of Public Comments

Numerous public comments were received from over a dozen nuclear utilities, several nuclear vendors and consultants, and other organizations such as NUMARC, EPRI and the BWR Owner's Group. Overall, the responses were supportive of the Review Group effort and recommendations. There were several "major" comments submitted that are addressed below.

Most of the commentors disagreed with the "generic grouping" discussed in the Graded Implementation approach of regulations. This part of the report has been expanded to clarify the Review Group's recommendation. It is felt, by the commentors, that a plant-specific categorization of important SSCs should be allowed and plants should not be "penalized" by other plants. The Review Group did not intend the generic group to be a penalty but one suggestion for addressing the issue of completeness, uncertainties, and the subjectivity that can be imbedded in a PRA. The Review Group encourages plant-specific analyses and recommends a plant-specific categorization of SSCs where it is supported by a solid plant-specific PRA (e.g., plant-specific data, technical analyses supporting assumptions).

Several commentors indicated that the consequences associated with potential core damage accident sequences should also be included in the classification of important SSCs. The Review Group is in agreement and this issue was clarified in the report.

The criteria in the report regarding the HRA screening values was felt to be too restrictive by several commentors. The Review Group encourages licensees to perform detailed human reliability analyses. The report was clarified to indicate that the screening values are applicable when a solid technical analysis is not provided.

It has been recognized by the Review Group that, although PRA has been used to reduce burden in certain areas, the NRC has primarily used PRA insights to support adding requirements on the industry. Public comments have been received on the Review Group Report that indicate, in regards to the use of PRA, the NRC staff are *"unwilling to allow for flexibility"* and that *"an endorsement by the Commissioners and/or senior NRC staff for developing or allowing flexibility in regulations does not mean that this flexibility will ever make its way into the manner utilities are regulated."* A recent example (by a commentor) used to illustrate this point was a discussion between the NRC staff and industry (i.e., Cooperative Efforts Group) regarding their intention in responding to Generic Letter 89-10 that appeared (to the respondent) to preclude the possible use of probabilistically-based analyses to develop a graded response to Generic Letter 89-10. The Review Group recognizes that what has been suggested in this report represents a change in approach in many ways, and such change may, on occasion, be difficult to implement. Over-reaction, however, based on preliminary discussion is cautioned against. Once a formal submittal has been made by a licensee (or group of licensees) containing concrete proposals, the staff will respond and a technical position taken relative to the proposal's acceptability. If a licensee feels that the NRC staff response is inconsistent with Commission policies and procedures, the licensee is encouraged to raise their position to NRC senior management.

One recommendation by the Review Group is the publication, by the NRC staff, of a handbook in the form of a regulatory document or regulatory guide regarding methods for optimizing Technical Specifications. It needs to be recognized that this work is the culmination of an effort expanding over multiple years in which ongoing discussions between the NRC and industry regularly occurred. The Review Group agrees that continued dialogue is necessary (and is encouraged) in the implementation of the methods discussed in the handbook. Delaying the issuance of the handbook, however, for additional dialogue is not necessary.

4.11 ACRONYMS, ABBREVIATIONS AND REFERENCES

4.11.1 Acronyms and Abbreviations

AEOD	Office of Analysis and Evaluation of Operational Data
AOT	Allowed Outage Time
ARI	Alternate Rod Insertion
ASP	Accident Sequence Precursor
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
CDP	Core Damage Probability
CDF	Core Damage Frequency
CRD	Control Rod Drive
EDO	Executive Director of Operations
EPRI	Electric Power Research Institute
HEP	Human Error Probability
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating and Air Conditioning
IPE	Individual Plant Examination
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
MSIV	Main Steam Isolation Valve
NEA	Nuclear Energy Agency
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
OECD	Organization for Economic Cooperation and Development
PRA	Probabilistic Risk Analysis
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RES	Office of Nuclear Regulatory Research
RHR	Residual Heat Removal
RPS	Reactor Protection System
SLC	Standby Liquid Control System
SSC	Structure, System, Component
SSW	Standby Service Water
STI	Surveillance Test Interval
STNP	South Texas Nuclear Plant
STS	Standard Technical Specification
TMI	Three Mile Island

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APPENDIX A

SAFETY IMPORTANCE OF REGULATIONS

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A.1 INTRODUCTION

This appendix provides the results of a pilot feasibility effort to assess the relative "safety importance" of regulations. This was accomplished by determining the impacts (in terms of core damage frequency for two commercial nuclear power plants) associated with hypothesized changes in the contents and implementation of Chapter 10 of the Code of Federal Regulations, Part 50 (10 CFR 50) *[Ref. A-1]*.

A.1.1 Background

On January 4, 1993, the Executive Director for Operations of the Nuclear Regulatory Commission (NRC) appointed a Regulatory Review Group. The group conducted a comprehensive and disciplined review of power reactor regulations and related NRC processes, programs, and practices with special attention placed on the feasibility of substituting unnecessarily prescriptive requirements and guidance with performance based requirements and guidance founded on risk insights. The probabilistic risk assessment (PRA) subgroup of the Regulatory Review Group investigated utilizing a PRA type analysis to provide additional flexibility in the regulations and their implementation. This subgroup contracted the Idaho National Engineering Laboratory (INEL) to provide assistance in identifying a quantitative measure of the impact of a specific regulation and assessing whether this impact could be analyzed.

A.1.2 Technical Approach

To determine the potential impact by a rule on the safety of a plant, an approach was developed that examined the potential effect of a rule on a plant's core damage frequency. The Surry and Peach Bottom NUREG-1150 models were used to determine this impact. The development of the approach involved the following basic tasks:

- Sample regulations were selected for use as "test cases."
- Specific changes to the selected regulations were hypothesized to determine their potential safety impact.
- A framework was developed to systematically relate regulations to PRA model changes.
- The impact of the hypothetical regulation changes on the plant design, operation and maintenance was determined using expert elicitation.
- The results of the expert elicitation was propagated through the NUREG-1150 PRA models to determine the impact on the core damage frequency.

As a feasibility study, a limited number of hypothetical regulatory changes were investigated, and an analysis of the core damage frequency impact was performed for each change.

A regulatory change could lead to changes in the input parameters (e.g., component availability and reliability) of the PRA or changes to the PRA model (e.g., system fault tree) itself to account for new failure paths or the elimination of safety barriers. For this feasibility study, it was necessary to focus on the most core damage significant components, systems, initiating events and human actions in the PRA; structural changes were not investigated. The impact of changes in the characteristic parameters for these components, systems, initiating events and human actions were then quantified using the IRRAS 4.0 code *[Ref. A-2]*.

A.2 SAMPLE REGULATIONS AND CHANGES

A.2.1 Selection of Sample Regulations

A classification scheme was used to group and evaluate the 10 CFR 50 regulations to generate a sample of regulations for use in this feasibility study. This classification scheme employed a number of variables that characterized the impact a given regulation can have on a PRA model. In particular, they indicate the mechanism by which the regulation can impact the PRA model and the scope (extent) of this impact. The variables are binary and are defined as follows (the possible values are indicated in the brackets):

<u>Mechanism of Impact</u>

X_1:	Directness of impact mechanism	[Indirect/Direct]
X_2:	Potentially affects numerical values of PRA model parameters	[Yes/No]
X_3:	Potentially adds new parameters to PRA model	[Yes/No]
X_4:	Potentially removes parameters from PRA model	[Yes/No]

<u>Scope of Impact</u>

X_5:	Impact extent	[Localized/Pervasive]
X_6:	Potentially affects PRA dominant sequences	[Yes/No]
X_7:	Potentially affects non-dominant sequences in PRA	[Yes/No]
X_8:	Potentially affects systems/components/failure modes not in PRA	[Yes/No]

By rating the 10 CFR 50 regulation with these variables, 14 natural groupings of rules were identified (see Appendix A.1). A representative sample set of four regulations were chosen for this study:

- 10 CFR 50.62 [The ATWS (anticipated transient without scram) Rule]
- 10 CFR 50 Appendix B [The Quality Assurance (QA) Rule]
- 10 CFR 50.120 (The Training Rule)
- 10 CFR 50.65 (The Maintenance Rule)

The impact of the ATWS rule, 10 CFR 50.62, is in a grouping that has a direct impact (on the PRA model), potentially affects model inputs, and may remove parameters in the PRA. The scope of the rule is considered local and changes would impact both dominant and non-dominant sequences. As an additional point the rule was chosen because its risk significance is expected to differ for boiling water reactors (BWRs) and pressurized water reactors (PWRs). It also provides a case where a number of detailed PRA studies have been done to analyze alternative strategies for compliance.

The QA Rule (10 CFR 50 Appendix B) and the Maintenance Rule (10 CFR 50.65) are in a grouping that has a direct impact and potentially affects model inputs. They have a pervasive effect on numerous components, systems, and structures in the plant. Appendix B has been in effect since 1970 while the maintenance rule has only recently been adopted. It has not yet been completely implemented.

The Training Rule (10 CFR 50.120) is still in draft form. It will be in a grouping that has a direct impact, potentially affects model inputs and may add parameters to the PRA. It will have a pervasive impact on numerous components, systems, and structures in the plant. Changes in this regulation will impact the numerical input values of both the dominant and non-dominant sequences in the PRA.

A.2.2 Identification of Regulation Changes

The regulation changes were restricted to either total elimination, total implementation, or never adopted. Total elimination refers to total removal of the regulation and the resulting impact. Total implementation refers to full execution of those regulations that have been recently adopted by the NRC but are not implemented by the plants at the present time. The impact of the regulation being implemented at present was examined. The impact of a current regulation being removed is different than the impact that results from the regulation never having been adopted. The concept of a regulation never having been adopted attempts to determine the impact of the regulation as if it had never existed. This would have a different impact than the total elimination of the regulation, since other activities (e.g., learning) usually accompanying the adoption of a regulation will have an influence even after the regulation is eliminated.

For each regulation change hypothesized the average plant response was also a consideration. A basis of the average plant response was developed for each regulation change in the elicitation. The experts were then allowed to discuss and develop a final working definition of the average plant for use in the elicitation process. The changes were defined as follows:

- 10 CFR 50.62 (The ATWS Rule)

  The impact of the ATWS rule was elicited as if the regulation had never been adopted. The expert was to imagine that in 1984 the regulation was never adopted into law. The average plant response will be different for BWRs and PWRs.

  Prior to adoption of the ATWS rule BWRs were designed with standby liquid control system (SLC), alternate rod insertion (ARI), and recirculation pump trip (RPT) features but these were not credited in the design bases accident analysis. The reliability, redundancy, and diversity of actuation logic varied from plant to plant. There was not extensive thermal hydraulic analysis of ATWS behavior nor a general industry awareness of the implications of an unmitigated ATWS event. Not all BWRs had emergency procedures dealing with symptoms and operator actions to control and mitigate ATWS events. Not all BWRs had the capability to inject sodium pentaborate in time to avoid excessive suppression pool heatup.

Prior to adoption of the ATWS rule PWRs did not have the auxiliary mitigation systems actuation circuitry (AMSAC) logic to provide turbine trip and auxiliary feedwater system (AFW) startup via diverse logic from the reactor protection system (RPS). There was not extensive thermal hydraulic analysis of ATWS behavior nor a general industry awareness of the implications of an unmitigated ATWS event. Few PWRs had emergency procedures to control and mitigate ATWS events.

- 10 CFR 50 Appendix B (The QA Rule)

For Appendix B the impact was elicited for two different scenarios.

The first scenario elicited the impact of the QA rule as if the rule had been eliminated. The expert was to imagine that the regulation was adopted into law in 1970 and had recently been eliminated.

In this case, the average plant response to elimination of the regulation will be to maintain (at a minimum) standard industrial practices for non nuclear electric power plants. Equipment and components subsequently purchased will be within specifications and have manufacturers warranties and certifications. Paperwork and recordkeeping will be eliminated including pedigree documentation. Vendor inspections will be eliminated.

The second scenario elicited the impact of the QA rule as if it had never existed. The expert was to imagine that in 1970 the regulation was never adopted into law. What would be the resultant differences in availability of various components and frequency of initiating events today? The average plant response will be to maintain (at a minimum) standard industrial practices for non-nuclear electric power plants. Equipment and components will be within specifications but no manufacturers' warranties, certifications, or pedigree document will have ever been available.

- 10 CFR 50.120 (The Training Rule)

  For the Training Rule the impact was elicited for two different scenarios.

  The first scenario elicited the impact of total implementation of the training rule. This rule was adopted in 1993 and has not been fully implemented by the plants yet. For this elicitation the experts were to imagine that they were in the future, 2001, and determine the resultant difference that total implementation of the regulation has had on the availability of various components and frequency of initiating events.

  The second scenario elicited the impact of the training rule as if the intent of the regulation had never existed. There is a very involved and convoluted history associated with this regulation and a Memorandum of Understanding between the Institute of Nuclear Power Operations (INPO) and NRC. The Memorandum of Understanding required that INPO establish training accreditation programs with NRC guidelines. This was NRC's attempt to escape adopting a training regulation. A more thorough understanding of this history provides a different viewpoint on the intent and importance of this regulation.

  The average plant for this second objective, will be a plant prior to the adoption of the intent of the regulation as contained in the Memorandum of Understanding. Previous to this time plants did not have INPO accredited training programs.

- 10 CFR 50.65 (The Maintenance Rule)

  Determine the impact of total implementation of the maintenance rule. This rule was adopted in 1991 and does not have to be fully implemented by the plants until 1996. For this elicitation the experts imagined that they were in the future, 2001, and determined the resultant difference that total implementation of the regulation has on the availability of various components and frequency of initiating events.

A.3 MODELING FRAMEWORK

The largest technical difficulty in the study was associated with developing credible linkages between a given regulation and the PRA model itself. To systematically understand the relationships between regulations and the impact on the core damage frequency, a complex framework was developed by the INEL (Appendix A.2). The framework adopted for identifying these linkages is shown in Figure A.2-1. Basically there are only four classes of model parameters that impact the results of a risk assessment model: initiating event frequency changes, component unavailability changes, recovery probability changes, and changes to PRA model structure. The changes in these parameters associated with a given rule change were determined using an expert elicitation process (see Section A.5). This process was designed to make the experts consider multiple mechanisms by which a particular parameter might be affected.

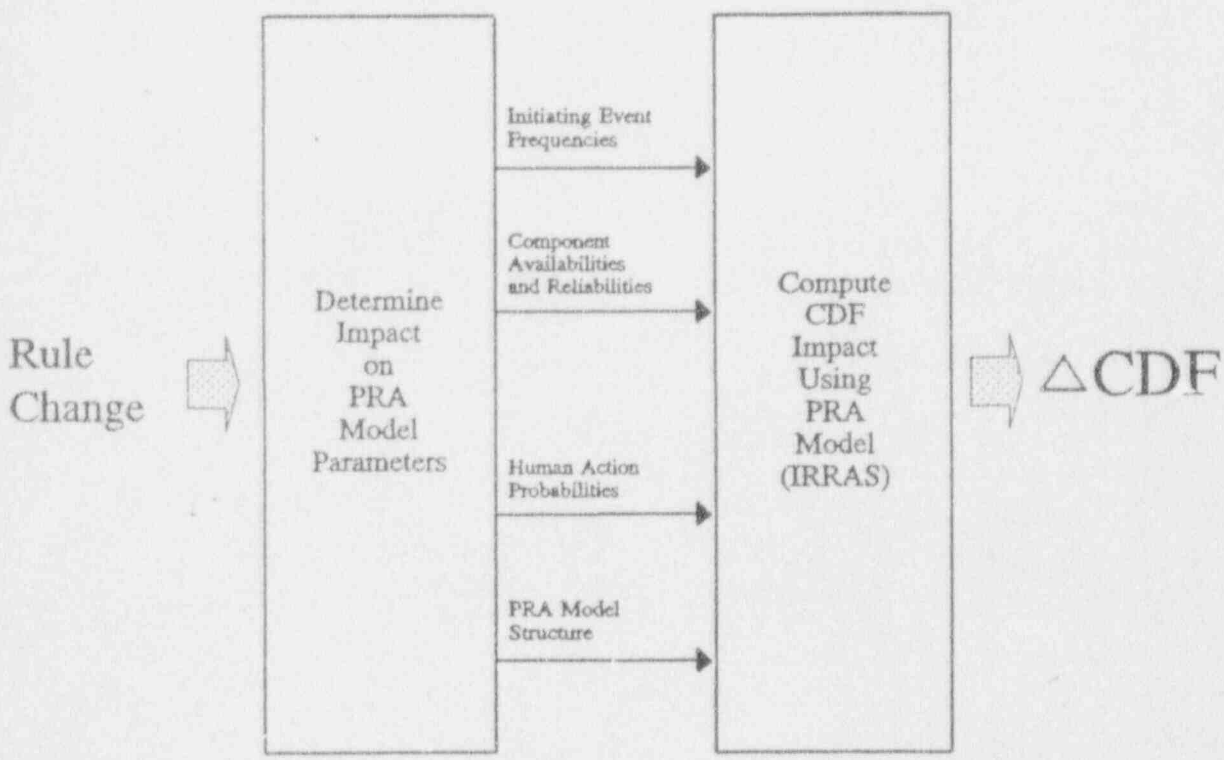

Figure A.3-1. Modeling Framework.

Due to the limitation of this feasibility study only three of the four classes were elicited. First, importance measures were systematically developed for these PRA model parameters using the IRRAS 4.0 code. These measures are useful for predicting the effect of a change in a single PRA model parameter (e.g., a failure rate) or of small changes in a number of parameters. However, in this project, the simultaneous impact of a single regulation change on multiple basic events must be evaluated. Moreover, if the impact is significant, it can be expected that the magnitude of change in a given parameter may be one or more orders of magnitude greater than the original parameter value.

Second the dominant accident sequences and the importance measures were used to develop elicitation questions based on the regulation change hypothesized. The questions were developed to yield the greatest change in impact possible due to the time limitations of this study. Table A.3-1 contains examples of the types of questions concerning the PRA parameters that were elicited.

Table A.3-1
Example of Parameters Elicited

- Availability of scram system (e.g., RPS and ARI for BWR).
- Ability to achieve reactor subcriticality (for BWRs).
- Operator likelihood to initiate emergency actions (e.g., SLC for BWRs).
- Frequency of general plant transients and support system plant transients.
- Availability and reliability of active valves, diesel generator, turbine driven pumps, motor driven pumps, and batteries to perform as needed.
- Likelihood of proper actions by non-licensed operators.
- Likelihood of latent errors by maintenance personnel.

A.4 QUANTIFICATION

For each regulation change discussed in Section A.2, the following steps were performed.

1. The specific scope of impact (i.e., which systems, components, basic events are affected by the regulation change) was determined and the qualitative impact documented in a summary discussion for the expert elicitation. Once the impacts were identified, a walk through example was presented by a normative expert for each elicitation that illustrated how the regulatory change may affect the PRA parameters. For example, in the case of quality assurance, the discussion could cover the different ways that QA impacts the procurement, installation, and testing of equipment.

2. The mechanism of the impact (which PRA parameters might be modified) was determined for the change and elicitation questions were developed.

   For each regulatory change, selected PRA parameters in the dominant sequences were grouped. The elicitation questions for each regulation were designed to directly address these PRA dominant sequence groups. The dominant sequence groups are presented in Appendix A.3.

3. Qualitative measures of the expected change in groups of PRA parameters as a result of the benefit or elimination of selected regulations were elicited. (See Section A.5)

   Both the direction and magnitude of change were elicited. The magnitude elicitation was limited to a descriptive scale (obvious, noticeable, subtle). Figure A.4-1 was presented as a reference for this descriptive scale during the elicitation.

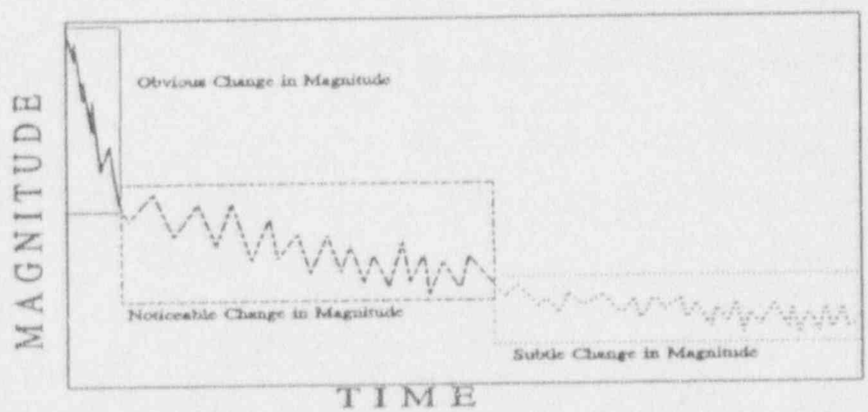

Figure A.4-1. Representation of Descriptive Magnitude Scale.

Direction of change was indicated by an increase, decrease or no change.

No change. This implies there is absolutely no relationship or correlation between the regulation and the PRA parameters. If no change is anticipated then the magnitude is not questioned.

Magnitude of change in effect was elicited if there was either an increase or a decrease in the parameter (e.g., component unavailability).

Subtle A subtle change is anticipated. A subtle change implies a trend which requires a long period of time to discern (e.g., the trend if it exists, is within the range of random fluctuations in data.)

Noticeable A noticeable change is anticipated. A noticeable trend is one which can be detected <u>over time</u>.

Obvious An obvious change is anticipated. An obvious trend is <u>immediately observable</u> (i.e., marked and dramatic).

4. Quantification was accomplished by transforming the descriptive scale (obvious, noticeable, subtle) into numerical values. In this case the descriptions were transformed into values based on the judgement of our PRA experts.

5. The transformed numerical values were combined into a single factor (estimator) for each question; this factor was used to modify the dominant sequence groups, and then the PRA was requantified.

One of the weaknesses in this work concerns the connection between the qualitative data and the development of quantitative values. This is because each expert may have a differing opinion of the numerical values associated with their qualitative responses. Despite this weakness, it is expected the output of the estimation process is useful (in this feasibility study) to indicate the safety significance of these regulations.

A.5 EXPERT ELICITATION

Evaluation of the magnitude of PRA model input parameter changes was based on an expert elicitation process. The types of questions to be elicited centered on cause and effect issues of regulations on plant reliability type parameters. Because of the types of questions being considered the composition of the expert panel was given careful consideration. Most PRA experts tend to work for consulting firms, research organizations, or government bodies. They are not typically involved in the day to day operation of facilities which see the impacts of changes on equipment failure and human error rates. For this reason it was felt that the experts, rather than being PRA experts, should have strong backgrounds involving operational safety assessment (e.g. familiarity with nuclear power plant operations; existing training of licensed and non-licensed operators, maintenance personnel and other support personnel; familiarity with root cause analysis from actual equipment failure incidents; and familiarity with the regulatory process as it impacts day to day plant operations.) To assure diversity in the possible opinions of the experts it was felt that there should be individuals with both NRC and nuclear utility backgrounds. Given these general guidelines for the desired backgrounds of the panel members, an expert panel was assembled consisting of the following:

- An NRC Senior Resident Inspector (familiar with nuclear power plant operations, training, maintenance, root cause investigations, operational safety assessment, and regulatory impact)

- An NRC Senior Licensing Project Manager (familiar with regulatory process, operational safety)

- A nuclear utility Nuclear Engineering Department Manager (familiar with operational safety, PRA, regulatory process, root cause investigations, reliability trend data, familiarity with both BWR and PWR reactors)

- A nuclear utility Licensing Manager from a different utility in a different NRC Region (familiarity with nuclear power plant operations, regulatory process, root cause investigations, familiarity with both BWR and PWR reactors)

- An NRC Research Division Level Manager (familiarity with operational safety, PRA, regulatory process)

A formal training and normative session were held and the experts were formally polled on the direction and magnitude of regulatory driven changes in key parameters impacting the PRA models. The normative session raised general issue statements for each rule were introduced, and an overview of the linkages between plant functions and reliability was explained. Each issue statement summarized the objectives of elicitation, notes background information provided, and defined the suggested baseline plant to be used. Issue statements also defined direction and magnitude of the changes to be elicited, explained the attached elicitation tables, and defined the suggested primary impacts of each rule.

For each hypothetical rule change, experts were asked to judge both the direction and the magnitude of the impact (i.e., Would the rule change increase, decrease, or not change reliability? Would the impact be subtle, noticeable, or obvious?). Specific components, systems, and functions to be considered were specifically named on the elicitation tables provided. The experts were not elicited on either a change in the actual core damage frequency or the source terms but in general terms on the availability of a certain component or group of components and on the likelihood of a human action.

The experts were asked to first write their own estimates of the impacts of rule changes, and the basis for those impacts, on the forms provided. After all experts had completed making their estimates, they were asked in random sequence to disclose and explain their initial estimates to the rest of the group. The experts were then given the opportunity to privately change their estimates, and to provide any additional reasoning on the provided forms. For each rule, a brief closing discussion was held.

For the expert elicitation the qualitative results were summarized for each regulation as shown in the sample expert elicitation table shown in Table A.5-1. The final summarized expert elicitation tables are contained in Appendix A-4. There was excellent agreement between the experts concerning the direction of the regulation change impacts, and reasonably good agreement concerning the magnitude of impacts.

Table A.5-1
Example of Expert Elicitation

	ELICITATION QUESTION/EXPERT RESPONSE	EXPERT A	EXPERT B	EXPERT C	EXPERT D	EXPERT E
	10 CFR 50.62 BWR ELICITATION					
1	Has the availability of the automatic scram system (RPS, ARI) increased or decreased?	Subtle Increase	Noticeable Increase	Subtle Increase	Subtle Increase	Expert not elicited
2	Has ability to achieve reactor subcriticality (via boron injection and RPT) increased or decreased?	Noticeable Increase	Subtle Increase	Noticeable Increase	Subtle Increase	Expert not elicited
3	Has the likelihood of the operator to initiate SLC (given an ATWS event) increased or decreased?	Noticeable Increase	Noticeable Increase	Obvious Increase	Noticeable Increase	Expert not elicited

A.6 RESULTS

As described in Section A.1, this feasibility study investigated four hypothetical regulatory changes and analyzed the impact to core damage frequency from each change. The method used to quantify the impact of regulation changes was based on identifying appropriate changes to the PRA model, eliciting qualitative measures, and then quantifying the impact of these changes as described in Section A.4.

After the expert elicitation, the qualitative results from the elicitation were summarized. These qualitative results were transformed into quantitative values by the PRA experts in the study. To transform the qualitative results as shown in Table A.5-1, the descriptive scale for both availability and frequency of an event was given a quantitative value as summarized as follows:

	Frequency	Availability or Likelihood
No Change	1.0	1.0
Subtle	1.2	2.0
Noticeable	2.0	5.0
Obvious	3.0	10.0

The PRA experts based these values on the knowledge of what constitutes a magnitude of change in the PRA. The transformation values will differ depending on whether the impact is either the initiating event frequency or in the availability of a basic event. For example a subtle change in magnitude in the number of transients would require a lower factor of change (1.2 versus 2.0) to see a resulting change in the PRA, than a subtle change in the availability of the basic event.

The quantitative values were then combined into a single factor (estimator) for each question. Each factor was used to modify the dominant sequence groups of events from the dominant sequences and the PRA was requantified to yield a new core damage frequency (core damage frequency after). The results as given in Table A.6-1 and shown in Figures A.6-2 and A.6-3 demonstrate that the methodology can distinguish between impacts of different regulations. The experts elicited determined that the implementation of each regulation had some positive effect in reducing core damage frequency.

It is important to note that the experts were careful to indicate that it was difficult to determine the impact of a regulation change because of the potential influences from other policies, regulations, and general knowledge. It was beyond the scope of this elicitation to investigate the interaction of combinations of regulations and other policies. The short time frame only allowed for preparation of both the elicitation questions and background on each regulation at a gross level of detail. It is felt, however, that an expanded elicitation could provide the experts guidelines as to how to determine the contribution from each specific regulation, or in some instances, combinations of regulations and other policies as appropriate.

Table A.6-1
Summary of Core Damage Frequency Impacts

	BWR BASE CASE CDF = 3.6E-6			PWR BASE CASE CDF = 3.2E-5		
REGULATION CHANGE	CDF AFTER	DELTA CDF	FACTOR	CDF AFTER	DELTA CDF	FACTOR
ATWS (impact of the rule)	1.6E-5	-1.3E-5	4	4.0E-5	-8.2E-6	>1
APPENDIX B (impact if the rule had never existed)	6.9E-6	-3.3E-6	2	1.2E-4	-9.0E-5	4
APPENDIX B (impact if the rule were eliminated)	4.8E-6	-1.2E-6	>1	6.6E-5	-3.4E-5	2
TRAINING (impact if the rule were implemented)	3.4E-6	2.4E-7	0.9	2.7E-5	4.5E-6	0.8
TRAINING (impact of the intent of the rule)	1.9E-5	-1.6E-5	5	3.8E-4	-3.5E-4	12
MAINTENANCE (impact if the rule were implemented)	2.8E-6	8.2E-7	0.8	2.0E-5	1.2E-5	0.6

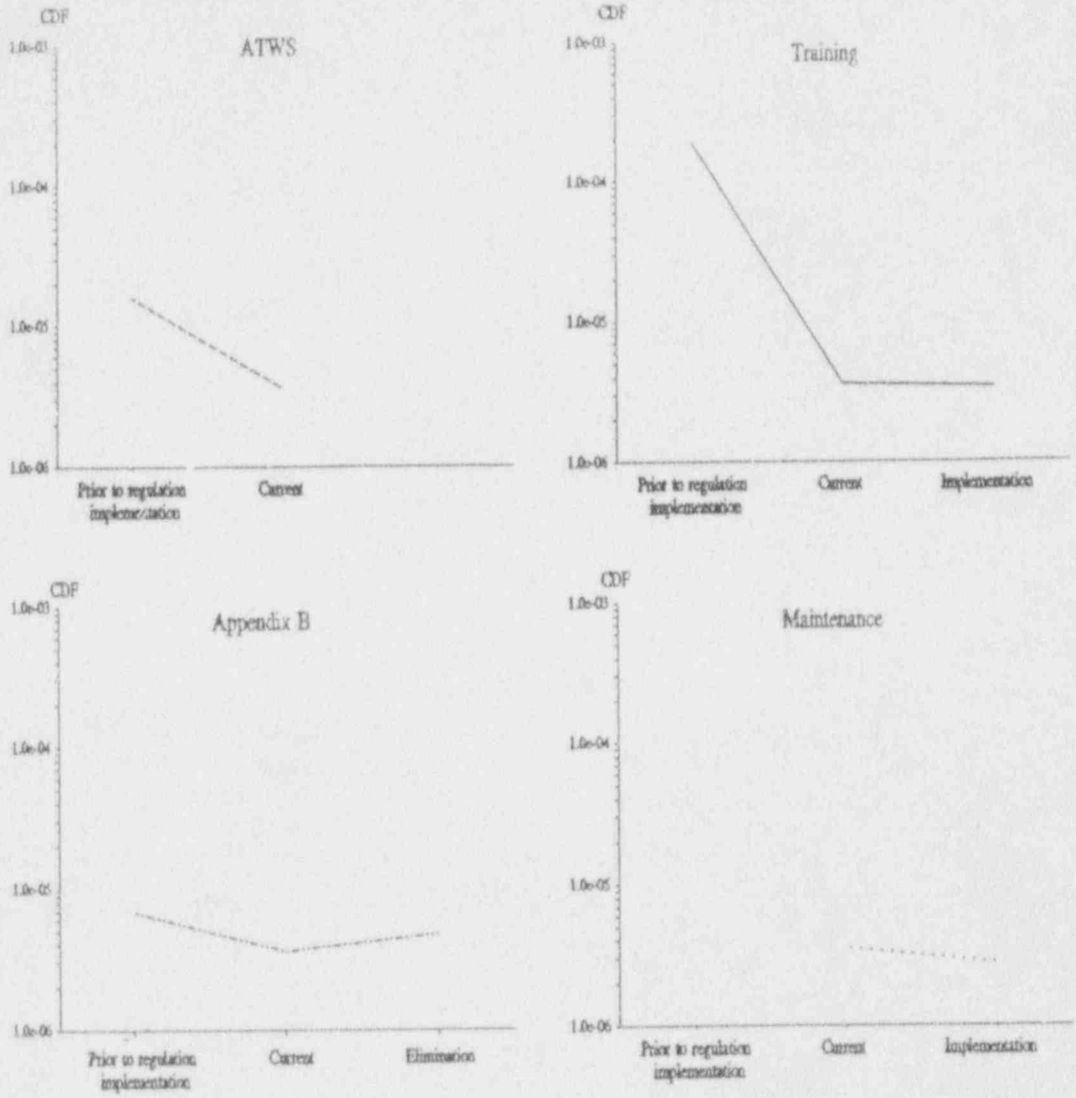

Figure A.6-2. Resultant BWR Core Damage Frequencies Due to Changes in the Regulation.

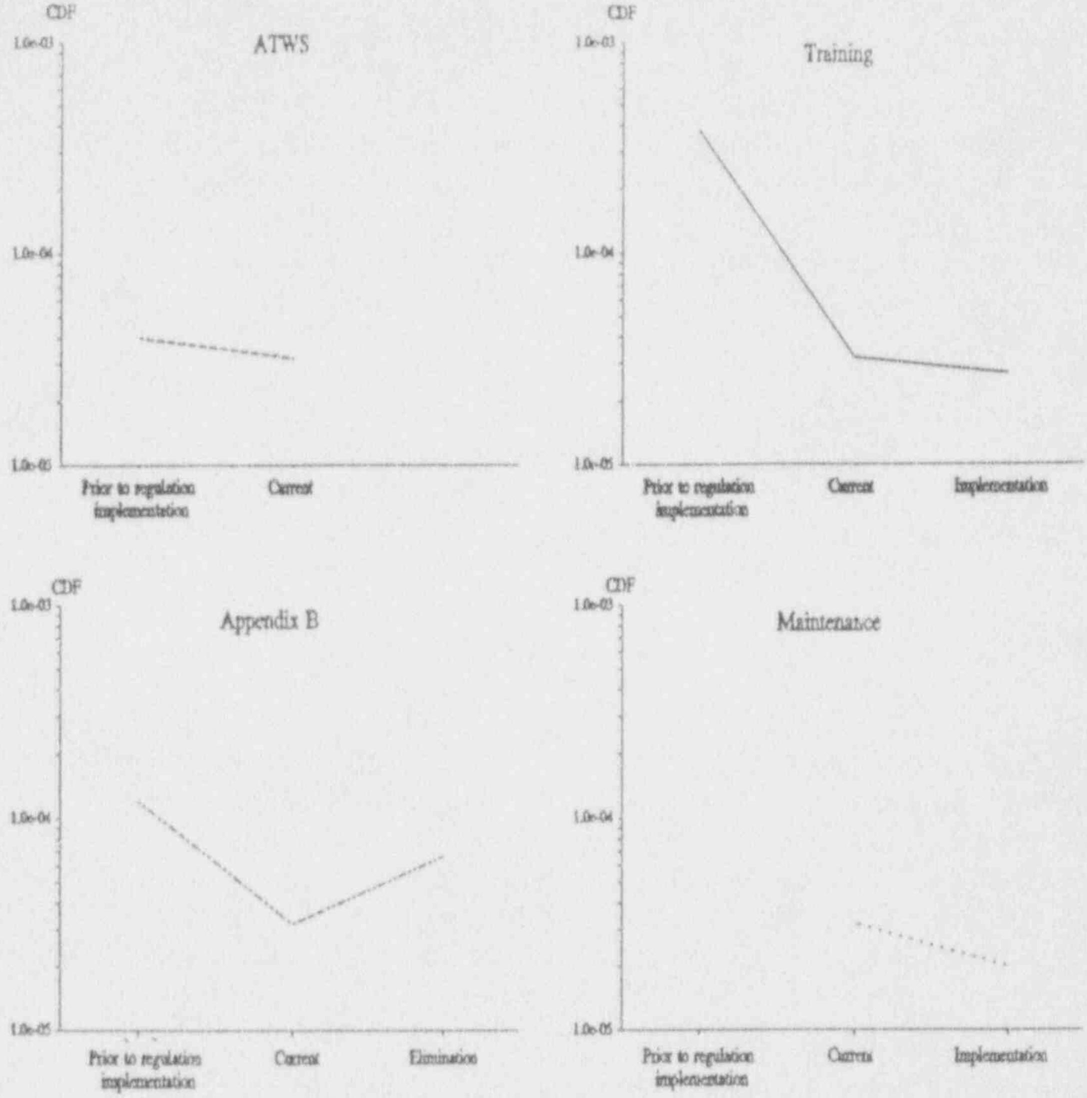

Figure A.6-3. Resultant PWR Core Damage Frequency Due to Changes in the Regulation.

The training rule had the greatest impact of any of the regulation changes on the dominant sequences in both the BWRs and PWRs. The training rule was elicited for two impacts. The impact of total implementation of the rule from the current 1993 situation and the impact of the intent of the regulation prior to the 1982 time frame. As is shown in Figures A.6-2 and A.6-3, the plant core damage frequencies, prior to implementation of the regulation, currently, and after total implementation are presented from left to right. The experts commented that the implementation of a systematic approach to training (this was initiated in 1983) had a large positive influence. They felt it was difficult to tell how much of this change was actually due to the regulation and how much is related to other influences.

The ATWS rule was elicited only on its impact prior to implementation. For the BWR it provided a greater factor of reduction in core damage frequency (4.4) than in the PWR (1.3). The experts felt that its greatest contribution was in increasing the knowledge base concerning reactor behavior and as a result the operators have an increased likelihood to initiate pentaborate injection given an ATWS.

The QA rule, 10 CFR 50 Appendix B, was also elicited for two impacts. The impact of total elimination of the rule from the current 1993 situation and the impact of the intent of the regulation prior to the current time frame. The results of the elicitation on Appendix B demonstrate that some benefit (a factor of 2 in BWRs and 4 in PWRs) was obtained from the original institution of this event. The benefit appears to produce a slightly better reduction in core damage frequency for the PWRs. The experts discussed that with the elimination of this regulation some benefit would be lost (a factor slightly greater than 1 in BWRs and 2 in PWRs) but that the average plant would maintain a level of QA that would maintain the core damage frequency at much the current level. They also stated that information added to the knowledge base would not be lost with subsequent elimination of the regulation and this would also contribute to the maintenance of the current core damage frequency.

Information on the maintenance rule was elicited only on its impact after full implementation in the future. All the experts determined that the maintenance rule contributed the least to reduction of core damage frequency. For the BWR, it provided a greater factor of reduction in core damage frequency (4) than in the PWR (slightly greater than 1). The elicitation brought out that the maintenance rule's greatest impact was an increased focus on safety significance of support systems. The reduction in core damage frequency will be subtle and largely due to the increase in the knowledge base.

A.7 STUDY LIMITATIONS AND ISSUES

For this feasibility study it was necessary to limit the number of regulations, the number of changes, and the questions that could be considered for expert elicitation and analysis was limited in scope due to time and budget constraints.

A.7.1 Limitations

By characterizing the mechanism and scope of the impact to the plant analysis, the analysis was limited to four regulations. These regulations are selected from a larger set which has been assessed by NRC as being potentially safety significant. The classification and selection of regulations for impact analysis is performed in such a manner that the impact assessment methods developed during the analysis should be applicable to regulation changes not evaluated in this project.

It was necessary to limit the elicitation to the response of an average plant. It is anticipated that a more dramatic change in impact would be expected for an elicitation directed at the poor performer.

Regulation changes were restricted to either total elimination or implementation of the regulation. Changes in the regulations were evaluated one rule at a time. Thus, combinations of regulations changes were not addressed. It is expected that the methods developed in this project could be easily extended to handle multiple rule changes.

An analysis of the dominant accident sequences was performed to determine changes in terms of core damage frequency. It is recognized that some of the rules are focused on ensuring worker and public health and safety. The methodology (if not necessarily the specific models) employed in this analysis is expected to be directly applicable towards the analysis of the public health risk importance of the rules; the treatment of worker safety may require some additional methodological developments. This study was directed at obtaining core damage frequency values and did not directly address any source term values.

Delta core damage frequencies were computed using the NUREG-1150 models for the Surry *[Ref. A-3]* and Peach Bottom *[Ref. A-4]* plants. Regulations aimed specifically at proposed plants, plants under construction, or plants being decommissioned, were not be evaluated.

A.7.2 Issues

The largest technical difficulty identified in this study was associated with developing credible linkages between the regulations and the PRA model itself. In an attempt to understand this issue, an overall framework was adopted for identifying these linkages. During this process several concerns were expressed. One concern is the limitations in the ability to internally generate and integrate the diverse processes linking a regulation change to potential changes in a PRA model. To assist in this mental process it may be possible to use a tactic of decomposition and employment of an "influence diagram" modeling framework illustrating causal mechanisms for PRA model changes. This conjectural framework is presented in Appendix A.2.

The level of response to a change in an existing regulation may differ greatly between plants. It is possible that a range of plant responses are possible. It is anticipated this range of responses will extend from the poor performer to the above average plant. Assuming that most regulation is written to directly address weakness specific to the poor performer plants the result of the implementation of a regulation should be an overall increase in the average safety across plants. An elicitation directed at the poor performer would most likely result in a larger change than at an average or above average plant. Eliciting the range of plant responses may have yielded important information on the safety response to the regulation. It can be postulated that a graph of the level of implementation of a rule and the change in safety (i.e., core damage frequency) would result in an S shape curve as shown in Figure A.7-1.

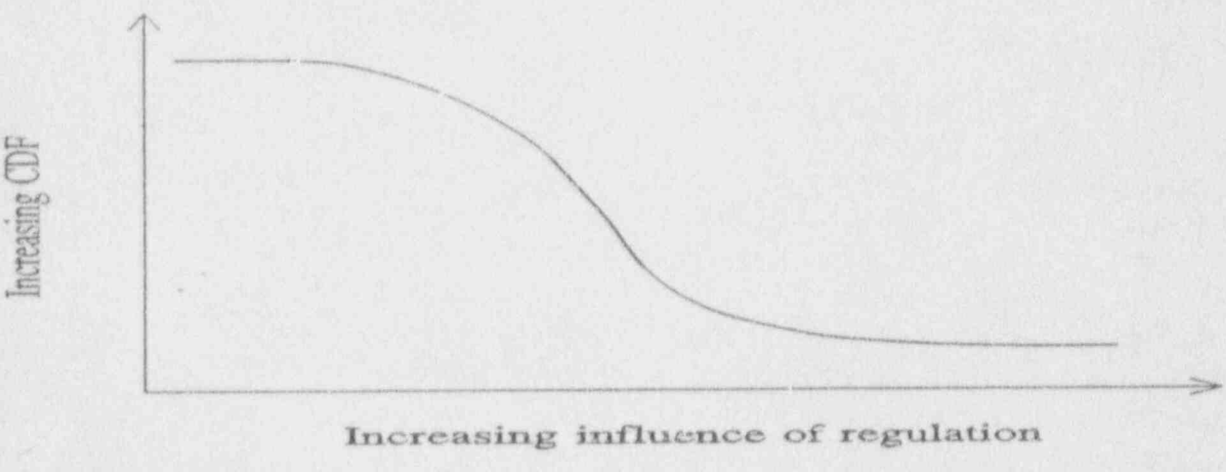

Figure A.7-1. Implementation Curve.

The curve will flatten out at the bottom, because there may be a level of implementation after which no increase in safety may be seen and even conversely a decline in safety may be possible. For example, human factors studies have shown that there is a point at which maximum benefit is obtained from training. There may be a point where training will no longer benefit and in fact takes enough time away from other important plant responsibility to actually reduce safety. It could be postulated that an above average plant response may be close to this area of the curve. The curve will flatten out at the top because there should be some level of safety that without that regulation implementation a plant cannot fall below. For example, there must be some minimum amount of training that a plant must maintain just to continue functioning. If formalized training were not available, informal on the job training would no doubt quickly replace this function to some extent in the plant. It could be postulated that a plant that was poor performer may come close to this area of the curve. Additional, no one regulation can account for 100% of the safety inherent in a plant. Therefore the total elimination of this rule would have to allow for some contribution to safety from other sources.

The total elimination or implementation of the regulation are the most extreme situations with the potential for wide ranging affects on safety and were used in this study. This type of regulation change is applicable to any regulation. Removal, revision or the reduction of scope of specific sections may be also be applicable in certain cases. For example, the ATWS rule has sections that deal specifically with either PWRs or BWRs and it is possible that the section could be removed, revised or the scope reduced without significantly affecting safety. Additionally, proposed changes in the use of the Regulatory Guides and other documentation that provides guidance for implementation of the regulation may prove to be important.

A.8 CONCLUSIONS AND RECOMMENDATIONS

Based on the results of this limited scope feasibility study the following conclusions can be made:

- While some biases may exist in the interpretation of the impacts of regulations on PRA model parameters (event frequencies, component reliabilities, etc.) the relative directions and relative magnitudes of changes are not generally disputed by either utility or NRC experts. This result was unexpected.

- Utilizing the adjusted PRA model parameters obtained by the expert elicitation, it is possible to quantify and differentiate the impacts of specific NRC regulations. This can be done by identifying in a systematic fashion how regulations impact the frequency of potential initiating events, component reliability, probability of recovering failed systems, and PRA model structure (number of barriers available).

In the course of performing the study, several issues were identified that warrant further consideration in interpreting the absolute values obtained. The results obtained are a reflection of the specific modeling approaches taken in the Surry and Peach Bottom NUREG-1150 PRA studies. As an example, differences in the approaches taken for modeling the reactor protection system unavailability have an effect on the magnitude of the ATWS Rule impacts which will differ between PWRs and BWRs. Consideration of how to deal with these subtle differences could be the subject of future work. As an additional issue, the use of expert elicitation to estimate the likely changes to PRA model inputs is potentially biased by the inability to separate out the effects of numerous rule changes and industry initiatives which have been underway for the last decade. Each of the experts commented on this problem and this should be given further thought.

No attempt was made at addressing the uncertainty that is present in PRAs. This uncertainty comes from a variety of sources including random variation in component failure rates and initiating event frequencies. A final uncertainty was analyzed for the total core damage frequency for both the Surry and Peach Bottom PRAs. Error factors of 10 and greater are generally calculated for final results. The delta core damage frequencies that were observed in this study are within the noise of this variability. Despite this, the results are consistent, and can be valuable as a tool for further study.

To improve the ability to differentiate the impacts of a wider body of regulations, it will be necessary to eventually consider the impacts on PRA "back-end" parameters (e.g. those parameters which impact source terms and public exposure). This is a recommendation for future work that would lead to the ability to evaluate rules like the Combustible Gas Control Rule (10 CFR 50.44). The evaluation of these inputs can be done as an extension of the basic methodology put together for this feasibility study.

A.9 ACRONYMS AND REFERENCES

A.9.1 Acronyms

AFW	Auxiliary Feedwater
AMSAC	Auxiliary Mitigation Systems Actuation Circuitry
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA	Quality Assurance
RPS	Reactor Protection System
RPT	Reactor Pump Trip
SLC	Standby Liquid Control

A.9.2 References

[A-1] Chapter 10 of the Code of Federal Regulations, Part 50, Domestic Licensing of Production and Utilization Facilities, revised as of January 1, 1993.

[A-2] K. D. Russell et al., "Integrated Reliability and Risk Analysis System (IRRAS)," Version 4.0, NUREG/CR-5813, January 1992.

[A-3] R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah, Unit 1 Internal Events," NUREG/CR-4550, Volume 5, Parts 1 and 2, April 1990.

[A-4] A. M. Kolaczkowski et al, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events", NUREG/CR-4550, Volume 4, Revision 1, August 1989.

APPENDIX A-1 REGULATION ANALYSIS SUMMARY TABLES

Table A-1.1
Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
50.5	Deliberate misconduct	Indirect	O	A	I	both	I	✓			P	✓	✓	✓	potential influence great
50.7	Employee Protection	Indirect	O	A	I	both	I	✓			P	✓	✓	✓	potential influence great
50.9	Completeness & Accuracy	Indirect	O	A	I	both	I	✓			P	✓	✓	✓	potential influence great
50.34	Contents of application; technical information	Marginal	Op	M,A	EP,Se,D,Pr,M	both	I	✓	✓	✓	P	✓	✓	✓	is this of concern to CDF
50.34a	Design objectives for equipment of control releases of radioactive material in effluent - nuclear power reactors	Indirect	Op	M	Pr	both	I				L			✓	effects worker safety not concern to CDF
50.36	Technical Specifications	Substantial	O	M	D,Pb,F,b,C,Pa,U	both	D	✓*	✓*		P	✓	✓	✓	*a,b demand failure & T&M *will have cutsets with simultaneous T&M
50.44	Standards for combustible gas control system in light-water-cooled power reactors	Substantial	Op	M	D,M,C	BWR	D				L	✓	✓		BWR risk is greater since it has greater CDF implications
						PWR	D	✓*	✓		L	✓	✓		*does the H2 explosion in core affect CDF

Table A-1.1
Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
50.46	Acceptance criteria for emergency core cooling systems for light water nuclear power reactors	Substantial	Op	Pb	U	both	D	✓		✓	L	✓	✓		
50.47	Emergency Plans	Indirect	Op	Pb	U	both	I								is this of concern to CDF
50.48	Fire Protection	Indirect	[illegible]	Pr	Pr,M	both	D	✓	✓		P	✓	✓	✓	
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants	Substantial	O	Pr	Pr,Pb, M,Pa	both	D	✓	✓*		P	✓	✓	✓	*if things were screened because designed to fail under certain environmental conditions
50.51	Duration of license, renewal	Indirect	O	A	A	both	D	✓	✓*		P	✓	✓	✓	*piping and components
50.55	Conditions of construction permits	Indirect	O	A	A	both	I	✓			P	✓	✓	✓	allows changes to the QA program
50.55a	Codes and standards	Substantial	O	Pr	M,Pb	both	D	✓	✓*		P	✓	✓	✓	determines the quality of the system *boundary condition assumes no piping or reactor failures
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards	Indirect	O	A	A	both	I	✓			P	✓	✓	✓	learn from experience

Table A-1.1

Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
50.59	Changes, tests and experiments	Substantial	Op	Pb	Every-thing	both	D	✓	✓	✓	P	✓	✓	✓	Dependent on change or specific application
50.60	Acceptance criteria for fracture prevention	Indirect	Op	Pr	Pr,Pb	both	D	✓	✓		P	✓	✓	✓	Licensee require to meet Appendices G&H
50.61	Fracture toughness requirements protection against pressurized thermal shock events	Indirect	Op	Pr	Pr,Pb	both	D		✓[a]		L			✓	[a]would add a term for failure
50.62	Requirements for reduction of risk from ATWS events for light-wa..r-cooled nuclear power plants	Marginal	Op	M	D,Pr,F	BWR	D	✓		✓[a]	L	✓	✓		[a]may make a different whether a new or existing plant important for BWR
						PWR	D	✓		✓[a]	L		✓		[a]may make a different whether a new or existing plant
50.63	Loss of alternating current power	Indirect	Op	Pb	Pa	both	D	✓		✓[b]	L[a]	✓	✓		[a]affects only new plants [b]electric power
50.65	Requirements for monitoring the effectiveness of maintenance at nuclear power plants	Substantial	Op	Pb	Pr,M	both	D	✓			P	✓	✓	✓	Maintenance Rule

Table A-1.1
Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
50.72	Immediate notification requirements for operating nuclear power reactors	Indirect	Op	Pr	A	both	D	✓			L	✓	✓	✓	could affect when support arrives
50.73	Licensee event report system	Indirect	Op	Pr	A,D	both	I	✓	✓		P	✓	✓	✓	Learning issue, safety culture, maintenance worsens, human error value change, do we find new failure mechanisms
50.80	Transfer of licenses	Indirect	O	A	A	both	I	✓			P	✓	✓	✓	Safety culture, unqualified licensee could run plant poorly
50.82	Applications for termination license	Indirect	O	M,A	A	both	I								Plant is down, exclude shut down risk
50.90	Application for amendment of license or construction permit	Indirect	O	A	A	both	I	✓	✓	✓	P	✓	✓	✓	allows changes to the license
50.92	Issuance of amendment	Depends on Amendment	O	A	A	both	D	✓	✓		P	✓	✓	✓	dependant on change, same impact as TS, need to be more careful about changes in reg

Table A-1.1
Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
50.100	Revocation, suspension, modification of licenses and construction permits for cause	Indirect	O	A	A	both	I	✓			P	✓	✓	✓	safety culture issue
50.101	Retaking possession of special nuclear material	Indirect	O		S	both	I								Plant already lost its license & is not operating
50.103	Suspension and operation in war or national emergency	Indirect	O	O	S	both	D	✓		✓	L	✓	✓	✓	Affects CDF if external conditions cause marginal plant operating conditions or to minimize the risk due to sabotage
50.109	Backfitting	Indirect	O	Pr	A	both	D	✓	✓	✓	P	✓	✓	✓	depends on what backfit is
50.120	Training Rule	Indirect													
APP.A	General Design Criteria for Nuclear Power Plants	Substantial	Op	Pb	D,Pb,M, Pr,Pa, U,C	both	D	✓	✓	✓	P	✓	✓	✓	
APP.B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	Substantial	Op	Pb	D,Pr, Pb,Pa, M,F,C, U	both	D	✓			P	✓	✓	✓	
APP.G	Fracture toughness requirements	Indirect	Op	Pr	Pr,Pb	both	D	✓	✓		P	✓	✓	✓	

Table A-1.1
Summary of 10 CFR 50 Regulatory Review Forms

Regulation No.	Title/Subject	Contribution to Safety	Rule Basis[1]	Rule Type[2]	Rule Group[3]	Rule for PWR or BWR	Impact Mechanism[4]				Impact Scope[5]				Comments
							D	M	A	R	Extent	D	N	O	
APP.H	Reactor Vessel Material Surveillance Program Requirements	Indirect	Op	Pr	Pb	both	D		✓		L			✓	

Notes:

1. P = PRA Based, Op = Operating Experience, O = Other.
2. Pb = Performance Based, Pr = Prescriptive, M = Mixed, A = Administrative, O = Other.
3. EP = Emergency Planning, S = Security, D = Defense in Depth, Pr = Prevention, M = Mitigation, Pb = Pressure Boundary, F = Fuel Boundary, C = Containment, Pa = Power Availability, U = Ultimate heat sink, A = Administrative Control, E = Everything in plant, I = Integrity.
4. D = Directness, I = Indirect, D = Direct, M = Modify Parameters, A = Add Parameters, R = Remove Parameters.
5. L = Local, P = Pervasive, D = Dominant Scenarios, N = Nondominant Scenarios, O = Outside PRA.

Table A-1.2
PRA Grouping of 10 CFR 50 Rules

Impact Mechanism				Impact Scope (PRA)				
X_1 Directness	X_2 Modify Parameters	X_3 Add Parameters	X_4 Remove Parameters	X_5 Extent	X_6 Dominant Scenarios	X_7 Non-Dominant Scenarios	X_8 Outside PRA	10 CFR 50 Rules
I								50.47,50.82,50.101
I				L			Y	50.34a
I	Y			P	Y	Y	Y	50.5,50.7,50.9,50.55, 50.58,50.80,50.100
I	Y	Y		P	Y	Y	Y	50.73
I	Y	Y	Y	P	Y	Y	Y	50.34,50.90
D		Y		L			Y	50.61,APP.H
D	Y		Y	L		Y		50.62
D	Y	Y		L	Y	Y		50.44
D	Y		Y	L	Y	Y		50.46,50.62(BWR),50.63
D	Y			L	Y	Y	Y	50.72
D	Y		Y	L	Y	Y	Y	50.103
D	Y			P	Y	Y	Y	APP.B,50.65
D	Y	Y		P	Y	Y		50.120
D	Y	Y		P	Y	Y	Y	50.36,50.48,50.49,50.51, 50.55a,50.60,50.92,APP.G
D	Y	Y	Y	P	Y	Y	Y	50.59,50.109,APP.A

I = Indirect D = Direct L = Local P = Pervasive

APPENDIX A-2 MODELING FRAMEWORK

A-2.1 Introduction

The largest technical difficulty identified in this study was associated with developing credible linkages between the regulations and the probabilistic risk assessment model itself. In order to assess the impact a given regulation change has on core damage frequency, it is useful to delineate specific mechanisms on the two. In an attempt to understand this issue, an overall framework was adopted for identifying these linkages. During this process several concerns were expressed.

One concern is the limitations in the ability to internally generate and integrate the diverse processes linking a regulation change to potential changes in a probabilistic risk assessment model. A possible method to assist in this mental process is a tactic of decomposition and employment of an "influence diagram" modeling framework illustrating causal mechanisms for probabilistic risk assessment model changes. Such a framework, while it may provide a possible biasing mechanism, may be beneficial for focusing discussion and should allow judgments to be made.

It was felt that the complexity of the framework would be too difficult to present during the elicitation, given the time constrains and the type of expert that was elicited in this feasibility study. To avoid this the model framework was used to assist in the formulation of questions for the expert elicitation.

It is important to note that the model was used successfully in this limited analysis. In attempting to obtain a more robust and detailed analysis this model may not adequate. Many of the comments from the experts elicited related to the complexity of the questions and the difficulty in developing a qualitatively or quantitatively assessment of impact.

A-2.2 General Modeling Framework

A general modeling framework postulating the linking between the regulation and the probabilistic risk assessment model is shown in Figure A-2.1. Of the three transfer function boxes in that figure, the last one (the IRRAS computation) is straightforward in principle in that it describes the four basic classes of model parameters that input to a risk assessment model: initiating event frequency changes λ, component unavailability changes, recovery probability changes, and changes to probabilistic risk assessment model structure. This section briefly discusses the how the first two boxes were designed to consider the multiple mechanisms by which a regulation change can affect a particular probabilistic risk assessment model parameter.

Impact on Safety-Related Entities and Processes

In principle, a given regulation change will have a direct (primary) impact on some subset of the nine entities and processes shown in Figure A-2.1 (the definitions of these entities and processes are provided in Table A-2.1). Further, a primary impact on any one can cause subsequent secondary impacts on the others. In order to determine the total impact of a given rule change, it is useful to represent the relationships between the entities/processes. Table A-2.2 shows a matrix representation of the significant interrelationships; Figure A-2.2 shows the somewhat convoluted influence diagram representation.

To read these figures, consider the following example. Assume that a given rule change has a primary impact on Paperwork and Recordkeeping. According to our model, this change can have a secondary impact on both Training and Administrative Controls (to implement the primary change). The magnitude and significance of the secondary impact depends on the particular rule change involved.

Propagation of Entity and Process Changes to probabilistic risk assessment Model Changes

The primary and secondary impacts of a regulation change shown in Table A-2.2 and Figure A-2.2 provide the output from the left-hand box in Figure A-2.1. To couple these impacts to the probabilistic risk assessment parameters and model structure exiting the middle box, additional linkages have to be provided. Figure A-2.3 shows these couplings in terms of key plant safety functions (the boxes), desired outcomes (the large circles), and the different types of probabilistic risk assessment parameters (the small circles). Thus, for example, a degradation in maintenance capabilities brought about by a regulation change could lead to degraded equipment conditions, and therefore to increases in the frequencies of certain initiating events and component unavailabilities.

The linkage between Table A-2.2 and Figures A-2.2 and A-2.3 is through the function boxes in Figure A-2.3. A change in one of the nine entities and processes will have an impact on the functions. Note that some of the entities/processes in Table A-2.2 and Figure A-2.2 correspond directly to functions in Figure A-2.3 (e.g., Training). In other cases, a number of functions can be affected by a single entity and process (e.g., Paperwork and Recordkeeping). A generic diagram illustrating the connection between Table A-2.2 and Figures A-2.2 and A-2.3 is not provided because it is expected that the connection is fairly specific to the rule change being considered.

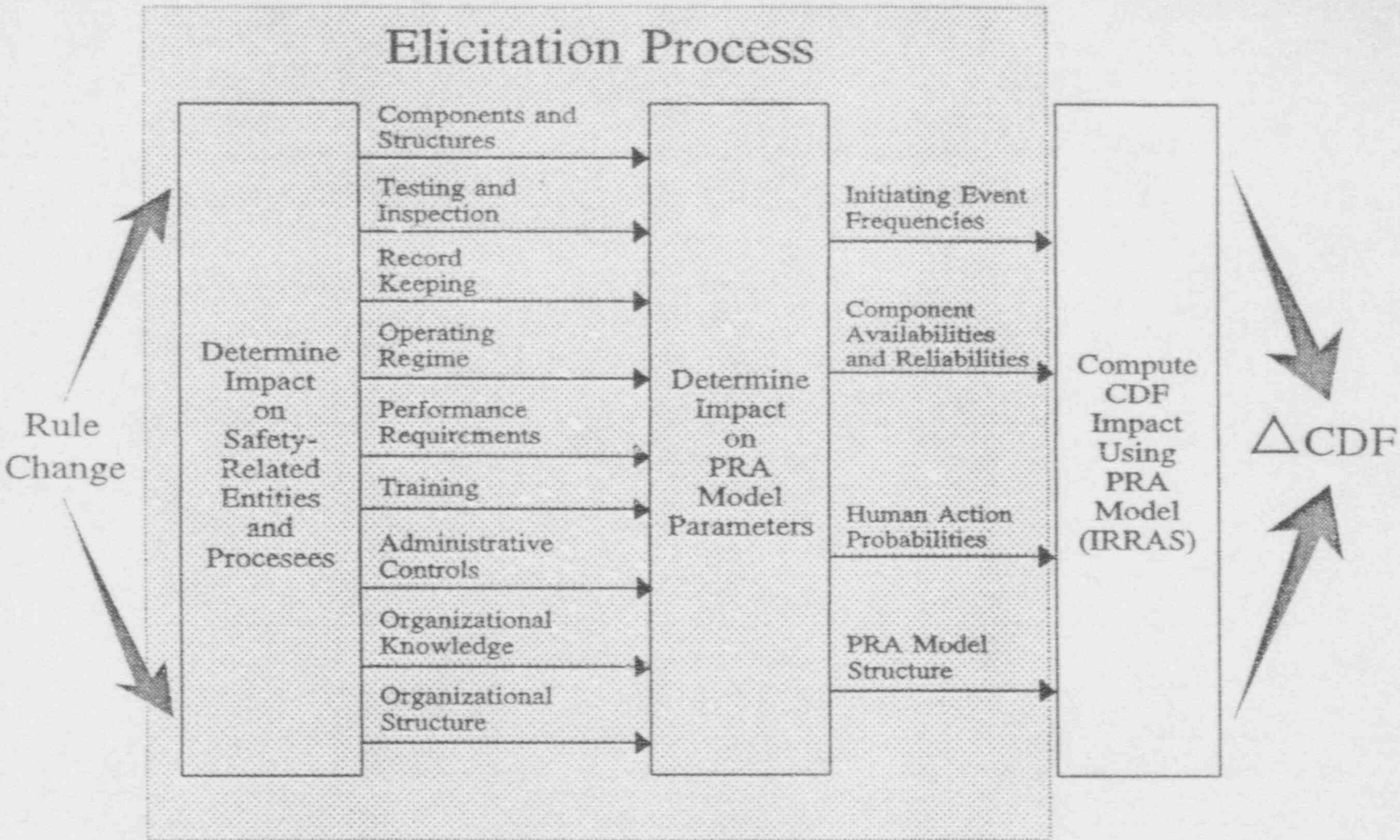

Figure A-2.1. General Modeling Framework.

Table A-2.1
Suggested Definitions of Primary Impacts Due to Regulation Changes

CHANGES TO	DESCRIPTION OF IMPACT
Components and Structures	This is the most obvious impact caused by a regulation and involves addition (or removal) of a component or structure.
Required Tests/Inspections	Tests and inspections (including the procedures for carrying them out) provide verification of system operability or functional integrity. Examples include: relay logic test, inservice tests of pumps and valves, steam generator tube eddy current inspections.
Allowed Operating Regimes/Procedures	Certain regulations cause changes to the allowed regimes of plant operation. Typically such limits are defined as LCOs in Technical Specifications or as procedural limits in operating, maintenance, or emergency procedures. Examples include: allowed power levels, power distribution limits, ultimate heat sink temperatures, allowed torus water levels/temperatures boiling water reactors (BWRs), and air ejector radiation levels.
Performance Requirements	Certain regulations deal with monitoring (and in some cases trending) required levels of performance in systems or components and actions which must be taken when deviations are found. Examples include: diesel starting reliability (Regulatory Guide 1.109), system availability monitoring per the maintenance rule, trends in pump bearing noise signatures.
Training	Regulations can impact the training of operators, PEOs, I&C and maintenance personnel, general plant staff (security, health physics, emergency plans etc.), fire brigades, and emergency operations staffs.
Administrative Controls	In many cases administrative controls (administrative procedures) provide limitations on equipment status and personnel activities not covered in the LCOs in Technical Specifications. Examples include: limitations on allowable shift overtime levels, vital area access, controls on flammable materials in vital area, material controls in containment to avoid sump strainer blockage, and signoff approvals for : lockouts/tagouts of active equipment and maintenance work orders.
Studies/Knowledge Acquisition	Certain regulations require evaluations or investigations which may alter the knowledge base and prompt other followup actions. Examples include: root cause investigations required in Licensee Event Reports and performing Independent Plant Evaluations.
Organization	Regulations have resulted in creation of or changes to organizations such as independent safety engineering groups, PORCs, SORCs, Nuclear Review Boards, and dedicated fire brigades.
Paperwork/Recordkeeping	Many regulations require recordkeeping to provide an auditable trail allowing the Nuclear Regulatory Commission to confirm that regulations are in fact being followed.

Table A-2.2
Matrix Representation of Relationship Among Entities and Processes

	PRIMARY IMPACT DUE TO CHANGE IN RULE	SECONDARY IMPACTS ON PROCESSES								
		1	2	3	4	5	6	7	8	9
1	Components and Structures	--	x	x	x	x	x			x
2	Tests and Inspections		--			x	x	x		x
3	Operating Regimes and Procedures		x	--		x	x			x
4	Performance Requirements		x		--	x		x		x
5	Training					--				x
6	Administrative Controls					x	--			x
7	Studies, Knowledge and Acquisition	x	x	x	x	x	x	--	x	x
8	Organization					x	x		--	x
9	Paperwork and Recordkeeping					x	x			--

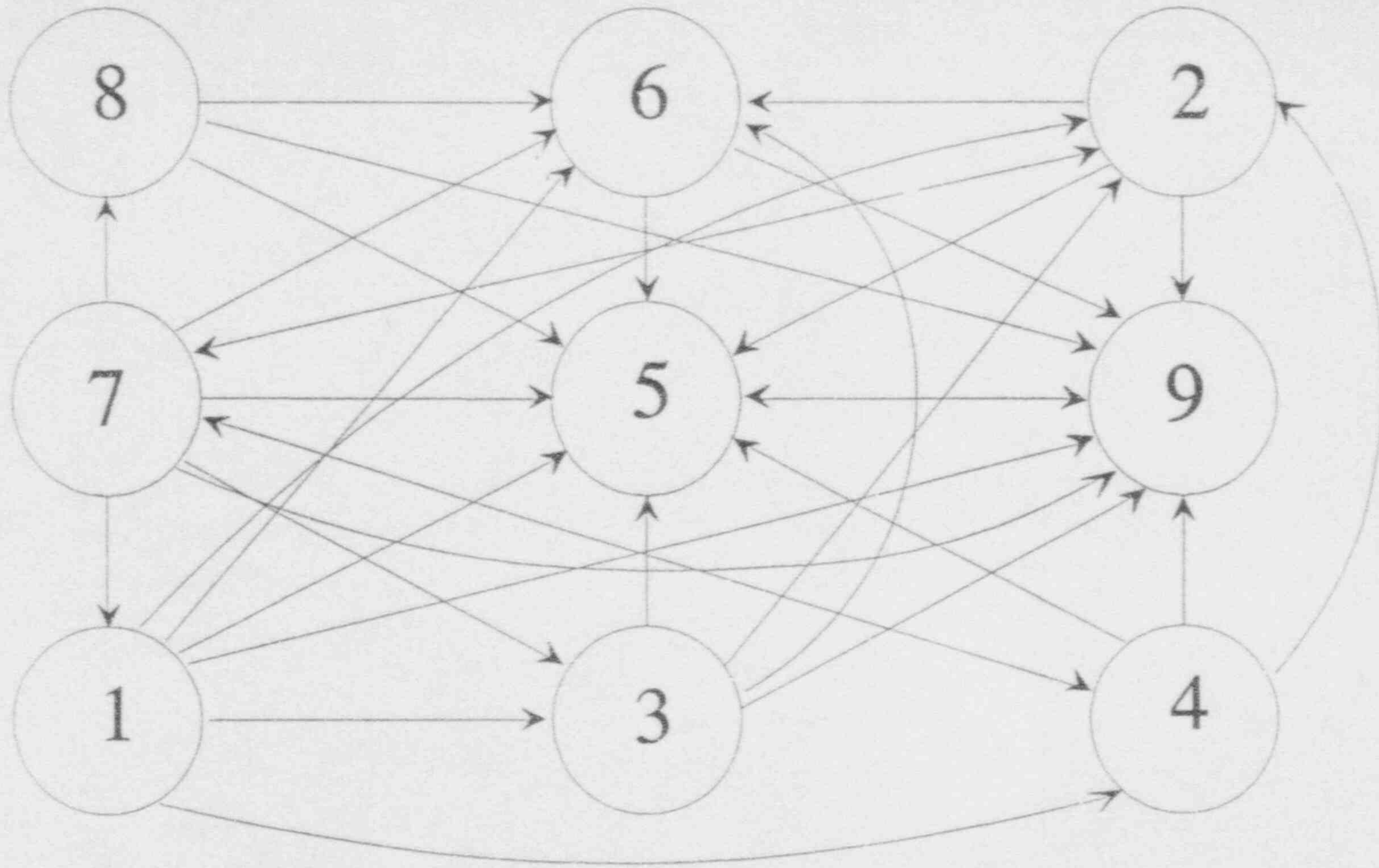

Figure A-2.2. Influence Diagram Representation of Relationships Among Entities and Processes.

Table A-4.3
Summary of Events Affected by the Impact of the Rule

#	AVERAGE	*/÷	BWR Basic Event Groups Modified	PWR Basic Event Groups Modified
3	1.8 (+)	*1.8	Active Valves Group	Active Valves Group
4	1.6 (-)	*1.6	No Events Qualify	AFW-CCF-LK-STMBD
5	1.8 (+)	*1.8	Diesel Generator Group	Diesel Generator Group
6	1.8 (+)	*1.8	Turbine Driven Pump Group	Turbine Driven Pump Group
7	3.8 (+)	*3.8	No Events Qualify	Recovery containing non-licensed actions
8	3.0 (+)	*3.0	Recovery containing operator actions	Recovery containing operator actions
9	2.6 (-)	*2.6	All groups, events with Test & .aintenance	All groups, events with Test & Maintenance
10	3.0 (-)	*3.0	Latent Maintenance Group	RMT-CCF-FA-MSCAL
10 CFR 50.65 ELICITATION				
1	1.2 (-)	÷1.2	T2, T3A, T3B, and T3C	T, TN, and T2
2	1.1 (-)	÷1.1	No Events Qualify	T5A and T5B
3	1.4 (+)	÷1.4	Turbine Driven Pumps Group	Turbine Driven Pumps Group
4	1.6 (+)	÷1.6	Active Valves Group	Active Valves Group
5	1.4 (-)	÷1.4	No Events Qualify	AFW-CCF-LK-STMBD
6	1.2 (+)	÷1.2	Diesel Generator Group	Diesel Generator Group

Volume Five

Regulatory Review Group

Public Comments

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FINAL
AUGUST, 1993

PUBLIC COMMENTS

Accident Prevention Group
BWR Owners' Group
Carolina Power & Light Company
Centerior Energy
Commonwealth Edison
Entergy
Electric Power Research Institute
GPU Nuclear Corporation
Nuclear Management and Resources Council
Ohio Citizens for Responsible Energy, Inc.
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Dr. Vojin Joksimovich
President/Chief Executive Officer

June 28, 1993

Mr. James M. Taylor
Executive Director for Operations
USNRC
One White Flint North
Rockville, MD 20555

Dear Mr. Taylor:

Our company is pleased to provide comments to the Nuclear Regulatory Commission on the subject of risk-based regulation (RBR). Our comments are summarized as follows:

A. Ten assertions form the paper titled "Where Do We Go From Here in U.S. Nuclear Safety Regulation". Revision 3 of this paper, which is now APG Report #28 (copy enclosed), was presented to the ACRS on June 11th, while Revision 1 was presented to the NRC's Regulatory Review Group on April 13th.

B. General comments on Volume 4, "Regulatory Review Group Risk Technology Application".

C. Specific comments on Volume 4.

D. Comments dealing with the treatment of Human Reliability Analysis (HRA).

O&M costs largely attributable to regulatory requirements and how the utilities have responded to them, are driving competitiveness of nuclear utilities right into the ground. RBR is a solution to the problem, as we visualize it and as documented in Section A.

The NRC's Regulatory Review Group's vision most certainly represents a modest step in the right direction, but falls short of providing what is badly needed, *i.e.*, laser focus on legitimate nuclear safety issues rather than diluted effort emphasizing numerous peripheral issues which has driven the costs to unacceptable levels, and which continues to permeate nuclear safety regulation.

EDO --- 009079
93-08799-A-00

Mr. James M. Taylor
USNRC

June 28, 1993
Page two

The above has motivated our small company to expend significant resources to provide the enclosed comments. Should you require any clarifications of our comments, please do not hesitate to contact me.

Sincerely,

Dr. Vojin Joksimovich

VJ:dm\NRC-RBR\LETTER.vj
Enclosure

A. <u>TEN ASSERTIONS FROM APG REPORT #28</u>

1. No other industry has invested more resources in public safety than the nuclear industry. O&M costs largely attributable to regulatory requirements and how the utilities have responded to them, have escalated to unacceptable levels and are driving competitiveness of nuclear utilities right into the ground. Long-term sustainable and manageable cost reductions are imperative for saving the nuclear option.

2. For this large investment, the industry has achieved a remarkable safety record. Nevertheless, there is no room for complacency; the level of safety achieved has to be maintained and continuously looked to be enhanced.

3. The nuclear industry worldwide has been preoccupied with the hardware to the point of obsession. TMI action plan alone resulted in thousands of hardware changes costing the rate payers billions of dollars. As a result, existing hardware is good enough. There is now clear recognition that many hardware considerations, initially incorporated into the design or later backfitted, as well as elaborate plant security arrangements, are of peripheral impact to public risks associated with NPP operations.

4. Since TMI, readiness of operating crews to respond to complex accident scenarios has been greatly enhanced. Simulator training and emergency operating procedures are probably the most instrumental. However, more needs to be done, not in terms of quantity, but quality of training. The simulator offers much more before it reaches its full potential.

5. With almost all NPPs operational, the emphasis has to shift from traditional engineering considerations into entirely operational ones. In order to maintain and enhance the existing level of safety, our understanding of operational risks has to be vastly expanded. Plant operations have to be seen as a collection of systems, human actions and process requirements in a highly interactive mode which requires a cultural change in the industry. The "4M" aspects, discussed briefly in this paper, should receive due attention. Core damage frequencies can undergo large changes over time due to changing plant hardware configurations, human reliability and organizational factors.

6. Full benefits should be derived from currently under-utilized and sufficiently in-depth researched disciplines such as PRA in both integral and time-dependent mode, human reliability and safety culture.

7. Risk-based regulation and plant management, as advocated in this paper, is an answer. Greater self reliance and more self regulation through instillation of enhanced safety and risk culture via advanced self assessment programs should be

key ingredients. A good example for an advanced self assessment program is the Integrated Risk Management (IRMP) proposed in this paper.

8. Risk technology applications as proposed by the NRC's Regulatory Review Group represent a step in the right direction. Subsequently, the NRC should gear up its resources to respond expeditiously to nuclear utility initiatives. Furthermore, a regulatory culture reform will be needed to reflect some points made above.

9. Regulatory culture reform should address two fundamental issues: the proper role of the regulator, *i.e.*, cooperative, like in many European countries vs. competitive, and change of binary (OK/Not OK) compliance thinking to a highly interactive systems performance perspective and its associated reduction in variabilities.

10. A massive instillation of risk education in the whole industry via management and personnel training has to be initiated, and the sooner the better. In addition, rule-based culture has to be substituted with knowledge-based culture. Regulatory and utility-sponsored research must continue with emphasis on the human and organizational factors in particular.

B. GENERAL COMMENTS ON VOLUME 4, "RISK TECHNOLOGY APPLICATION".

1. The document probably represents the most significant step the NRC has taken towards risk-based regulation (RBR) since issuance of NUREG-1050. Group 1 (graded implementation) and Group 2 (configuration control) applications have a potential for regulatory burden reduction if accompanied with regulatory practices aimed at de-emphasizing issues peripheral to nuclear safety.

2. It is commendable that the NRC crossed the U.S. boundaries in search of state-of-the-art in RBR. The Torness example, as well as the U.K. example of an operational system for on-line configuration control are clearly demonstrating that the U.S. is behind. However, it appears that this non-U.S. survey is more of a cosmetic rather than substantive nature, *i.e.*, plough back of these types of experiences is not recommended anywhere.

3. NRC correctly states that its use of PRA needs to be changed to "allow PRA based insights to reduce regulatory burden when it is shown that such a reduction does not reduce the safety envelope of the plant". On the other hand, the document fails to provide examples how such burden might be reduced. To the contrary, it suggests that removing SSCs from the Q list is inappropriate. It provides such obstacles for use of PRA in Group 3 applications that no licensee in the pragmatic state of mind would ever attempt to use PRA for this optimal application in a regulatory framework. The NRC staff should give a serious consideration to including Operational Risk Model type of applications as discussed in APG Report #28 as Group 3 with existing Group 3 becoming Group 4.

4. Too much concern is given to "uncertainties" which has the flavor of trying to discredit PRA; the entire concept of risk analysis is based on decision making under uncertainty. Advocates of deterministic analysis are deluding themselves if they do not recognize the inherent uncertainty in setting limits in codes and standards, the uncertainty that some QA procedure <u>will</u> be carried out, and so on. Quantitative risk analysis attempts to put uncertainties in myriad potential threats to defense-in-depth in a common basis.

5. The NRC staff's understanding of the importance of HRA (Human Reliability Analysis) is inadequate and should be of profound concern to the nuclear safety community. The staff is preoccupied with a concern that "inadequate HRA" could distort ordering of dominant sequences and therefore spoil proper Group 1 and Group 2 applications. To preclude such possibility, the staff prescribes some arbitrary and unsupported set of screening values. The NRC staff does not seem to recognize that many operating crew actions could be as important as SSCs and therefore should be identified, ranked and their reliability ensured through established training programs, EOP validations and use of advanced operator aids. Furthermore, the document ignores the existence of developed human reliability technologies such as the ten-year old EPRI-sponsored Human Reliability Program.

6. The document provides no apparent evidence that a thorough process was followed to investigate all of the potential applications of PRA to regulations. While the resulting recommendations and examples of near-term applications of PRA are generally good, they seem very narrow in perspectives. Even some of the current NRC applications are not addressed; *e.g.*, risk-based inspections; risk-based equipment qualification. Section 4.7 seems a bit broader in scope in addressing past and current rule making issues. Lacking is a sense of "top down" or comprehensive study of how PRA can affect regulation.

   A more persuasive treatment would have surveyed the entire set of regulations and interpretive documents that licensees have to address for potential applications or insights from PRA methodology. The survey should include 10CFR50 including Appendices, the body of Regulatory Guides, Generic Letters, Unresolved Safety Issues, inspections, SALPs, etc. With such a survey, documented to show reasons why PRA can or cannot affect a given area, one would be led to a list, and schedule, of potential regulatory reforms or augmentation using PRA. While the Regulatory Review Group may have performed such a survey that lead to the recommendations provided in the report, there is no documentation or arguments to this effect.

7. The document (*e.g.*, Section 4.1, Introduction) provides neither historical nor current uses of PRA by the NRC. For example, in history, several NPPs used PRA to gain waivers from backfits. Well known to us is Big Rock Point's use of PRA to avoid some of the post-TMI actions that were shown to have little or no risk reduction potential at that plant. Several plants were examined in the 1980 is under the SEP (Systematic Evaluation Program) and used PRA methodology, in part, to determine if certain design changes were warranted (*e.g.*, NUREG-0829, Integrated Plant Safety Assessment, Systematic Evaluation Program, San Onofre Nuclear Generating Station, Unit 1).

   In addition, PRA has been applied in other regulatory areas. For example, PRA was used to assess the equipment qualification issue for electrical equipment (NUREG/CR-5313, Equipment Qualification (EQ)- Risk Scoping Study).

   Currently, the NRC has published a series of reports on use of risk-based inspections (*e.g.*, NUREG/CR-5865, Generic Service Water System Risk-Based Inspection Guide). Such applications are not mentioned in the report.

8. The Commission should form an external group to supplement the NRC staff's knowledge, which would provide advice to the Commission regarding exploitation of the full potential of PRA techniques for reducing regulatory burden while maintaining or enhancing nuclear safety, state-of-the-art in human reliability and organizational factors technologies. A NUREG-1050 type of approach, *i.e.*, representation both from the NRC and industry or purely an external group such as the Lewis Committee for review of WASH-1400 are both viable options.

C. Specific comments on Volume 4

1. Section 4.1 Introduction: Historical summary and references (Section 4.11) should note at least three other significant events:

   a. 1974-1978 application of PRA to the high-temperature gas-cooled reactor (HTGR) as concurrent to WASH-1400; moreover, that study produced several advancements in PRA methodology including development of the beta-factor approach to common-cause failure modeling and explicit modeling of operator responses in accident sequences.

   b. The 1978 report by the "Lewis Committee" entitled "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission should be acknowledged and summarized. The Kemeny and Rogovin Commission reports on TMI are noted; Lewis addressed WASH-1400 *per se*.

   c. The use of PRA by Big Rock Point (circa 1980) to gain exemption from many post-TMI regulatory requirements while committing to other risk reducing modifications.

2. With regard to the Lewis Committee report, their recommendations included "re-evaluate NRC's inspection and quality assurance system and licensing criteria...."

3. The concern about reliability of solid state C&P and associated software (p. 4-6) might take cognizance for advancements in the US space program.

4. Paragraph at top of the page (p 4-10) is confusing and seems contradictory; It say that importance rankings are more robust but then says "weakest insights" are from quantitative rankings alone...

5. The description of accident sequences (p 4-12) does not explain how PRA sequences differ from those addressed as Design Basis Accidents for the same classes of initiating events.

6. The discussion on different fault tree methods (p. 4-12, 4th paragraph) is not accurate and tends to confuse . Why not delete? If there is a need for a discussion of PRA methodology, refer to NUREG/CR-2300 or other.

7. *Initiating Event Analysis* (p 4-13), modify end of first sentence to read "...frequency of events of the same class."

8. *Data Analysis*, Hardware Faults (p 4-16): Crud buildup seems more like an example of systematic, causal failure and not a random failure.

9. *Data Analysis*, Common Cause Faults (p 4-16): The example of defective replacement parts as a CCF is not very good; it means ineffective QA program --this is an organizational factor that could be analyzed. Perhaps a better example would be miscalibration of several redundant instruments by a common technician due to a) lack of training; b) attitude; c) flawed procedures; or 4) flawed calibration of calibrating instrument. Examples should also mention spatial interactions which have not been eliminated by configuration or barriers.

10 Section 4.3, PRA Application Definition:

a. The second paragraph of 4.3.1 and the "charter" in section 4.3.2 are quite clear and set the stage for some progressive thinking. However, beginning with the third paragraph of 4.3.1 and the rest of Section 4.3, clarity and progressive approaches vanish. The second sentence of the third paragraph of 4.3.1 mentions a wide range of potential regulatory support. The paragraph continues by introducing "other possible uses (described below)" as if these are secondary. However, the remaining parts of 4.3 plus Sections 4.4 through 4.6 make these the <u>major</u> part of the report; the fact is obscured that the authors view these as "examples" rather than the only currently acceptable applications of PRA.

b. The first paragraph of page 4-22 introduces Figure 4.3-1 as "a possible structure" for the purpose of illustrating a spectrum of PRA ranging from generic to plant-specific. While we applaud the report's realization that such a spectrum exists, this section locks the report into this particular framework such that PRA criteria and supporting discussions go on and on to add a lot of substantiation to mere "examples". The bases for PRA criteria could be stated more generally to capture the notion that the degree of specificity in models and data must match the kind of regulatory decision to be made. The Conclusions and the Executive Summary regard these as examples yet the report, by treating the "examples" in such detail, does not stimulate imagination as to how similar approaches might be matched to other regulatory issues.

c. The labels given to the respective Groups seem too oriented to applications; why not call them by names that indicate degree of reliance on plant specific PRA and data.

d. Figure 4.3-1 and the other figures are difficult to read.

e. The text should define "average PRA" and "state-of-the-art PRA".

11. The third paragraph on page 4-23 seems to set a very negative tone on the whole effort. It sounds like the NRC is trying to find quasi-technical reasons for NOT WANTING to advance in risk-based regulation.

12. For both Group 1 and Group 2 applications, the SSC importance definition is based on standard measures such as Increase Importance Measure and Reduction Importance Measure (pages 4-19 and 4-24). It is important to note that one has to be cautious when calculating these measures by considering the potential impact of failure of one SSC on other SSCs and/or operator actions. For example, failure of a support system may cause failures or degradation of several frontline systems or failure of a primary safety system may impact reliability of specific operator actions by pushing them to use non-primary systems, contingency procedures, or may introduce new recovery actions.

    As a side comment, the equation on page 4-25 is not correct. The numerator should be the new core damage frequency estimate when unavailability of an SSC is assumed to be unity, rather than the Increase Importance Measure. The ratio itself is the Increase Importance Measure of an SSC.

13. Section 4.4.1 sounds very bureaucratic; it can be greatly clarified and simplified. Why carry a phrase like "plant-specific relatively non-important SSCs"? Does this mind-boggling phrase help us to regulate, make important decisions, or communicate with the public? In Section 4.4.3, SSCs are grouped into Groups A,B,C for a specific purpose; why not use similar kind of grouping or class labeling for SSCs, defined in a table, than the long phrase.

14. The text below Figure 4.4-1 and top of page 4-27 seems very repetitive.

15. The discussion in the last paragraph of page 4-26 suggests that SSCs that are on the Q-list due to deterministic considerations should remain on the list while others might be removed from PRA evaluations; has the NRC estimated how many SSCs are likely to remain and be subject to QA and the likely cost savings to NPP? If there is not much reduction in QA effort, there will not be much incentive for NPPs to use PRA this way.

16. The report uses Groups 1,2,3 for spectrum of risked-based regulation; Levels 1,2,3, apparently for graded QA; and Groups A,B,C ("or three levels") for importance of SSCs in graded rule application. Besides the tendency to confuse by having several taxonomies of order three, the latter two groupings seem to be two names for the same thing; if not, Level 1,2,3 in figure 4.4-4 should be defined.

17. The section 4.4 is about "graded implementation" which should be better defined in the first paragraph. Perhaps words from Sections 4.4.2 and 4.4.5 could be used in introductory paragraph. Further, section 4.4.5 is called "Graded Type Applications" but section gives the impression that it is <u>only</u> about QA.

18. The definition of "living PRA" equates it to "real time" PRA, but most definitions of living PRA are more in keeping with your definition of "PRA driven". We advocate use of current trend data in hardware and human reliability to provide frequent updates to PRA to trend the CDF; this requires a good plant-specific PRA

model but does not have to be "real time" nor "on-line". You should use the term "on line" PRA throughout the document if that is what is meant.

19. Figure 4.5-1 is not at all clear; its relationship to equation for AOT (Total) at top of page is not at all obvious.

20. The SSC unavailability expression on page 4-38 ($Q \approx \lambda T$) regarding STIs is too simplistic. It does not consider contributions such as failure on demand (*i.e.*, time-independent failure mechanisms), potential human errors during test, and operating type failures. This equation can be simply expanded to include such effects. For example, refer to References 21 and 22.

21. Section 4.6 is short and not very imaginative while Sections 4.4 an 4.5 are verbose and detailed. The prescriptive and essentially negative tone of Section 4.6 indicates that the NRC tried to find arguments to undermine the use of advanced methods. For example, the last paragraph states "...it is felt that the state of the art of PRA will not currently support this type of...on-line configuration control...", after praising an application in UK as an excellent aid to advise operations on delta-CDF for configuration change, but only as a Group 2. The description of the Torness (UK) application in Section 4.8 sounds much more plant-specific oriented, however, than the definition of Group 2 provided in Section 4.5.

22. Replace "most optimum" in sentence 1 of Section 4.6, or better yet, replace the sentence.

23. Several comments are made with regard to the feasibility study in Section 4.7.

    a. It is stated that the Training Rule, 10 CFR 50.120 will have a pervasive impact on numerous components, systems and structures in the plant (Page 4-44). No mention of the impact of this rule on human performance (and operator actions in particular) is made. How can the Training Rule have a pervasive effect on SSCs but not on operator actions? It is interesting to note that the Training Rule had the greatest impact of any of the regulation changes on the dominant accident sequences for both BWRs and PWRs. Also, the experts participating in the study commented that the implementation of a systematic approach to training (initiated in 1983) greatly reduced risk (Page 4-53).

    b. In Figure 4.7-1 (page 4-45), only recovery-type actions are included as human action parameters in a PRA. Pre-initiator human actions and post-initiator response-type human actions should also be included in the framework.

    c. The stated objective of 4.7 is to assess consistency with "safety goals" to examine relative safety importance of regulation, yet we get no appreciation from the body or conclusions that objectives are borne out or how. The section really addresses whether or not the NRC can estimate the change in risk due to adopting new or eliminating old regulations. "Safety goals" are not defined.

d. This section is too long; it dwells too much on process; summarize results and document the process (which is rather standard fare for PRA and HRA practitioners) in an appendix. By contrast, presentations in earlier sections use much jargon and concepts of regulation that confuse the readers (at least the PRA-oriented ones). In addition, this section seems rather shallow and gives the impression of being done in a short period of time and not a thorough study by NRC and national labs.

e. It is interesting that results in Section 4.7.3 are presented without discussion of uncertainties, while the topic is so prevalent as issue elsewhere in the document. So called quantification is really a "guess".

f. The transfer of qualitative to quantitative is interesting but has no factual basis. The process should be calibrated by finding data before and after some regulatory event. Perhaps some Performance Indicator like numbers of scrams or Safety Injection actuation before and after TMI action rules could be determined.

g. "Experts" on page 4-48 had to be trained in "linkages between plant functions and reliability" -- so how can they be called experts? The panel described on page 4-47 has no apparent PRA expertise; if they are skilled in PRA, this should be noted. Why were none of the well-known PRA consulting firms represented?

24. In Section 4.9, Paragraph 2 "both broad and narrow" is used; which is it? Reader cannot appreciate nuances among "use", "application" (which has been "broad"), and "utilization" (which has been "narrow").

25. Why are there no references in Section 4.9.3 to NRC-RES sponsored programs in HRA, or to newer research in organization and management or safety culture effects?

26. There seems to be a wide gap ("quantum jump") in Section 4.10 between applications, database usage and the NRC's perceived review requirements between Classes 2 and 3. As defined by the NRC, Class 3 seems like an ideal that may never be achieved. It seems that there is a virtual continuum of applications using plant-specific data, including quantitative measurement of HEPs that should be recognized. Application groups should be expanded to better include a Class 2A or a new Class 3, where Class 4 becomes the "unobtainable ideal".

27. CONCLUSIONS: This section describes the regulatory "Classes" or "Groups" (as they are termed elsewhere in the report) much more clearly than the rambling prose in earlier sections. Also, the listing of certain licensing applications in each class as **EXAMPLES** is clear, whereas earlier presentations give the impression that these are the only applications currently "approved".

D. Comments Dealing With the Treatment of Human Reliability Analysis (HRA).

The coverage of Human Reliability Analysis (HRA) in the document is of profound concern, knowing the fact that humans play an essential role in the safe operation of nuclear power plants and other complex industrial establishments. The document focuses on hardware-dominated PRA modeling and applications to the exclusion of the impact and importance of human and organization-related influences on nuclear safety. The following specific comments are made with respect to the treatment of HRA in the document:

1. The document recognizes the importance of human interactions and HRA in a PRA by making statements such as: "In summary, the strongest insights gained from a probabilistic analysis are derived from (1) the integrated and comprehensive examination that analyses of these types entail, (2) the attention devoted to interactions between systems, the operating staff, and the plant systems, and (3) the structured examination of operating experience." (page 4-9), "The HRA, therefore, not only impacts the estimated core damage frequency, but what are identified as the most likely contributions to realizing a core damage state." (page 4-17), or "HRA has the ability to impact the identification of the dominant sequences. Inadequate HRA could, therefore, erroneously result in identifying relatively important SSCs, as relatively non-important." (pages 4-32 and 4-40).

   In contrast, the document downplays the modeling and quantification of human interactions by just recommending three screening HEP values to be used for all human interactions regardless of what the operator action is and what the important influence factors (such as complexity, time constraint, quality of EOPs, training and MMI) are for both Group 1 and Group 2 applications (pages 4-32 and 4-40). This proposed approach does not reflect state-of-the-art.

   With the objective of advancing the state-of-the-art in HRA, EPRI launched a human reliability program in 1982. This program has covered important areas of development of a structured HRA framework to be used in PRAs and HRA quantification methods. These developments were supported by multi-year data collection efforts and development of computer software to facilitate both the processing of data collected using NPP training simulators and assessment of human reliability. The developed quantification methods cover both response-type actions and recovery-type actions as defined in this document (page 4-17). These developments are documented in References 1-13. A summary of EPRI-sponsored HRA activities can be found in References 14 and 15.

   A large database on operating crew performance on a variety of simulated scenarios for both PWRs and BWRs exist. Examples of these databases are the EPRI Operator Reliability Experiments (ORE) Program (References 6 and 7), RMIEP LaSalle Simulator data (Reference 16) and data collected by utilities to be used in IPEs or for training purposes (Reference 17). These databases have been used to derive generic (in ORE) and plant-specific (in RMIEP) estimates for reliability of operators when following AOPs/EOPs. These empirical sets of data include the effects of use of symptom-based EOPs, team skills of the crew, quality of training and other performance shaping factors. Therefore, rather than waiting for the day when the state-of-the-art is mature with respect to the treatment of "cognitive and comprehension errors" (page 4-7), one may use the existing integral data on the cognitive response of the control room operating crews to derive credible estimates for their reliability (or error probability). It should be noted that while the ORE database is proprietary to EPRI, the data may be used when the plants names and operators identities are not revealed. The ORE database was used in such a fashion for an NRC-sponsored project to provide support for updating the ANS-58.8 standard using simulator data (Reference 18).

As stated earlier, some utilities have already started systematic collection of data on human performance as part of their living IPEs, improve training and address regulatory issues (Reference 18), or as a way to reduce the number of human-caused Significant Events (Reference 19). It seems appropriate for the utilities the be encouraged by the NRC for such efforts, rather than downplaying the HRA and limiting this important area to only three guesses of 0.03, 0.1 and 0.5 (pages 4-32 and 4-40) when it comes to risk-based regulation. Incidentally, we have observed a large number of operating crews on simulated PRA-type accidents, and their unreliability is much less than 1 in 10 when responding to actions specified in EOPs. Use of plant-specific data on human performance inside and outside the control room, not only results in more realistic estimates for HEPs, but also reduces the uncertainties in the HEPs compared to those based on generic data and/or judgement.

2. The document suggests use of screening HEPs with a 10^{-3} threshold per accident sequence to preclude the possibility of identifying relatively important SSCs as relatively non-important (pages 4-32 and 4-40). It is noted, however, that use of overly conservative HEPs with an arbitrary and conservative threshold value may have a reverse effect, *i.e.*, it may cause identifying relatively non-important SSCs as relatively important SSCs, and furthermore, it may mask the impact of surveillance and maintenance as well as training. The latter point has also been made by Hirschberg, *et al.*, 1993 (Reference 20).

3. As stated before, the document is hardware dominated and extensively discusses SSCs. We believe that operator actions could be as important as SSCs, therefore, they should be identified, ranked and their reliability ensured through established training programs, EOP validations and use of operator aids.

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BWROG-93094
July 22, 1993

Mr. Frank P. Gillespie
Regulatory Review Group
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Docketing and Service Branch
Office of the Secretary

Subject: **COMMENTS ON REPORT OF REGULATORY REVIEW GROUP TO EXECUTIVE DIRECTOR (58 Fed. Reg. 29012 - May 18, 1993)**

Dear Mr. Gillespie:

The BWR Owners' Group (BWROG) welcomes the opportunity to comment on the subject report of the NRC's Regulatory Review Group (RRG). We fully support the RRG effort to identify regulatory requirements and practices that exceed what is required for adequate protection of the public health and safety. Revisions to overly prescriptive or redundant requirements and guidance would increase our member companies' flexibility in operating and maintaining their plants while maintaining and (in numerous cases) even improving overall plant safety.

Based on a very preliminary review of the report, we believe that the RRG has done an excellent job in their identfication of major areas of the current body of regulations and implementing guidance that warrant either revision or further study. We believe that a number of the items identified can be acted upon, based on data that has already been provided by the industry, without further study and urge expedited action by the NRC on those issues.

To that end, we believe it highly important that the NRC, with the assistance of the industry, move quickly to establish appropriate priorities for pursuing resolution of the various items already identified. We would be pleased to participate in and contribute to that process and would expect to do that under the auspices of the Nuclear Management and Resources Council (NUMARC).

Very truly yours,

L. A. England

L. A. England, Chairman
BWR Owners' Group

EXEC6T/LAE/LSG/rt

cc: KP Donovan, BWROG Vice Chairman
CL Tully, RRG Chairperson
BWROG Executive Oversight Committee
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CP&L

Carolina Power & Light Company

July 29, 1993

SERIAL: GLS-93-176

Mr. Frank P. Gillespie
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Gillespie:

USNRC REGULATORY REVIEW GROUP REPORT

Carolina Power & Light Company (CP&L) appreciates the opportunity to comment on the report of the Regulatory Review Group (RRG) published on May 28, 1993. Our general comments are provided below.

1. Carolina Power & Light Company supports the efforts of the RRG to identify areas in the regulations and other regulatory guidance where increased flexibility can be made available to licensees with little or no safety impact. This increased flexibility is a necessary first step in a nuclear utility's efforts to focus on safety-relevant problems.

2. Carolina Power & Light Company urges close cooperation between the NRC and NUMARC in the implementation of the recommendations noted in the report. For a number of regulatory areas (such as Appendix R, Appendix B implementation, and 10 CFR 21), it may be more cost-effective for both the NRC and the industry to pursue generic improvements rather than doing so on a plant-specific basis.

3. The current efforts of the task force led by Mr. Marsh to improve the review of utility-requested changes should either be continued and expanded, or integrated into the regulatory review process in such a way that licensees are encouraged to come forth with proposals for reducing regulatory burden.

4. Carolina Power & Light Company encourages the NRC's promulgation of appropriate guidance throughout the Regions to ensure that Resident Inspectors and other Region staff are in concert with potential license and commitment changes (general or plant-specific) resulting from implementing the RRG's recommendations.

5. Carolina Power & Light Company supports the increased use of PRA technology in the regulatory process, and encourages near-term cooperative NRC-NUMARC efforts to define criteria for utility use of PRA in the regulatory arena.

Thank you for the opportunity to comment on this important effort. Please contact me at (919) 546-7318 if you have further questions.

Yours very truly,

R. W. Prunty

R. W. Prunty
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July 29, 1993

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Mr. Frank P. Gillespie,
Director PMAS - Mail Stop 12-G18
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Washington, D.C. 20555

Subject: Comments on Regulatory Review Group Report

Dear Mr. Gillespie:

The Toledo Edison Company, operator of the Davis-Besse Nuclear Power Station, has reviewed the Regulatory Review Group Report which was placed in the NRC Public Document Room on May 28, 1993, for a 30-day comment period (later extended to a 60-day comment period ending July 29, 1993 by 58 FR 33285). Toledo Edison strongly supports the NRC's and industry's efforts to identify and eliminate regulatory requirements and commitments that are economically burdensome yet provide little or no safety value. Toledo Edison concurs that the elimination of burdensome regulatory requirements could indirectly benefit safety, in that the freed up resources that would result may be redirected to more safety-significant work.

As you are aware, the resources which Toledo Edison, or any utility, can make available to initiate burden reduction requests are limited. Toledo Edison recognizes that the NRC's resources are also limited. As our resources permit, Toledo Edison will continue to initiate or participate in industry activities designed to reduce regulatory burden, provided the potential benefits outweigh the cost, and provided that there is a reasonable opportunity for NRC acceptance. The Review Group Report provides an excellent summary of potential opportunities in this area.

Specific comments on the report are included in the attachment to this letter.

Comments on the
Regulatory Review Group Risk Technology Application, Volume 4

Use the term "Probabilistic Safety Assessment" (PSA) vice the term Probabilistic Risk Assessment throughout the report. While PSA techniques and applications are still relatively new, consistent terminology should be used within the industry and regulators. NUMARC, EPRI and utilities are using the term PSA; use of terminology with the word safety instead of risk may allow for a wider acceptance of these techniques.

In response to the first bullet on page 4-2 that discusses reliance on quantitative results from multiple plant-specific PSAs and the use of generic failure data, is one particular generic data base being proposed? Most PSA analysts have their own generic data base comprised of the generic sources available to them.

The second bullet on page 4-2 discusses the reliance on single plant-specific PRA quantitative results in selected areas. Examples are provided for this type of application. An important example that is not included in this listing and should be are justifications for continued plant operation (JCOs).

An important point to note regarding the human interactions write-up in the second paragraph on page 4-7 is the methodology developed by EPRI that utilizes simulator data for human interaction rates. Plant-specific operator training simulator exercises are observed and the data from these is used to calculate plant-specific human interaction rates. This methodology is described in EPRI NP-6937, Volumes 1, 2 & 3, Operator Reliability Experiments Using Power Plant Simulators.

The last paragraph on page 4-7 and continued on page 4-8 discusses that the use of PSA may be more appropriately applied to the potential for severe core damage or system availability than to public risk. This is certainly the case for most of the utilities who performed the minimum requirements for the IPE because, only a Level 1 PSA along with a containment analysis was performed i.e., a level 3 PSA was not performed.

Section 4.2.1 (page 4-12) and section 4.2.2 (page 4-13) discuss the elements of a PRA. The paragraphs that discuss initiating events do not mention steam generator tube ruptures or internal floods as initiating events. Both of these initiating events are included in current PSAs.

The Initiating Event Analysis portion of Section 4.2.2 (page 4-13) notes that Boolean models depicting various systems and components contributing to the initiating event are generally not developed. This is not necessarily always the case. For some BOP initiating events, like losses of specific/both trains of service water, component cooling water and makeup, Boolean models were developed and used as the basis for the initiating event frequency.

Table 4.2-1 is confusing. It seems to contain a lot of extra information if the only purpose is to identify some of the plant systems that are modeled in the PSA and those that are not. In some cases, the systems that are identified in the table as not explicitly modeled or evaluated are not consistent with several PSAs. For example, most PSAs do explicitly model component cooling water, normal service water and required ventilation systems.

Page 4-30 discusses how often the PRA needs to be updated for specific PSA applications. While it is certainly reasonable to update the PSA following an outage or in response to a specific design change, it is unreasonable to make it 'real-time driven'. This is especially true of the plant specific data analysis. Each application should consider the status of how up-to-date the PSA is; in the majority of cases, a 'real-time driven' PSA is not necessary.

On the top of page 47, an equation is provided that calculates the maximum pre-determined AOT extension for any single SSC. An example using real component data would be helpful; it is not obvious that the denominator would end up as a positive number.

Differentiation needs to be clarified with respect to PSA application criteria for one-time changes or exemptions and those that will be implemented permanently.

As discussed in several sections of this report, the PSA application criteria (sections 4.4, 4.5 and 4.6) will need to be applied in several pilot studies before it is implemented. Pilot studies are necessary to evaluate the existing review criteria and further redefine the guidelines as appropriate. Furthermore, similar to the efforts involved in maintaining a 'living' PSA, this Risk Application Technology will need to be updated to take into account new applications and criteria along with new state-of-the art PSA techniques and methodology.

If you have any questions regarding this matter, please contact Mr. Dale R. Wuokko, Manager - Regulatory Affairs (Acting), at (419) 249-2366.

Sincerely,

Louis F. Storz

MKL/dlc

Attachment

cc: J. B. Hopkins, NRC Senior Project Manager
J. B. Martin, Regional Administrator, NRC Region III
S. Stasek, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

July 29, 1993

Mr. James Sniezek

U.S. Nuclear Regulatory Commission
Deputy Executive Director
for Nuclear Reactor Regulation,
Regional Operations & Research

One White Flint North
11155 Rockville Pike
Rockville Maryland, 20852

Subject: Comments on Regulatory Review Group Report

This letter provides Commonwealth Edison's (CECo's) comments regarding the Regulatory Review Group Report. Commonwealth Edison sincerely appreciates the effort and the commitment of resources required to conduct this comprehensive and disciplined review of the NRC's regulations, process and programs. Moreover, CECo appreciates the opportunity to provide comments on this report.

Commonwealth Edison is generally in agreement with most of the reports recommendations. Many of the concepts and recommendations in the report are desirable and certainly feasible given the proper level of resources and commitment. CECo has provided specific comments in three attachments.

Attachment 1 - Provides general comments on Volume One of the report. These comments were selected to help identify those issues where CECo believes the greatest flexibility and benefits can be achieved. It also provides specific comments regarding enhancements or areas for reconsideration that could make this effort even more successful.

Attachment 2 - Provides detailed comments and suggestions for six areas: 1) petitions for rulemaking; 2) combined packaging of multiple modifications in a single license amendment; 3) safety evaluations for changes to commitments; 4) relief from non-binding documents inappropriately treated as requirements; 5) application of regulatory requirements as the only measure of specific plan changes, 6) graded approaches to certain regulatory requirements.

Attachment 3 - Provides specific comments regarding the potential applications of risk technology (Regulatory Review Group Report Volume 4).

EDO --- 009196
93-0560-H-00

In summary, CECo appreciates the efforts of yourself and your staff in developing an approach to achieve greater flexibility in the regulations and providing an opportunity to comment on this report. If you have any questions regarding the comments provided, please contact this office at (708) 663-7332.

Respectfully,

Robert J. Lezon
Nuclear Oversight & Regulatory Services

Attachment 1 - Comments Regarding RAG Report Volume One
Attachment 2 - Detailed Comments on Regulatory Review Group Report
Attachment 3- Comments Regarding RAG Report Volume Four

cc: Frank P. Gillespie

Attachment 1
Commonwealth Edison Comments
Regarding Regulatory Review Group Report
Volume One

Page 4

Obtaining NRC approval prior to implementing changes as related to Quality Assurance

CECo agrees with the recommendations and adds two specific comments for consideration.

Recommendation #1:

Even given the flexibility offered by the use of Topical Reports, significant resources are expended to satisfy the acceptance criteria for 'Organization' consistent with the Standard Review Plan Section 17.1. NRC Reviewer expectations have vary widely in the amount of detail to be provided for the title, responsibilities, authorities, and interfaces for persons that have any role in the implementation of the QA Program.

Given the general industry trend for more responsive and flexible organizations it is recommended that only those essential elements of organization such as the independence of quality verifiers, Stop Work and Unit Shutdown authorizations, and qualifications of the person responsible for a plant operating license should be required to be described and subject to prior NRC approval. All other details of organization should be at the licensees discretion.

Recommendation #2:

The acceptance criteria of the Standard Review Plans 17.1 and 17.2 may provide a barrier to adopting "Total Quality Management" and performance based quality systems. The expectations for the acceptance criteria sometimes impede the adoption and implementation of a quality culture based on personal accountability and empowerment of line departments.

Although SRP 17.3 does enable many current quality culture elements, it is a significant departure from current industry programs. It is recommended that the NRC consider endorsing parts of The Department of Energy's Order 5700.6c.

SRP 17.3 is similar to DOE Order 5700.6c, that presents a new set of 10 quality criteria instead of the 18 of 10 CFR 50 Appendix B. NRC and industry reviews have shown that there is little difference between the basic requirements of these two documents. The DOE publication of 5700.6c has a section following each requirement entitled "Discussion." This section provides insight into the intent and expectations and would provide the basis for SRP acceptance criteria. It is recommended that the Discussion section of 4700.6c be appropriately endorsed as the acceptance criteria for SRP 17.1 & 17.2. This will minimize the burden of rewriting entire QA programs and subtier documents to reflect 10 criteria instead of 18 while allowing more innovative implementation of existing requirements.

<u>Page 5</u>
Use of risk technology in Quality Assurance

While CECo agrees that the application of PRA's as the basis of quality assurance efforts is valuable, a transition phase might be valuable to the industry. The application of performance driven audit coverage inplace of the less flexible pre-approved schedules, currently required, would provide more immediate potential for improvement. It is recommended that the "Discussion" sections of DOE Order 5700.6c pertinent to assessments, be substituted for the acceptance criteria of SRP 17.1 regarding audit schedules and that the Technical Specification requirements to include an audit schedule be eliminated.

<u>Page 6</u>
Revision of Part 21 to recognize existing procurement practices.

CECo agrees with the proposed revision to the wording of Part 21. This is consistent with EPRI efforts and the direction our Materials Engineering program.

<u>Page 7 - Bullet 1</u>
Revision of 10 CFR 50 Appendix B

As previously stated CECo believes that the revision of SRP 17.1 acceptance criteria to include reference to DOE 5700.6c would be of value to the industry.

Apply the QA program in a graded manner to Fire Protection

The recommendation to apply the QA program in a graded manner to fire protection is definitely worthwhile and should be pursued. Based on past experience the NRC and industry should develop guidelines regarding such a program to prevent misinterpretation, confusion, and to provide a consistent implementation approach across all stations.

Page 7 - Bullet 2
Consolidation of Regulatory Guides

CECo agrees that the regulatory positions, endorsements of ASME Code Cases, and those elements of Generic Letters, Information Notices, and Bulletins that the NRC perceives to require quality assurance commitments should be in no more than two regulatory guides. There should also be a rigorous examination of whether any regulatory positions are needed if a licensee commits to a consensus standard such as NQA-1. Since the NRC participates in such standards setting activities, no modifying positions should be necessary.

Page 7 - Bullet 3
PRA methods for use in the application of 10CFR50.59

Section 2.3.19 probes whether PRA offers a viable method for unreviewed safety question determination and notes that 50.59 does not preclude the use of PRA in such application. The NRC recommendation that industry and NRC staff are encouraged to pursue the development of guidelines for such PRA use is a sound one and is endorsed. The application of PRA's to 50.59 reviews is discussed in Volume Four as a potential Group 2 application. However it could be debated that this application could be supported under the Group 1 approach. In addition, some revision of the wording of 10CFR50.59 may be required for effective use of PRA. An unreviewed safety question is defined there as one in which the probability or consequences of an accident may be increased, or the margin of safety is decreased. Changes in the plant should be allowed without prior Commission approval, also, when there is such an increase or decrease, but the amount of the increase or decrease is negligible based on PRA methodology.

Page 8 - Bullet 5
Ensure that there is a clear delineation of NRC expectations in the security area. This would mean either stating the 10 CFR 73.55(a) [which invokes 73.55(b) through (h)] is the NRC standard "effectiveness" in security or revising 73.55(b) through (h) if they are not a sufficient definition of NRC expectations. (See Section 2.3.18.)

CECo supports this proposal, however, care should be taken and industry participation should be sought for any revision to 10 CFR 73.55,so that the revision does not result in an additional staffing or financial requirements.

Page 8 - Bullet 6

Eliminate the requirement for submittal of quarterly security logs. This would eliminate a small burden that appears to have no benefit. (See Sections 2.3.16 and 2.3.18).

CECo fully supports the resolution of this item which is also being pursued by NUMARC.

Page 8 - Bullet 7

Revise 10 CFR 50.54(p) to eliminate the reference to safeguards effectiveness making the basic regulatory requirements for physical security consistent with performance requirement of 10 CFR 73.55(a). This would provide a more consistent level of protection at each facility. (See Section 2.3.18.)

CECo agrees that changes to the contingency plan which do not decrease the effectiveness below the basic regulatory requirements should be allowed. The revised performance requirements must be thoroughly reviewed during the comment period to ensure any changes do not add or require additional financial or manpower requirements.

Page 8 - Bullet 8

Revise existing guidance to provide an approach in security similar to that used for safety systems for compensatory measures such as allowed outage times. (See Section 2.3.18 and A.3.2.4 of Volume 3.)

CECo, supports this proposal at a minimum, however the NRC should give consideration to the NUMARC Alternate Protection Strategy (APS) which would go further in reducing requirements that provide little safety benefit.

Page 9 - Bullet 2

The rules governing items in which licensees are allowed to make changes to their facility or procedures (10 CFR 50.59), QA Program (10 CFR 50.54(a)), safeguards plan (10 CFR 50.54(p), fire protection plan (10 CFR 50.48), and emergency plan (10 CFR 50.54(q) should be made consistent. 10 CFR 50.54(p) would be amended such that changes to the security plan that do not decrease safeguards measures below the requirements of Part 73 could be made without NRC approval and submitted in accordance with 10 CFR 50.71(e).

CECo, supports this change. A large majority of the changes we process could fall under this new classification. This would allow updates on some established frequency without the burden of submitting a change within sixty days.

Page 10 - Bullet 2
The reporting requirement in 10CFR 26.71(d) should be modified to allow submittal of Fitness For Duty Performance Data on an annual basis.

Semi-annually, Commonwealth Edison submits seven Fitness For Duty Performance Data Reports (one for each nuclear station and one report for the Corporate Office) to the NRC. Changing to an annual reporting format would save significant resources and should be pursued as soon as possible.

Page 10 - Bullet 3
The Fitness For Duty Program audit frequency should be changed to allow audits to be performed at intervals no greater than every three years and areas of the program that are conducted by contractors should be audited at least every eighteen months.

Significant costs have been incurred for FFD Program and Testing Laboratory audits during the first three years of the program. This includes costs for outside consultants, independent toxicologists, Company audit teams and administrative burden.

Additionally, the aforementioned testing laboratory is biennially audited for certification by SAMSHA and also audited annually by the College of American Pathologists. Licensees should be relieved of the NRC requirement to annually audit SAMSHA certified testing laboratories.

Commonwealth Edison audits approximately 243 active contractor and vendor Fitness For Duty Programs each year. One Hundred Thirty-Eight of these companies are located out of state. Although, the audit is conducted in conjunction with an Access Authorization program review, additional savings could be achieved during a three year period by reducing the number of audits required from three to two.

Changing this requirement to conduct audits every three years would result in an significant annual savings. CECo strongly supports the recommendation to extend the period between audits.

Page 10 - Bullet 6
The Industry and staff should continue to build a consensus view in ASME Code Committees to revise the Code and address design basis testing and test frequency based on risk techniques.

CECo strongly supports the continued effort to reach consensus views to revise the ASME Codes, however, CECo does not endorse the adoption of requirements for design basis testing.

Design basis testing in some cases would be an extreme burden on utilities that have the potential to require plant modifications without a clear safety benefit. Design basis testing so remain a manufacturing effort to qualify equipment and designs. Test frequencies based on risk techniques have the potential to become a regulatory burden if utilities have to develop risk models and dedicate resources to perform extensive analysis.

Page 10 - Bullet 7
In-service Testing Program

CECo concurs with the recommendation to continue work on In-service Testing Program guidelines. CECo would like to recommend that licensees preserve the option to commit to updated Codes rather then being required to adopt the most recent addenda and additions referenced in the regulations.

Page 11 - Bullet 5
Application of 10 CFR 50.54 to fire protection plans (includes guidelines on plan revision) and elimination of fire protection license conditions

The recommendation to extend 10 CFR 50.54 to fire protection plans and eliminate fire protection license conditions is advantageous. This would more clearly define the maintenance and revision requirements for the fire protection plan that are now contained in NRC guidance documents.

Page 11 - Bullet 6
Revise NRC guidance to specify that alternative methods to fire code methods of compliance can be developed

CECO strongly supports this recommendation. Current NRC guidance documents reference various fire protection codes and standards as acceptable methods to meet specific guidance issues. However, they do not clearly indicated that alternatives are also acceptable

G.L. 86-10 provided guidance on this subject stating that deviations from fire protection codes and standards should be justified in the FSAR or FHA. Additionally, if the licensee states they "meet the intent" of the fire protection code or standard and does not identify any deviations, the NRC expects that the license conforms to the entire code or standard.

The current position does not provide adequate flexibility. Licensees should be able to define their alternate method of compliance and have it judged on its own merit. This would reduce the burden of conducting line by line fire code reviews which are very time consuming and resource intensive. This problem is more pronounced when dealing with administrative or programmatic codes as opposed to fire protection system design codes. Additionally, many codes and standards referenced in NRC guidance documents are not nuclear specific which leads to further complications and areas for conflicting interpretations.

Page 11 - Potential Improvements - Bullet 1
Replace the Guidance for Generic Letter 88-16 with a more generic flexible statement to reduce the number of license amendments required for referencing topical reports.

Comment # 1
Volume 1 Sect. 4.1.2 (pg.11) and Volume 2 Sect. 2.3.8 III (pg. 65)

The first item listed as a Potential Improvement concerns reducing unnecessary administrative license amendments which result from literal implementation of GL 88-16. Based on the generic letter's guidance, NRC approval of every new or revised Licensing Topical Report on vendor or licensee methodologies utilized in establishing the core limits (specified in the cycle-specific COLR) would require a Section 6 Tech. Spec. change to update the list of approved references prior to application of the approved methods. Since the issuance of the NRR Safety Evaluation Report (SER) for a Topical formally approves the proposed methods, there is no value added by a subsequent administrative amendment. CECo therefore fully supports the RRG suggestion that the Staff acknowledge this by revising GL 88-16.

Comment # 2
Volume 1 Sect. 4.1.2 (pg.11) and Volume 2 Sect. 2.3.8 III (pg. 65)

Another potential change to the implementation of GL88-16 which could reduce the number of administrative license amendments without impacting safety would be to allow all core limits which can change as a function of fuel type or cycle-specific parameters to be relocated to the COLR. The BWR MCPR Safety Limit, for example, has remained in the Tech. Specs. although it can change on a cycle to cycle basis with one vendor's methods and on a fuel product line basis with another vendor's methods. Since the Staff must approve the methodologies for determining the limit and can monitor the value of the limit via the COLR submittals, relocation to the COLR should not impact safety.

Comment # 3
Volume 2 Sect.2.3.8 (pg. 64 and 65)

Although Volume 2 Section 2.3.8 contains discussion of both GL 88-16 related improvements and GL 83-11 related improvements, the only item listed at the end of the section and in the corresponding summary list of improvement items in Volume 1 (Sect. 4.1.2) is the one related to GL 88-16 as discussed in Comment # 1. CECo believes that some of the RRG suggestions related to GL 83-11, i.e. concerning the streamlining of NRC Staff/Contractor reviews of Topicals for licensees performing their own reload nuclear design and reactor safety analyses, are very worthwhile and should not be overlooked by merely discussing them in the body of the more detailed Volume 2. They should be explicitly identified and listed as specific potential action items in Volume 1. Specifically:

a) As indicated in the pg. 64 Discussion Section II, there have been significant delays in review and approval of licensee methodology Topicals for several reasons including Staff insufficient resources and priority, lack of clear and consistent guidance on the content and level of detail expected in such Topicals, etc. The delayed approvals have resulted in utility costs to support parallel vendor efforts, delayed realization of savings associated with in-house analyses, and delayed realization of the improved in-house capabilities and technical expertise that comes with performance of safety analyses of record. CECo therefore strongly endorses the Staff plans to issue guidance on ways that the licensee can assure rapid review and approval. This should be called out as an RRG recommendation.

b) Also indicated in this section is the observation that licensees can simplify the approval process by using existing approved methods which allows a reduced scope review approach by the Staff or Contractor. CECo's experience (and other utilities) suggests that the benchmark/audit alternative to a full review is not used to the extent possible by the Staff even when a licensee is applying previously approved methods (or minor variations of approved methods) and essentially [illegible] needs to be approved as a qualified user of the previously approved methods. In some cases, the underlying methods have been required by the Staff's contractors to be rejustified repeatedly by each applicant, at significant additional cost and with no apparent benefit. CECo therefore recommends that the guidance to be issued by the Staff specifically recognize the distinction between licensee Topicals which [illegible]se new methods versus those which are only demonstrating qualification to use already approved methods. For the latter, the guidance should clarify both the reduced scope of the information needed and the limited type of review which should be necessary by the Staff and/or contractor.

Page 13 - Bullet 2
Policy on Design Basis Documents

CECO disagrees with the intent to modify the policy statement on design basis documentation. Where misunderstandings exist a communication effort should be made to ensure clarity. CECo believes that the current policy statement reflects our program objectives. Any changes in policy statement could significantly increase the costs associated with the current DBD program.

Page 13 - Bullet 3
An Information Notice or other suitable generic communication should be issued to remind licensees that the training and retraining in behavioral observation for the Fitness For Duty Rule should not focus solely on aberrant behavior resulting from substance abuse. This Information Notice should also provide guidance on what the staff considers to be appropriate action for aberrant behavior that is not substance induced.

Behavioral Observation training contained in Commonwealth Edison's N-GET and Supervisor FFD training and retraining programs addresses aberrant behavior resulting from any cause and does not solely focus on behavioral problems induced as a result of legal or illegal substance abuse. Appropriate actions, including management evaluation of the event, referral to physician or EAP services, treatment and or any resulting sanctions should be developed by licensees on a case-by case basis. NRC staff recommended generic "appropriate action" for non-substance induced aberrant behavior could increase licensee costs for lost time due to denial of unescorted access and added administrative burden resulting from pre-imposed generic requirements.

Page 14 - Bullet 2

Review the existing security requirements (particularly Appendix B to 10 CFR 73) to determine if they should be expressed in a more performance-based manner. (See Section 2.3.18.)

CECo agrees with this proposal. Care should be taken that any changes to express requirements in a performance based manner should be clearly articulated and understood by the inspectors to ensure that standards are not set that would increase the necessary level of staffing or financial commitment.

Page 15 - Bullet 2

Provide a consistent approach for making changes to "plans" such as the fire protection, physical security, emergency response, and quality assurance plans, within their proper regulatory and safety contents. Eliminate the regulatory requirement that compliance with physical security plans be imposed by a license condition. (See Volume III, Section A.3.2.2.)

CECo agrees with the findings represented in Volume III, Section A.3.2.2.

Page 15 - Bullet 5

Provide additional flexibility in the implementation of the physical security plans, such as providing Technical-Specification-type allowed outage times. This recommendation parallels one resulting from the Regulations Review Task. (See Volume III, Section A.3.2.4)

CECo supports this recommendation but believes that greater benefit could be obtained if additional adjustments were made consistent with the recommendations made in the NUMARC Alternate Protection Strategy (APS).

Page 17
Risk Technology

Specific Comments are provided in attachment 3 regarding the Regulatory Review Group Report Volume Four. The following general comment regarding Risk Technology is provided below:

> This report suggests ways that PRA methods could be used to improve regulation of nuclear power plant safety. Since the industry's wide spread use of this area is relatively new and complex, it is important that the criteria expressed in this document be treated more as guide than requirement. The NRC Staff should retain flexibility in working with individual utilities and with NUMARC. As individual applications are tried and as specific projects are performed, the precise criteria appropriate to effective use of PRA will become clearer.

Attachment 2
Commonwealth Edison Comments

Detailed Comments on Regulatory Review Group Report

This provides proposed comments by CECo on selected portions of the Nuclear Regulatory Commission's ("NRC") report by the Regulatory Review Group ("RRG"). The RRG has suggested that the NRC could provide licensees with regulatory relief by simplifying certain regulatory processes. In response, CECo has developed the following detailed suggestions for implementing such processes to help realize their benefits. Six areas addressed are:

- petitions for rulemaking;
- combined packaging of multiple modifications in a single license amendment;
- safety evaluations for changes to commitments;
- relief from treating non-binding documents as requirements;
- use of regulatory requirements as the only measure of changes to:
  - safeguards plans (50.54(p))
  - emergency plans (50.54(q))
  - fire protection (50.48); and
- adoption of graded approaches to:
  - quality assurance
  - outage times for security deviations
  - LCO's for operations/surveillance

Specific proposals for implementing more efficient regulatory processes in each of these areas are provided below.

Before describing the specific comments for each of these areas, however, CECo first responds to the RRG's recommendation set forth in Section A.3.2.3 (Vol. III) of the Report. Namely, as part of its effort to reduce regulatory burden and improve efficiency, the NRC invited licensees to provide candid insights on what actions are necessary -- from a practical perspective -- to make fully available and better use of opportunities to exercise flexibility in existing NRC regulatory processes. In response, CECo acknowledges that it has not always taken full advantage of existing opportunities to request NRC flexibility. Several reasons for this hesitation are delineated in the Regulatory Impact Survey. Its revelations were a good first step in determining why licensees have hesitated to request flexible treatment. The Commission should revisit that effort, especially its identification of the sources of underlying causal factors, and follow-up the RRG Report.

Petitions For Rulemaking

The NRC suggests that its process for considering petitions-for-rulemaking ("PFR") should be revised to encourage licensees to submit more PFRs than they have submitted in the past. Such encouragement would be provided by focusing NRC resources on PFRs which: (1) reduce regulatory burdens; (2) have no safety impact; and (3) are supported by analysis and data which the NRC finds complete and compelling. Where a PFR is found to satisfy these criteria, it would be published in the Federal Register as a proposed NRC rule. To implement this proposal, the RRG recognized that the NRC needs to issue guidance for the level of detail which it will find necessary and sufficient to meet the proposed acceptance criteria listed above.

CECo agrees that greater use of PFRs could improve the NRC's regulatory process. However, CECo believes that the RRG has proposed unnecessarily stringent criteria for giving expedited NRC consideration to PFRs. Few, if any, additional PFRs can be expected if these criteria are adopted. Accordingly, CECo proposes the following, alternative criteria which it believes will both implement the RRG's goals and satisfy the NRC's responsibilities. CECo also has provided guidance for meeting these criteria.

PFRs are regulated by 10 C.F.R. § 2.802. It sets out the required contents of a PFR. Essentially, a PFR should state the reasons for filing the PFR, suggest a solution to a problem, and provide the discussion and data necessary which support the action sought. 10 C.F.R. § 2.802(c). If the PFR is found by the NRC to contain this information, then the NRC is required to publish it. 10 C.F.R. § 2.802(e). Otherwise, the NRC is to request the petitioner to provide the additional information necessary. 10 C.F.R. § 2.802(f). This process, in CECo's view, provides adequate guidance for the preparation and prompt publication of PFRs.

Additional criteria for the publication of PFRs could be construed as proposed changes to the current rule. Alternatively, if those criteria are considered relevant only to the NRC's prioritization of PFRs, an internal administrative matter, they could be counter-productive. Essentially, the criteria would increase the threshold for the timely publication of PFRs if they propose to relax regulatory burdens.

No other PFR is required to demonstrate either that it has no safety impact or that the supporting analysis and data are compelling in order to be published in a timely manner. Any PFR published promptly under these criteria could be challenged for failing to meet them. Therefore, CECo suggests that these criteria not be adopted.

CECo appreciates the RRG's concern that criteria would help the NRC to prioritize licensees' use of the PFR process as a means for proposing improvements in the regulations. CECo believes that the only appropriate and necessary criterion is that the PFR addresses a regulatory concern in a way which meets the current requirements.

A licensee's concern need not be safety neutral if a justification for modifying the level of safety can be provided. Moreover, the analysis need not be compelling; a proposal which is not well-supported will evoke a large number of comments during the comment period required by the Administrative Procedure Act ("APA"). Licensees know that a large number of comments will require substantial time for the NRC to resolve. Therefore, licensees should bear the risk of going forward with a complete, but not compelling proposal.

Instead of adopting stringent new criteria for a specific category of PFRs, the NRC should simply ensure that adequate resources are available to review PFRs promptly for compliance with the completeness requirements in 10 C.F.R. § 2.802(d).

Combined Packaging of Multiple Modifications in a Single License Amendment

Section 2.3.17 of the RRG Report suggests, in part, that a single license amendment request ("LAR") could encompass numerous modifications that, overall, have no effect on the current level of plant safety. RRG Report, Vol. 2 at 132. This would enable the Staff to evaluate numerous issues in the context of a single LAR -- a process that would permit safety decreases to be balanced by safety increases as well as result in the realization of efficiencies and cost savings by both the NRC and its licensees. In conjunction with this recommendation, the RRG states that the Atomic Energy Act does not prohibit the Commission from making a no significant hazards finding on an LAR that proposes more than one change, as long as the amendment satisfies the criteria of 10 C.F.R. § 50.92(c).

The RRG correctly acknowledges that encouraging the submittal of and evaluating licensing actions in a combined manner, guided by an overall acceptance criterion of neutral safety impact, is a significant departure from current NRC policy and practice. Despite this, only a conclusory legal statement about Section 189.a. of the Atomic Energy Act is included in the RRG Report. That statement is insufficient to deter and/or withstand judicial challenge to any such regulatory standard(s) or licensee submittal(s). Unless greater legal support is provided through an official interpretation of relevant regulatory requirements in general, and §50.92(c) in particular, licensees will not have the confidence that there is a sufficient legal basis for combined license amendments. Such a basis is necessary to encourage and support the submission of integral LARs to the NRC for review and issuance.

In its report, the RRG states that a Policy Statement is the appropriate vehicle by which to advise licensees and the public of the manner in which the NRC will exercise its discretionary authority vis-a-vis the evaluation and issuance of license amendments on an integral basis. This assessment would be correct if the NRC's change in policy simply contemplated the "batch" processing of license amendments, each of which would satisfy the no significant hazards considerations test. But that is not the case. The proposed change in policy would permit the granting of license amendments which would not meet the NSCH test unless the term license amendment was redefined to cover a change in more than one license provision. Such a redefinition appears to have been applied to recent notices of amendments in the Tech Spec Upgrade program.

Although the NRC has already, through practice, adopted a broader interpretation of license amendment, licensees would feel more secure in using a broader definition of the term license amendment to submit combined license amendments if this definition were formalized by an interpretative rule explaining § 50.92.

In terms of its content, either a Policy Statement or interpretative rule should fully explain why a significant hazards evaluation need not be prepared on an individual basis for each change included in an LAR or Tech Spec change. This would be accomplished (1) through a more detailed statutory analysis of Section 189.a. of the Atomic Energy Act showing that it does not require an individual significant hazards analysis for each change set forth in a LAR; and (2) by a regulatory analysis of the express language in the existing regulations which similarly does not preclude the inclusion of multiple "changes" in a single LAR. Support for that analysis would include references to recent agency practice, noting that the NRC has not erected an absolute barrier to the consideration of multiple changes in a single LAR or package of LARs.

Apart from its legal content, the NRC should address -- from a policy standpoint -- the propriety of evaluating and assessing other licensee submittals, including commitment reductions, on a combined basis. CECo urges the RRG not to limit this important initiative to license amendments. Its benefits can be more widely realized if the concept is expanded to include the evaluation of other licensee efforts to achieve O&M cost reductions, especially safety evaluations of changes pursuant to 10 C.F.R. § 50.59.

In addition to providing legal comfort, the NRC must provide licensees with guidance to follow in demonstrating that a group of changes encompassed by an LAR or commitment reduction effort, for example, do not reduce overall plant safety. Critical to such guidance are the examples of aggregation which the NRC will recognize. For example, several changes in one system which overall improve its availability should be permitted in one LAR. Similarly, a group of modifications which together reduce the potential impact of an accident scenario should be permitted in one LAR. These and other, perhaps broader examples should be endorsed explicitly by the NRC.

Finally, such guidance could infuse innovative concepts into the significant hazards evaluation process. For example, the NRC could allow licensees to aggregate license amendments over time. Prior changes that yielded safety improvements could be "banked" so that credit for them could be taken at a later point in time, in conjunction with a significant hazards analysis for a change which is not safety neutral. Such guidance would assist licensees to reduce unnecessary regulatory requirements and commitments in a cost-effective manner.

Safety Evaluations for
Changes to Commitments

Licensees make numerous commitments to the NRC -- in the FSAR, in other licensing submittals, and in response to enforcement actions, Bulletins, and Information Notices. Commitments which are necessary for achieving compliance with regulations constitute legally binding requirements that are enforceable by the NRC. Alternatively, commitments which enhance a licensee's ability to achieve compliance but are not necessary for compliance are voluntary in nature and do not give rise to legally binding requirements. It is imperative that a licensee understand the nature of its commitments and follow appropriate procedures when changing them. The measures proposed in Section 2.3.2 of the RRG Report are an important first step toward a uniform understanding of commitments and of how they are to be treated by licensees.

In order to realize the potential for reducing the regulatory burden imposed both on the NRC and its licensees, the RRG correctly points out that the Staff and licensees should focus their resources on changes to commitments that may be significant contributors to plant safety. The proposed definition of "commitment" set forth in the Report is a step in the right direction but fails to adequately capture the tenor of the accompanying discussion and precisely distinguish between the two categories of licensee commitments distinguished above. Accordingly, CECo urges the NRC to consider adopting the following two definitions in 10 C.F.R. § 50.2:

> **Enforceable commitments** are those which have been made by the licensee and have been relied on by the NRC in writing in docketed correspondence for the purpose of finding that a facility satisfies legally binding requirements in either the facility license (license conditions and technical specifications) or regulations and orders issued by the NRC in accordance with the Administrative Procedure Act. This category of commitments gives rise to legally binding requirements which are enforceable by the NRC.
>
> **Voluntary commitments** are those which involve actions that are not necessary for compliance with NRC requirements or are outside the scope of existing regulations. They typically include licensee-initiated programs or measures taken to enhance the assurance of compliance.

Based on these definitions, licensees can be more clearly instructed (rather than simply on the basis of the commitment's "safety significance," as suggested in the RRG Report) as to what commitments should be submitted to the NRC Staff for review and approval and what commitments licensees are authorized to change without prior NRC approval. Such instructions would include a complete discussion of the change requirements attendant to each category of commitments in the Statement of Considerations accompanying any rulemaking to amend §50.2. In addition, the NRC is urged to issue regulatory guidance addressing the commitment change process and forego issuing the proposed amendment to §50.54.

Enforceable commitments should be grouped into two tiers based on their legal significance. The method for changing a commitment depends on which tier it is in.

> Tier 1 --
> License conditions and technical specifications. These may not be changed without a specific license amendment pursuant to 10 C.F.R. §§ 50.90 - 50.92.
>
> Tier 2 --
> Written commitments which have been relied on in writing by the NRC in either an SER or other docketed correspondence. Included are commitments described in the licensee's SAR and any other compliance commitments which could be incorporated in the SAR. These can be changed pursuant to the procedures specified in 10 C.F.R. § 50.59.

Voluntary commitments should be explicitly recognized as not requiring NRC approval to be modified. However, an informational letter to the NRC shortly after the change should be recommended by the NRC in order to fully satisfy the requirements of 10 C.F.R. § 50.9. Such a letter will ensure that licensees update the docket to assure "complete and accurate" information. In case of doubt, a licensee always has the option of providing the NRC with advance notice of a pending change in order to determine if it should be reviewed by the NRC as involving substantive regulatory compliance.

The definitions provided above, when coupled with corresponding guidance on changing commitments, would better serve both the NRC and its licensees in the commitment change process than reliance on the undefined and amorphous concept of a commitment's "safety significance," as set forth in the RRG Report. More importantly, this initiative should reduce the regulatory burden imposed on both the Staff and licensees by eliminating any confusion over the need to perform safety evaluations of "voluntary commitments." The end result should be a more clearly defined focus on changes to commitments that are necessary to meet binding legal requirements.

Application of Regulatory Requirements
As the Only Measure of Specific Plan Changes

Section 2.3.9 of the RRG Report identifies inconsistencies in the way existing regulations govern the evaluation and approval for certain changes made to licensee facilities and procedures (§ 50.59), QA plans (§ 50.54(a)), security procedures (§ 5054(p)), emergency plans (§ 50.54(q)), and fire protection plans (§ 50.48(a)). The differences inherent in these regulations do not accurately reflect the safety significance of underlying licensee activities. Thus, the RRG is correct in concluding that it would be advantageous and logical for the NRC to implement measures which would infuse consistency into these regulations.

CECo agrees that the rationale driving the evaluation of changes to programs or procedures encompassed by the regulations identified in Section 2.3.9 of the Report should be the continued satisfaction of regulatory requirements. A licensee should be allowed to make changes without prior NRC approval as long as the changes do not preclude the licensee from meeting the applicable underlying regulatory requirements of 10 C.F.R. § 73.55 for security, §50.49 (and Appendix E) for emergency plans, and § 50.48 for fire protection. It, therefore, is appropriate for the NRC to reinterpret its regulations to fully implement this concept.

Although linking the evaluation and approval for certain program and procedure changes to their respective regulatory bases is clear in principle, experience dictates that implementation of the idea will be more difficult in practice. One difficulty that likely will have to be overcome is defining the benchmark for those changes that cannot be directly tied to specific regulatory requirements or commitments. For example, the size of the security response force is not specified in the rules but only in the Security Plan. Guidance for calculating the size of the force in a specific plant is provided in a NUREG. However, reliance on the NUREG to judge the need for prior NRC approval of a security force reduction would not be appropriate because NUREGs should never be treated like rules. Therefore, to implement the good intentions of the RRG, CECo recommends that the NRC permit licensees to exercise flexibility to modify commitments where rules do not impose specific limits.

Relief From Non-Binding Documents
Inappropriately Treated As Requirements

The first RRG recommendation set forth in Section A.3.2.1 of the Report calls for reconsideration of the "practice of elevating Commission policy statements, regulatory guides, and other non-requirements to the status of legal requirements without following the disciplined rulemaking process." Vol. III at A-17. This recommendation should be adopted by the Commission. Only rules and regulations, promulgated pursuant to the Administrative Procedure Act, and NRC orders can impose binding legal requirements on NRC licensees. Commission policy statements, regulatory guides, and other non-binding communications (e.g., Bulletins, Information Notices, and Generic Letters) do not constitute legal requirements. Therefore, they should not be treated as regulations -- generically or otherwise.

In particular, the Commission should be more receptive to licensee-proposed alternatives to actions suggested as a matter of compliance in NRC generic communications where those suggested actions exceed regulatory requirements. Such NRC receptiveness to licensee alternatives should not be limited to prospective application but should also be applied retrospectively to both pending discussions and to closed matters which have imposed undue burdens. Prospectively, the Commission is urged to adopt the following proposals to streamline the generic communication process, to enhance the flexibility available to licensees in submitting their responses, and to reduce the regulatory burden imposed on both the NRC and its licensees.

(1) Each generic communication that requests licensee action should identify alternative courses of action available to licensees. The applicability of a particular alternative could be conditioned on a licensee's ability to make certain factual showings demonstrating the adequacy of the particular alternative chosen for its facility. The Staff should seek licensee input in developing the pertinent conditional factors prerequisite to the implementation of the alternatives provided for in a particular generic communication.

(2) Licensees should at least be offered an opportunity and, ideally, encouraged to present additional alternatives to the action requested in a generic communication.

(3) The NRC should provide licensees with a process (perhaps modelled on the backfit claim review process) by which to seek review of and relief from Staff rejection of either the applicability of an NRC alternative or the adequacy of a licensee-proposed alternative action.

The opportunity for mandatory NRC review of licensees' proposals is particularly important. If structured properly, it ensures full and fair NRC consideration of all relevant factors. It could be a review process similar to that employed in connection with the rejection of a §2.206 petition or backfit appeals. Appropriate review would establish the necessary indicia of objectivity by requiring decisionmaking by Staff members other than those responsible for the development, issuance, and review of generic communications.

Retrospectively, licensees should be encouraged to provide alternative means for implementing non-binding suggestions even where those alternatives may already have been rejected. Here, too, independent senior NRC management review is critical.

Graded Approaches to
Certain Regulatory Requirements

The RRG has recommended that the NRC evaluate the feasibility of employing a graded approach to the applicability of the technical provisions of certain LCOs and surveillance requirements and in the implementation of specific review committee functions. Section A.3.2.4, Vol. III. Alternatively stated, this recommendation centers on the implementation of a more performance-based (i.e., a graded approach) that takes into account the relative safety significance of the different areas and items subject to NRC regulation. Because such an approach would provide licensees with more flexibility and is consistent with regulatory requirements, CECo endorses adoption of the graded approach recommended by the RRG.

For example, Appendix B to 10 C.F.R. Part 50 authorizes the NRC and its licensees to employ a graded approach to quality assurance. The QA program requirements set forth in Criterion II specifically provide that "[t]he quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, and to an extent consistent with their importance to safety." Consistent with the graded approach inherent in this statement, the full panoply of QA requirements should not be applied to non-safety systems, such as ATWS.

Similarly, licensees should be authorized to implement a graded approach to outage times for security deviations and LCOs for operations/surveillance based on safety significance. For security deviations, the concept of "importance to safeguards" should be used in the same manner as important to safety is used in Appendix B. Some guidance on how to determine "importance to safeguards" has already been provided by the NRC in the enforcement process. Additional guidance should build on this experience by using realistic estimates of the exploitability of safeguards deviations. As for LCOs, insights from PRAs should explicitly be applied.

Miscellaneous Recommendations

A. Current Licensing and Design Bases

Section 2.3.10 of the RRG Report contains a compendium of miscellaneous comments pertinent to various NRC regulations. In particular, the RRG notes that there is "no clear understanding of the scope and depth of the term `design bases'" and that, although Part 54 defines the term "current licensing basis," (CLB) there is no companion definition in Part 50. Therefore, the RRG has recommended that a definition for CLB be "developed . . . and that the scope and depth of the term `design bases' be clarified." RRG Report (Vol. II) at 72.

Turning first to the definition of CLB in Part 54, although it was adopted for renewing plant operating licenses, its concept is not limited to the license renewal context. The definition (10 C.F.R. § 54.3), as opposed to the application of the CLB in the license renewal process (10 C.F.R. § 54.21), is not unique to Part 54. Indeed, the definition must be equally applicable to Part 50 because the CLB derives from and is the product of initial plant licensing and operation under Part 50. The CLB carries forward into the renewal term for purposes of Part 54 licensing. Therefore, the definition of CLB in Part 54 must be identical to any definition of the term included in Part 50.

The RRG also has called for the NRC to clarify the definition of "design bases" set forth in 10 C.F.R. § 50.2. This is not necessary. Any change to the § 50.2 definition would disrupt recent NRC and industry efforts to reach a common understanding of the term. Moreover, any such amendment would unnecessarily undermine industry efforts to enhance design documentation programs. Utilities have spent considerable sums developing such programs using the existing definition.

The definition was adopted in 1968 and has not been substantively revised since then. See 33 Fed. Reg. 18,610 (December 17, 1968). There is no serious question regarding the use of the term within the existing regulatory framework, especially for reporting purposes under 10 C.F.R. §§ 50.72/73.

Industry guidance on the scope of design bases is provided in NUMARC 90-12, "Design Basis Program Guidelines," October 1990. This guidance has been found generally acceptable by the NRC Staff. See SECY-91-364, "Design Document Reconstitution," November 12, 1991. In SECY-91-364, the NRC Staff distinguished the terms "design basis" and "licensing basis." In addition, NUREG-1397, "An Assessment of Design Control Practices and Design Reconstitution Programs in the Nuclear Power Industry," April 1991, states (p. 8) that "[t]he licensing commitment tracking system and the FSAR are important source documents to support the design-bases documents."

Finally, contrary to the RRG's suggestion, the NRC need not take action, such as rulemaking, to remedy any confusion resulting from the NRC policy statement on "Availability and Adequacy of Design Bases Information at Nuclear Power Plants." See RRG Report, Vol. I, § 4.1.2 "Potential Improvements" at 13 (second bullet). NUMARC's letter to NRC dated October 2, 1992, states the belief that "utility efforts with respect to design basis reconstitution have been very productive under the current regulatory environment." CECo agrees with the NUMARC position and does not believe that further regulatory action is necessary.

B. Line-Item Tech Spec Improvements

CECo endorses the RRG's recommendation that the NRC should permit line-item improvements in accordance with the Improved Standard Tech Specs to be made available to individual licensees on a plant-specific basis. RRG Report, Section B.3.2.4 (Vol III). This option should be made available in addition to lead and subsequent plant licensees who opt to adopt Standard Tech Specs at their facilities. Such limited ad hoc relief will significantly benefit older plants by making their operation more efficient and, therefore, should be made available.

Resource constraints have limited the NRC's ability to grant such line-item improvements. Such resource constraints could be eased substantially if the NRC would consider periodically identifying line-item improvements in a generic communication and providing a simplified process by which all affected licensees could adopt them.

Attachment 3
Commonwealth Edison Comments
Regarding Regulatory Review Group Report
Volume Four

General Comments on Volume Four

CECo is interested in the prospects for the future deployment of IPE applications as generally envisioned in the RRG's PRA application proposals. CECo has additional tasks to complete our IPE program commitments and tailor specific application plans for deployment at each site. However, when complete, some of the guidance and limitations provided in Volume Four, may create barriers to full utilization, as previously described in CECo's letter to J. Sniezek dated May 20, 1993. Although some changes were made to the initial draft report resulting from the public meeting comments (such as treatment of HRA), others were not addressed and are discussed below.

Many concepts described in Volume Four are desirable and feasible given the proper levels of resources and commitment. Utilities should be allowed to explore the potential PRA uses envisioned in Volume Four, as well as, others as they are identified. However, the report should clearly state that the degree of implementation or departure from the current conduct of business should rest with the individual utility.

In some cases, Volume Four is overly prescriptive and absolute. The report should be generally conditioned to allow utilities to offer alternative approaches where plant-specific justifications are provided. The realities are that once stated in the report, it will be difficult to deviate from any prescriptive attribute. A major example of this is the proposal to structure the Graded QA requirements around "plants of similar design." While doing so might simplify the NRC review process, the technical justification for doing so appears to be lacking. Those SSC's that have not been found to be relatively important in the plant-specific PRA should not be penalized or conditioned in any way because of a PRA result of a "plant of similar design." We know that individual plants within such a class of similar design can differ substantially, and that was the original rationale for performing IPE's. The report should allow plant-specific justifications to take precedence when valid plant-specific assessments are available.

Another example concerns the requirement for plant-specific data in certain PRA applications. The treatment of generic data versus plant-specific data for a given PRA application should be decided on plant specific basis where valid reasons exist, and not as prescriptively defined in Volume Four.

The logistical problems in maintaining an active data base of the "similar plant" results to accomplish the above is of concern. Volume Four page 4-66 states that "The data base will allow users to gather information both by plant and across plant." As indicated in a recent Commissioner briefing (SECY-93-110), the NRC is making progress on development of a database to capture plant-specific information from the IPE submittals. However, it is not clear how such a data base will function at the system or component level of interest, how such information will relate from one plant to the next, utility accessibility, etc. In any case, plant specific justifications should prevail, as discussed above.

Volume Four should consider one additional application of PRA :an extensive re-writing of 10CFR50. The Design Basis Accident (d.b.a.), with all of its conservative assumptions, calls for redundancy, etc., was established with the best tools of the time. Now available is a much greater understanding of severe accidents, better tools for calculation, and many years of operating experience. This can justify more realistic design basis accidents, more precise inclusion of conservatism for safety margins, and correction of such misleading assumptions as instantaneous release of fission products from the core at the start of an accident. This is in part a PRA application and in part an application of the other things that have been learned in severe accident analysis since the TMI accident. Not only in the interests of effective use of utility and NRC resources, but also to improve safety.

Specific Comments on Volume Four

Page 4-7 Introduction 5th para: While the Introduction is a reasonable summary, overall, some statements are made which are incorrect. Text on page 4-7 states that performance-based criteria may be more "appropriate" and "robust" when applied to core damage or system availability rather than public risk. Containment and source term analyses have more substance and are of more value than implied by this statement. These statements should be deleted. Use of source term analyses should not be dismissed. It is reasonable to focus on system performance or core damage frequency, however, for the established reasoning of it being more desirable to prevent core damage than to have to "contain" it.

Page 4-10 Introduction 1st para: It is stated, "While the quantitative results are important, they should be considered as most useful for a screening of the results to identify important accident sequences and plant features, and to give indication of areas with relatively little or relatively high importance in a probabilistic context." This statement implies that quantitative analysis is useful only for "scoping." Statements like this should be deleted. Every use of the PRA relies on "numbers," whether it is in prioritizing MOV's, evaluating modifications, or some other use. The "quantitative" aspect of PRA's is their strength.

Page 4-19 PRA Results: The results of PRA's are much more than just the importance measures listed here. The discussion should also include such other PRA results as core damage frequency, the nature of the scenarios most important for core damage, the timing of accident progression, and the minimum equipment necessary to accomplish various safety functions. Additionally, PRA is a way to quantify the merits of changes to the plant. Limiting this section's discussion to Importance Measures implies that they are the only important results of a PRA. Also, the discussion in this section refers to "component" importance. The Importance Measures discussed apply to "basic event" importance. One component is often represented by more than one basic event. A sentence should be added stating that basic event importance can be used to assess component importance, and then the words "basic event" importance should be used exclusively in describing the traditional importance measures.

Page 4-22 PRA Applications Figure 4.3-1: MOV prioritization for G.L. 89-10 should be added to the list of Group 1 applications. Assessment of operational occurrences should be added to Group 2.

Page 4-25 Importance Definition: The equation for Increase Importance appears to be incorrect. Risk achievement importance is defined as the CDF with that basic event failure probability set to 1.0, divided by the base case CDF. In addition the discussion should permit use of other importance measures, like the effect of doubling a component failure rate, for making such assessments.

Page 4-26 Importance Classification: The discussion employs the undefined terms "probabilistically unimportant" and "deterministically important." Besides lacking precise meaning, these terms are misleading. PRA's use a lot of deterministic methods, and traditional rules often use probability-related arguments (like the rejection of reactor vessel rupture as a d.b.a. because it is too unlikely). Perhaps terms like "traditionally important, based on d.b.a. rules" would better describe the situation.

Page 4-29 Graded Implementation Requirements: See discussion provided above detailing our concerns with "plants of similar design."

Page 4-32 PRA Criteria: The treatment of HRA utilizing screening values is an improvement from the initial draft report which recommended Human Error Probabilities (HEP's) set equal to 1.0. However, assuming even these screening values yields a distorted view of the plant risk profile. The proposed screening values are still too high to be a reasonable representation of operator reliability, for most operator actions. Both the NRC and the industry have spent considerable resources in improving procedures and operator training. To correctly represent risk, proper credit must be given for those procedures and that training. Operator actions are crucial to successful response to accidents, and most would be accomplished with a high degree of success. Many operator actions, with much time available and with good procedures and training, are certain to have failure probabilities much less than the suggested screening values. This document should not specify HEP values to be assigned. The Licensee should justify his choices in the same way that he justifies every other aspect of his PRA. Uncertainty should be treated via sensitivity studies of particular important human actions.

Page 4-35 Configuration Application Regarding AOT's: This discussion suggests permitting a plant to remain at power operation when the risk of staying in that mode beyond an AOT is lower than the risk of shutting down. The methodology should also permit staying at power when the risk of doing so is negligible, regardless of the risk of shutting down.

Page 4-38 Configuration Application Regarding STI's: The discussion of unavailability is oversimplified. The equation shown is based on a failure model that assumes that failure probability is proportional to time since last test. That is true, in practice, for only certain failure modes, and it is not true for others. The model is often used because it provides a first-order (linear) estimate. But it is only one kind of approximation, and it must be recognized to be only an approximation. This issue needs more attention, explanation, and flexibility in the report.

Page 4-41 PRA Application for On-Line Configuration Control: It is stated that the state of the art will not support this type of application. That is not true. This can be done with only slight further development of current methods.

The discussion on p. 4-41 appears to emphasize the need for requirements, audits, and inspections, at all levels, if such a tool is employed. It must be recognized that there is always going to be some uncertainty when such tools are used. Given the nature of the tool, it is probably not practical to require the kinds of rules, audits, and inspections traditional for other systems. This should not be a reason to avoid use of such tools. These tools can uniquely make it possible to identify daily plant configuration changes which may have high risk. Anticipating all such combinations is not possible with, and cannot be completely specified in, written documents such as Technical Specifications. Therefore, the tools have the potential of greatly increasing safety. If rules are too prescriptive or too burdensome, those tools will not be feasible. The safety benefits of such tools should not be lost simply because there may be some uncertainty concerning their modeling. It may not be wise to put such applications in rigid documents like Technical Specifications. The NRC Staff should, however, recognize these kinds of tools as useful ones to support safe plant operation, should give credit to utilities who use them effectively, and should avoid regulations that make them impractical.

ENTERGY

Entergy Operations, Inc.

John F. McGaha

July 21, 1993

Regulatory Review Group
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Frank P. Gillespie, OWFN 12 D21

Subject: Comments on NRC Regulatory Review Group Draft Report

CNRO - 93/00026

Dear Mr. Gillespie:

Entergy Operations, Inc. has reviewed the NRC Regulatory Review Group (RRG) draft report concerning the review of power reactor regulations and related issues (58 FR 29012 and 58 FR 33285). We wish to submit the following on behalf of Arkansas Nuclear One Units 1 & 2, Grand Gulf Nuclear Station, and Waterford 3 Steam Electric Station.

Overall, the report is of a high quality and contains a number of excellent recommendations. The NRC management and staff deserve recognition for such an in-depth review and the resulting innovative alternatives to current regulatory approaches. We believe that this initiative, if continued to be pursued, will ultimately result in a significant benefit to safety for the commercial nuclear power industry. While difficult to quantify, the current level of regulatory complexity and burden, coupled with overcommitment by licensees, represents a source of distractions to both our personnel and our allocation of resources at the expense of more safety significant concerns. Removing unnecessary requirements and streamlining regulatory processes will result in allowing the industry and the NRC to more clearly focus on operational safety issues.

We urge the NRC to pursue these and additional efforts related to regulatory burden reduction. Such efforts are extremely important to the future of the commercial nuclear power industry. Because of the limited lifetime of the RRG, organizational and policy changes must be expeditiously implemented by the NRC to ensure that the work of the RRG continues. We also believe that in order for the RRG effort to continue a continuing charter and champion within the NRC will be required.

In addition to NRC efforts, we recognize that many changes recommended by the RRG will require licensee support and participation. Entergy Operations is ready to provide needed support in the further development of recommended initiatives and would appreciate the opportunity to participate with the staff in this effort.

Additional comments are provided in the attachment. We appreciate this opportunity to express our views on the report and the Commission's consideration of our comments. Please contact Mr. Kenneth Hughey (601-984-9756) or Mr. Herbert Kook (601-984-9766) of my staff should you have any questions or desire additional information regarding this matter.

Sincerely,

W.K. Hughey
for
J. R. McGAHA

JRM/wkh
attachment

cc:

Mr. T. W. Alexion	Mr. J. L. Milhoan
Mr. R. P. Barkhurst	Mr. P. W. O'Connor
Mr. R. H. Bernhard	Mr. N. S. Reynolds
Mr. R. B. Bevan, Jr.	Ms. L. J. Smith
Mr. J. L. Blount	Mr. D. L. Wigginton
Mr. S. D. Ebneter	Mr. J. W. Yelverton
Mr. E. J. Ford	Central File (GGNS)
Mr. C. R. Hutchinson	DCC (ANO)
Mr. H. W. Keiser	Records Center (WF3)
Mr. R. B. McGehee	Corporate File [12]

Additional Entergy Operations, Inc. Comments On the NRC Regulatory Review Group Draft Report

__Volume One__

Page 5, para 2 The statement that "full implementation of an alternative approach" to Appendix B will not require a rule or license change is not necessarily true for all possible alternatives to the existing approach, although such a statement does appear to be true for those approaches discussed in the report.

Page 6, section 4.1.1 Care should be exercised with this and other sections so that new definitions do not lead to a new set of requirements which are more burdensome than the existing requirements. New definitions should be carefully made and fully coordinated among NRR, the NRC Regions, and the industry before their adoption.

Page 7, first bullet It may be premature to state that Appendix B itself should not be changed or modified. More review and consideration, and discussion with the industry, should occur before this is stated definitively.

Page 7, second bullet Clarify that the Regulatory Guides 1.84, 1.85, and 1.147 are those which appear to require revisions on a periodic basis.

Page 10, first bullet Clarification should be added to the phrase "maintaining appropriate control of changes to material that is removed from Technical Specifications...". EOI assumes this refers to Core Operating Limit Reports, certain surveillance requirements, equipment lists, etc. placed under licensee administrative controls, but a more explicit description is needed.

Pages 17-19, section 4.3 The ideas in this section represent a significant potential for reductions in unnecessary burdens for the industry. However, the industry and the NRC have many obstacles to overcome before the results outlined in this section are achieved. A recommendation should be considered for the NRC to evaluate offering introductory training in risk technology for the staff.

The fourth bullet on page 18 recognizes a key obstacle to risk technology use by the industry, i.e., the NRC embracing risk technology where it shows additional requirements are necessary but rejecting it when it contradicts a preferred staff position. A Policy Statement, SRM, or other process might be used to improve this situation. For example, the staff might be required to specifically address risk considerations in the documented backfit evaluation or analysis for new requirements.

<u>Volume Two</u>

Pages 9-10 The discussion of commitments and changes to commitments should be clarified. The term "equivalence in safety" and "equivalent to the original commitment" should be replaced with discussion such that commitment changes are permitted without NRC approval so long as the changed commitment does not constitute an unreviewed safety question per 10 CFR 50.59.

Page 10 In the first sentence, "Proposed " should be deleted (also in the third sentence) and "shall be" changed to "are" since 10 CFR 50.71(e) already requires the reporting of changes discussed but does not require reporting prior to the change being made. In section II, the use of "prospectively" is unclear; if it refers to applying the new change method and definitions to future commitments only, the NRC, industry, and licensee use of multiple commitment definitions and processes may be extremely cumbersome.

Pages 25-26, sections IV and V/VI Renumber the second section V to VI and the existing VI to VII. While considering changes to the NRC FFD program, the NRC rejected a two-tier system of 50% testing for employees and 100% for contractors for a single system at a 50% rate. Other industries regulated by the DOT have a significantly higher positive test rate than nuclear employees or contractors; the DOT considers these rates adequate to propose to lower their required testing rate below 50%. Therefore, the recommendations in section V/VI should be changed to include that the rulemaking discussed in section IV should be changed (or a new rulemaking proposed) to allow 50% testing for all nuclear workers.

Pages 25-26, sections V-V/VI Overall, the proposed recommendations appear positive. However, the NRC RRG should also consider the burden caused by the scope of audits, auditing HHS-approved laboratories, and/or the re-auditing of activities and information pertinent only to laboratory selection and establishing initial contracts with laboratories.

Page 26, section VI/VII There is no basis for the recommendation provided under Potential Improvements and the need for this action is not evident. In addition, this appears to recommend establishing a new regulatory requirement via generic communications and contrary to 10 CFR 50.109. New regulatory requirements should be established through rulemaking and in compliance with 10 CFR 50.109 as discussed elsewhere in the report.

Pages 27-30 Generic communications are inconsistently used, which leads to NRC and industry confusion. Additional action is needed to resolve prior industry concerns in this area. New definitions should be adopted by the NRC for the types of generic communications, for example:

-Information Notice: A mechanism to share NRC and/or industry experience issued for information only; new/different NRC positions or new regulatory requirements would not be established by Information Notices

-Bulletin: A communication of action(s) in regard to a specific issue which the NRC would clearly expect licensees to either complete or provide justification for alternative action(s); these expectations may involve a new or different NRC position regarding previously existing regulations but new regulatory requirements would not be imposed by a Bulletin; confirmation of actions or alternatives may be requested under 10 CFR 50.54(f) (note much of this role has been recently performed by generic letters)

-Administrative Letter: A communication of an administrative or informational nature which would not require a response and would not address: new/different NRC positions, new regulatory requirements, or generic safety issues

-Generic Letter: A communication about generic safety issues, or a generic request for information which would be reasonably expected to exist (possibly under 10 CFR 50.54(f)), and which might involve new/different NRC positions. Generic letters would not: establish new regulatory requirements, request confirmation of the completion of new actions/programs, or require the generation of significant amounts of new information.

Page 29 We believe that the NRC staff biweekly letter listing pending generic communications could be changed to a monthly letter without impact on the industry while possibly reducing NRC burden.

Pages 64-65, section 2.3.8 The regulatory basis for "requiring" the submittal and review by the NRC of the computer code verifications is unclear. The only basis mentioned is Generic Letter 83-11, which is not a regulatory requirement. Information should be provided for the basis of this requirement, the recommendation changed so that the NRC staff no longer imposes this burden upon the industry, or rulemaking initiated to properly establish the requirement.

Pages 68-69, section IV The text of section IV states that the rules governing changes to QA, security, fire, and emergency plans should be the same. The changes in approved QA programs to be submitted for prior NRC approval in the proposed 10 CFR 50.54(a)[3] do not appear consistent with the changes requiring prior NRC approval recommended for 50.54(p)(2) and 50.54(q). Change "provided the change does not reduce the commitments in the program description previously accepted by the NRC" to "provided the change does not reduce the commitments in the program below the requirements of 10 CFR Appendix B."

Page 71 section III Consistent with a philosophy of eliminating unnecessary regulation and regulating only where real need exists, this section should not be included unless real situations occur in which a utility fails to provide adequate space for NRC personnel.

Pages 73-74 See comments regarding pages 9-10 and page 10.

Pages 75-81 As inferred in this section, Policy Statements addressing licensee issues are frequently considered by the NRC staff as regulatory requirements. Such requirements should properly be imposed upon licensees as backfits in accordance with 10 CFR 50.109. Add that Policy Statements should address only internal NRC policies, i.e., practices and expectations, rather than those expected or demanded of licensees. Items in section A should be deleted, and items in sections C, D, and E should be deleted or replaced by rulemaking.

Pages 90 - 92, section II These are strong statements, which we endorse. They do, however, highlight the need for the NRC to be willing to adopt changes in attitude and culture to accept the new approaches discussed in the report.

Page 98, section XII In this section and many others, the term "performance-based" is frequently used but is never well defined or explained. As discussed previously, various people or organizations in the NRC and the industry may make a number of different interpretations of any term which is not well defined. The report should include a clear definition of performance-based which is agreeable to the industry and the NRC.

Page 107 The reference for "update of the FSAR" should be 10 CFR 50.71(e)(4).

Page 145-146 The discussion on this subject is excellent. The recommendation should be modified to encourage licensees to begin considering the use of risk methods to provide additional bases for USQ determinations; this would allow the NRC and licensees to gain experience and insight in using these tools as soon as possible; in addition, some concerns not apparent by more traditional methods might be identified.

Volume Three

No comments.

Volume Four

General There is a considerable amount of new and different material in this section. It should be carefully considered and reviewed before pursuing full implementation. And as mentioned previously, care should be exercised so that new definitions do not lead to a new set of requirements which are more burdensome than the existing requirements. New definitions, especially in this relatively new area, should be carefully made and fully coordinated among NRR, the NRC Regions, and the industry before their adoption.

Section 4.2.3

Human reliability analysis (HRA) offers valuable insights to a PRA study. By using HRA analysis, past PRAs have identified a number of problems in procedures and training that had not been found using "traditional regulatory" approaches. HRA techniques can help to measure many effects which influence the reliability of an action, thereby establishing the relative importance of human failure events. This assists in prioritizing training activities and showing the effectiveness of verification activities, allowing the most effective use of resources. HRA techniques offer the best, if not only, means of evaluating the significance of training, stress, hesitancy, conflicting priorities, etc

If an Human Error Probability (HEP) of 1.0 is used, the effectiveness of using safety prioritizations determined by the IPE is weakened. Most application work and sensitivity work done (by utilities) with PRA is done by manipulating the cut set results. Therefore, many cut sets are lost (i.e., truncated) once an HEP has been added to the cut set. Evaluations done by the manipulation of dominant cut sets usually can be performed fairly quickly. However, requantification of an entire PRA is much more time consuming.

If the PRA should be requantified with all HEPs (both pre-accident and post-accident) set to 1.0, this may well cause important cut sets to be lost because of limitations of the computers used in quantifying the accident sequences.

That is, important hardware cut sets may be truncated because of the large number of cut sets generated because of the overly conservative HEP of 1.0. Rather than setting all HEPs to 1.0, a reasonable screening value should be used. Different screening values should be used for pre-accident and post-accident HEPs. The pre-accident screening value could be relatively low as these actions are usually controlled by procedure and well understood. Post-accident screening values of approximately 0.5 could be used since these are usually cut set dependent and time considerations come into play. Values or ranges of values should be specified by the NRC or the industry for acceptance without specific additional plant-specific justification.

Screening values are often used in PRA studies to solve the plant model in order to protect against dependencies, high stress, and other influences which could invalidate a human failure event probability. An adequate consensus should exist resulting from this experience to develop industry screening values. The NRC Accident Sequence Precursor study method values for human failure events might also be used; these are regularly used by the NRC staff in regulatory evaluations. Another option would be the development of conservative probabilities developed for actions required within various time intervals. Such an approach would allow the distinction between actions required to be performed in ten minutes from those required to be performed in several hours.

Section 4.4

Consideration of a graded approach to QA programs based on risk is a very positive development in allowing the industry to utilize resources commensurate with importance to safety. However, the benefits to be gained are a function of how the QA groups are defined. The proposed groupings throughout section 4.4 appear overly conservative and will limit the benefits from adopting such an approach. Specifically, including any SSC found in "A PRA," that is, in any similar plant, in the most important QA group will unnecessarily reduce the benefits of a graded program and will conflict with PRA application to the Maintenance Rule.

Although the NSSS design of plants are similar to other plants of their class, there are significant design differences, particularly in support and balance-of-plant systems, such that the validity of applying the conclusions of one plant PRA to another is doubtful. For example, a support system at Grand Gulf may exhibit a significantly different risk measure than the same system at another BWR/6. Requiring Grand Gulf SSCs to be determined by the characteristics of another plant is inappropriate. An attempt to differentiate these SSCs is made in the document but results in the creation of another category of SSCs. SSCs that are shown to be not important on the basis of a plant specific IPE do not warrant being included in a graded QA category higher than SSCs not important in any PRA.

In addition to the above concern, some plants such as Grand Gulf have utilized their IPE results to rank SSCs for the Maintenance Rule implementation. Adopting a different approach to a graded QA program would create inconsistencies between these two programs, while they should have the same basis.

If the goal of including risk results from other plants is to ensure that all necessary SSCs have been included with a minimum of review effort, the same result could be attainable through a licensee comparison of plant-specific risk rankings against "class-average" rankings and requiring justification for any discrepancies which might be of concern. Most plants have already compared their IPE results with other facilities of their class and understand the reasons for such differences. Therefore, little additional burden would be created. This appears to be a much more preferable approach than mandating unnecessary conservatisms which would be in effect for the remaining life of a plant.

Section 4.4.1

The need to be able to normalize the importance measure is understandable but this appears to put plants with better core damage frequency numbers at a disadvantage to those with poorer results. For example:

Plant #1 has baseline CDF of 1E-5 with an SSC that has an Achievement Importance Measure of 4E-4. The unavailability impact for this SSC is a factor of 40 which would be considered relatively important for Plant #1.

Plant #2 has baseline CDF of 1E-6 with an SSC that has an Achievement Importance Measure of 4E-5. The unavailability impact for this SSC is again a factor of 40 which would also be considered relatively important per the definition.

However, the change in CDF associated with Plant #1's SSC (Achievement Importance Measure - CDF) is 3.9E-4, while the change in CDF associated with Plant #2's SSC is 3.9E-5. There is an order of magnitude difference in the two but the defined Achievement Importance Measure impact ratio makes them appear equal. There should be another measure to capture importance without penalizing those plants that have a relatively low CDF. Possibly, the above could be offered as an example with some desirable attributes, such as providing a method to ensure that existing levels of safety are maintained.

Section 4.4.2 - 4.4.3

On page 4-27, second para The first two sentences would be clearer if changed to: "Those SSCs of a plant which are not included in the Q list have been determined to be deterministically unimportant, and are therefore not currently required to be subject to QA requirements."

As discussed above under Section 4.4, the inclusion of all SSCs found to be important in "a PRA" is not necessary if justified. Also, as now written, the description of Group A SSCs appears to envelop all Group B SSCs, which appears confusing. A possible alternative phrasing (which also addresses the preceding comment) might be:

"Group A - Those SSCs meeting one of the following criteria:

-Found to be important in the plant-specific PRA

-Found to be important through deterministic methods.

Group B - Those SSCs meeting one of the following criteria:

-Found to be important in a PRA of a plant of similar design <u>and</u> which has not been shown to be not important by plant-specific means

-Found to be important to plant or system availability through deterministic or other means."

For Group A SSCs, generally the same standards as now are used for SSCs requiring "full QA" would be used for design, procurement, receipt, storage, installation, testing, maintenance/surveillance, etc. For Group B, only limited activities such as design control and initial testing should be imposed. Other programs and requirements (i.e., the Maintenance Rule) would ensure that these SSCs would continue to operate to the level identified as appropriate by the IPE. For Group C, these would be subject to the same minimal requirements as now applied to "nonsafety-related" or nonnuclear equipment.

Section 4.4.4

No basis is provided for the statement that "Generally, updating the PRA at every refueling outage will provide this [necessary] currency." As the industry matures, significant changes to existing nuclear power plants are expected to be greatly reduced if not eliminated. Evaluating changes as they occur, as part of the design and licensing process, for possible impact on a PRA would appear to be a more reliable and less burdensome process than specifying an arbitrary period for updating a PRA. Depending on the changes and PRA use at a given plant, there may be occasions when updating once an outage is not frequent enough or occasions when updating is not needed for a number of outages. At this point in the development of risk technology, such a statement appears to be contrary to many of the industry and RRG concerns about overly prescriptive and arbitrary regulatory requirements.

Section 4.5.2

Although the concept of using CDF to increase STIs for relatively non-important SSCs appears sound, the assumption of unavailability as directly proportional to time x failure rate may be flawed. Some component failure rates, such as those for diesel generators and normally nonenergized solenoid valves, may be more valid when computed per number of demands rather than per time period. Also, the equation used does not address unavailability due to out-of-service time for maintenance and surveillance activities, which might be greater in some cases than that due to failure rate x STI. For example, the equation seems to imply that reducing an STI by one-half would reduce unavailability one-half; there does not seem to be any consideration of a "break-even" point for continuing to reducing the STIs of important SSCs.

Sect[illegible]

See previous comments on Section 4.4.4 and 4.5.2.

EPRI

Electric Power Research Institute — Leadership in Science and Technology

Monday, July 26, 1993

Mr. Frank Gillespie
Regulatory Review Group
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Concerning: Volume Four of the Regulatory Review Group Report

Dear Mr. Gillespie:

EPRI is pleased to comment on Volume Four of the Regulatory Review Group's draft final report. These comments are meant to complement the ones we have already submitted in a letter date May 12, 1993, and are again meant to clarify parts of the report in a way that will be helpful to industry in responding to the NRC initiative in the area of risk-based regulation.

Our additional specific comments follow:

- In Section 4.4 of the report, a suggestion is presented as to how a licensee might approach a graded implementation of Appendix B. As currently visualized, a classification would be created for SSCs which have been found to be relatively important in *any* PRA of a particular class of plants, such as a BWR 1-4. Our feeling is that such an approach, while intellectually appealing, will in practice be difficult, if not impossible, to implement. However the concept of scrutinizing similar plants is an important one that can and should bear on judging whether or not an individual plant's analysis is robust. Our suggestion is to do away with this category as a plant-specific category for graded QA implementation, and use as part of the validation and verification process of a plant's SSC prioritization process a specific comparison step with other plants in the same class. This approach might achieve the NRC goal of assuring completeness without burdening a licensee with including SSCs unimportant in his plant.

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- In several places, the statement is ma:'.. that "In addition, any application should not violate the defense-in-depth philosophy (c.f. page 4-34)." While we agree, it is also true that one has to be careful that defense-in-depth is not unevenly and illogically applied, as it has occasionally in the past.

- In Section 4.5, PRA Application for Configuration Analysis, a clarification would be helpful concerning whether "tradeoffs" between certain actions, for example AOT extensions, would be acceptable and under which circumstances. The report seems to implicitly imply that AOTs could be extended if "the current estimated average level of safety " does not decrease. It would help our understanding if the question of optimizing AOTs so that net overall safety is improved could be explicitly discussed.

In general the Regulatory Review Group's report is a significant step forward toward the utilization of risk-concepts in nuclear plant regulation and the application of probabilistic, deterministic and operational knowledge in improving safety while at the same time increasing operational flexibility. We wish to thank the NRC for the opportunity to comment at this time.

Sincerely yours,

Frank J. Rahn

Frank J. Rahn, Manager
Risk-Based Applications

cc: J.-P. Sursock
B. Chexal

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U. S. Nuclear Regulatory Commission
Attention: Mr. Frank Gillespie
Regulatory Review Group (OWFN, 12D21)
Washington, D.C. 20555

Dear Sir:

Subject: NRC Regulatory Review Group Report - Comments

GPU Nuclear respectfully submits the enclosed comments on the subject report. We believe that this report represents a thorough assessment of regulations, technical specifications, and regulatory processes. The NRC staff is to be commended, and we look forward to working with NRC in future implementation of some of the recommended changes.

If you have any questions or additional clarifications are needed please contact me at (201) 316-7334.

Sincerely,

J. C. Fornicola
Licensing and Regulatory Affairs
Director

JCF/DJD/amc

Enclosure

cc: Mary Drouin - USNRC
NUMARC

ENCLOSURE

VOLUME I

Summary - Page 1 and 2:

1. Certain areas appear not to have been included by the review group that may warrant consideration.

   a) Enforcement Policy, 10 CFR 2 Appendix C
   b) SALP Process
   c) ISI

Recommendations

1. Page 6, 4.1.1:

   Recent NUMARC Petition for Rulemaking on 10 CFR 21 should be recognized as a specific industry recommendation related to a graded QA approach.

2. Page 7, Section 4.1.1, fourth "bullet":

   Regulatory Guide 1.123 should be included in the list of documents since this Regulatory Guide is still committed to by GPUN and other utilities.

3. Page 7, Appendix B, first "bullet":

   Consideration should be given to revise or retitle Appendix B to facilitate restructuring the culture of ANSI standaros, regulatory guides, and practices which have been developed around Appendix B QA requirements.

4. Page 7, second "bullet":

   Recommend that out-dated Regulatory Guides be voiued rather than left in place. They are a source of confusion if left as is.

5. Page 8:

   The four security issues are valid, but should be expanded to include endorsing the NUMARC initiative. Physical security has been emphasized in the past rather than the broader perspective of plant safety. Significant flexibility can be achieved in the security area (reference NUMARC initiative) without any adverse impact on plant safety.

6. Page 9:

   Adding requirements to 10 CFR 50.2 and 50.54 is unnecessary and adds a burden. We agree that commitments need to be reviewed as discussed. Changes to licensing basis commitments should continue to be evaluated and reported to NRC under the existing criteria and requirements of 10 CFR 50.59.

7. Page 10, third "bullet":

   We agree 10 CFR 26 audits should be every 3 years; however, this should apply to both licensee and contractor programs. Vendor audits for safety-related hardware are every 3 years or longer. Fitness for duty requirements need not be more restrictive.

8. Page 10, seventh "bullet":

   Review Group recommendation for IST should be extended to include ISI.

9. Page 11, sixth "bullet":

   The term "performance-based" is used several times. Experience has shown this term is widely-interpreted. Examples would be useful when using or defining this term.

10. Genera.

    NRC should consider revising its method of issuing generic letters. Recommend inviting the industry to propose actions required on an issue versus NRC developing a generic letter and then having the industry potentially disagree on implementation. The NRC would always be able to override industry if deemed appropriate.

<u>VOLUME II</u>

1. Section 2.3.1:

   GPUN strongly supports the NUMARC Petition for Rulemaking, which should be acted upon promptly.

2. Section 2.3.2:

   GPUN agrees with the approach; however, the reporting requirements proposed are an unnecessary additional burden.

3. Section 2.3.5:

   a. The Sample Frequency should be 50% for all personnel. The differentiation between contractor vs. licensee adds burden without commensurate safety improvement.

   b. Audit frequency should be 3 year r licensee and contractor programs.

4. Section 2.3.6:

   NRC needs to explicitly cancel generic communications when they are superseded by regulaticus or other actions. Old, often conflicting generic communications cause confusion for the licensee and NRC inspectors.

5. Section 2.3.9, Page 68:

   The language for the QA Plan changes should be the same as changes to other plans. "Changes can be made without NRC approval if the changes do not reduce the licensee's measure below the requirements of 10 CFR 50 Appendix B." Not to change this section would continue to give regulatory weight to the licensee and NRC excesses of the past that do not contribute to safety.

6. Section 2.3.11:

   "Policy Statements". The NRC should be prompt at updating/eliminating policy statements that are no longer applicable due to new regulations or changes in circumstances.

7. Section 2.3.12, Page 87, 10 CFR 50.59:

   GPUN endorses and utilizes the guidance in NSAC-125. We recommend that NRC similarly endorse NSAC-125 or expeditiously work with NUMARC to resolve concerns. Also, additional effort to develop criteria for defining scope of activities subject to 10 CFR 50.59 would concentrate licensee resources on safety significant evaluations.

8. Section 2.3.13:

   It may be necessary to change 10 CFR 50 Appendix B so that the old culture and mystique surrounding QA can be erased. The NRC has not considered that in their assessment.

9. Section 2.3.15:

   The NRC recommendations do not go far enough. The industry has made extensive comments on NUREG 1022 and the NRC has not acted on those comments. For example, why does a required report to the state environmental agency require another report to the NRC?

10. Section 2.3.18:

   "Security"; reiterate the NUMARC position. Security should focus on plant safety and not have a narrow focus only on Physical Security. The Training and Qualification Plan and Contingency plans can be eliminated. These should be incorporated in licensee procedures, not plans.

VOLUME III

1. NRC policy should be to permit Technical Specification line-item improvements which are consistent with new Standard Technical Specifications (STS). Such line-item improvements need to be independent of full implementation of the STS.

2. Risk assessment methods and IPE results should be endorsed by NRC as appropriate sources of justification to extend Technical Specification allowable outage times and surveillance intervals.

General Comments:

More emphasis should be placed on PRA determinations when attempting to arrive at quality/safety class of systems and components. The report is rather silent on grading quality issues based on their PRA consequence. If endorsed more fully by the NRC, the PRA could be used to reduce/focus QA scope of review and save resources where warranted.

References

1. Weerakkody, S.D., and Phillabaum J., "Use of PC-Based Integrated Core Damage Models for Decision Making at Nuclear Power Plants," Presented at the PSAM Conference, February 1991.

2. Andre, G. R., "Risk Based Technical Specification Program,"EPRI Report TR-101894, January 1993.

3. Samanta P.K., Kim, I.S. (BNL) and Vesely W.E.(SAIC), "Risk Based Configuration Control: Application of PSA in Improving Technical Specifications and Operations Safety," Proceedings of PSA International Meeting, January 1993.

4. Atefi, B. (SAIC) and Wohl M. (USNRC), "A Real Time Risk Based Approach to Technical Specifications," Proceedings pf PSA International Meeting, January 1993.

5. Weerakkody, S.D., "Diversity of PRA Techniques Used in Utilizing PRA Insights in Project Prioritization" (work in progress).

NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

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July 29, 1993

Mr. Frank P. Gillespie
Regulatory Review Group
Office of the Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Gillespie:

On May 18, 1993, the NRC published in the *Federal Register* (58 Fed. Reg. 29012) a notification that a report by the Regulatory Review Group would be available for public comment on May 28, 1993. On behalf of the nuclear industry, the Nuclear Management and Resources Council hereby submits the following comments.[1]

We commend the efforts of the RRG in producing a report that is both comprehensive and insightful in its evaluation of current problems with individual regulations and with the overall regulatory process. We also found many of the proposed resolutions to be innovative.

We endorse the central theme in the report that most of the apparent inflexibility in regulatory requirements does not reside in the regulations, but rather in the implementing practices and associated guidance documents. The over-emphasis on implementing practices often has created an inflexible environment where the methods of compliance have taken on a greater significance than the legal requirements themselves. This environment, in turn, results in unnecessary expenditures of NRC and industry resources without a commensurate safety benefit. Before performance-based regulatory approaches can be developed and successfully implemented, a clear distinction between formal regulatory requirements and informal regulatory guidance must be established, both in principle and in practice.

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[1] NUMARC is the organization of the nuclear power industry that is responsible for coordinating the combined efforts of all utilities licensed by the NRC to construct or operate nuclear power plants, and of other nuclear industry organizations, in all matters involving generic regulatory policy issues and on the regulatory aspects of generic operational and technical issues affecting the nuclear power industry. Every utility responsible for constructing or operating a nuclear power plant in the United States is a member of NUMARC. In addition, NUMARC's members include major architect/engineering firms and all of the major nuclear steam supply vendors.

We also endorse the implicit finding of the report that, all too often, informal regulatory mechanisms are employed as substitutes for formal regulatory requirements. Any actions on the part of licensees that the NRC considers necessary to establish an adequate level of protection of public health and safety should be imposed by formal regulatory requirements (rules, orders and license requirements). Conversely, burdens should not be imposed on licensees that are not necessary to ensure an adequate level of protection or where the costs, both direct and indirect, of a proposed NRC action do not provide a commensurate safety benefit, thereby distracting licensees from more safety-significant actions.

Due to the length of the RRG Report, we have organized our comments as follows:

- **Volume One: Summary and Overview**

  Our specific comments on the recommendations contained in this volume are included in our comments on the remaining volumes.

- **Volume Two: Regulations**

  Attachment 1 to this letter provides our comments on the nineteen position papers in this volume. Attachment 2 provides our comments on the review forms for selected regulations.

  The NRC's Elimination of Requirements Marginal to Safety Program, the industry's December 21, 1992, letter to the Commission and the Draft Regulatory Review Group Report all identify potential improvements to regulations. The industry is in the process of prioritizing the recommendations in these three reports and will communicate the results to the NRC upon completion.

- **<u>Volume Three</u>: Operating Licenses**

  On the basis of the results of the four plants reviewed in this volume, we endorse the recommendation that each licensee should review their license to identify instances of overly prescriptive or unnecessary requirements. Indeed, a number of plants have already embarked on such reviews. The industry and the NRC should work collectively to develop an approach to process identified license changes in a manner that is efficient for both the NRC and the industry. We believe the NRC has an obligation not to impose burdensome requirements that do not contribute to safety. Where this has occurred in the past, the NRC is equally obligated to establish a management system that provides the necessary resources to remedy this condition.

- **<u>Volume Four</u>: Risk Technology Application**

  NUMARC, through the Regulatory Threshold Working Group, will provide the industry's focal point for interaction with NRC management on risk technology applications. The Working Group will also coordinate industry pilot projects that demonstrate the viability of generic applications of PSA technology in the regulatory process. Currently, the Working Group is identifying those applications that are of most importance to the industry. Volume Four of the RRG report will be a primary source of information for this effort. Significant industry resources have already been expended in developing a blended (risk-based/deterministic) method for prioritization and categorization of motor operated valves (MOVs). This work will provide the basis for a graded approach for ensuring the performance of MOVs in response to Generic Letter 89-10. We expect to discuss this work with the NRC staff in August.

  Attachment 3 to this letter provides our detailed comments on this volume.

While we believe the contents of this report are of high value, we recognize that similar studies in the past have not resulted in substantive changes. We urge the NRC to establish a formal steering committee to direct the implementation of the recommendations in this report. In doing so, particular attention should be given to ensuring the efforts of the Office of Research are properly coordinated with those of the Office of Nuclear Reactor Regulation. The nuclear industry is committed to provide the necessary support to effect the changes needed to ensure that its resources focus on regulatory matters of safety significance.

Please contact me with any questions you have concerning these comments or if you need additional information.

Sincerely,

Stephen D. Floyd

Stephen D. Floyd
Manager - Special Projects

SDF/sdf
Attachments

Attachment 1

NUMARC COMMENTS ON VOLUME TWO OF THE REGULATORY REVIEW GROUP REPORT

2.3.1: COMMERCIAL GRADE PROCUREMENT AND 10 CFR PART 21

The Regulatory Review Group Report recognizes that significant improvements are necessary to ensure that Part 21 and its approach to dedication of commercial grade items is effective and capable of proper implementation. The report correctly identified the bases of these problems as the changed procurement needs of nuclear utilities (now primarily involving replacement parts for existing equipment rather than the purchase of major new pieces of equipment), the difficulty in meeting the criteria contained in the definition of commercial grade items (10 CFR 21.3 (a)(4)), and the lack of a flexible generic process for dedication of commercial grade items to ensure that they will perform their intended safety function.

Over the past several years, procuring replacement parts has become a difficult problem because many of the original suppliers and manufacturers are no longer in business or no longer maintain Appendix B qualified programs. This means that, as a practical matter, it is more difficult to obtain parts from Appendix B-qualified vendors, the costs of such parts are higher (often dramatically), and delays that frequently result from the current process could have an adverse effect on plant safety.

The nuclear industry shares the NRC's concerns and, on June 22, 1993, submitted a petition for rulemaking seeking NRC action to improve those provisions of 10 CFR Part 21 related to commercial grade items and their dedication for safety-related applications. The industry's petition for rulemaking requests that the NRC amend 10 CFR 21.3 and 21.7 to effect the following changes: (1) replace the existing definition of commercial grade item with a practical definition, (2) allow for a flexible, generic process for dedication of commercial grade items and (3) clarify that the entity performing the dedication of a commercial grade item is responsible for evaluating and reporting deficiencies pursuant to Part 21 reporting requirements.

The Report recommends that the definitions of commercial grade item and dedication be broadened. While the industry seeks to achieve the same result as is intended through the Report's recommended substitution of "or" for "and" in the definition of commercial grade item, we would urge the Commission to adopt the modification suggested in the petition for rulemaking because it is a more straightforward standard, and therefore, easier to use. Further, the petition's suggested approach to the dedication process clearly meets the Regulatory Review Group's intent to define dedication as the action necessary to provide reasonable assurance that a commercial grade item will perform in a safety-related application.

With respect to changes to the definition, the industry has determined, and the NRC's Regulatory Review Group apparently agrees, that the current definition of commercial grade items is so limiting as to essentially prevent its effective use. The industry has proposed to expand the commercial grade item definition to include any item obtained on the commercial market, without regard to whether or not the supplier maintains an Appendix B program. Overall, this change makes much more likely a reasonable price and delivery time which, in the long run, will enhance safe plant operation.

As noted above, the petition also requests that the NRC codify a flexible approach to dedication. The NRC has, in essence, agreed to such an approach through the agency's endorsement[1] of the EPRI *Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Applications.*[2] Dedication is no longer appropriately viewed as the point in time when a commercial grade item becomes a basic component, but rather has come to represent the process for determining that there is reasonable assurance that a commercial grade item will perform its intended safety-related function. In particular, this process should entail more than simply establishing or following a paper trail for a given item. The possible methods for evaluating whether the reasonable assurance standard is met include (1) testing and/or inspection; (2) surveying the commercial grade supplier to determine that appropriate quality controls are in place; (3) observing the manufacturing process to ensure that the item will have the necessary attributes, and (4) analyzing the historical record of the item for acceptable performance. To address the NRC's interest in being able to review the dedication methods and processes, the industry's petition proposes that documentation be maintained by the dedicating entity for the purpose of an NRC audit or inspection. As was noted in the petition, the dedication process contemplated by the petition imposes upon the user of the commercial grade item the responsibility for dedicating that item. This is appropriate, because the user understands the safety significance of the proposed component and is best able to identify the characteristics necessary for it to perform its intended function.

Although the Regulatory Review Group did not address reporting requirements directly, it is important to clarify that the responsibility for reporting deficiencies in commercial grade items rests with the entity performing the dedication process. This is appropriate for a number of reasons, including that suppliers and sub-tier suppliers do not necessarily know whether a commercial grade item is destined for a safety-related application, and that there is no time limitation upon Part 21 reporting responsibility for suppliers.

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[1] See Generic Letter 89-02.

[2] The NRC conditioned its endorsement of the guidelines upon implementation by licensees with a few minor modifications in acceptance methods involving the supplier survey and supplier/item performance history. Thereafter, the NRC issued 92-05, which the industry believes went beyond the agency's previous position, to one that significantly exceeded a standard to provide reasonable assurance that commercial grade items that are dedicated will be suitable to perform their intended safety-related function.

In conclusion, the changes to and clarification of 10 CFR Part 21 suggested in the industry's petition for rulemaking will improve the efficiency and effectiveness of obtaining replacement parts for use in safety-related applications and make more effective the process to ensure their proper functioning. The industry's petition closely tracks the Report's recommendations both in concept and in some of the detail. We believe prompt and favorable NRC rulemaking action should now result.

2.3.2: COMMITMENTS

As the Regulatory Review Group properly concluded, the term "commitment" is not defined in NRC regulations and no guidance has been issued by the NRC on how a commitment should be treated in the regulatory process. It is clear what commitments are not: they are not regulatory requirements, which are legally binding requirements established by a statute, regulation, license condition, technical specification, or order. The most direct explanation of the NRC's expectations is found in 10 CFR Part 2 Appendix C which states that "NRC expects licensees and vendors to adhere to any obligations and commitments resulting from these processes [*i.e.*, administrative mechanisms such as Bulletins, Generic Letters, Information Notices, Notices of Deviation, Notices of Nonconformance] and will not hesitate to issue appropriate orders to licensees to make sure that such commitments are met."

The Regulatory Review Group Report stated that the regulations make a clear distinction between a commitment and the plans that are implemented to satisfy the commitment, but no support was provided for that conclusion. We believe there is real significance in treating commitment implementation differently from the commitment itself. As long as a licensee meets the commitment as originally discussed with the staff, the process should be immaterial and should be allowed to be changed at the licensee's discretion. Experience suggests that no such distinction is made in practice. We agree with the Regulatory Review Group that this distinction should be clarified. As the Regulatory Review Group correctly concluded, it is not clear how licensee commitments and changes to those commitments could best be accomplished and that a common understanding among the NRC and licensees would be very useful.

The Regulatory Review Group Report recommended that the NRC commence a rulemaking activity (1) to modify 10 CFR 50.2 to establish a definition for commitment, and (2) to add a new section to 10 CFR 50.54 to formalize the process by which changes to commitments may be made. The industry is currently in the process of assessing the potential impact of this approach. We believe this is an important area that requires a thorough evaluation before steps as formal as rulemaking are pursued. We are working to develop an industry position on this subject and will communicate our position to the NRC in the near future.

2.3.3: DECOMMISSIONING

The Regulatory Review Group report recognizes certain changes are needed to regulatory documentation associated with Decommissioning. The report should also include important updates needed to accommodate the possession only license. The following specific comments are offered:

(a) The report discusses the need for revision and reissuance of Regulatory Guide 1.86 to reflect the current NRC organization and address areas of inconsistency with 10 CFR 50.82. In addition, this guide should be updated to include the definition of a possession only license and premature shutdown related issues. The guide also contains site cleanup criteria that are currently used as guidance, but are being reviewed by NRC as part of a rulemaking to codify cleanup criteria.

(b) A discussion in Section 2 addresses certain difficulties with the current version of Regulatory Guide 1.86 including, for example, frequencies of required surveillances. The discussion should be expanded to include certain other important aspects of the decommissioning process, such as, emergency planning, training, maintenance, fitness for duty, insurance, annual fees, Appendix R, and environmental qualification.

(c) Recommendations included in item IV -- We believe that the issues identified in NUMARC 92-02 should also be included in this section to correct inconsistencies.

2.3.4: FIRE PROTECTION

The body of regulations and associated guidance documents related to Fire Protection should be improved to incorporate performance-based and risk-based methods as part of the NRC's Marginal to Safety Program. We have developed improvements to these regulations and will discuss them with the cognizant NRC staff in the near future prior to the industry filing a Petition for Rulemaking.

2.3.5: FITNESS FOR DUTY

We generally agree with the Regulatory Review Group's comments on Fitness-for-Duty (FFD). The following specific comments are offered:

(a) The access authorization rule (§ 73.56(g)) requires contractor/vendor programs to be audited twice as often as licensee programs. We believe that the period between audits should nominally be 3 years, and it should be the same for all groups. The periodicity should be based on performance. If there are significant shortcomings in program performance, the audit frequency could be increased to ensure that appropriate corrective action has been taken. If the contractor/vendor program meets all requirements, the next audit could be scheduled up to 3 years in the future. We suggest establishment of a set of criteria to be used to determine when, and if, audit intervals for contractor/vendor programs should be reduced. While this would need to be designed to ensure sufficient licensee attention, it should reward satisfactory contractor/vendor performance.

(b) The Regulatory Review Group report refers to several SECY documents, implying that other issues related to the FFD rule are already being addressed. We are unable to determine which other FFD issues are in this category because one of those documents, SECY-92-308, has not been made public. All activity associated with publishing the proposed changes in this SECY are apparently being held in abeyance while resolving backfit issues. Three NUMARC letters to the NRC (April 17, 1991, July 24, 1992, and December 21, 1992) provided comments on several fitness-for-duty issues. The changes identified in those letters would provide significant financial benefits to the nuclear power industry without diminishing safety. If implemented, our suggestions would relieve many of the inefficiencies experienced by the industry with the FFD rule during the three years since implementation. These issues need to be addressed in the context of providing flexibility for licensees in meeting performance-based criteria.

(c) We encourage the Regulatory Review Group to emphasize the recommendation made on the review form for 10 CFR 73.56: *"Continue efforts to review fitness-for-duty and access authorization programs and entertain requests to provide relief in areas where program results indicate such can be done without affecting safety."*

2.3.6: GENERIC COMMUNICATIONS

Generic communications (NRC Bulletins, Generic Letters, and Information Notices) are intended to assist the NRC in gathering information on emerging issues to determine if formal regulatory action is warranted. In principle, the NRC's generic communication system satisfies an important need: to communicate promptly with licensees on issues of potential significance or to advise licensees of important events or issues so that they can determine what action would be appropriate based upon lessons learned from other licensees or NRC experience.

While in principle generic communications are an appropriate mechanism to disseminate information to and request information from licensees, in practice the system does not effectively achieve those goals. The industry agrees with the Regulatory Review Group's assessment of the three major weaknesses in the NRC's generic communication process: (1) often there is no clear tie between the requested actions and a regulatory requirement; (2) the industry rarely has an opportunity to participate in the technical resolution of an issue until after issuance of a Generic Letter; and (3) the staff inadequately considers the cost of the request in the Generic Letter. An elaboration of our view of these weaknesses and others follows.

First, responding to generic communications often is very costly in actual dollar amounts as well as time and personnel resource allocation, but a commensurate safety or efficiency benefit is rarely achieved. The issuance of any generic communication, including an Information Notice, carries an implicit and, sometimes, explicit NRC expectation that licensees will review and analyze the subject matter even if action is not required. This imposes a significant burden on licensees. It requires a continual reallocation of resources and thereby diverts resources from those matters which objectively deserve a higher priority. Second, the current process frequently is used to accomplish informally that which the NRC staff cannot or chooses not to accomplish formally through a rulemaking or an order. Generic communications are frequently used to solicit utility commitments to resolve emerging generic issues of particular concern to members of NRC staff or to impose regulatory "requirements" without using the formal process required by the Administrative Procedure Act. Third, generic communications often are not founded on regulatory requirements. As noted, they may be issued merely if the subject at issue is a matter of concern to one or more members of the NRC staff. Thus, the cumulative burden imposed upon licensees through generic communications is significant.

The Commission should also consider another area of weakness in the generic communications process -- overall NRC management. It is not clear what role, if any, NRC senior management plays in controlling NRC staff involved in the determination to prepare and issue generic communications. To illustrate the point, the monthly list of proposed generic communications, dated July 1, 1993, identified 84 generic communications as currently "in progress." A number of the proposed generic communications have been pending for a lengthy period of time, in some cases even years. There is no indication that each issue is accorded an appropriate priority, and that criteria have been established or are used to allocate staff resources to address issues through generic communications. Described more broadly, neither the process nor the resulting generic communications appear to be receiving appropriate NRC management oversight. In fact, the NRC no longer even includes a scheduled completion date on the monthly list referred to above, seemingly signifying the NRC's acknowledgment of its unwillingness to manage the generic communication system.

The Regulatory Review Group identified 219 Generic Letters and supplements to Generic Letters that were issued in the period 1983 - 1993. By the Group's analysis, only 66 included requests for actions to be taken, programs to be developed, or schedules to be provided for completed actions. On average, however, that means that every two months a licensee was required to reallocate resources to address a new issue and, potentially, take substantive action (*e.g.* a modification of plant hardware or a procedure) to respond to the NRC's requests.

Of the remaining 153 Generic Letters identified for the 1983-1993 period, some were administrative in nature and required no action by licensees. The majority, however, (roughly one per month) provided substantive information which each licensee was expected to analyze to determine the significance and applicability to that licensee. As such, even "information only" communications impose a significant burden on licensees. We therefore disagree with the Regulatory Review Group's conclusion that 10 CFR 50.54(f) information requests "appear to be applied in an appropriate manner." That process is still subject to misuse. The burden imposed merely to collect information for the NRC's further analysis is a costly one because it is often resource intensive and time consuming.

We believe that the Commission's Staff Requirements Memorandum dated December 20, 1991, regarding SECY-91-172, "Regulatory Impact Survey Report -- Final" was a very positive step toward ensuring better management of the generic communications process. First, for those generic communications in which a new NRC staff position is articulated or through which the staff seeks additional licensee commitments, the Commission will be apprised of such communications prior to their issuance unless it is an urgent safety matter. The Commission now will be better able to assess both the burden imposed by a particular generic communication and the cumulative effect of that burden in relation to others already imposed upon licensees. In addition, the Commission will now be in a better position to determine if the priority proposed for that communication by the staff is appropriate. Second, the staff will now solicit the views of interested groups on generic communications prior to issuance. This should assist the staff in both formulating its position and describing exactly what it seeks though its request for information. Finally, the staff is now required to provide a rationale for addressing the issue through a generic communication rather than through rulemaking or individual orders. These additional steps should significantly reduce the use of generic communications as a means to informally establish requirements and should encourage the NRC to adhere to the provisions of the Administrative Procedure Act when seeking to impose regulatory requirements.

Because the principles upon which the generic communication process are sound, we do not advocate any change to the principles. However, we strongly advocate the imposition of management discipline over this process. The problems identified above can all be resolved simply and effectively through the imposition of internal NRC management controls.

We support the recommendation to develop a coherent system for generic communication (generic letters and bulletins) that clearly differentiates between those communications that require action by licensees, whether it be a request for information or more specific hardware related action, and those that disseminate information, e.g. information notices. We believe the use of an administrative letter as described in NRC Administrative Letter 93-01 to replace generic letters that only provide information or are administrative in nature is appropriate.

2.3.7: INSERVICE TESTING (IST)

We agree with the Regulatory Review Group's conclusions that the industry and the staff should continue to build a consensus position in the Code committees, particularly to address incorporation of risk techniques to address test frequency (as well as addressing selection of risk-significant components to which tests should be applied) and to focus on performance-based testing. We would urge caution, however, both by the code committees and by the staff, in expanding the scope of testing into design basis conditions.

The primary purpose of inservice testing is to assess operational readiness of active components and detect degradation from the as-installed condition so that corrective actions can be taken. Assurance that a pump or valve is initially capable of performing its design functions is provided by startup testing and by the controls applied during design, manufacture and installation, and in some cases, by qualification testing of the class of components. A recent study by EPRI - Plant Support Engineering considered safety benefits of additional design basis testing as proposed by the NRC in September 1991. This study, which was reported to the ASME Operations and Maintenance (O&M) Committee in March 1993, concluded that the proposed changes to the code testing requirements would result in minimal safety benefits. We recommend that any proposed changes to inservice testing requirements to expand the scope into design basis testing be carefully evaluated in terms of safety and cost benefit, and that only changes with significant safety benefit be considered.

We also encourage the staff to implement the Regulatory Review Group recommendations to continue work on the IST program guidelines to allow licensees to take advantage of generic approval to use the most recent addenda and editions. The NRC should consider issuing the draft NUREG containing this guidance for comment. Additionally, the NUREG should not be finalized until the proposed regulatory changes are finalized. This will assure that the NUREG and the regulations are consistent. The referenced Inspection Procedure should also be revised as necessary and made available for public comment to assure consistency.

2.3.8: LICENSEES PERFORMING THEIR OWN SAFETY ANALYSES

We concur with the recommendations of the Regulatory Review Group to allow a general reference to the basic topical report in the technical specifications that would obviate the need for a purely administrative license amendment as later revisions to topical reports are made.

2.3.9: ACTIVITIES PLACED UNDER LICENSEE CONTROL

This section discusses changes to various regulated activities that are under licensee control. These changes would make consistent the treatment of updating changes to programs such as the QA Plan, the Security Plan, Safeguards Contingency Plan, Guard Training and Qualification Plan, the Emergency Plan, and the Fire Protection Plan.

We agree with the discussion that there should be coherence in the reporting and updating periodicity, as well as a clear statement as to what actually constitutes a change that does not decrease the effectiveness.

2.3.10: MISCELLANEOUS FINDINGS AND RECOMMENDATIONS

Section 2.3.10(V) - Revised and Additional Definitions

We do not believe development of a new definition of the term, "current licensing basis," for application in 10 CFR 50 warrants industry and regulatory attention and resources in that it will not result in significant regulatory burden reduction. As for the term, "design bases," we believe the 10 CFR 50.2 definition is quite clear, and that the scope and depth of the term is adequately discussed in NUMARC 90-12, "Design Bases Program Guidelines," which has been accepted by the NRC staff and the Commission. Changing the scope and depth of this term would almost certainly result in a greater burden to licensees who have used NUMARC 90-12 in tailoring their design bases programs. Thus, we believe this recommendation should be dropped in its entirety from the report.

Section 2.3.10(VI) - Control Of Material Removed From Technical Specifications

(a) The staff SECY on Technical Specification Improvement Program, although not specifically endorsing NSAC-125, states that current licensee programs to control changes are strong enough to support the relocation of requirements inherent in TSIP conversion. The SECY statement should be included in the discussion.

(b) The report makes some generalizations about FSAR content, updates and control that may not be universally true. For example, it states that since most licensees only update FSARs based on what was originally in the FSAR, the current UFSAR does not describe in sufficient detail the facility or procedures, if at all. Therefore, 10 CFR 50.59 would not act as a control on the licensee's ability to change items not specifically addressed in the FSAR without reporting the changes to the NRC. The report goes on to state that any changes to the Technical Specifications that relocate material to the FSAR that does not describe the facility or procedures would also not be controlled by the 10 CFR 50.59 change process.

2.3.11: POLICY STATEMENTS

We disagree with the recommendation to delete the Severe Accident Policy Statement until the issue of rulemaking for advanced light water reactors is resolved. While Generic Letter 88-20 adequately addresses existing plant severe accident challenges and insights, the staff is considering a severe accident rule for the

Advanced Light Water Reactors (ALWRs). We oppose generic severe accident rulemaking for ALWRs and agree with SECY 92-070 that "severe accidents should be used to complement traditional design basis accidents to enhance the level of safety at nuclear power plants, rather than making severe accidents design basis accidents."

2.3.12: CURRENT PROGRAMS

Under the discussion of 10CFR50.59, we believe that NSAC -.125 should receive full endorsement by the NRC, rather than the limited endorsement referred to in the report, once NUMARC and the NRC resolve pending issues under discussion.

2.3.13: QUALITY ASSURANCE AND PROCUREMENT AT OPERATING NUCLEAR POWER PLANTS

The NUMARC Appendix B Working Group has reviewed the relevant segments of the Regulatory Review Group Report associated with quality assurance.

NUMARC, through its Appendix B Working Group, agrees that in the area of quality assurance there is significant potential for improving the efficiency and effectiveness of implementing the NRC's regulations. Interpretation of the regulatory requirements and implementation practices are the major areas for further assessment and potential improvements.

The industry concurs with the statements in the Summary that the recommendations define a good starting point for further discussions and assessments into the issues associated with the implementation of the quality assurance regulations, 10 CFR Part 50, Appendix B. Considerable work is required to define the criteria and refine the implementation guidance to assure such concepts are well understood and can be incorporated into the working environment in the most practical manner. A comprehensive review of the language in the rule and the resultant implementation practices derived from the numerous industry standards and regulatory guides that attempt to interpret the regulation is also essential, if the overall objectives of a more effective and efficient regulatory regime are to be realized. These reviews and activities would build on the initial recommendations in this report using the experiences gained in implementing these regulations over 20+ years.

The industry agrees that the theme and content of the too numerous guidance documents (industry and regulatory) has compounded what was originally intended to be a straightforward regulatory requirement; that every facility should have a quality assurance program to provide control over activities affecting the quality of identified structures, systems, and components to an extent consistent with their importance to safety. In addition, we maintain that the continual embellishments added and demanded by individual auditors and inspectors (industry and regulatory) have resulted in an engrained but inappropriate set of implementation practices, concepts and cultures. These embellished practices do not add to, or assure, a more defined, acceptable or sustainable level of quality. Rather, the implementation practices tend to detract from the fundamental intent of the regulation by over emphasizing documentation and processes. New guidance and, if necessary, amendments in the regulations are required to refocus the emphasis of the quality assurance regulations on performance and product/activity quality attributes rather than pure documentation and administrative processes.

The industry believes that it is premature to conclude that a rule change is not required. The rule, drafted when there were numerous plants under construction, was focused on design and construction elements. In the past 20 years, there has been a shift in emphasis towards operations and decommissioning while the number of suppliers willing to commit to a 10 CFR Appendix B program, with all its legal and financial implications, has significantly decreased. Over the years, regulatory interpretations have manipulated the language in the rule, so that the rule has become a "forced fit" to operational QA implementation practices. After 20+ years of working with the current language and with th[illegible] implementation emphasis, it is appropriate to undertake a complete review of [illegible] language in the regulations as well as the associated implementation practices to determine if changes are warranted. We believe that such a review, if carried through to its logical conclusion, could result in an amendment to the rule, combining all quality assurance requirements into one rule. A natural outflow from such an exercise would be a new set of practical and more effective guidance documents that would build on the experiences of the past.

Several companies are already evaluating and adopting some of the performance based concepts referenced in the report. In addition, as indicated through the recent industry - NRC interactions on the implementation of the maintenance rule, there is a general movement towards a more practical, performance based regulatory implementation regime. Quality assurance is another area that already appears to be eminently suitable for such treatment. NUMARC, through its Appendix B Working Group, will interact with the NRC staff to assess, refine and, where justified, develop performance based quality assurance implementation criteria, practices and strategies to implement NRC requirements in a more efficient and cost effective manner.

2.3.14: QUALITY ASSURANCE PROGRAMS FOR FIRE PREVENTION

We have no additional comments to those provided for section 2.3.13.

2.3.15: REGULATORY GUIDES

We do not concur with the recommendation that resources not be applied to revising or updating these guidance documents. Regulatory guides are used during the inspection process as defacto requirements. In many cases, licensees commit to these documents and, upon doing so, such commitments become part of the licensing basis for the plant. Similar to generic communications, we believe a coherent system should be established that defines the NRC practice and process for development and issuance of regulatory guides. We believe the regulatory guides should be reviewed for consistency with regulations, updated to incorporate more current NRC management and staff positions, and be flexible enough to allow alternative methods that may ultimately be acceptable to the staff after further review.

2.3.16: EVALUATION OF REPORTING REQUIREMENTS

NUMARC agrees with the six recommendations and believes that their implementation should be consistent with the implementation of the conclusions from the NRC task force for reporting requirements.

However, we believe the following additional routine reports, that are not discussed; should also be included: 10 CFR 50.54(p)(2), 10 CFR 50.54(q), 10 CFR 140.15(A), 10 CFR 140.15(b)(1), 10 CFR 140.17(b), 10 CFR 140.21, and 10 CFR 50.71(e). We agree with the recommendation that for reports that are determined to be still appropriate, their content should be reviewed to ensure all the individual reporting items are appropriate.

It is also recommended that this review be applied to situational reports. There are many situational and periodic reports that were discussed in our letter to the NRC on reporting requirements that were either not discussed, or the staff felt were still appropriate reports. For example, the annual report of insurance and financial security required by 10 CFR 50.54(w)(3) was identified by the RRG as an appropriate reporting requirement. This information is always available upon request, and serves no on-going purpose other than providing information to the staff. If the staff is interested in a particular utility's financial or insurance data, they can request it. In addition, 10 CFR Part 140 also contains three annual reports concerning financial and insurance information that are required to be submitted to the staff. Besides being redundant to 10 CFR 50.54(w)(3), it is also information that can be obtained by request if required.

2.3.17: RULEMAKING PROCESS

The NRC's current rulemaking process unnecessarily consumes time and resources for which the agency, the industry and the public all pay a price. In addition to the actual costs associated with the rulemaking proceeding, an even more significant price is paid in the form of regulations on safety-related issues that should be promulgated to protect the public health and safety not being acted upon quickly enough, and regulations that are identified that unnecessarily burden the licensee and divert attention from more important safety concerns being allowed to remain in place for years. The NRC can address the problems created by these circumstances through the adoption of a more efficient rulemaking process. A streamlined process will enhance overall safety by allowing regulatory improvements to be implemented expeditiously, while satisfying all applicable laws and regulations and continuing to provide for meaningful public participation.

Stating the problem succinctly, the NRC's rulemaking process simply takes too long. As a matter of fundamental logic, there is no reason why, for example, a time limit of one year should be considered "timely" for a decision as to whether a petition for rulemaking should be denied or a rulemaking activity should be commenced on that subject. And, past experience demonstrates that in actuality the average time required has been closer to two years. Similarly, to set a goal for the bureaucratic process that a final rule should be issued within two years of the decision to commence a rulemaking activity sets far too low a goal. If an issue has been identified that the NRC concludes should be the subject of a rulemaking, regardless of whether identified through a petition for rulemaking or by NRC staff, it is difficult to justify how the public interest is served with a process that takes three, or too often four or five years to complete.

The Review Group has analyzed the current rulemaking process including a summary of the pertinent NRC staff Internal Guidance (IG), the Regulations Handbook (NUREG/BR-0053, Revision 2) and the EDO Procedures Manual (NUREG/BR-0072). Unfortunately, the Review Group's analysis starts and ends with a premise that the current process, as established in those internal staff manuals, is an appropriate process. The result is that the Review Group's recommendations and identified areas for potential improvements will not result in significant changes consistent with the goals of the federal government's *National Performance Review* or even come close to achieving the benefit to the public interest that would flow from a more efficient and effective rulemaking process. As Chairman Selin stated in his speech at the NRC's April 1993 workshop, *Elimination of Requirements Marginal to Safety:* "the more we bleed our resources into inefficiency and non-productive ends, the less resources are available for safety." Just as it is in the public interest for issues of potential safety significance to be promptly resolved, so it is in the public interest for regulatory amendments that could achieve significant cost reductions to the industry and, ultimately, the consumer, at no diminution of safety, to be promptly processed.

The industry is in the process of developing a proposal to streamline NRC rulemaking procedures. The proposal will provide detailed suggestions on how this process could be improved. Our recommendations will be provided to the Commission in the near future. While we believe our recommendations have significant merit, there is no substitute for the NRC performing its own in-depth review of its rulemaking process in order to achieve a revision and reformation of the process. In light of the gravity of this issue, we recommend such a wholesale change rather than beginning and ending, as the Regulatory Review Group's analysis did, with the basic premise that the current process is the most effective way to conduct NRC rulemakings. The NRC staff is in a better position than the industry to pinpoint all of the inefficiencies and bottlenecks in the current process -- the rulemaking process has evolved to its current status over the last twenty-plus years -- but there is no indication that the staff has ever taken stock and objectively analyzed whether its current process is the best way to accomplish its goal. The current economic climate and realities demand that both private industry and the federal government must find better ways to accomplish their responsibilities. We encourage the NRC to refrain from defending the status quo and really move to establish a goal of greater excellence. Streamlining the NRC's rulemaking process is an opportune place to start.

2.3.18: REVIEW OF REGULATIONS AFFECTING SECURITY

The Regulatory Review Group comments and recommendations provide some refreshing insights to security regulation. We encourage the RRG to go further. In addition to the following comments on specific recommendations in the RRG report, we suggest that this opportunity be utilized to widen the scope of the security regulations review.

In the introduction to the discussion of security, the report notes that the general performance objective is delineated in 10 CFR 73.55(a). Specifically, *"The physical protection system is designed to protect against the design basis threat of radiological sabotage. Section 73.55(a) also states that the onsite physical security system and organization must include, but not necessarily be limited to, the capabilities to meet the specific requirements contained in paragraphs 73.55(b) through (h). Paragraphs 73.55 (b) through (h) are considered performance-based."* In Section III we note the statement: *"The prescriptiveness of some of the rules related to security at power plants is in striking contrast to the rules in many other* *areas."* The industry finds security regulations to be prescriptive and conducive to regulation by inspection. We support the last conclusion in Section III that, *"...if the anticipated reassessment were to lead to the re-writing of all or part of the security rules, this would afford the opportunity to recast the prescriptive sections of the rules in a more performance-based approach."*

The RRG report notes that several initiatives affecting security were described in SECY-92-272, and that these are being re-evaluated by the staff. Enclosure 1 to SECY-92-272 is the NUMARC Alternative Protection Strategy (APS) provided by NUMARC letter dated June 24, 1992. In the APS document and again in our December 21, 1992, letter, we recommended the elimination of several requirements that we consider provide no benefit to safety. Specifically, the following regulatory requirements should be removed:

- vital area door locks and alarms
- guard at containment entrance
- escorting cleared vehicle even with a driver granted unescorted access
- re-searching on-duty, armed security guards

We believe these subjects should not be discounted by the RRG because they are being re-evaluated by the staff. They seem to be held in abeyance at the NRC while the design basis threat (DBT) re-evaluation is in progress. These issues are germane to this review and should be included as part of the process.

On June 24, 1993, the Commission was briefed by the staff with a recommendation to modify the DBT (SECY-93-166) *"...to include a land vehicle for the transport of personnel, hand-carried equipment, and/or explosives,"* and to make *"...appropriate modifications to 10 CFR 73.55 to reflect the change..."* The Regulatory Review Forms for 10 CFR 73.55 and 10 CFR 50.13 are germane. The former contains the recommendation: *"NRC requirements for security should be reviewed. If the present delineation of them is not sufficient, the Commission's expectations should be articulated clearly. Licensees should then have the freedom to modify their commitments at will as long as they met the NRC articulation of expectations in the security area."* The review form for § 50.13 shows that the Office of General Counsel (OGC) was asked to determine if § 50.13 conflicts with the requirements of Part 73. Currently, licensees are required by Part 73 to protect against paramilitary sabotage threats. But, § 50.13 states: *"An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is not required to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person,..."*

It is not clear how OGC came to the conclusion that *"...there is no conflict."* We believe this needs to be further explained because it is relevant to the current Phase 2 of the DBT review described in the March 11, 1993, memorandum from the Executive Director for Operations (EDO) to the Commission.

Our interest is in reducing unnecessary security requirements that have minimal value in providing for public health and safety. Our APS document explained previous industry thoughts on insider measures. With the advent of requirements for barriers to protect against vehicular intrusion and the establishment of adequate standoff for a design basis explosion, the total plant security posture needs review. If barriers designed to deny unauthorized vehicular entry into the protected area perimeter were required, the access capabilities of a potential paramilitary force are even more limited. If the NRC has a basis for requiring preparation for a paramilitary threat, it is not clear when 10 CFR 50.13 would apply.

Fifteen years of experience should be factored into this review of plant security posture. While a threat composed of a paramilitary force could be justified as the design basis in past years, it may no longer be appropriate in the 1990's, We understand the Commission's desire to set a DBT above the perceived actual threat to provide some margin. Still, this DBT should be based on realism. It is important that security measures at the plant be limited to those that add value in protecting public health and safety. This consideration should be an essential ingredient in the current DBT rulemaking process. Although one recommendation of the RRG is to *"Review existing security requirements (particularly Appendix B to Part 73) to determine if they should be expressed in a more performance-based manner,"* the staff may treat these ongoing security issues separate from the DBT issues and thus avoid a comprehensive review of security regulations.

We recommend that the RRG reword its recommendations to advocate a comprehensive and integrated review of security regulation to make it more performance based and realistic in order to allow the most efficient method of protecting the health and safety of the public.

2.3.19: USE OF RISK TECHNOLOGY ASSESSMENT IN UNREVIEWED SAFETY QUESTION DETERMINATIONS

The increasing use of Probabilistic Safety Assessment (PSA) in the plant decision making process, including its role in 50.59 unreviewed safety question determinations, presents new issues or questions relative to the proper perspective in utilizing risk insights. These issues include:

- Implementing severe accident management guidance.
- Integral analysis of the net effect of a number of changes.
- Status and treatment of PSA cutsets/sequences relative to "design basis accidents" and "analyses described in the FSAR."
- Other regulatory requirements (50.54x, 10 CFR 50 App E) that also address controls on the change process.
- Configuration control and status of PSA-driven plant changes relative to the current licensing basis.
- Level of risk change considered to be negligible.
- Treatment of uncertainties.

A concerted dialogue between industry and NRC staff will be needed on this aspect, along with others, as we define a mutually agreeable understanding as to the proper role of PSA in regulatory applications. We should not prejudge whether the necessary clarifications require a rule change or merely adjustments in regulatory guidance, such as NSAC - 125.

Attachment 2

NUMARC COMMENTS ON VOLUME TWO, APPENDIX A OF REGULATORY REVIEW GROUP REPORT

10 CFR Part 20

The RRG review did not include 10 CFR Part 20. This regulation has significant impact on plant O&M costs directly and indirectly. The recent revision of Part 20 appears to provide enhanced flexibility in implementation. However, the rule may impose unnecessary burdens marginal to safety on the nuclear power industry because the rule has general applicability to all NRC licensees. Hence, the rule addresses many types of licensed activities that may be less mature and less rigorous in compliance than the nuclear power industry. The revised Part 20 rule ought to be addressed by the RRG. Recognizing that the revised Part 20 is essentially a new rule and experience with its implementation is minimal, the RRG should recommend that a review of the revised Part 20 should be performed by the NRC approximately two years following Part 20's required implementation date of January 1, 1994.

10 CFR 50.2 Definitions

The regulatory review group's report recommends a revision to 10 CFR 50.2 "Definitions" to include specific definitions. In particular the report recommends the term "current licensing basis" be defined. By letter dated October 2, 1992, NUMARC submitted comments to Chairman Selin identifying a number of concerns regarding the policy statement on availability and adequacy of design basis information. Additionally, NUMARC submitted comments on April 2, 1993 to the a draft generic letter (ref. Fed. Reg. 15885, March 24, 1993) expressing the belief that a generic letter is unnecessary and unwarranted. We also recommended reconsideration of the policy statement in terms of the long term impact on future self-initiated licensee efforts. We recommend no further action in redefining design basis.

The Office of Policy Planning examined the CLB issue for operating plants and their report, OPP-92-02, provides a draft definition for CLB. The following comments and concerns are offered on the OPP definition of CLB:

- The report suggests that the CLB should not be changed without NRC review and approval. This is contrary to 10 CFR 50.59 and would be inconsistent with a definition of CLB added to 10 CFR 50.2.

- The second paragraph on page 5 of the report; states that "...CLB should only specify how applicable regulations will be met..." Formulating the CLB as a set of requirements causes it to be more of an encumbrance on licensees.

- It is not certain if the proposed definition offered on page five is broader or narrower than the definition in 10 CFR 54.3. Phrases like "...relied upon as the basis for meeting applicable regulations..." are open ended but could be interpreted either way. A broader definition is beneficial for renewal in terms of what can be challenged. However, a broader definition may be a disadvantage in other regulatory areas, 50.54(f) for example.

<u>10 CFR 50.9</u>

The Regulatory Review Group has determined that 10 CFR 50.9, which was promulgated to ensure that complete and accurate records are maintained by licensees and discrepancies or changes identified to the NRC within two working days of identification, should not be the subject of further review. The industry agrees with the NRC's conclusion because the regulation relates to the NRC's ability to carry out its mandate to protect the public health and safety and imposes a reasonable regulatory burden on licensees.

<u>10 CFR 50.12</u>

The Regulatory Review Group has determined that 10 CFR 50.12 should not be the subject of further review. The industry supports the NRC conclusions.

The NRC may not exempt itself from a regulation as has been suggested in various proposals related to this regulation. To the contrary, and as a general precept of administrative law, regulations are intended to prevent agencies from doing exactly that. Even if the agency seeks to exempt itself from its regulations to accomplish what might be appropriately categorized as "laudable aims," the D.C. Circuit has expressly ruled that this action "cannot be sanctioned" (<u>Reuters Limited v. Federal Communications Commission</u>, 781 F.2d 946, 950-51, (D.C. Cir. 1986)). Thus, the industry strongly opposes any modification to or use of 10 CFR 50.12 to allow the NRC to bypass strictly adherence to the regulations by which it is bound.

<u>10 CFR 50.34 Contents of Applications, Technical Information</u>

50.34 contains requirements for the preparation of PSARs and FSARs. The types of information required include the following: design basis, licensee organization, facility operation, preoperational testing, emergency planning, technical specifications training, security, safeguards and TMI items.

It is expected that future applicants for combined licenses (COL) will be required to meet the intent of 50.34. However, the application content requirements of 50.34 are quite general in nature and can be expected to be met in due course through implementation of the design certification and COL subparts of Part 52.

Additionally, specific requirements for security, EP, etc., are located elsewhere in the regulations such that the focus of 50.34 is principally on what must be covered by license applications, not how these matters must be addressed. The notable exception to this is 50.34(f), *TMI-related requirements*, which are specified in this section of the regulation. However, consideration of TMI items will be reflected in design certifications referenced by COL applications.

Supporting Regulatory Guidance

The major piece of supporting regulatory guidance for 50.34 is Regulatory Guide 1.70, "Standard Format and Content of SARs" (Revision 3, November 1978). It is likely, given its vintage, that this format and content guidance will need to be substantially revisited relative to COL implementation.

Summary and Conclusions

Based on the above, the existing 50.34 should not significantly impact future COL applicants. We concur in the conclusion that 50.34 is not a likely candidate for further review or revision. It is likely, however, that RG 1.70 will not provide an appropriate basis for COL applications and would require a revision.

Review of Sections 50.34a, 50.36a, 50.36b and Appendix I

The RRG concludes that these items are not candidates for further review. This recommendation is counter to the industry's letter to the Commission dated December 21, 1992 regarding industry review of NRC regulations and regulatory processes. In that letter we included recommendations for improving 10 CFR Part 50, radiation protection standards, in particular, related general design criterion GDC 19 and Part 50.34a, 50.36a, and Appendix I as well as related regulatory guidance, for example, Reg Guides 1.21 and 1.109. We continue to recommend these documents be updated consistent with the newer concepts and methods of the revised Part 20, additionally considering opportunities to enhance performance based aspects and relieve burdens marginal to safety.

10 CFR 50.47

The Regulatory Review Form for 10 CFR 50.47 appears to be incorrectly completed. The group responded "no" to the question, "is it feasible to make the rule more performance based?" We believe the correct answer is "yes," with the explanation that a more performance based rule could allow for some appropriate reduction in the requirements for training and testing. The RRG responded "no" to question K, regarding whether the rule could be modified to provide flexibility without compromising adequate safety. We believe the answer should be "yes," with the explanation that reduction in the requirements for back up response personnel would not affect emergency response and could provide significant relief for some licensees, especially those who have five staff rotations of their emergency response organization.

10 CFR 50.49, Environmental Qualification

We concur with the assessment of this regulation and support the staff efforts in developing improvements as part of the Marginal to Safety Program. We believe that the application of risk-based methodologies to the Q List will reduce the set of Class 1E equipment currently within the scope of this rule to those that indeed contribute to safety.

10 CFR 50.54, Conditions of License

We strongly support the suggested changes to this regulation and the removal of Fire Protection reports as license conditions. These recommendations should be factored into the NRC's Marginal to Safety Program.

10 CFR 50.55a:

We agree with the Regulatory Review Group's recommendations that both the ASME Section XI and O&M Codes work toward incorporating risk-based approaches in inservice inspection and testing. We encourage efforts by the staff to make the rule consistent with the codes and to provide adequate flexibility in the rule and supporting guidance and implementing documents to incorporate risk-based techniques as they are developed and incorporated into the codes.

10 CFR 50.62, ATWS

While we would agree the rule does not need to be changed, we believe it is important to note that the rule is clearly *prescriptive* rather than *mixed* (as noted in the worksheet in Volume II). The basis and requirements of the ATWS rule were "hardware fixes" to accomplish specific functions in lieu of defining design basis ATWS events to be protected against. Little was left to the discretion of licensees in developing criteria and designs to meet the intent of the rule. Also, we concur with the RRG recommendation that the somewhat customized ATWS protection system QA requirements should be reassessed as part of the overall graded approach to QA.

10 CFR 50.63, Loss of All Alternating Current Power

Generic risk studies were used as the basis for the requirements of this rule. As the result of industry actions relative to Individual Plant Examinations and applications of Probabilistic Safety Assessments, a much more accurate assessment of loss of power risk exists. This rule should be considered for further review to determine if the improved state of knowledge warrants changes to a more performance-based regulation.

10 CFR 50.109

Because the backfit rule includes within its scope any means used by the NRC "to create an obligation upon licensees to change the design, construction or operation of a facility..." 49 Fed. Reg. 47034, 47035 (1985), many of the rulemaking changes anticipated to result from the NRC's evaluation of its regulations and its regulatory approaches have led to questions concerning the applicability of the backfit rule. That is, as the NRC performs its evaluation, it is likely to identify regulations and/or requirements that are no longer necessary or need modification to better serve their intended purposes. The NRC also may propose to change a regulation to attain enhanced efficiency or decrease licensees' costs although such changes would not produce a substantial increase in overall safety. Further, to better accomplish the intent of a current regulation, the NRC may propose additional requirements despite the fact that they would not provide a commensurate benefit to licensees or a substantial increase in overall protection of the public. Both the industry and the NRC have questioned how 10 CFR § 50.109 would apply in, *inter alia*, in these anticipated circumstances.

The industry has concluded that it is not necessary to modify § 50.109 to allow the NRC to adopt modifications that reduce or eliminate regulatory requirements or that constitute regulatory improvements. The industry's view is that such actions meet the

Rather than promulgate the new regulation in the alternative, the NRC may wish to replace the old regulation with the new for consistency and ease of administration. To meet the intent of the backfit rule, the NRC may make an explicit finding that existing licensee programs (in conformance with the prior regulation) are deemed to satisfy the provisions of the new regulation. As long as no change is required in licensee programs, there would be no backfit.

Suggestions have been made that the NRC may waive the backfitting rule or exempt certain rulemaking actions from the rule in particular cases. Specifically, this was the staff's recommendation in the May 13, 1993, briefing to the ACRS on the backfitting rule. For both legal and policy reasons, this approach is not appropriate. As a legal matter, the NRC's authority to waive the backfitting rule is limited. Under well established principles of administrative law, an agency lacks authority to depart from its own rules. Also, as a matter of policy, if the NRC could exempt particular actions from the standards of § 50.109 on an ad hoc basis, the same uncertainty and unpredictability that led to the rule's promulgation would again be created. Licensees and the public would never be certain about when the rule would be applied. Such ad hoc waivers or exemptions would destroy the very discipline the rule was intended to impose on the regulatory process. The NRC's Office of General Counsel (OGC) recently registered its objection to this approach at the above referenced ACRS meeting. OGC expressed concern about the difficulty in distinguishing the kinds of issues for which an exemption likely would be sought and that, therefore, the result would be that exemption requests would appear arbitrary.

Overall, the purpose of § 50.109 is not be served by applying an excessively broad interpretation of the rule so as to preclude changes that would be beneficial to licensees.[3] To do so would be to elevate form over substance. This is not legally required nor do public policy arguments support such an approach. Thus, the industry endorses the NRC's Regulatory Review Group's recommendation that no action be taken to modify 10 CFR 50.109. Further, the industry's position and that of the Regulatory Review Group are consistent with the Commission's Staff Requirements Memorandum dated June 30, 1993, SECY-93-086, *Backfit Considerations*, wherein the Commission determined that the backfitting rule continues to provide a valuable measure of discipline to the rulemaking process and that no change to the rule is necessary.

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[3] The notice and comment rulemaking process will afford licensees an opportunity to comment on the impact of the proposed change and whether it will actually improve efficiency and reduce costs. If licensees do not agree that such is the case, and the change would increase the burden on licensees, the standards of § 50.109 would apply.

10CFR50 Appendix A, General Design Criteria

We agree with the conclusions of the report. However, in applying the design criteria during inspections, the staff often defines new design basis scenarios by considering variations of design basis accidents, e.g. partial loss of offsite power, and then uses these variations in compliance/enforcement actions. The General Design Criteria should be interpreted with respect to design basis accidents and other scenarios considered in a manner consistent with Probabilistic Safety Assessments (PSA) for risk significance before taking regulatory actions on individual licensees.

10 CFR 50 Appendix B, Quality Assurance

There is currently an effort underway to publish Revision 4 to Regulatory Guide 1.28, "Quality Assurance Program Requirements" (DG-1010), that is inconsistent with the spirit of the recommendation calling for revision of implementing documents and guidance related to Appendix B that includes performance-based and graded approaches. Accordingly, this independent effort of RG 1.28 should be discontinued pending a more comprehensive review of Appendix B. NUMARC submitted comments to a draft of this regulatory guide on February 4, 1993 in response to a request for public comments (57 Fed. Reg. 59362, December 15, 1992).

10 CFR 50 , Appendix E, Emergency Exercise Frequency

Section 4.1 of the report lists recommendations for changes to regulations and regulatory guidance. Absent from that list is an important petition for rulemaking by Virginia Power Company regarding reduction in the exercise frequency from annual to biennial. The Virginia Power Company petition provides a sound basis for a performance-based approach that should be included in the RRG's recommendations.

SPECIFIC COMMENTS ON REGULATORY REVIEW GROUP RISK TECHNOLOGY APPLICATION - VOLUME 4

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Human Reliability Analyses (HRA) is an integral part of any Probabilistic Safety Assessment (PSA) analysis and as such, can have an impact on the importance measures of many of the Systems, Structures, and Components (SSC) in the PSA model. The SSC impacted are not limited to only those SSC involved with or directly related to the probability of a particular human event. Therefore, both increasing and decreasing a human event would be necessary to establish the risk significance of an SSC. An example of the need to examine both aspects is illustrated using the probability of failing to depressurize in Boiling Water Reactors (BWRs). The importance of the high pressure systems will increase from the base value when the probability of the human event is increased but at the expense of lowering the importance of the low pressure systems. The opposite effect will be observed when lowering the human event probability. The relative change in importance between the extremes must be weighed in order to assess the risk significance of an SSC.

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The differences between PSAs on similar type plants could be affected not only by plant design but could be due to assumptions made during the analysis, success criteria, truncation effects, data, and a myriad of other PSA modelling factors. The prescriptive requirement of using plant-specific data when determining SSC importance based on a SSC found to be risk significant in another similar plant PSA is unfounded. Plant-specific data is not necessarily better than generic data for many reasons. An alternative to requiring plant-specific information would be the use of sensitivity analyses to bound the impact of changes in SSC probability and availability, both in terms of the particular SSC importance and the corresponding change in importance of other SSC.

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The SSC unavailability associated with the fault exposure time and thus the surveillance test interval may not contribute significantly to the overall SSC unavailability; therefore, consideration of the impact of other factors (i.e., unavailability due to maintenance, failure to start, etc.) must be accounted for when optimizing the surveillance test interval.

Attachment 3

NUMARC COMMENTS ON VOLUME FOUR OF THE REGULATORY REVIEW GROUP REPORT

NRC REGULATORY REVIEW GROUP REPORT

VOLUME FOUR

RISK TECHNOLOGY APPLICATION

General Comments

NUMARC commends the Regulatory Review Group's (RRG) effort in assessing the application of risk technology to the regulatory process. In particular, we strongly support the RRG determination that, "PRA methods provide an integral tool that can be used to help ensure coherence and consistency in the regulatory process, and provide a means of converting diverse deterministic requirements to performance-based requirements. This provision can occur with equivalent protection to public health and safety while offering increased flexibility to licensees, provided the risk-based criteria are met." We believe the important point recognized in this determination by the RRG is that PRA (PSA) is a tool or a means to an end, and is not an end in itself. This tool can be used effectively together with the other assets we have available, namely thirty-plus years of operating and licensing experience as well as the increased knowledge base of nuclear energy technology that exists today. It is through the complementary use of all of these that the strengths of each can be maximized and the weaknesses or limitations minimized to produce a more effective and efficient regulatory process.

NUMARC is coordinating industry activities to apply risk technology to the design and operation of operating plants. We have established the Regulatory Threshold Working Group as the principal mechanism to perform this function. The Working Group is chaired by Jack Skolds, Vice President - Nuclear, South Carolina Electric & Gas Company. The membership consists of twenty executives and managers, representing utilities, NSSS vendors, EPRI and INPO. The Working Group's fundamental purpose is to promote the use of PSA technology and other new approaches to regulation as an aid to focus industry and regulatory attention and resources more effectively. The Mission Statement of the Working Group is enclosed for your information.

The Working Group will develop guidance on generic applications of PSA technology, as well as a framework of PSA attributes that will support such applications. This framework will address the types of issues delineated in Volume 4, e.g., PSA updating, configuration control, data bases, human reliability analysis, quantitative risk criteria, etc. Although we may not view each of these attributes in the manner described in the report, we believe the ideas expressed in the report provide a reasonable context upon which to base further industry/NRC staff discussions.

The Working Group will also coordinate industry pilot projects that demonstrate the viability of generic applications of PSA technology in the regulatory process. Currently, the Working Group is identifying those applications that are of most importance to the industry. Volume Four of the RRG report will be a primary source of information for this effort. Significant industry resources have already been expended in developing a blended (risk-based/deterministic) method for prioritization and categorization of motor operated valves (MOVs). This work will provide the basis for a graded approach for ensuring the performance of MOVs in response to Generic Letter 89-10. We expect to discuss this work with the NRC staff in August.

NUMARC's intent in all of these activities is to apply the PSA tools that currently exist at utilities. Applications will be focused on using risk models consistent in detail and makeup to that requested for the IPE. While we generally support further research efforts to refine PSA technology and to gain additional analytical insights, the Working Group's effort is intended to maximize the benefits using current available tools over the next twelve to eighteen months.

It is obvious that extensive interaction between industry, NRC, and the public will be needed as efforts to apply PSA technology to the regulatory process move forward. NUMARC, through the Regulatory Threshold Working Group, will provide the industry's focal point for interaction with NRC management. To this end, <u>we believe it is imperative that NRC management provide an integrated structure within the NRC staff to build on and continue the excellent work done to date by the RRG and to interact effectively with industry.</u>

Finally, we note that the RRG's expectation, relative to the impact of Volume Four, is that it provides a starting point for specific applications to quality assurance implementation in the short term and to the next phase of possible technical specification improvements in the longer term. We agree that these areas are certainly amenable to applications of PSA technology, and welcome the RRG's work as a foundation for future discussions between industry and NRC. In this spirit, it is not our intent nor desire to provide detailed, critical comments on subsections or specific methods or approaches discussed in Volume Four. We view Volume Four as a positive contribution and as the start of a productive dialogue. However, this does not mean that the industry endorses the criteria for the application of risk technology identified in Volume 4. We do note, however, that the scope of industry efforts to develop generic PSA applications is potentially much broader than the two areas noted by the RRG. We trust that the RRG's emphasis on these two areas was not meant to preclude other potential short term or longer term efforts in other areas.

Comments on Conclusions/Recommendations

- The three general classes for utilization of PSA-based techniques, as categorized by similar boundary conditions, assumptions, as well as similar requirements for the depth and breadth of review by NRC staff, is a good starting point for discussion of ways to characterize and treat different PSA applications. Consideration of different factors may yields other characterization schemes. For example, recognizing that PSA is only one tool or input to a decision-making process, we believe it is possible to blend the PSA's insights with other data in a particular application that may not require a more detailed PSA model or a more in depth review by NRC. Industry pilot projects on specific PSA applications will be done with the objective to identify any technical, legal or policy issues for resolution. It is our intent that through a series of pilot projects on different applications of PSA, any major issues will be identified and will serve to establish appropriate classes of applications.

- We support the RRG's recommendation that the Commission elicit licensee proposals for risk-based regulatory approaches. Our intent at NUMARC is to provide industry coordination of such efforts through the Regulatory Threshold Working Group. We believe our efforts will ensure that industry resources are applied effectively and that the potential benefits to be gained through generic PSA applications can be realized by all licensees. We also trust that through interaction with NUMARC, NRC can apply its resources to support these efforts in a more effective manner.

- With regard to the development by NRC of a handbook for optimizing Technical Specifications using risk-based techniques, we believe that dialogue between industry and NRC should begin in the near future, rather than wait for release of the handbook for public comment as suggested in the report. This application would certainly be of interest to licensees and is also within the scope of our Working Group's efforts. Early dialogue on this topic would ensure that industry and NRC efforts are complementary, and not redundant, and that our resources are applied effectively.

- We strongly agree with the RRG's recommendation that an integral agency plan be established to effectively manage NRC resources relative to the development and implementation of risk-based methods in regulation. We believe this plan should include provisions for communication with industry at the senior management level as well as at other levels, as required by the topic under consideration.

- NUMARC recognizes that the transition to the use of more risk-based approaches to regulation will be evolutionary, and not revolutionary. At the same time, however, we believe that through the commitment of, and effective communication between, all parties involved that real progress can be made in the short term. We firmly believe that without some demonstrated successful applications, it will be difficult to sustain both the management attention and resources needed for success in the longer term. The industry, through NUMARC, has organized to begin this transition. As indicated previously, industry pilot projects are an integral part of our strategy for the transition, with the goal of demonstrating success in the short term.

- We concur with the RRG's conclusion that NRC primarily uses PRA insights to add requirements to the industry. There is a general concern in the industry of the potential for risk-based regulation to become another layer of regulation atop the current deterministic framework. Such an occurrence would be diametrically opposed to the concept of risk-based regulation, which uses quantitative and qualitative insights derived from PSA to focus attention on issues commensurate with their impact on safety. We believe that risk-based regulation can result in reduced regulatory burden <u>and</u> also serve to enhance overall plant safety, and we are committed to demonstrating this concept.

- Finally, we offer a general comment on the notion that risk insights from a plant-specific PSA apply to all similar plants. Although the NSSS design of plants are similar to other plants of their class, there are significant design differences, particularly in support and balance-of-plant systems, such that the validity of applying the conclusions on one plant's PSA to another is doubtful. While we assume this approach was suggested in the report as a means of addressing the "completeness" issue, we believe there are other means, such as blending PSA insights with other analytical tools and operating experience, that can adequately address this issue.

NUMARC Regulatory Threshold Working Group

MISSION STATEMENT

Purpose:

1. Promote the use of probabilistic safety assessment (PSA) technology and other new approaches to regulation as an aid to focus industry and regulatory attention and resources more effectively.

   a. Identify and facilitate generic applications of PSA technology in the regulatory process that sustain or enhance current levels of safety and improve plant performance.

   b. Coordinate industry activities to assess and develop generic applications of PSA technology.

2. Provide unified nuclear industry interaction with NRC management.

Goals:

1. Develop industry guidance on generic applications of PSA technology.

2. Establish a common industry/NRC understanding of the framework of PSA attributes (e.g., level of review, data, maintenance, quality, etc.) necessary to support generic applications of PSA technology.

3. Demonstrate through industry pilot projects the viability of generic applications of PSA technology in the regulatory process. Identify any technical, legal or policy issues associated with such applications.

4. Provide recommendations on any changes to regulations or regulatory practices that are necessary to support generic applications of PSA technology or other new approaches to regulation.

5. Consider the NRC's Safety Goal Policy Statement and provide industry input to NRC on revisions to its regulatory analysis guidance.

6. Interface proactively with NRC so as to resolve both industry and regulatory concerns regarding applications of PSA technology.

REEDY ASSOCIATES, INC.
ENGINEERING MANAGEMENT CONSULTANTS

July 9, 1993

Frank Gillespie
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Gillespie,

I have read the report prepared by the Regulatory Review Group with great interest. I found the conclusions disturbing for the important reasons of safety and costs.

The Review Group's conclusion that Appendix B is "performance-based" is without foundation for the reasons identified in this letter. I do not find their conclusion surprising, because NRC staff members informed NUMARC of this conclusion three weeks prior to the NRC Workshop on April 27-28, 1993. This Workshop was purported to be for the purpose of obtaining industry input regarding "elimination of requirements marginal to safety." One of the main topics of discussion was changing Appendix B to be "performance-based." The apparent fact that the Review Group had already determined that Appendix B need not be changed before all the facts were discussed at the Workshop indicates that the NRC does not appreciate the depth of the problems caused by Appendix B.

With regard to cost, I am currently gathering more financial data to show that the cost of following Appendix B, as opposed to an actual "performance-based" quality assurance program, costs our country more than one billion dollars each year. The one billion dollars is based on estimated unnecessary costs at the 100 nuclear plants in this country. Unfortunately, this cost is paid by the rate-payers who cannot afford it in our current economy.

To assess whether or not the quality assurance requirements of 10CFR50 Appendix B are "performance-based," it is necessary to identify what "performance-based" means. I feel that any "performance-based" quality assurance program must meet the following provisions:

1. The primary objective must be achieving product quality, rather than documentation quantity. Quality is assured when the product consistently meets or exceeds specifications, regulations, and client expectations.

2. Management must provide all the means and encouragement to empower workers to do the best they can to achieve the desired product quality. This requires allowance for innovation and identification of goals rather than overly-detailed, prescriptive procedures.

   (a) The organization must strive to improve the quality of their work. Managers should ensure that each employee shares the overall vision of the organization's purpose with regard to quality.

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(b) Each employee must take responsibility for the work he performs.

(c) Management must provide employees with the material and training necessary to perform their tasks.

(d) Management must ensure that employees are constantly encouraged to improve the quality of their work.

(e) Management must ensure that employees are competent to do the work they perform.

(f) Management must seek and use relevant experience based on new technologies or lessons learned.

3. Assessments must be made to assure that the product quality goals have been met or exceeded. The primary means of assessment is to verify that the quality program is being implemented in a way that ensures product quality. This is accomplished by reviewing work performance, not documentation of activities.

4. Performance-based assessments of the work and the product must be made by people who understand the work they are reviewing. Auditors must be competent to review the work they audit. Assessors must be concerned with the goal, rather than concentrating on the documented means to reach the goal. Some documentation may be reviewed, but documentation is subjective evidence, not objective evidence. The NRC and QA auditors continually treat documentation as objective evidence in spite of the fact it is only subjective and easily modified.

Section XI of the Regulatory Review Group's Report on page 98, states, "In practice, QA is not performance-based. There is a long-term tendency to use paper (the pedigree) as a substitute for engineering judgment." This is certainly true and the inappropriateness of this approach should be identified in Appendix B. Substitution of documentation for understanding and judgment started in 1971 and reached a peak as soon as nuclear construction reached its peak. The NRC reinforced this interpretation of their requirements through their inspection program, and industry standards (such as ANSI N45.2 and ASME NQA-1) have been written to reinforce this precedent.

The Foreword of ANSI N45.2 stated that it was intended to supplement the provisions of Appendix B. The Foreword of NQA-1 states, "The Introduction, Basic Requirements, and Supplements together are intended to meet and clarify the criteria of Appendix B of 10CFR Part 50, dated January 20, 1975." The NRC participated in the development of these documents and endorsed them. These industry standards were never intended to be "performance-based" quality assurance. They were written to help people understand the provisions to be met to comply with Appendix B. There is now activity to change industry quality programs to be "performance-based," but encouragement from the NRC is needed. Changing Appendix B would provide that encouragement.

If you read Appendix B, the phrase in almost each of the criteria is, "measures shall be established," (documented), but never does it state that the product shall meet established requirements. In fact, five of the eighteen criteria are exclusively concerned with paperwork and another four criteria address assessment functions. Obviously, Appendix B was written from the viewpoint of inspection, enforcement and oversight. There is little, if any, emphasis on either product or service quality, or on the management and work processes that ensure it. Appendix B stresses independence of inspectors rather than requiring inspectors and auditors to be knowledgeable or qualified to assess what they are evaluating. If the report is correct, why would the NRC discuss whether or not QA should be changed to be "performance-based" at the April Workshop? I believe the facts overwhelmingly support the conclusion that Appendix B is not "performance-based," but is inspection driven and compliance-oriented.

Why is it, that if Appendix B is "performance-based," as the Review Group's report states, no one knows it? Even assuming that the NRC always intended Appendix B to be "performance-based," it is obvious that industry does not understand that a "performance-based "approach is permitted. The fact that no one understands that a "performance-based" QA approach is allowed by Appendix B is reason enough to rewrite Appendix B.

The most compelling argument for rewriting Appendix B is that it does not assure product quality. In fact, it sometimes detracts from product quality. In most cases, product quality is achieved in spite of Appendix B. I have previously provided supporting facts for this premise in a letter to Mr. Richard Vollmer of the NRC. I have also heard industry experts testify to this fact in litigation and in NRC hearings.

Several years ago, EPRI and the Nuclear Construction Issues Group tried to obtain NRC agreement on an approach to drastically reduce documentation, but the final program was never agreed to by the NRC. The approach was to keep only essential data required for plant operation. One utility, on its own, has now used this approach and estimates that the annual savings per site is more than $7.5 million. That would amount to an approximate annual savings in the United States of $500 million. Everyone knows documentation is necessary, but saving all data is expensive and does not enhance plant safety.

Preliminary numbers show that changing Appendix B to encourage a practical approach to QA would save much more than one billion dollars each year, when considering all nuclear plants now operating in this country. I am currently gathering more information regarding changing to a "performance-based" QA program. There are several examples of "performance-based" QA programs currently available. One is the Department of Energy's program in DOE Order 5700.6.C., which is soon to be published in the Code of Federal Regulations as Title 10 Part 830.120. Also, the International Atomic Energy Agency has moved to endorse a more practical approach to QA in that they have prepared a draft document of QA requirements with supplementary guidelines. Additionally, I understand that the ISO-9000 Committee, which publishes QA requirements for all kinds of products (not only nuclear), is considering a similar approach. If we likewise move to change and update Appendix B to reflect the past twenty-plus years of experience, it will enhance our country's ability to compete overseas and reduce the trade deficit.

If the NRC will initiate changes such as these, there will be a direct and positive impact on all international trade. As you know, most suppliers in the U.S. have gotten out of the nuclear business because of the limits and punitive nature of Appendix B and the costs involved. However, the suppliers still make the identical products for the commercial (non-nuclear) industry. I believe this clearly illustrates the wastefulness of the current QA requirements as reflected by Appendix B. Industry QA standards must also change if we are to compete in the international arena. It will be hard to change these U.S. industry standards if Appendix B does not change. I have assessed QA programs and found they fully complied with Appendix B and industry standards, but unfortunately, they did not foster product quality and effective use of resources. Anyone who honestly appraises most Appendix B QA programs will also reach this conclusion.

At the NRC Workshop in April, Chairman Ivan Selin made the following very important statements.

> "I am pleased to be here to welcome each of you personally to this, the first major NRC public workshop on eliminating requirements that are marginal to safety. This event marks a new approach to nuclear regulation in which the NRC, jointly with the public and the nuclear industry, seeks continually to improve its regulations."
>
> "This topic represents something near and dear to my heart. There is no excuse for a government agency inflicting any greater burden on its supporting public than is absolutely necessary, which would be reason enough for the major push which this conference is kicking off. But there is an even more basis reason for trying to make our regulation more systematic, predictable, and efficient in its impact on our licensees."
>
> "The major reason for seeking to remove inefficient regulations is safety -- to free up resources which can be shifted to more productive safety uses. Programs that result in a better allocation of resources for competing risks are worthy of staff resources and are consistent with the mission of the agency."

For the sake of safety, quality and economics, Appendix B must be changed. A good way to start that change is to consider the new practical approaches outlined in the documents I mentioned above. The principles identified in these approaches to QA reflect proven techniques and are based on the knowledge that comes from practical experience.

I believe that the NRC and industry can and should work together to improve safety and quality and eliminate unnecessary costs. The cost savings and enhancements to safety that will come from implementing a truly "performance-based" set of QA requirements is established in data that I will send to the NRC in the near future. In the meantime, please feel free to call me if I can provide you with any additional information.

Very truly yours,

Roger F. Reedy

Roger F. Reedy
President

RFR:k

cc: Vice President Gore, White House
Ms. Hazel O'Leary, DOE
Commissioner Selin, NRC
Mr. Richard Vollmer, NRC
Dr. Moni Dey, NRC

June 28, 1993

12-D-22

**** BY FAX ****

M. Frank Gillespie
Regulatory Review Group
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Gillespie:

These are the comments of the Ohio Citizens for Responsible Energy, Inc. ("OCRE") on the report of the Regulatory Review Group.

A. General Comments on the Program

OCRE does not object to improving efficiency, removing obsolete requirements, clarifying requirements, and eliminating redundant requirements. However, OCRE does object to actions which are tantamount to deregulating the nuclear industry and which decrease public participation opportunities and access to information. It is essential to ensure that efforts to streamline regulations and enhance efficiency do not have these negative impacts.

OCRE is concerned that the primary motive for undertaking this program appears to be cost reduction for licensees. This is not a proper matter of concern for the NRC. See Union of Concerned Scientists v. NRC, 824 F.2d 108 (D.C. Cir. 1987), in which the Court held that the NRC may not consider costs to licensees in either setting or enforcing "adequate protection" standards. If licensees are paying $200-$300 for setscrews and washers, that is a matter appropriate for investigation by state rate regulatory agencies, not by the NRC. OCRE suspects that such high costs are not the result of the NRC's regulatory programs; rather, they probably result from the licensees' poor management in cost control.

The only basis for eliminating or relaxing regulatory requirements is competing risk: that complying with a requirement introduces other risks of a greater magnitude than the risk which the requirement was intended to reduce.

While it could be argued that devoting resources to matters posing minimal risk diverts resources from more risk-significant areas, and thus produces competing risk, other means exist for rectifying this problem, if it exists. Deregulation is not the answer. If NRC licensees have insufficient financial resources to comply with all regulatory requirements, then that is an issue which should prompt NRC enforcement action. The licensees may also seek relief from rate regulatory agencies. OCRE is very concerned about the interface between safety and economics and believes that rate regulatory commissions should carefully consider the impacts of their decisions on nuclear safety. Such commissions should also ensure that monies earmarked for nuclear safety are in fact spent there and not diverted to non-safety uses.

It is instructive that the Marginal to Safety Program, after years of work, has failed "to identify a regulation that warranted complete elimination because it was so burdensome on operating reactors and so marginal to safety." "Elimination of Requirements Marginal to Safety," 57 FR 4166, 4167 (February 4, 1992). It therefore does not appear that the Regulatory Review Group should be able to find much that needs fixing either.

It appears that the Regulatory Review Group program is a resource intensive process. OCRE thus questions whether this is an appropriate use of limited NRC resources. The NRC should devote its resources to regulation, to protecting the public from the hazards posed by the nuclear industry, rather than deregulation, which could be defined as protecting the nuclear industry from the costs of the public's expectations that it be safe. The nuclear industry is quite capable of protecting and promoting its own interests and does not need the NRC's help.

Finally, some historical perspective is in order. Industry complaints about the burdens of regulation have been perennial, dating back to the 1960s, when the then AEC's regulatory requirements were minimal. This is documented in the book by Dr. David Okrent, a former longtime member of the ACRS: Nuclear Reactor Safety: On the History of the Regulatory Process (1981, University of Wisconsin Press). Dr. Okrent notes (p. 227) that in 1967 industry complaints began about "snowballing" regulatory requirements, particularly requirements for redundancy in safety systems, which were, according to industry, leading to the point where overall safety was degraded. Similarly, the industry as early as 1963 was looking for evidence of an actual source term lower than the 1962 TID-14844 values (p. 307). Today, of course, these are familiar refrains. Rather than accepting these complaints as factual, the NRC should first establish whether they have any more basis than they did in the 1960s.

B. Specific Comments

1. Regulatory Guidance

The NRC's regulatory program can be divided into two parts: binding regulations, contained in 10 CFR, and regulatory guidance, such as regulatory guides, the Standard Review Plan, bulletins, generic letters, information notices, NUREGs, and policy statements. It has long been held that such regulatory guidance documents are not regulations, and compliance with them is not mandatory. Rather, they represent a method acceptable to the NRC staff for complying with regulations. Licensees are free to propose alternative approaches for complying with regulations. Gulf States Utilities (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).

Thus, any industry complaints about the burden of regulatory guidance are non-sequiturs. Rather than complain about the burdensome NRC, licensees should develop well-justified alternatives and aggressively pursue their interests through the mechanisms which are available both to licensees specifically and to any person in general. The former category includes exemptions and operating license amendments, while the latter category includes petitions for rulemaking and commenting on proposed rules and draft regulatory guidance. That the industry has not been as successful as it would desire when it has participated in the development of the NRC's regulatory process is probably more attributable to invalid positions than to a lack of responsiveness by the NRC.

2. Quality Assurance

The Regulatory Review Group's report correctly notes that Appendix B to 10 CFR 50 is already performance-based and utilizes a graded approach. The report clearly states that "the industry had not taken advantage of this flexibility." It is therefore clear that any excessive QA-related costs are not attributable to the NRC but to the industry's failure to manage the QA programs effectively. So this is not a matter of concern to the NRC.

3. Public Participation

OCRE is concerned that changes to regulatory requirements and programs have the potential to eliminate opportunities for public participation and access to information. Two trends now occurring indeed have this effect. Removal of items from the plant Technical Specifications diminishes the universe of potential operating license amendments. The operating license amendment process is the only post-licensing opportunity for public participation in which there is a right to a hearing under Section 189a of the Atomic Energy Act.

Reducing reporting requirements makes the operating nuclear power plant even more of a "black box." This trend is apparent in NRC bulletins and generic letters issued over the past five years. These documents, rather than instructing licensees to send to the NRC detailed documentation and analyses, now instruct them to send conclusory statements that they have completed the requested actions, while keeping the detailed records on the plant site, where it is accessible only to NRC inspectors. If these materials are never sent to the NRC, they will not get into the Public Document Room, and the public will never have access to them. They will not even be obtainable under the Freedom of Information Act, unless an NRC inspector just happened to make a copy of the requested documents and still had the documents in the agency's possession at the time the FOIA request is filed. Ironically, the 1989 revisions to the NRC's rules of practice place an even greater burden on public petitioners to provide as much documentation and factual basis for contentions as early as possible, while access to detailed information is diminishing.

One solution which will reduce reporting burdens and still provide for public access to information would be to establish an electronic docketing system and PDR. Licensees would submit their reports in electronic format, either through a BBS or by sending in floppy disks, and these materials, once processed by the NRC, would be accessible by the public in the same manner. This would save the licensees the cost of copying and postage for reams of paper, while allowing more rapid and convenient access

The NRC's "Principles of Good Regulation" clearly establish that nuclear regulation is the public's business. We do not have a rational regulatory process when meaningful public participation rights effectively cease after a reactor is licensed and operating, which is precisely when it becomes dangerous, and when a licensee cannot correct a typographical error in its Tech Specs without a formal operating license amendment, complete with Federal Register notice and an opportunity for a hearing, while major programs such as the IPE are conducted with no opportunity for public involvement, except for the woefully inadequate process of 10 CFR 2.206. OCRE is gravely concerned that the outcome of the Regulatory Review Group program will make the regulatory process even more irrational and inscrutable, from a public participation perspective.

4. Plant Operating Licenses

The Regulatory Review Group program includes a review of four operating licenses. OCRE questions the purpose of this review, since licensees have substantial control over the content of their licenses: they can always request an amendment to revise any provisions which they deem unnecessary, burdensome, lacking in basis, etc.

5. Safety Margins

Implicit in the Regulatory Review Group program is the assumption that exceeding the minimum regulatory requirements is to be avoided. E.g., among the findings in the review of the Seabrook OL is that several items appeared to exceed the applicable regulatory requirements.

OCRE is concerned that this program will lead to the erosion of safety margins. Ironically, the NRC has recently issued Information Notice 93-35, "Insights from Common-Cause Failure Events," which concluded that the action (for alleviating common cause failures) that had the highest potential benefit was using equipment with larger design margins.

General Electric in its 1975 Reed Report recognized the importance of margins:

Natural phenomena are never completely understood; material properties cannot be completely specified; materials processing is not completely reproducible; dimensions are never precise or completely measured; and mathematical analysis is never complete or exact. Engineering practice recognizes these circumstances by incorporating margins between calculated performance and design requirements. These margins are often large, as in code prescriptions to design to no more than half the nominal material yield stress. The size of the margin requirements is related to the degree of analysis performed: the more phenomena included in the analysis, the smaller the margin prescription. . . . Whenever margins are small, elaborate and frequent inspection and maintenance procedures are imposed to insure against equipment failures. Engineering experience relates small margins to unreliability either through performance fall off, actual failures, or inspection requirements.

The utilities and the architect-engineers have to be convinced that more conservative designs, which may cost more, are the desired work-horses for power generation based on projected improvement in availability/capacity.

GE Nuclear Systems Task Final Report, pp. 7, 51.

It is instructive that new advanced reactor designs are incorporating larger margins than those in the current generation of operating plants.

Eroding safety margins to cut costs is in fact not cost-effective in the long run, as this will lead to more failures and more inspection and testing requirements which may increase wear. It may even cause serious accidents which will induce more public opposition to nuclear power. One more severe core melt accident, especially in the United States, will probably mean the end of the nuclear industry.

If the reactor population increases, each individual reactor must become safer if the aggregate risk is to remain constant. This means that improving safety, not cutting corners, will be most beneficial to the survival and expansion of the nuclear industry.

6. Risk-Based Regulation

OCRE believes this is an area of legitimate inquiry, as it falls into the category of "competing risks" noted above. Using risk-based allowed outage and repair times, rather than the rigid times set forth in the Tech Specs, exceedance of which requires a plant shutdown, may yield a net safety benefit. It is appropriate to balance the risk of plant shutdown vs the risk of continued operation with a failed component. Likewise, the "risk meter" or automated living PRA approach and technology deserve further study.

However, the NRC needs to remain cognizant of the limitations and potential for abuse of PRA: manipulation through "fudge factors" or unrealistic input values, the degree to which the PRA models the actual, as-built plant, and uncertainties in the degree of completeness (has every potential failure sequence actually been considered?) and consideration of external events.

A good example of the apparent lack of complete consideration of the consequences of component failure was the discussion at the May 6, 1993 PRA Subgroup meeting of the safety significance of an oil cover. The comment was made that if an oil cover fails, what are the consequences? So some oil sprays around, but the pump still runs. This discussion neglected the potential for that oil spraying around to cause a fire. What are the implications for plant safety if that oil cover is deemed to be unimportant to safety and the NRC weakens fire protection regulations?

OCRE believes that careful NRC oversight is needed with the risk-based approach to avoid the potential for licensees to become essentially self-regulating.

Respectfully submitted,

Susan L. Hiatt

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FAX COVER SHEET

Date: 6-29-93 Log No.

Time: 4:20 PM Number of Pages: 7
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TO:
Mr./Ms. FRANK GILLESPIE

Of: ~~REGULATORY REVIEW GROUP~~, NRC

FAX #: 301-504-1672

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PHILADELPHIA ELECTRIC COMPANY

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July 29, 1993

NUCLEAR SERVICES DEPARTMENT

Mr. Frank P. Gillespie
U. S. Nuclear Regulatory Commission
Regulatory Review Group
Washington, DC 20005

Subject: Philadelphia Electric Company
Comments on the NRC's Regulatory Review Group Report
to the Executive Director for Operations

Dear Mr. Gillespie:

This letter is being submitted in response to the NRC's request for comments on the Regulatory Review Group's report to the Executive Director for Operations published in the Federal Register (i.e., 58FR29012, dated May 18, 1993). Philadelphia Electric Company (PECo) appreciates the opportunity to comment on the subject report.

PECo endorses the concept of using Probabilistic Safety Assessment (PSA) methods and performance based measures in the regulatory process. We believe increased flexibility while maintaining or improving the protection of the health and safety of the public is essential in prioritizing issues and limiting resources from both the Nuclear Regulatory Commission (NRC) and industry perspectives. PECo also fully supports the development of an integrated plan to assure the risk based methods used in regulation are consistent throughout all divisions within the NRC.

Although testing of the methodologies and pilot programs are an essential facet of gaining acceptance and initiating change, they must be performed expeditiously to gain the benefit in the increasingly competitive environment we all face. Equally important is the use of risk-based analyses to address current issues that are based on deterministic criteria and analyses.

Again, we firmly agree that risk-based technology application and regulation is a necessary and positive step toward optimizing the operation of our nuclear facilities while maintaining adequate protection of the public health and safety.

In addition, we fully support Nuclear Management and Resources Council's (NUMARC's) position and comments regarding the subject report.

PLG

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July 28, 1993
EPRI-1469-PLG-09

Mr. Frank P. Gillespie
Regulatory Review Group
NRC Headquarters
11555 Rockville Pike
Rockville, MD 20852

Dear Mr. Gillespie:

COMMENTS ON NRC REGULATORY REVIEW GROUP REPORT

Enclosed are our comments on the final report of the NRC Regulatory Review Group.

We have restricted our comments to Volume 4, Risk Technology Application. Most of them are technical in nature to reduce the potential for confusion or misinterpretation. However, a few do provide alternative definitions, approaches, and goals.. We recognize that the reviewed document was produced in a very short period of time and is acknowledged to be a first step. Since the NRC Regulatory Review Group is under pressure to complete their report, these comments are offered as suggestions with the recognition that some may be more appropriately addressed as work progresses to implement their recommendations.

In summary, the document represents an excellent step by the NRC toward a practical risk-based regulatory philosophy.

Very truly yours,

Hugo Pomerahn

Hugo P. Pom-ahn
Senior Vice President

Enclosure

PHILADELPHIA ELECTRIC COMPANY

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July 29, 1993

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In addition, we fully support Nuclear Management and Resources Council's (NUMARC's) position and comments regarding the subject report.

Specific comments on Volume 4 of the subject report are provided in the attachment to this letter.

If you have any questions, please contact us.

Very truly yours,

M. C. Kray for

G. A. Hunger, Jr.
Director
Licensing Section

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July 28, 1993
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NRC Headquarters
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Very truly yours,

Hugo P. Pomrehn

Hugo P. Pomrehn
Senior Vice President

Enclosure

Page 4-5: The introduction will be very useful to the those people not familiar with the evolution of PRA in the United States. However, there is an important additional set of experience that should be addressed, if not in this document, then as the program to integrate PRA into the regulatory environment progresses. Since the TMI-2 accident, there have been many industry initiatives to incorporate risk-based arguments into the support of specific license actions; e.g., the UCS petition to shut down Indian Point and Zion, restart of TMI-1, licensing of Diablo Canyon, and the Seabrook emergency planning issue. The NRC has spent much time reviewing the models and basis for these submissions. This would be an excellent set of case studies to provide useful insight into the problems that need to be solved for the success of risk-based regulation. One such problem is setting the decision criteria for addressing risk-based arguments in actual submissions. As we move forward, we should continually look for any lessons learned from these initiatives that can assist in formulating a reasonable and consistent decision framework for using risk-based arguments.

Page 4-6: The wording in the fourth paragraph on common cause failures has a split thought. It would read better if the last sentence were brought forward and modified as follows:

> "Detailed methods have been developed for evaluating the significance of dependent failures, which address not only the quantitative aspects of the analysis but, more importantly, the qualitative knowledge gained that can help prevent their occurrence. As a result of this effort, comprehensive guidance on acceptable ways of analyzing the raw data for dependent failures has been published jointly by the Electric Power Research Institute (EPRI) and the NRC as NUREG/CR-4780. At the present time, the lack of readily accessible root cause data.....(continue with paragraph)"

Page 4-11: The fourth paragraph explicitly states that the final report will focus on applications suitable for Level 1 PRAs. While we agree that many applications can be addressed with the Level 1 portion of the PRA, it may be appropriate to add at the end of the paragraph: "Some sequences important to core damage may not contribute to release categories that are a more direct measure of public safety. These questions may be able to be addressed with the Level 2 scope required by the IPE process. They should be included in risk-based applications where appropriate."

Page 4-11: We understand that Section 4.2 is intended to provide a general overview of the various methods that are generally used by licensees in their PRAs. To put the overview into perspective, we believe that the following should be added to the last paragraph of page 4-11:

> "The summary provided in the following subsections by no means represents the level of detail, scope, and assumptions that make up a PRA. It provides a person not familiar with risk assessment a reasonable perspective of the scope of considerations that need to be addressed. Adequate understanding

of these requirements can be obtained only through the study of the considerable literature on this subject and the application to actual problems."

Some of the examples used in the summary refer to specific methods that could be misleading when applied to PRA models, in general. Specifically, we recommend that the following modifications be made to reduce the potential for confusion:

- The paragraph on page 4-12 regarding types of fault trees uses some specific definitions of trees that may not be widely used. The entire paragraph goes into more detail than warranted by other parts of the summary. This paragraph should be deleted.

- Page 4-13: Change the last sentence of the first paragraph on initiating events to read:

  "Most initiating event frequencies are developed directly from operating data, but logic models for key support system initiators are often created in many PRAs, for example to model loss of service water, component cooling water, or electrical buses."

- The footnote on page 4-13 addresses a detail of initiating event analysis and should be deleted. (Exceptions to the footnote occur because system failures that by themselves do not cause initiating events may still have to be processed as initiators if other system failures in combination with them can cause a plant trip. See, for example, the BNL review of the Diablo Canyon PRA.)

- Page 4-16: The discussion of data analysis on the bottom of the page suggests that one must make a choice to use either generic or plant-specific data. In practice, both generic and plant-specific data can be incorporated to reflect the relative strength of the two types of evidence by using Bayesian updating.

- Page 4-17: The definition of recovery actions given here is not universally accepted by the PRA community. It is also inconsistent with that given on page 4-32. We strongly prefer the definition of recovery actions given on page 4-32.

- Page 4-18: The last paragraph before the beginning of 4.2.3 may be misinterpreted to mean that the 1E-8 truncation limit should be used. Guidance later in the document (page 4-39) states that the truncation value must ensure that 95% of the core damage frequency is captured. We believe the 95% criteria is more valid, and in that case, the 1E-8 value has been found to be too high for large linked event tree sequence models. It can also cause fault tree linking models to save an incomplete set of cutsets. We suggest that the 95% criteria be used here as well.

- Page 4-20: The Fussell-Vesely Importance measure should be defined in accordance with Vesely's definition in NUREG/CR-4377, pages 72-73; that is, it is the likelihood of the occurrence of an event times the difference between risk measure for the conditions under which the event occurs and the event does not occur. In the context of a Level 1 PRA, it expresses the contribution of the event to the core damage frequency. When divided by the total core damage frequency, th importance measure is normalized to be expressed as a fraction of that risk measure. It is applicable for determining risk importance when the component is not known to be failed or down. A typical application might involve prioritizing inspection programs. (Incidentally, the use of cutsets is only one method of evaluating Fussell-Vesely Importance.)

Pages 4-22: One consideration not addressed in recommendations regarding the three general groups of applications is how the "generic plant" requirements of Group 1 can be modified for a plant that has made the effort to achieve the detailed PRA, satisfying a Group 2 or 3 application. The logistics of identifying and tracking important SSCs for other plants within a generic group (especially the Westinghouse plants) may involve a considerable expenditure of resources, and this effort could actually increase as the PRA model becomes more detailed. This may reduce the motivation for plants to accomplish more detailed PRAs and pay for NRC reviews. Criteria need to be established so that a "Group 2" or "Group 3" type PRA can apply Group 1 graded implementation without having to address requirements that compensate for the shortcomings of the Group 1 PRAs.

Pages 4-24 to 4-28: Important SSC Definitions. It would be useful to formulate a more operative definition of the term "important SSC" before talking about how the PRA importance measures can be used. This is especially important because the number of classifications made in this section become confusing. Now is the opportunity to require a specific technical reason for having an SSC on a "Q" list. We suggest that two categories encompass the requirements.

- **Deterministically Important** - SSCs whose failure would result in core damage, but for reasons of careful design and continued plant control are judged to have very low failure rates. These components may not be included in the PRA explicitly, but, if they are not considered explicitly, the boundary conditions under which the PRA is formulated make it clear that they are assumed not to fail. Consequently, the QA, QC, testing, and maintenance programs necessary to maintain high confidence in those low failure rates are firm requirements that cannot be violated.

- **Important in a PRA** - SSCs that, in the context of the integrated plant system and the relative frequency of the challenges it faces, are required to function in order to maintain an acceptable core damage frequency or limit the frequency of the release of significant amounts of radionuclides. These SSCs are identified by both examination of the dominant sequences contributing to core damage and release and by exercising various importance measures, as dictated by the problem being addressed.

From these definitions, the following points should become clear and would be appropriate to include in the discussion:

- The boundary conditions, success criteria, and assumptions in a PRA actually refer to our understanding of how the plant physically behaves and the completeness with which we address it in the PRA. When considered in conjunction with the individual plant design, they provide a good basis for making plant-specific determinations of relative importance. The PRA results and importance measures should never be used without verifying that they encompass the question being addressed and are physically reasonable.

- The existing "Q" lists would be subjected to systematic scrutiny to justify their classification as important SSCs as suggested in Section 4.4.2. As was stated by the authors, the exact methods and criteria may yet need to be worked out, but by using a physical basis that all parties can identify with on a measurable basis.

- Given the above two thoughts, an orderly process to achieve a categorization of groups of importance can be formulated. Those mentioned on page 4-29 are a good starting point.

Page 4-25: The idea of using importance results from other PRAs, even for similar plants, to establish a graded QA list for a specific plant must be approached very cautiously. Plant design differences make it difficult to determine how specific equipment at one plant should be interpreted for another; i.e. they will not have the same name or number. This could place an unnecessarily heavy burden on a utility that has taken the effort to accomplish a detailed plant-specific PRA. At a minimum, such comparisons should be restricted to the NSSS systems; i.e. exclude support systems.

Pages 4-32 and 4-40: The motivation for the suggested screening criteria for human error probabilities is unclear. Conservative human error probabilities are likely to artificially raise the importance of some equipment while lowering the importance of others, possibly dropping the SSCs below the significance threshold. A wiser course of action would be to continue to use realistic values for human error, with appropriate consideration of the uncertainty resulting from nature of the action and the evaluation methods.

Section 4.5 is much more brief than Section 4.4, yet there is considerable more literature on risk-based Technical Specification applications. Many of these applications are discussed in Section 4.9 on existing NRC efforts. Rather than provide one example of a potential method for evaluating AOTs and providing an overview of one aspect of the STI considerations, it may be wise to reference that work and discuss the general goals that the NRC hopes to achieve by applying risk-based methods to evaluate Technical Specifications.

Page 4-36: We recommend that the concept of risk neutral Technical Specification changes made in the fourth paragraph be made a goal rather than an absolute requirement. That goal should be evaluated in the context of what is physically happening at the plant and how that is being quantitatively addressed within the PRA. In other words, it must be applied with common sense in the context of the plant's safety requirements. The goal should recognize that there are safety benefits that cannot be immediately quantified within a given PRA model. These benefits can be expressed in terms of operational and administrative improvements that could be recognized as valid reasons to declare a small increase in quantified risk as being insignificant and offset by other factors. Finally, it should be flexible enough to recognize that the calculated core damage frequency may increase or decrease as the quantitative model evolves to better represent our understanding of the integrated performance of the plant.

people understand the provisions to be met to comply with Appendix B. There is now activity to change industry quality programs to be "performance-based," but encouragement from the NRC is needed. Changing Appendix B would provide that encouragement.

If you read Appendix B, the phrase in almost each of the criteria is, "measures shall be established," (documented), but never does it state that the product shall meet established requirements. In fact, five of the eighteen criteria are exclusively concerned with paperwork and another four criteria address assessment functions. Obviously, Appendix B was written from the viewpoint of inspection, enforcement and oversight. There is little, if any, emphasis on either product or service quality, or on the management and work processes that ensure it. Appendix B stresses independence of inspectors rather than requiring inspectors and auditors to be knowledgeable or qualified to assess what they are evaluating. If the report is correct, why would the NRC discuss whether or not QA should be changed to be "performance-based" at the April Workshop? I believe the facts overwhelmingly support the conclusion that Appendix B is not "performance-based," but is inspection driven and compliance-oriented.

Why is it, that if Appendix B is "performance-based," as the Review Group's report states, no one knows it? Even assuming that the NRC always intended Appendix B to be "performance-based," it is obvious that industry does not understand that a "performance-based "approach is permitted. The fact that no one understands that a "performance-based" QA approach is allowed by Appendix B is reason enough to rewrite Appendix B.

The most compelling argument for rewriting Appendix B is that it does not assure product quality. In fact, it sometimes detracts from product quality. In most cases, product quality is achieved in spite of Appendix B. I have previously provided supporting facts for this premise in a letter to Mr. Richard Vollmer of the NRC. I have also heard industry experts testify to this fact in litigation and in NRC hearings.

Several years ago, EPRI and the Nuclear Construction Issues Group tried to obtain NRC agreement on an approach to drastically reduce documentation, but the final program was never agreed to by the NRC. The approach was to keep only essential data required for plant operation. One utility, on its own, has now used this approach and estimates that the annual savings per site is more than $7.5 million. That would amount to an approximate annual savings in the United States of $500 million. Everyone knows documentation is necessary, but saving all data is expensive and does not enhance plant safety.

Preliminary numbers show that changing Appendix B to encourage a practical approach to QA would save much more than one billion dollars each year, when considering all nuclear plants now operating in this country. I am currently gathering more information regarding changing to a "performance-based" QA program. There are several examples of "performance-based" QA programs currently available. One is the Department of Energy's program in DOE Order 5700.6.C., which is soon to be published in the Code of Federal Regulations as Title 10 Part 830.120. Also, the International Atomic Energy Agency has moved to endorse a more practical approach to QA in that they have prepared a draft document of QA requirements with supplementary guidelines. Additionally, I understand that the ISO-9000 Committee, which publishes QA requirements for all kinds of products (not only nuclear), is considering a similar approach. If we likewise move to change and update Appendix B to reflect the past twenty-plus years of experience, it will enhance our country's ability to compete overseas and reduce the trade deficit.

If the NRC will initiate changes such as these, there will be a direct and positive impact on all international trade. As you know, most suppliers in the U.S. have gotten out of the nuclear business because of the limits and punitive nature of Appendix B and the costs involved. However, the suppliers still make the identical products for the commercial (non-nuclear) industry. I believe this clearly illustrates the wastefulness of the current QA requirements as reflected by Appendix B. Industry QA standards must also change if we are to compete in the international arena. It will be hard to change these U.S. industry standards if Appendix B does not change. I have assessed QA programs and found they fully complied with Appendix B and industry standards, but unfortunately, they did not foster product quality and effective use of resources. Anyone who honestly appraises most Appendix B QA programs will also reach this conclusion.

At the NRC Workshop in April, Chairman Ivan Selin made the following very important statements.

> "I am pleased to be here to welcome each of you personally to this, the first major NRC public workshop on eliminating requirements that are marginal to safety. This event marks a new approach to nuclear regulation in which the NRC, jointly with the public and the nuclear industry, seeks continually to improve its regulations."
>
> "This topic represents something near and dear to my heart. There is no excuse for a government agency inflicting any greater burden on its supporting public than is absolutely necessary, which would be reason enough for the major push which this conference is kicking off. But there is an even more basis reason for trying to make our regulation more systematic, predictable, and efficient in its impact on our licensees."
>
> "The major reason for seeking to remove inefficient regulations is safety -- to free up resources which can be shifted to more productive safety uses. Programs that result in a better allocation of resources for competing risks are worthy of staff resources and are consistent with the mission of the agency."

For the sake of safety, quality and economics, Appendix B must be changed. A good way to start that change is to consider the new practical approaches outlined in the documents I mentioned above. The principles identified in these approaches to QA reflect proven techniques and are based on the knowledge that comes from practical experience.

I believe that the NRC and industry can and should work together to improve safety and quality and eliminate unnecessary costs. The cost savings and enhancements to safety that will come from implementing a truly "performance-based" set of QA requirements is established in data that I will send to the NRC in the near future. In the meantime, please feel free to call me if I can provide you with any additional information.

Very truly yours,

Roger F. Reedy

Roger F. Reedy
President

RFR:k

cc: Vice President Gore, White House
Ms. Hazel O'Leary, DOE
Commissioner Selin, NRC
Mr. Richard Vollmer, NRC
Dr. Moni Dey, NRC

JUL-30-1993 09:27 FROM SOUTHERN.NUCLEAR.OPER.CO. TO 913015042259 P.01

Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201-1295
Telephone 205 868-5000

Southern Nuclear Operating Company
the southern electric system

July 30, 1993

Docket Nos. 50-348
50-364

Mr. Samuel J. Chilk
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Docketing and Service Branch

Regulatory Review Group; Publication
of Report to Executive Director
(58 Federal Register 29012 on May 18, 1993)

Dear Mr. Chilk:

Southern Nuclear Operating Company has reviewed the publication of the Regulatory Review Group's report to the Executive Director for Operations, which appeared in the Federal Register on May 18, 1993. Southern Nuclear Operating Company is in agreement with the NUMARC comments, which are to be provided to the NRC.

In reference to NUMARC's comments on Volume Four, we recognize it is not NUMARC's intent to provide detailed, critical comments on specific methodologies. However, this should not be taken to mean that the criteria is endorsed. Southern Nuclear Operating Company feels that certain specific criteria will need to be addressed in more detail. For example, there is a general concern over implementation of a screening value of 3E-2 for pre-initiator human events. There are an infinite number of conceivable pre-initiator human actions, and only a few are usually modeled.

Additionally, Southern Nuclear Operating Company believes that a successful conclusion of this effort is extremely important if there is to be a viable nuclear option in the coming years. Operations and Maintenance costs rose at alarming rates during the 1980s and early 1990s, primarily due to new regulations and licensee commitments. Many of these commitments resulted in significant cost increases to the nuclear industry but had little to no safety benefit. Both the NRC and the licensees must put into place a more disciplined process for evaluating the benefits of new regulations and commitments. Performance-based initiatives should be considered whenever possible, old regulations and commitments which have little to no overall benefit to public health and safety need to be eliminated. The Regulatory Review Group initiative provides an important framework for the initiation of such an effort.

Should you have any questions, please advise.

Respectfully submitted,
SOUTHERN NUCLEAR OPERATING COMPANY

DM Morey

Dave Morey
Vice President
Farley Project

DNM/DSC

cc: Southern Nuclear Operating Company
R. D. Hill, Plant Manager

U.S. Nuclear Regulatory Commission, Washington, DC
T. A. Reed, Licensing Project Manager, NRR

U.S. Nuclear Regulatory Commission, Region II
S. D. Ebneter, Regional Administrator
G. F. Maxwell, Senior Resident Inspector

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

Mark O. Medford
Vice President, Technical Support

July 29, 1993

Mr. Frank P. Gillespie
Regulatory Review Group
Office of the Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Gillespie:

This letter provides comments on the report of your Regulatory Review Group. A May 18, 1993, Federal Register notice (58FR29012) announced the pending availability of this report for public comment. TVA has reviewed the report, and strongly supports its basic conclusion that the means by which existing regulatory requirements have been implemented contributes to burdens on nuclear utilities which do not result in commensurate safety benefit. TVA has not had time to perform a detailed review of all of the recommendations in the report.

It is very clear that TVA's nuclear plant operating and maintenance costs have been unnecessarily increased by the regulatory process. As just one example, TVA spends over $600,000 per year to ensure that security officers are available to implement immediate compensatory actions for inoperative equipment because security requirements do not include allowed outage times for equipment repair. The unnecessary costs are seldom the result of specific, detailed requirements included in the Code of Federal Regulations. Instead, they usually result from concerns expressed during NRC/TVA interactions (including inspections), commitments made by TVA in response to (or to head off) such concerns, application of consensus standards implementing various regulations, etc. The common element seems to be that each individual action, once implemented, takes on a life of its own. They appear, at least, irreversible. The resulting collective increase in operating staff and efforts has raised the cost of nuclear generation in the U.S. to the point where it is often unable to compete with alternative means of generation. This situation must be redressed, or there will be no more U.S. nuclear power industry.

TVA intends to review activities at each of our nuclear plants to identify unnecessary regulatory-induced burdens. We will take action to relieve these burdens. Where such action can be accomplished without prior NRC approval, we will do so. Where it is appropriate to notify the NRC of an intended change in a previously-established commitment, we will so notify. Where NRC approval is required (e.g., Technical Specification change, exemption), we will seek it. In any event, our activities will be fully open to review by the NRC resident inspectors and others. We ask that NRC support these activities and continue its own efforts to address those areas in which NRC action is required to permit improvements.

TVA encourages NRC to carry forward with the efforts recommended by the Regulatory Review Group. The Review Group report is an insightful commentary on the current state of NRC activities, and its recommendations appear to be focused in the right direction. They are, however, only a first step. Many of these recommendations require further action within NRC to change rules, review current activities, etc. NRC management should take steps to implement them. NRC management should also ensure that resources are made available to approve those burden-reduction actions which require prior NRC approval. It is equally important that NRC management promulgate throughout the staff the message that changes to reduce burden are acceptable.

TVA endorses the specific comments on the Regulatory Review Group report being submitted concurrently by NUMARC.

Sincerely,

Mark O. Medford

Mark O. Medford

TU ELECTRIC

Log # TXX-93274
File # 10035

August 2, 1993

William J. Cahill, Jr.
Group Vice President

Mr. Frank P. Gillespie
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: COMMENTS TO REGULATORY REVIEW GROUP REPORTS

Dear Mr. Gillespie:

TU Electric is pleased with the opportunity to comment on the Regulatory Review Group's (RRG) recommendation to revise 10CFR50.2 and 10CFR50.54 to include the definition of a commitment and the method of reporting a change in commitment to the NRC.

TU Electric does not believe that revisions to 10CFR50.2 and 10CFR50.54 are needed for the reasons stated below. The RRG recognizes that commitments can be classified in two main categories. First, are those commitments of such safety significance that an upgrade from commitment to regulatory status of a license condition is justified. Second, are those remaining commitments that are not safety significant enough to be elevated to a license condition.

The RRG acknowledges that, for the latter, the licensee should be allowed to make changes using the licensee's administrative controls. TU Electric agrees with the RRG that some commitments are safety significant such that they do or should reside in a license basis document, i.e., Operating License, Technical Specifications, Security Plan, QA Plan. A change to these commitments and the reporting requirements associated with the change would be governed by the requirements for changing the parent license basis document. The proposed amendment to 10CFR50.2 and 10CFR50.54 would be unnecessary for the commitments that have a safety significance since existing regulation change processes are already codified.

The remaining commitments are those having no unreviewed safety questions and for whom safety equivalence, defined as remaining in the accident analysis space of the Safety Analysis Report, can be shown by a 10CFR50.59 safety evaluation. Changes to the remaining commitments therefore, can be governed by the licensee's administrative program and the 50.59 process. Notification of changes to commitments is accomplished by the annual 50.59 summary submittal; the RRG recommendation to change 10CFR50.54 to require notification is therefore unnecessary.

The RRG recommends that a definition of commitment be codified by including it in 10CFR50.2. The recommendation seeks to make the promise by the licensee a legal obligation. The proposed definition of a commitment states that the promise on the part of the licensee was relied in whole or in part by the Commission as a basis for a safety decision. TU Electric believes that promises made to the NRC are part of the current licensing basis and therefore should be evaluated under 50.59 before changing, and reported as required under that regulation. Therefore, addition of a definition of commitment and an additional change process is unnecessary.

Sincerely,

William J. Cahill, Jr.

By: JSMarshall
J. S. Marshall
Generic Licensing Manager

JDR/

1901 Chouteau Avenue
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St. Louis, Missouri 63166
314-621-3222

UNION ELECTRIC

July 30, 1993

Mr. Samuel J. Chilk
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Attn: Docketing and Service Branch

ULNRC-2831

Dear Mr. Chilk:

SOLICITATION OF PUBLIC COMMENTS ON
THE REGULATORY REVIEW GROUP'S
REPORT TO THE EXECUTIVE DIRECTOR
Ref.: Federal Register Volume 58 Number 94 dated
May 18, 1993 (58 FR 29012)

Union Electric Company strongly supports the overall intent of the Regulatory Review Group and the majority of the recommendations made in the subject report. Any effort to reduce the burdensome regulations which have little or no direct safety impact is welcomed. We feel that several of the recommendations may have significant immediate or potential value to licensees. Among them are:

- The revision of 10 CFR Part 21 to recognize existing procurement practices and conditions (report section 2.3.1);
- The revision of 10 CFR Part 50.54 to make consistent the rules governing methods by which licensees are allowed to make changes to their facilities, plans, and programs (section 2.3.9).
- Ensuring that there is a clear delineation of NRC expectations in the security area (section 2.3.18).

- Elimination of the requirement for submittal of quarterly security logs (section 2.3.18).

- The revision of some of the implementing documents for Appendix B to Part 50 (section 2.3.13).

We appreciate the opportunity to comment on this issue. Please contact us if there are any questions regarding this letter.

Very truly yours,

Alan C. Passwater

Alan C. Passwater
Manager, Licensing & Fuels

RMD/kea

cc: T. A. Baxter, Esq.
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VIRGINIA POWER

July 29, 1993

Mr. Frank Gillespie
Regulatory Review Group
Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 93 - 425
NLP / RBP

Dear Mr. Gillespie,

COMMENTS ON REGULATORY REVIEW GROUP REPORT

On the May 28, 1993 the NRC made available a report containing the Regulatory Review Group's (RRG's) recommendations and the bases for those recommendations concerning the revision, evaluation or elimination of certain power reactor requirements, regulatory guidance and licensing processes.

Virginia Power commends the NRC for undertaking this effort and encourages timely implementation of the report's recommendations. We also urge the NRC to continue their effort to eliminate or simplify regulatory requirements marginal to safety. The following comments are provided below.

Virginia Power supports a regulatory environment that includes performance and risk based considerations. A change in emphasis from strict compliance of regulations to more performance and risk based could result in significant savings to both the NRC and the licensees. The strict adherence to prescriptive regulations can require extensive efforts in maintenance, testing, documentation, reporting, and inspection activities that do not necessarily contribute to safety or the intended purpose of the regulations. Also, the utilization of a more performance based regulatory environment would encourage high levels of performance and could ultimately improve the effectiveness of the regulations by encouraging the development of innovative approaches that may result in higher safety and lower costs.

Virginia Power recommends that NRC reviews for regulatory reduction take place on a periodic basis to ensure continued necessity, consistency with other requirements and continued safety benefits. The NRC routinely considers anticipated costs and safety benefits before issuing many of its regulations. However, implementation of a regulation may result in higher costs or lower benefits than originally estimated. In addition, without periodic review, combinations of requirements may result in conflicting requirements or unanticipated higher costs. As delineated in the RRG's report, quality assurance, equipment qualification, fitness for duty, and security are some examples of regulations where the accumulation of requirements has resulted in unwieldy reporting and documentation, high costs, and marginal safety improvements.

Very truly yours,

M L Bowling

M. L. Bowling, Manager
Nuclear Licensing and Programs

cc: Mr. D. Modeen
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Mr. A. Marion
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July 28, 1993

Northeast Utilities (W-133)
107 Selden Street
Berlin, CT 06037

Ms. Mary Drouin
US Nuclear Regulatory Commision
Office of Research
NL-S 324
Washington D.C. 20555

Dear Ms. Drouin:

I am a senior nuclear engineer in the PRA section at Northeast Utilities. Recently, I had an opportunity to read a draft of the NRC report entitled "Regulatory Review Group - Risk Technology Application." At Northeast Utilities, I have worked in the risk application area on a day to day basis for approximately five years.

Enclosed, please find some of my thoughts on the subject report that may interest you. Also enclosed are some thoughts on the recent EPRI report entitled "Risk Based Technical Specification Program," since that report was extensively used as an input to section 4.5 of the subject report. I would like to emphasize the followin

i. Please treat these comments as "A PRA engineer's opinion" rather than any technical position formally adopted by Northeast Utilities. My objective here is to share with you the insights that I have gained for the benefit of the PRA Application Technology area.

ii. Each comment provided here precipitated from the experiences from one or more actual PRA applications. The detailes of these applications are not provided here.

Please call me at 203 -665 3594 if you need any clarifications on the comments.

Sunil D. Weerakkody
Senior PRA Engineer

GENERAL COMMENT

The NRC proposed Risk Technology Application document classifies different types of PRA applications to three groups and sets standards for the PRA modeling requirements for each application group in terms of

a. level of detail of PRA model,

b. PRA update frequency,

c. level of NRC review, and

d. level of plant specific data usage.

While the above attributes are significant, all of these items limit their role to setting standards on the PRA model used for a particular application without setting any standards at all on the process used to derive PRA based conclusions or recommendations.

There are a large number of assumptions, approximations, and boundary conditions that control the results from PRA quantifications and conclusions derived from those calculations. None of the items in the list above emphasize the significance of a appropriate methods or procedures that must be used to re-visit and recover the impact of assumptions, approximations, and boundary conditions used in PRA analysis on specific PRA based decisions. Standards on the process of deriving PRA based conclusions is at least as important in deriving accurate risk based conclusions.

Figures 1 and 2 are used to further illustrate the above point. Figure 1 shows how risk significant information is lost when PRA models are constructed. Figure 2 illustrates a process that would be appropriate to re-visit and recover this lost information into the PRA based decisions, conclusions, or recommendations.

The four attributes a,b,c, and d (level of detail of PRA model, ..etc.) relates mostly to step 3 of figure 2. The standards or requirements which are proposed here applies mostly to steps 2 and 4 of figure 2.

Step 2 of figure 2 "formulate PRA problem" requires indepth knowledge of the plant and insights on the plant specific PRA. In almost all cases where PRA provides significant inputs, the PRA supervisor and one or more PRA engineers who has considerable knowledge of the plant under consideration participate in this step. The meeting that takes care of step 2 does not follow any specific procedure or format and in many ways similar to a brain-storming session. It is this brief, however, extremely crucial

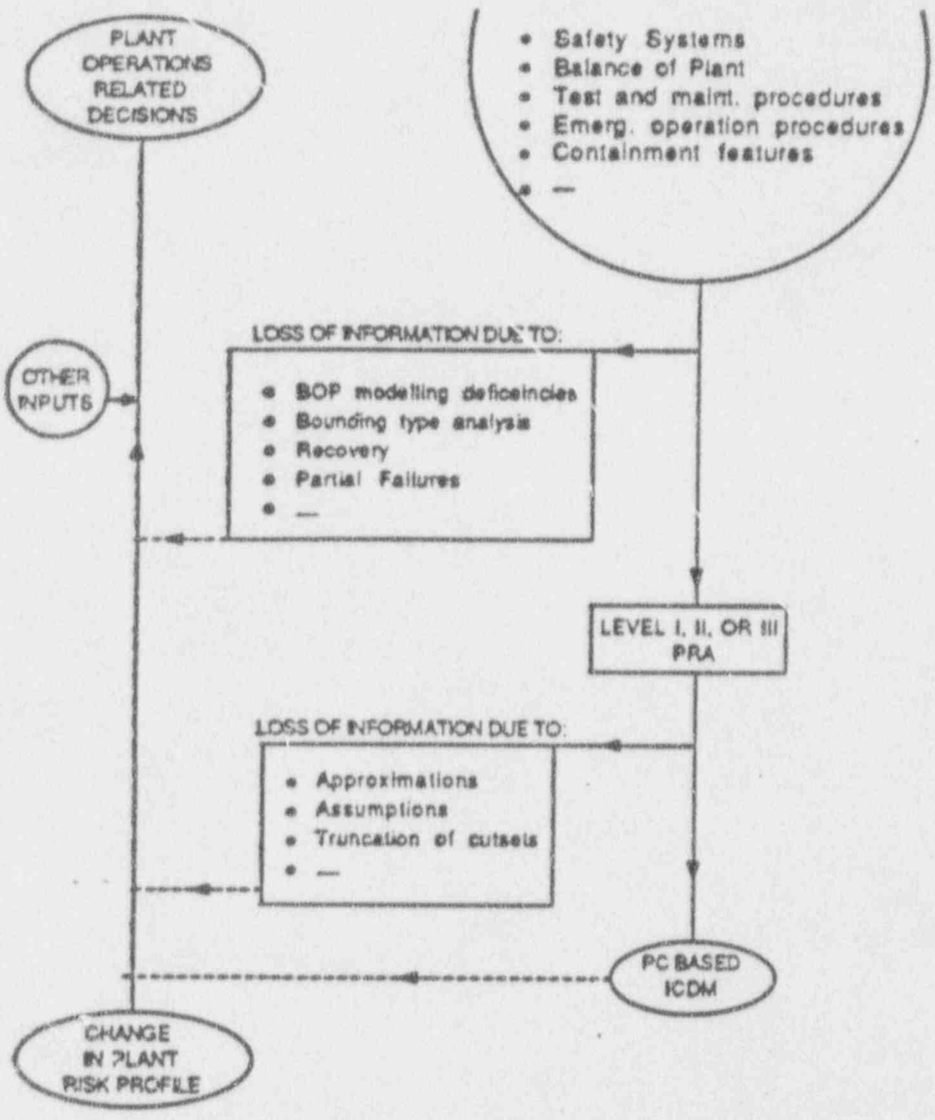


FIGURE 1: INFORMATION FLOW IN PC BASED INTEGRATED CORE DAMAGE MODEL DEVELOPMENT AND USAGE

SPECIFIC COMMENTS ON EACH PRA APPLICATION TYPE

The proposed report provides three types of applications and suggest requirements for PRA modeling used for each application type. Comments on each application category are as follows:

I. TYPE OF APPLICATION

- o Performance-based response to the Maintenance Rule and risk-based approaches

- o Graded quality assurance

REQUIREMENTS

A. IPE type PRA model

B. IPE type NRC reviews

C. Generic failure rate data could be employed

D. Frequent updates of PRA not required

COMMENTS

No comments since preparer has no hindsight on this type of applications.

II. TYPE OF APPLICATION:

- o optimization of selected technical specifications

- o evaluations of unreviewed safety questions under 10 CFR 50.59,

- o use of pre-calculated configuration management analyses to support extension of allowed outage times under certain circumstances)

REQUIREMENTS

A. Requires average PRA modeling

B. Generic failure data would be sufficient in most instances but it would need to be augmented with plant specific data in those selected areas where heavy reliance was placed on the plant specific results.

C. For greater than one time use, the PRA would have to be modified as necessary to reflect any changes in the current design and operational practices.

D. Usage would likely require updating at least each refuel step that sets the agenda for rest of the PRA analysis and assures an accurate risk quantification.

Participation of the PRA supervisor guarantees overall technical expertise in the PRA area and cumulative knowledge of plant systems during the discussions that are to follow. Presence of a PRA lead engineer/s ensures (a) additional knowledge of plant familiarity, (b) knowledge of intimate details of assumtions, approximations, that went into the model development and (c) carryover of insights from step 2 to step 3. Although for typical PRA applications this process (implementation of step 2) is not time consuming, it is during this step that one ensures all risk impacts are captured into the PRA analysis.

At the end of the meeting, in general, the following problem attributes are decided upon:

a. what risk elements (initiators, systems, operator actions, containment features) are impacted ?

b. is the PRA model on PC detailed enough to perform calculations to support the particular application under consideration ? (For example, for the Millstone Unit 2 (MP2) and Millstone Unit 3 (MP3) RHR autoclosure removal projects it was found out that detailed RHR system models for shutdown are needed)

c. what bounding type hand calculations will be necessary ? (For example, for the Tech Spec change request for one time extension of STIs of MP3 for ESF and RTS instrumentation, it was determined that bounding type hand calculations are adequate)

d. are generic data adequate ? (For MP3 DC power supplies for load sequencer, NPRDS search of power supply data provided valuable insights). Are plant specific data necessary ? (For MP3 ESF STI extension Tech Spec change we concluded plant specific data to be crucial). Should additional data sources be re-visited ? (To determine auto vs. manual operation of PORVs for MP3 the generic test data generated and documented by EPRI was crucial).

e. do known PRA methodolgy limitations have major impact ? (During some applications over-estimated common cause failure rates and assumptions regarding constant failure rate for standby components lead to overly conseravative or non-conservative conclusions.

f. what are the anticipated results ? (Forming an opinion on the expected results based on engineering judgement helps step 4 of figure 2 and highlights assumptions that must be re-visited during PRA quantifications).

outage.

COMMENTS

Based on some specific instances where PRA was used to support STI and AOT extensions, the following comments are made on items A, B, C, and D above:

Item A
Although an average plant specific PRA can be adequate for some application needs, in general PRA bases for this type of application requires expanded analysis of a specific area in the PRA that was not modeled in detail before.

Average PRA modeling is adequate for STI or AOT evaluations of major components or system trains such as diesels, service water pumps, and accumulators, etc.. However, frequently there are instances where a typical PRA model is not detailed enough (eg: individual analog channels, slave and master relays associated with instrumentation). In these cases either additional bounding type hand calculations or conclusions based on a knowledge of fundamental PRA input components such as failure rates, redundancy, diversity coupled with overall insights derived from extreme familiarity with plant operations (systems, and procedures) and the plant PRA has been more than adequate to provide the necessary justifications.

Items B and C

The preparers experience is consistent with these conclusions. The importance of additional insights that were gained from plant specific data in the several of our applications of this type cannot be over emphasized.

Item D

Updating PRAs shortly following each refuel is desirable and necessary. However, this "not updating per refuel" does not in general in the way of reaching accurate PRA based conclusions relating to JCOs or Tech Spec changes (Except for rare major design or procedural changes that impact important system fault tree or event tree logic).

Plant changes that impact the area of concern associated with the Tech Spec or design change can easily be factored in to the PRA based conclusion although not necessarily through the use of the PC based computer model.

III. TYPE OF APPLICATION

o Risk-based technical specifications requiring on-line updating of PRA models

REQUIREMENTS

A. Comprehensive analytical efforts since minor changes in assumptions or boundary conditions may significantly affect regulatory decisions

B. Requires a level of detail that either stretches or exceeds the current state of the art.

C. Requires a comprehensive plant specific data analysis

D. Requires that the PRA be reviewed by NRC staff at a depth equivalent to that afforded to a final safet analysis report in the course of a part 50 operating license review.

COMMENTS

The preparer's experience is in total agreement with the basic theme (not necessarily each requirement) underlying the above observations.

Items A, B
An on-line PRA model that would provide adequate and accurate insights for plant operation configuration control must be capable of accommodating sensitivies associated with the large number of assumptions, boundary conditions, etc.. The technology of the existing PRA models is nowhere near this capability since there is a significant loss of information during the PRA model creation process (illustrated in figure 1). For selected applications the above may be possible. However, any fullblown application must be preceded by pilot type applications that confirms the validity of techniques used.

For a PRA model to acquire this capability, the whole process illustrated in figure 2 must be structured, and computerized.

At the present time, the impact of lost informtion on the PRA based conclusions is recovered by PRA engineers during the decision making process by re-visiting the existing PRA model, specifically those assumptions and boundary conditions that potentially impacts the decision made. Discussions among engineers within PRA and communications with plant operations further ensures that appropriate assumptions and boundary conditions are re-visited and accommodated to the PRA based conclusions. An on-line PRA model must computerize this process of re-visiting assumption. Sophisticated PRA models that can emulate this process definitely exceeds the current state of the art.

Reaping the benefits of risk based requlations to relieve unnecessary burdens must not await the creation of such sophisticated PRA models.

Items C and D

no comments

ADDITIONAL COMMENTS ON SECTION 4.5

Section 4.5 has drawn considerably from the recent EPRI document (Andre, G. R., 1993). This section provides some thoughts on the above document.

TECH SPEC ACTION ALTERNATIVE

SUMMARY OF METHOD:

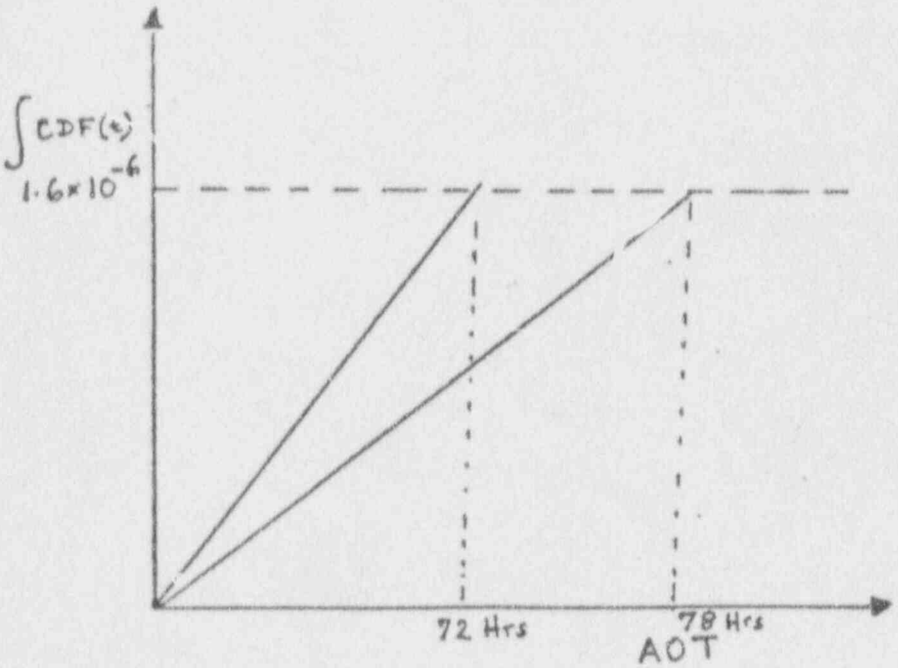
Develop alternative actions to allow same risk. Based on the CDF(t) integrated over the AOT duration.

NEGATIVE/POTENTIAL IMPROVEMENT:

Based on ∫CDF(t) only. Therefore, can result in too conservative AOT extensions (e.g., EPRI report example, AOT extension from 72 to 78 hours when ~1 week could have been justified.) No AOT extension must be based on TECH SPEC ACTION ALTERNATIVE method alone without due consideration to the impact on CDF(avg). For the example chosen, even if the AOT was extended by 1 week, the CDF(avg) increases by only a couple of percents.

POTENTIAL IMPROVEMENT:

- Limit application of the approach to cases where CDF(t) exceeds a minimum CDF(t).



NEGLIGIBLE RISK APPROACH

JMMARY OF METHOD:

stifies AOT extensions based on negligible increase in CDF.
at is based only on change in yearly averaged CDF.

OSITIVE:

'RI report TR-101894 does an excellent job in identifying how one should go about anging appropriate PRA parameters to assess CDF.

EGATIVES/AREAS FOR IMPROVEMENT:

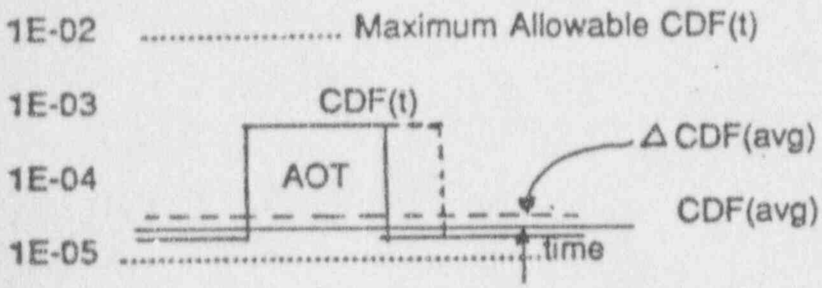
Does not accommodate CDF(t) (instantaneous CDF). This can result in AOT extensions which appear to be reasonable based on PRA analysis, but which are really not. In the long-term, such requests hurt the credibility of the PRA based AOT extensions.

Does not prompt/require the PRA engineer to examine appropriate plant specific failure data. Again, the result would be somewhat similar to item above.

OTENTIAL IMPROVEMENTS:

Limit application of the approach to cases where CDF(t) does not exceed a specified maximum? See illustration below.

Look for and utilize appropriate plant specific data.



RISK TRADEOFF APPROACH

SUMMARY OF METHOD:

Compares risk of staying at power with risk of shutdown.
Integrated CDF (integrated over the AOT based).

POSITIVE:

EPRI report provides excellent analysis method that incorporates risks of shutting down, increased potential for tripping during power reduction.

NEGATIVES/AREAS FOR IMPROVEMENT:

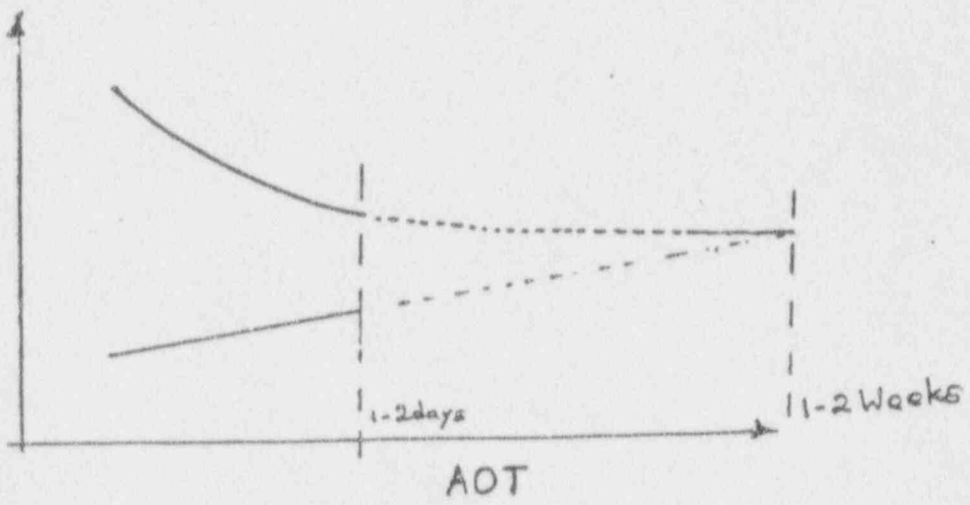
Use of past statistical data (on pump repair experience) rather than the actual situation for which information is available to support AOT extension.

Report recommends increase AOT Ø → 1-2 day (Based on statistical analysis of past pump repair experience, only a few pumps are recovered after a 1-2 days. Therefore, AOT must be 1-2 days.)

Use same basis to increase AOT Ø → > 1-2 Weeks depending on the actual circumstances. For example, the Tech Spec may require going to shutdown if there is no reasonable confidence that the failed pumps can be repaired within 1-2 weeks.

POTENTIAL IMPROVEMENTS:

- Re-visit results and basis for conclusion.
- Limit application to AFW, RHR, LTOP (systems needed during shutdown).



OVERALL DRAWBACKS/POTENTIAL IMPROVEMENT

Provide limitations/guidance on applicability for each method. No guidance on which of the three methods is most appropriate for application.

Use more than just one risk measuring tool.

- CDF
- CDF(t)
- $\int$CDF(t) dt

Try factoring in known plant specific information.

- Failure rates, drift rates.
- Random vs. common cause potential.
- Posterior rather than prior knowledge of repair durations.

Insufficient uncertainty analysis to identify dominant parameter. Identifying the physical parameter that drives the AOT extension (low failure rate, pump repair time, opertor error probability, etc.) ensures a robust AOT (an AOT that does not jump from 1 day to 20 days when a minor variation is made in an uncertain variable) that makes engineering sense.

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

ET-NRC-93-3934

July 28, 1993

Mr. James M. Taylor,
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NRC Draft Regulatory Review Group Report Request for Comment

These comments are submitted by the Westinghouse Electric Corporation ("Westinghouse ") in response to the United States Nuclear Regulatory Commission ("NRC") request for public comments on the draft Regulatory Review Group Report.

First, Westinghouse applauds the NRC for taking the initiative to perform the necessary research and analysis to support the development of this report. This report will help serve as one of the fundamental steps in the overall nuclear industry initiative to reduce the regulatory burden while maintaining plant safety. We strongly support this initiative and believe that the continued viability of nuclear power depends highly on the success of this and other such initiatives.

Based on recent industry meetings, we understand that the NRC has identified a lead individual to serve as a point of contact for all utility regulatory burden reduction programs. We support this action and believe that a single point of contact within the NRC is necessary for the long term success of these programs.

EDO --- 009194
93-0955-A-00

Our general comments on the Draft Regulatory Review Group Report are as follows:

Based on our involvement with the industry technical specification improvement programs, utilities are not quick to implement the line item improvements associated with these programs due to lack of a near term return on investment including the regulatory review time periods expected.

The report focus on core damage frequency as being the final bottom line by which plant changes are to be evaluated. Although this is true in most cases, there are many cases for which containment release (fission product release) or public health risk should be used to evaluate the impact of proposed plant changes. For example, only addressing core damage frequency neglects the impact on containment performance in mitigating the consequences of an accident.

As the report is currently written, the report does not acknowledge the previous PSA applications already performed by the industry and approved by the NRC. For example, the Commonwealth Edison Byron Limiting Conditions for Operation Study, the Westinghouse Owner's Group RPS/ESFAS, Turbine Valve Testing, and Auto Closure Interlock Removal Studies, and numerous other utility sponsored studies.

Additional Westinghouse specific comments on the Draft Regulatory Review Group Report are provided in Attachment I to this letter. If you have any questions regarding these comments, please contact Mr. K.J. Vavrek of my staff at (412) 374-4302.

Very truly yours,

NJLiparulo

N. J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/p

Attachment

WESTINGHOUSE COMMENTS ON NRC
REGULATORY REVIEW GROUP DRAFT REPORT
SPECIFIC COMMENTS

Volume I

Page 7, Paragraph 8. It should be noted that the industry has previously attempted to develop the use of PRA for use in the application of 10 CFR 50.59 via NSAC-125.

Page 9, Paragraph 4. There appears to be an inconsistency between paragraphs 3 and 4. In paragraph 3, the licensee is allowed to incorporate proposed changes to commitments without prior NRC approval provided they do not constitute an unreviewed safety question. In paragraph 4, licensees must receive prior approval from the NRC for any "reductions" in commitments in the QA plan. Permitting changes without prior NRC approval to these plans provided an unreviewed safety question doesn't exist would give the licensees more flexibility and reduce regulatory review burden.

Volume II

Page 146, Paragraph 1. The application of "risk" to unreviewed safety questions determination should be focussed on the impact to the health and safety of the public, not necessarily on a parameter such as core damage frequency.

Page 146, Paragraph 3. The use of risk technology should neither be barred nor required as part of unreviewed safety question determination, but it should be allowed as a tool for the licensee.

Volume III

Page 18, Section 3.3.2 The report uses the new improved technical specifications (MERITS) as a standard for judging potential regulatory overburden. However, the MERITS specs themselves are not optimized fully because most allowed outage times and surveillance test intervals are taken directly from earlier Standard Technical Specifications. Optimization beyond MERITS is possible and with additional risk or deterministic safety analyses this can be realized.

Volume IV

Page 7, Paragraph 5 and Page 8, Paragraph 1. The "bottom line" is the health and safety of the public, and therefore, parameters like core damage frequency and system unavailabilities should not be made regulatory compliance criteria. For example, only addressing CDF neglects the impact on containment performance in mitigating the consequences of an accident.

Page 13, Paragraph 6. The discussion should include the applicability and use of plant response tree models, which integrate containment performance, in the accident sequence analysis.

Page 13, Last Paragraph and Page 14, Paragraph 1. This is a very conservative definition of core damage. Localized fuel damage, although highly undesirable, does not necessarily result in endangering the health and safety of the public.

Page 18, Paragraph 5. If a SSC was not included in the PRA, it was already qualitatively judged to be risk insignificant based on the engineering judgement of the analyst.

Page 19, Paragraph 2. If the HRA is applied consistently and systematically across all accident sequences and realistically models the EOP actions, it should not unduly influence PRA results.

Page 22. Figure 4.3-1 is inconsistent with the accompanying text. The figure indicates that updates for group 1 applications will be required on a refueling basis while the text (page 4-23, paragraph 1) indicates that updates will be done "only when there was a major redesign of one of the plant systems or a major modification in the basic operational principles". The text for group 2 applications (page 4-23, paragraph 3) indicates that updates "would be desired each refueling outage".

Page 24, Paragraph 3. The approach described in this section for graded QA implementation will not work for systems whose primary function is to mitigate releases from containment, such as containment spray system, containment cooling system, and containment isolation. Based on the proposed approach, these systems may be defined as relatively non-important when using the core damage criteria, but based on a containment release criteria, this may not be true.

Page 27, Paragraph 4. Just because a SSC is identified as risk significant for a group of plants does not necessarily imply that it is risk significant for a given plant. If the model assumptions resulting in the difference are based on valid analysis, then the SSC should be able to be moved to the graded QA list.

Page 32, Paragraph 4. Setting all human error probabilities to a pre-determined value would provide erroneous results. Systems that are backed up by operator actions, with the operator action HEPs all set to the same HEP, then could incorrectly be identified as more important to safety then systems with no operator action backup. In addition, the relative importance of operator actions would be lost. To address this problem the operator actions could all be increased by a common factor, this would maintain the relative significance of the operator action HEPs.

Page 35, Section 4.5. More detailed information should be provided in the report concerning the information required in a submittal requesting Technical Specification changes. Is the impact to average yearly core damage frequency enough or will additional information involving time dependent risk profiles, importance calculations, and/or conditional core damage frequencies be required?

Page 35, Paragraph 5. Releases from containment should be considered, instead of core damage frequency, for systems that are of primary importance to preventing containment releases.

Page 36, Paragraph 2. The core damage probability associated with shutting the plant down should consider at least two regimes of mode change. The first is from the at-power condition (100% power) to the point where the auxiliary feedwater pumps are started and the second is from this point to the targeted mode. The risk associated with the first regime is due to the potential of a reactor trip while the reactor power is being reduced. The risk associated with the second regime is due to the potential of the auxiliary feedwater system failing. In addition, the probability of repairing the failure within the given AOT needs to be considered, since if the repair cannot be completed within this time period, then a shutdown will still be required. This method is discussed in detail in EPRI report TR-101894 "Risk-Based Technical Specification Program".

Page 39, Section 4.5.3. The report does not address what level of risk degradation, if any, is acceptable for changes in AOTs and STIs. Some definitive statement would help. Also, do changes to AOT and STIs need to be reassessed when the model is updated? Data or assumptions within the model may invalidate some changed AOTs or STIs at a later date.

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July 29, 1993

Mr. Frank P. Gillespie
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Solicitation of Public Comments on the Regulatory Review Group Report

Dear Mr. Gillespie:

These comments on the Regulatory Review Group (RRG) Report are submitted on behalf of the Nuclear Utility Backfitting and Reform Group (NUBARG),[1] in response to the request for public comment in 58 Fed. Reg. 29,012 (May 18, 1993) as modified by 58 Fed. Reg. 33,285 (June 16, 1993). NUBARG strongly supports the RRG initiative. Our comments focus solely on the continued need for changes to 10 C.F.R. § 50.54(f) and to Staff procedures governing such information requests.

The RRG report addressed the question of whether the procedures in 10 C.F.R. § 50.54(f) concerning information requests required revision (this issue was previously identified in the Marginal-to-Safety Program). The RRG noted that the NRC had recently improved its procedures for processing generic communications, among other things, by requesting public comment on certain forms of such correspondence prior to issuance. The RRG concluded that, because of the improved procedures, no change was needed to the provisions of Section 50.54(f). NUBARG agrees that the new NRC procedures provide improvement in Staff use of information requests. NUBARG believes, however, that additional improvements can be made to ensure that Section 50.54(f) requests do not impose excessive burdens on licensees.

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[1] NUBARG consists of the nuclear utilities listed in the Attachment hereto, each of which owns or operates a power reactor licensed by the NRC. NUBARG actively participated in the development of the NRC's backfitting rule and has closely monitored its implementation.

The standard established in Section 50.54(f) for Staff issuance of information requests is that the burden imposed be justified in view of the potential safety significance of the issue under consideration. Despite the existence of this standard, concerns remain regarding the potential for information requests to be used to bypass the rulemaking process and the disciplined cost-benefit process required by the backfitting rule. Specific examples on which these concerns are based include the use of information requests to impose new Staff positions interpreting regulatory requirements (e.g., Generic Letter 89-06 on SPDS criteria), to request extensive analyses and programs (e.g., IPE and IPEEE), and to impose new programmatic requirements (e.g., MOV testing under Generic Letter 89-10).

A primary means for addressing these concerns is by changing Section 50.54(f) itself. NUBARG recommends that the NRC raise the threshold for issuance of an information request by (1) revising Section 50.54(f) to cross-reference Section 50.109 for cases in which the request involves a new program or an extensive analysis for which the backfitting rule should be invoked; and (2) for compliance issues, requiring identification of the specific existing regulatory requirement for which verification of compliance is sought. To accomplish these changes, NUBARG recommends that Section 50.54(f) be revised as follows by adding new third and fourth sentences to read:

> Where the information is sought to verify licensee compliance with the current licensing basis, the Staff will identify the specific regulation or other provision of the licensing basis for which verification of compliance is sought. Where the information request would result in the establishment of a new program, including testing or analysis, or an extensive study using new criteria, in order to develop the information required, the provisions of 10 C.F.R. § 50.109 will be followed.

Another effective means for addressing the concerns associated with information requests is to make changes to Staff procedures to incorporate pertinent recommendations of the Administrative Conference of the United States (ACUS). The Administrative Conference is the body that makes recommendations to the President, the Congress and the agencies on matters concerning the implementation of the Administrative Procedure Act. The problem of agency use of nonbinding issuances (such as generic correspondence) to impose new requirements was discussed in

Recommendation No. 92-2 of the ACUS, codified at 1 C.F.R. § 305.92-2. Included in Recommendation No. 92-2 were the following three improvements: (1) rulemaking should be used in lieu of nonbinding issuances if new regulatory requirements are intended; (2) the agency must maintain flexibility to accept alternative actions proposed by the affected parties in response to nonbinding issuances; and (3) a process allowing affected parties to appeal or otherwise seek relief from such non-binding documents should be made available. NUBARG concurs in these recommendations and supports their incorporation into Staff procedures through the following:

1. Provide guidance to the Staff that alternative actions and schedules must be considered in response to generic communications, including those issued under Section 50.54(f).

2. Provide an informal process (e.g., through revisions to the NRC's Management Directive 8.4 on backfitting) for a licensee to seek relief from the requested actions of a generic communication, where, for example, the actions would impose a substantial burden on the licensee without a comparable safety benefit.

NUBARG appreciates the opportunity to comment on the RRG's important initiative. Please contact us if you desire further information.

Sincerely,

Jan A. MacGregor
Counsel to the Nuclear Utility
Backfitting and Reform Group

NUBARG Members

Carolina Power & Light Company

Centerior Energy Corporation
(representing Cleveland Electric Illuminating Company and Toledo Edison Company)

Commonwealth Edison Company

Entergy Operations, Inc
(representing Arkansas Power & Light, System Energy Resources, Inc., and Louisiana Power & Light)

Florida Power & Light Company

Florida Power Corporation

New York Power Authority

Niagara Mohawk Power Corporation

Northeast Utilities

Pennsylvania Power & Light Company

Philadelphia Electric Company

Rochester Gas & Electric Corporation

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NRC-93-106

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July 28, 1993

Mr. Frank P. Gillespie
Regulatory Review Group
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Gillespie:

Comments on the Regulatory Review Group Report

On May 28, 1993, the NRC made the subject report available for public comment. Wisconsin Public Service Corporation (WPSC) appreciates the opportunity to review this report and has provided specific comments to Volume 4 of the report in the attachment to this letter. Overall the subject report serves as an excellent voice for change in the current regulatory process. Wisconsin Public Service Corporation firmly supports the conclusions made by the Regulatory Review Group.

If you have any questions, we would be glad to discuss our comments with you or a member of your staff at your convenience.

Sincerely,

Charles A. Schrock

Charles A. Schrock
Manager - Nuclear Engineering

PMF/cjt

Attach.

cc - US NRC Senior Resident Inspector
US NRC Region III
US NRC Document Control Desk

LIC\NRC\28.WP

ATTACHMENT 1

Wisconsin Public Service Corporation

Comments on Volume 4 of the

Regulatory Review Group Report

I. Overall Comment:

Overall this document serves as an excellent voice for change in the current regulatory process. Wisconsin Public Service Corporation (WPSC) firmly supports the conclusions made by the Regulatory Review Group (RRG). Currently probabilistic risk assessment (PRA) insights are used only to add requirements in the nuclear industry; however, as the RRG states often in their report, PRA-based insights also need to be used to reduce regulatory burden when it is shown that such a reduction does not reduce the safety envelope of the plant. In fact, graded type approaches to issues such as that outlined for Quality Assurance (QA) by the RRG can actually improve the overall safety of nuclear plants by allowing utilities and NRC to focus their resources on equipment, systems, programs, etc. that truly impact the health and safety of the public.

The RRG discusses the flexibility that presently exists within the regulatory environment to improve plant safety while reducing undue regulatory burden. The flexibility referred to by the RRG may exist in a few select groups whom report to high levels at the NRC, but it does not exist at the working level in the NRC staff. The RRG recognizes this as evidenced by the candor expressed in this document and the public meetings that they have held over the past several months. One example of the absence of flexibility on the part of the NRC staff took place in a meeting between the Cooperative Efforts Group (a group of 4 utilities who share resources when possible in dealing with issues that are of mutual concern) and the NRC staff on June 7, 1993.

The Cooperative Efforts Group presented a graded approach for the testing of motor operated valves (MOV) in response to Generic Letter 89-10. The intent of the graded approach for MOV testing was developed to more effectively use resources to protect the health and safety of the public. The approach developed by the Cooperative Efforts Group combined deterministic and probabilistic analyses along with testing results and maintenance histories for each valve currently being tested in the respective utility's MOV programs to determine the appropriate level of testing for each valve. Although NRC personnel applauded the efforts made by the Cooperative Efforts Group, they were unwilling to allow for flexibility in responding to this regulatory requirement.

This example is provided to make an important point which is that an endorsement by the Commissioners and/or senior NRC staff for developing or allowing flexibility in regulations does not mean that this flexibility will ever make its way into the manner utilities are regulated. This is because when specific programs, rules, etc. are debated, the staff members responsible for each area will argue that their areas are safety-related and that we cannot "compromise" safety. The point that we must all try to remember is that there are different degrees of safety-related which must be taken into account when issues are regulated. The concept of something being safety-related or not safety-related ignores this continuum of safety and greatly impares the staff's flexibility. The program outlined by the RRG in applying a graded approach to QA recognizes this fact and it is for that reason that we are optimistic that change is possible.

The RRG report references the conclusions made in a 1989 Nuclear Energy Agency report. In summary the four conclusions are:

- The application of PRA provides plant management with a tool that generates insights not readily available from the traditional deterministic safety and licensing analyses.

- The existence of a PRA capability within a plant operator's organization provides for a logical framework of regulatory discussion and negotiation to be created.

- The benefits derived by plant operators are generally greatest when there is a full commitment to development and maintenance of an internal PRA capability, with minimal dependence on outside experts except for an initial technology transfer.

- The application of PRA to an existing plant has always resulted in the identification of effective ways of achieving plant safety, and has thus contributed to the overall effectiveness of plant operation.

The final conclusion made in the Nuclear Energy Agency report was that the implementation of PRA as an aid to plant safety management is directly beneficial to those implementing it in support of their plant designs or operations and to all those concerned with ensuring nuclear plant safety. We reiterate these conclusions because of our full agreement with them and to applaud the RRG for recognizing the insightfulness of these conclusions and the effort that they put forward in developing Volume 4 of this report. WPSC encourages an aggressive effort by the NRC and utilities in the recognition and implementation of these conclusions at all levels of their organization.

II. Specific Comments:

A. WPSC feels that the category termed Group 3 in the report is one which is probably not worth directing any NRC resources into at the current time. As the RRG stated, the current state of PRA technology does not support on-line configuration control through the use of a "risk meter." We are aware that some domestic and foreign utilities are pursuing this type of tool; however, we would not feel comfortable basing operational decisions primarily on plant risk (core damage or containment failure) alone. There are too many other factors that need to be considered when making operational decisions. Instead PRA should be used as one of the factors in decision making, sometimes the driving factor. We feel that the current state of PRA technology supports this philosophy, as does the RRG.

B. The RRG provides generic human reliability analysis (HRA) data for use in generic PRA applications. The purpose of supplying the generic data is to preclude the possibility of using inadequate HRA data which could erroneously mask important sequences. We agree that HRA has a dramatic effect, and that careful attention needs to be given to it's effective use. We would recommend that the RRG consider modifying their recommended screening values to be more in line with what is typically calculated using current HRA methods. Namely, we would recommend a value of at least 1E-2 be used for pre-initiator human events, a value of at least 1E-2 be used for response type post-initiator human events, and a value of 0.1 be used for recovery type post-initiator human events. We believe that the bottom threshold value of 1E-3 for all post-initiator human events per accident sequence is appropriate. Typical post-initiator human event probabilities range from 0.1 to 1E-4; therefore, the use of 1E-2 for response type actions and 0.1 for recovery type actions seem appropriate. [Pages 4-32, 4-40]

C. The RRG discusses how frequently PRA models need to be updated based on the type of applications the PRA is being used for. We feel that some combination of the outage driven and PRA driven update categories described is appropriate; however, clarification should be added on the extent of updating that is necessary. Items such as plant modifications, emergency operating procedure changes, Technical Specification changes, maintenance frequency changes, etc. need be incorporated per the outage driven or PRA driven categories depending on the application. However, the maintenance history database should not need updating on as frequent of basis. The primary reason for this is that component failure rates are typically not going to change significantly on a per year basis; therefore, the considerable personnel resource expenditure for updating equipment failure rates should be performed less frequently, for example every three years. [Page 4-30]

D. When discussing the plant data analysis, a somewhat misleading statement is made that should be modified. The RRG document states that if adequate plant documentation exists, then plant-specific failure rates, etc. are computed; however, if inadequate plant documentation exists, "generic" data must be used. Typically, generic data is used because the plant being analyzed has not experienced a particular event or the specific piece of equipment in question has yet to experience a failure that rendered it inoperable. In these cases it is much more appropriate to use "generic" data, than to use an initiating event frequency of zero or a failure probability of zero. Therefore, using the terminology of inadequate plant documentation in the report is misleading. [Page 4-16]

E. For the case of generic grouping the RRG initially recommends grouping plants in 5 categories of designs. The category that we are most familiar with is Westinghouse PWRs. This is a broad category, and it may be more appropriate to break the category down into plant size and/or vintage. One such category could be two loop Westinghouse PWRs. The four plants (6 units) that fall into this category are of very similar design allowing for easy comparisons for both the utilities and the NRC. From our perspective, a smaller category such as this would make projects/comparisons more manageable and easier to perform. This would increase the likelihood of utilities or small groups of utilities performing analyses, submitting regulatory burden requests, etc. without the need for contractor support. [Page 4-25]

F. When discussing initiating event analysis, the RRG document states that typically initiating events are modeled by a single event. This is true for many events, but there is a category of initiating events that is typically modeled differently. Namely, support system failure initiating events such as loss of service water, instrument air, etc. are modeled as fault trees, not as point values. [Page 4-13]

G. While making a point that certain deterministically important equipment is determined to be probabilistically unimportant, the RRG report uses the reactor pressure vessel as an example. We feel that this example may be inappropriate, at least for the Kewaunee PRA and certain others that we have reviewed. Failure of the reactor pressure vessel is modeled as an initiating event that leads directly to core damage and is not truncated from the final results. [Page 4-27]

H. When discussing the advances of PRA technology, the RRG report states that much, if not all, of the analysis of internal events can be performed on personal computers. The analysis of external events could be added to that sentence. [Page 4-6]

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July 23, 1993
FYC 93-018
SPS 93-069

Dr. Frank T. Gillespie, Chairman
Regulatory Review Group
Office of the Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555

Subject: Request for Comments on the Regulatory Review Group's Report to the Executive Director for Operations (58FR29012) and Comment Period Extension (58FR33285)

Dear Dr. Gillespie:

Yankee Atomic Electric Company (Yankee) appreciates the opportunity to comment on the subject report. Yankee is the owner of the nuclear power plant in Rowe, Massachusetts. Our Nuclear Services Division also provides engineering and licensing services to sponsor companies in New England and other nuclear power plants in the United States. Yankee has been specifically charged by our sponsors to aggressively pursue revisions to regulatory processes as well as relaxation of specific requirements. Many of the specific areas identified by our sponsors as deserving attention have been discussed in the subject report.

The Regulatory Review Group (RRG) is to be congratulated for an outstanding effort. The scope of review is comprehensive, and, for the time available, the depth of review of many issues is noteworthy. The innovations presented as alternatives to some of the current obsolete regulatory paradigms are both original and refreshing. Finally, the extension of the comment period to 60 days from the original 30 was a relief for those reviewing this extensive document.

Discussion

The Review Group noted, at the beginning of its report, that, during the review, special emphasis would be placed on the feasibility of substituting performance-based requirements whenever reasonable. To this end, one of the most profound statements contained in the Review Group's report appears in Section 2.3.9 under III "Regulatory Coherence" (p. 67).

> "We believe that a commitment is to the basic articulation of the requirement and not to the method by which the licensee initially stated a commitment would be met."

This statement highlights the essence of a performance-based regulatory system and at the same time the major barrier to its achievement which exists currently. Although the obligation of licensees to meet all "requirements" (i.e., rules) is unquestionably clear, the differentiation between satisfaction of the objective of a requirement and the method of carrying out this obligation or commitment has historically been very difficult. A further complication, which makes characterization of commitments much more difficult, is the practice (which has grown in the recent past) of issuing requirements in forms other than as rules (e.g., BWR Mark I containment "hard" vents).

Clarification of the distinction between compliance with a requirement and the individual licensee's prescription as to how compliance is to be achieved is a fundamental attribute, we feel, of a performance-based regulatory system. The Review Group Report provides a practical example of an approach to that problem. (Section 2.3.29 - P. 133).

> "The security plan is required. However, the plan is a demonstration of how the applicant will comply with 10 CFR 73 and is not therefore, a requirement in and of itself. Changes to the plan that do not reduce the licensee's ability to meet 10 CFR 73.55 effectively but actually reduce the resources [required to do so] should be acceptable without prior approval. The burden is then on the staff to establish not that the plan commits less resources, but that 10 CFR 73 is not being effectively met."

The variety of "commitments" that exist today makes the achievement of uniformity in this area, consistent with the above example, a difficult matter. Our contribution to this discussion is included in Attachment 2 in the response to the recommended definition for "commitment". We commend the Review Group for their insight and for their effort to deal with this issue.

Structure of the Response

There are 60 recommendations presented in the report. Most appear to be very good suggestions and should, we believe, be implemented as soon as reasonably practicable. The ones which we strongly support in the form presented are listed in Attachment 1. In this and the other attachments, the page reference is to Volume I and the section references are to the other volumes.

Several of these recommendations, however, warranted specific comments, in our opinion, because they dealt with very significant issues for which we would like to suggest alternatives or additions to the proposed action. The discussion of "commitments" is an example of this kind of issue. These comments are provided in Attachment 2.

There were a few comments contained in the report which we feel deserve qualification or rebuttal. These are discussed in Attachment 3.

Finally, one aspect of the report deserving attention are the issues not substantively addressed or fully developed. The three prominent ones in this category, in our opinion, are: Inspection Program Improvements, Seismic Design Requirements, and Generic Treatment of Relief Requests. Each of these are mentioned but none, other than relief requests, briefly discussed in conjunction with a revision to 2.802 (p.8 and Section 2.3.17), is developed to the degree that we believe they should. The following sections of this letter summarize discussions and recommendations for these issues which are included as separate enclosures.

Issues Deserving of More Attention in the Report

Inspection Program Improvements (Enclosure 1)

The inspection process is "where the rubber meets the road" so to speak for implementation of the regulatory process and, to a large degree, for manifestation of regulatory philosophy. Any modifications to achieve a performance-based system and to reduce regulatory burden must, ultimately, be expressed as changes in the NRC's inspection program.

It would be interesting and informative to obtain the Review Group's comments on the inspection program. To be effective, many of the Review Group Recommendations must have an inspection program component. Additionally, based on discussion at the Regulatory Information Conference, it is not clear that the NRC Task Force report on inspection program reform will be made available for public comment. Hence, the availability of the Review Group's comments may also be questionable. Because of the significance of the inspection program to reform, it is imperative that both of these reports be made public as soon as they are available.

Without any indication of what might be contained in either report, we have developed a series of proposals for improvement in what is acknowledged as an overly burdensome inspection process. Fully developed in Enclosure 1, these include the following suggestions:

1. More realistic criteria and controls for major team inspections should be established.

2. The NRC should perform regular reviews of the effectiveness of special issue major team inspections.

3. The NRC should improve management oversight of inspection activities to provide assurance that individual inspectors do not impose inappropriate regulatory actions or positions.

4. There should be a strong correlation between licensee performance and the inspection burden.

5. Proposed rulemaking packages and changes in regulatory guidance should also include the related changes to the NRC inspection manual.

6. The expenditure of resources for each inspection should be clearly documented.

<u>Seismic Design Requirements</u> (Enclosure 2)

Over the last 15 years, the Lawrence Livermore National Laboratory (LLNL), at the direction of the NRC, has developed and refined probabilistic models to assess the likelihood of exceeding the seismic design basis at existing nuclear facilities in the eastern United States (EUS). During the eighties, with encouragement from the NRC staff, the Electric Power Research Institute (EPRI) also developed a state-of-the-art seismic hazard methodology. Comparisons of the 1985 and 1989 EPRI and LLNL results at typical EUS sites showed dramatically different perceptions of seismic hazard, with the EPRI results being much more favorable. As a result of continued discussion between scientists, as well as evolutions in the state-of-the-art, today's LLNL results (1992) compare favorably to the EPRI results at most EUS sites.

Significantly, while the plant-specific hazard associated with seismic events computed by LLNL and, thus, the perception of seismic hazard that it connotes, has been steadily evolving in a direction of reduced hazard, the staff has continued to press for detailed seismic reviews of all plants (the seismic IPE). In fact, with the older vintage plants are being required to perform not one, but two seismic review programs (IPE/EE and a review to satisfy Unresolved Safety Issue A-46).

This topic not only should be included in the report, but deserves special commentary. As is explained in Enclosure 2, there has been a transition in the Staff's belief that the seismically induced design basis accident was a low probability event (pre-1980), to the belief that it might be highly likely (early 80's), to a consensus (LLNL and EPRI results) that this design basis accident is, in fact, a severe accident of very low probability. This issue constitutes a unique opportunity to relieve an enormous regulatory burden with no impact on the current level of plant safety.

Generic Relief Process

The Review Group Report highlighted the value of a generic safety evaluation as the means to improve the response time for licensee relief requests (p.5). A generic safety evaluation which could "envelope" many plants is a very worthwhile tool which deserves immediate development to its full potential. A few changes to the generic Technical Specification or license improvement process currently employed could result in rapid action in most cases with a significant reduction in duplication of staff efforts. This process is described in Enclosure 3.

Briefly, the process would involve:

1. Issuing a draft generic letter containing a generic safety evaluation and offering a specific improvement opportunity for public comment.

2. Asking in the draft for an expression of interest by licensees and for an estimate from them of the time required to prepare plant-specific submittals.

3. Issuing the final generic letter containing a filing deadline by which interested licensees could be involved in a "batch" for generic treatment.

4. Reviewing and processing of the individual requests by a single Issue Manager who would prepare the staff's "no significant hazard" analysis utilizing the elements of the submittals received.

5. Presenting a single notice for public comment in the Federal Register listing all the plants for which the notice has been prepared.

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6. Resolving of comments, if any, and issuing the revised pages to all the plants included in the notice.

It is our belief that nothing in the "Sholly" amendment to Section 189a of the Atomic Energy Act precludes the treatment of more than one plant at a time in the steps required to satisfy the legal requirement of the Act. Additionally, such a procedure would enhance rather than inhibit public involvement. In effect, the proposed change would be publicly available in draft form and final form as a Generic Letter and once as a public notice for all of the plants involved.

We suggest that the staff consider applying the proposed process on a trial basis. Specifically, we suggest that removal of the requirements for the Post-Accident Sampling System be utilized as the test case. The NRC has already developed the necessary technical justification in NUREG/CR-4330. We believe that relief on this issue will afford many licensees significant cost savings with respect to on-going training and maintenance requirements.

Conclusion

We are most hopeful that the Regulatory Review Group's Report heralds the onset of a fundamentally new approach to regulation; that is one in which the regulator and the licensees can work to establish a common set of performance objectives. This would set the stage for a process in which real advantage can be taken of the insights from risk analysis. The first step of this approach is acknowledgement that the current body of regulations constitute a set of requirements which go beyond the assurance of adequate protection of public health and safety. Two existing regulatory programs (Marginal to Safety, and this report) focusing at regulatory excess, constitute this acknowledgement. The next step is appropriate adjustment of the regulatory burden on the regulated community. Progress on this next step must begin immediately.

Very truly yours,

D.W. Edwards

D. W. Edwards
Director, Industry Affairs

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Enclosures

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ATTACHMENT #1

SIGNIFICANT RECOMMENDATIONS WE STRONGLY SUPPORT

- Do not revise or modify 10 CFR 50 Appendix B (p. 7) (Sections 2.3.13, 2.3.14, and 4.4).

  The implementation of Appendix B in a performance-based and graded manner can readily be accomplished by revising the guidance and implementing documents using insights from risk assessments.

- Revise 50.54(a)[3], 50.54(p)(2) and 50.54(g) to make the program change process for QA, security and emergency plans consistent (p. 9) (Section 2.3.9). These are also manifested in plant-specific recommendations (p. 15 Section A.3.2.1 and p. 16 Section B.3.2.2).

  A standardized treatment for changes to licensee programs which results in allowing changes that do not decrease program effectiveness without NRC prior approval is a worthwhile improvement. However, this endorsement should not be construed as supporting the addition of other programs to this set that are not currently included in regulation.

- Revise the frequency of fitness for duty performance information submitted from semiannual to annual (p. 10) and program audits from annual to triennial for licenses and 18 months for contractors (p. 10) (Sections 2.3.4 and 2.3.6)

  Both the reporting frequency and audit frequency are unnecessarily burdensome.

- Reconsider the need for numerous reporting requirements which are generic and plant-specific (p. 11) (Section 2.3.16).

  The three recommendations listed in this regard should be pursued to end unnecessary and burdensome reporting.

- Replace the guidance in Generic Letter 88-16 with guidance directing that the reference of specifically-approved topical report revision be in the core operating limits report (p. 12) (Section 2.3.8).

  This would eliminate an unnecessary Technical Specification change submittal.

- Upgrade the Regulatory Agenda to provide clear scheduled and accurate status information for all rulemaking activities (p. 13) (Section 2.3.17).

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A clear and current rulemaking record (at least in abstract) would enhance the value of the Regulatory Agenda for all users.

- Review existing security requirements to determine if they can be expressed in a more performance-based manner (p. 14) (Section 2.3.18).

  Security offers a good opportunity to employ performance-based requirements.

- Ensure there exists a disciplined control process to preclude introducing nonrequirements into licenses as a matter of practice (p. 14) (Section A.3.2.1, Vol III).

  This past practice has circumvented all safeguards such as the Backfit Rule, 10 CFR 50.109, and should not be permitted to continue.

- Permit line-item improvements in accordance with Improved Standard Technical Specifications for all plants (p. 16) (Section B.3.2.4., Vol. III).

  The Technical Specification Improvement Program provides flexibility to licensees on a line-item-basis without compromising safety.

- Expand the use of performance-based requirements for both license conditions and Technical Specifications (p. 16) (Section 3.2.4, Vol. III).

  The use of performance-based requirements allows the incorporation of prescriptive provisions into licensee-controlled documents where administration can remain entirely the purview of the licensee.

ATTACHMENT #2

RESPONSE TO SIGNIFICANT RECOMMENDATIONS WARRANTING COMMENT

- Define "Current Licensing Basis" and "Design Basis" (p. 6) (Details Section 2.3.10)

  The proposed definition of "Commitment" may be useful in refining the definition of current licensing basis (CLB) which appears in Part 54. The definition of CLB contained in 10 CFR 54 (Section 54.3) conveys a scope that is overly broad. (See discussion of the definition of "commitment" on Pages 6-8 of this attachment).

  Recommendation - The proposed definition of "commitment" should be adopted and the current definition of CLB should be modified to be consistent with it.

  Clarification of the term "design basis" may not be necessary. However, the results of the industry-wide design basis recovery initiative sponsored by NUMARC will both answer the question as to the need for revision as well as provided the appropriate wording changes.

  Recommendation - Any clarification of the term "design basis" should await completion of at least the majority of design basis recovery programs currently underway.

- Revision of 10CFR Part 21 to allow gradation of the rigor applied to dedication of parts based upon safety significance (p. 6) (Section 2.3.1).

  This is an important change because it will increase greatly licensees' ability to procure and utilize commercially available parts. The dedication process which qualifies them for use in safety-related applications can be tailored to the relevant safety function and the degree of safety significance for each part instead of a blanket uniform review. The result is a much more cost-effective program. NUMARC has proposed a specific set of wording changes which accomplish the review group's recommended changes in a very effective manner (Petition for rulemaking filed June 21, 1993).

  Recommendation - Adopt the changes proposed by NUMARC into 10 CFR 21.

- Develop PRA methods for use in 10 CFR 50.59 evaluations (p. 7) (Section 2.3.19).

  This recommendation, though made more clear in Section 2.3.19, could be interpreted as an ADD-ON to 50.59 evaluations. We believe that 50.59 evaluations are currently being performed adequately and need no more required analysis. The development of <u>guidelines</u> (not "methods" as stated on page 7) might be helpful for the use of PRA as an <u>optional</u> technique for portions of the evaluations. We have found that often the best use of PRA insight is as an overview mechanism to check for reasonableness of the conclusion after a 50.59 evaluation has been completed. This has not been needed, however, for all evaluations.

<u>Recommendation</u> - Clarify the above recommendation in the report th... PRA should not become an add-on requirement.

- Redirect the Marginal to Safety Program to be only a response to petitions program (p. 8) (Section 2.3.17).

The report recommends that the Marginal to Safety Program focus on and be responsive to "specific and detailed petitions for rulemaking that are performance-based, propose to eliminate regulatory burden, and are safety neutral." The report further recommends that "the industry should take advantage of the petition for rulemaking process and submit complete, technically sound petitions in accordance with the NRC Regulatory Analysis Guidelines that the staff could publish expeditiously as proposed rules and request public comments." We concur that it is reasonable to look to the expected beneficiaries of improvements in the regulations to play a major role in the development of proposals and supporting justification for said proposals.

However, regardless of the quality and degree of detail provided in a rulemaking petition, commitment of NRC resources is still required to evaluate and process such petitions. At the recent Marginal to Safety Workshop and at the Regulatory Information Conference we were unsuccessful in obtaining any clear indication of the resources that the NRC can make available to support this activity. If licensees are to expend their resources in developing rulemaking proposals, there must be a reasonable expectation that these proposals will be processed in a timely manner. Our concerns have been heightened by a recent newsletter account of a meeting between the NRC Staff and NUMARC representatives to discuss the Marginal to Safety Program. The newsletter quoted the Staff as indicating that they only have resources to work on proposed changes to 10 CFR 50, Appendix J and <u>one</u> other issue.

With respect to the question of developing rulemaking proposals which have sufficient detail and bases to expedite the rulemaking process, the NRC Staff has been working on revisions of NUREG/CR-0058, "Regulatory Analysis Guidelines for the U.S. Nuclear Regulatory Commission" and NUREG/CR-3568, "A Handbook for Value-Impact Assessment" for an extended period of time.

<u>Recommendation</u> - Timely completion and publication of pending revisions to NUREG/CRs-0058 and -3568 would significantly assist those who intend to develop rulemaking petitions. In

addition, NRC resource commitment and accountability for results (i.e., relief actions completed in a timely fashion) must be a prominent part of any program modifications.

- Utilize an "integral approach" to licensing actions (p. 8) (Section 2.3.17)

On Page 132, the report suggests that amendments could contain a number of modifications that "... overall have no effect on the current level of safety at the plant." The idea appears to be a very good one. It would allow the tradeoff of various requirements to maintain a level of safety represented by the plant as currently licensed, but would permit increased flexibility in operational decisions. An example might be that the allowed outage times for a particular piece of safety equipment could be extended to permit maintenance with insignificant impact on at-power calculated risk. This, in turn, would increase availability of that equipment during shutdown. The net effect would be to reduce shutdown risk and, thereby, reduce overall plant risk.

However, this type of example presumes that the safety envelope of importance is the total risk presented by the plant and that inter-mode tradeoffs are possible. However, review of Volume Four "Risk Technology Application" indicates that the type of tradeoff envisioned by the Staff may only be intra-mode. (pages 4-36 and 4-37.) Since PRA methodology represents an effective way to quantitatively estimate the "level of safety" for a plant (and changes thereto), and, since the specific treatment of tradeoffs in the PRA section of the report appears limited to single mode (e.g., during operation), the concept of an "integrated approach" does not appear to have nearly the usefulness expressed in the summary explanations in Vol. I.

Recommendation - Clarify the concept of "integrated risk approach" in a manner that is consistent throughout the report.

- Revise 10 CFR 2.802 to clearly distinguish between Petitions for Relief versus those for Health and Safety reasons (p. 8) (Section 2.3.17)

The suggestion, which is detailed on P. 131 would "... draw a distinction between a petition for rulemaking proposed for public health and safety reasons and one that is made to eliminate burden." The recommendation goes on to describe that the accompanying guidance for the regulation should outline an acceptable threshold of information and the detail needed for each type of petition. Although not stated, the inference could be drawn that there would necessarily be higher information threshold to introduce a petition for change to the regulations which would provide relief to licensees in some way as compared to the threshold to petition for a change which might impose further regulatory burden. This implied dual standard, presumably tilted toward increased regulatory requirements, seems entirely inappropriate. The justification necessary to support the change in the regulation should be measured to a single standard for level of information and detail, independently of the possible effect of the change. The level of detail and the nature of information required for backfitting of a requirement is clearly established in regulation. It would seem that, with regard to level of detail and the kind of information necessary, a petition for removal of a requirement should conform to a similar standard. We have never experienced difficulty in developing a petition in conformance with 2.802 as it is currently written.

This is not to say that a well-founded and adequately prepared petition for relief should ever be "hostage" to a process which, by its nature, delays appropriate actions. We would endorse a change to 2.802 which had, as its result, assurance of expedited relief from an unnecessarily burdensome requirement which has little or no safety benefit.

Recommendation - This suggestion should be modified as suggested in the paragraph above.

- Add a definition for "Commitment" to 10 CFR 50.54 (p.9) (Section 2.3.2)

The term "commitment" does appear in the regulations. Though not defined there, a "commitment" is mentioned, incidentally, in Appendix C to 10 CFR 2 (VI.D. Related Administrative Actions). The inference in this section is that a commitment is different from an "obligation" and may not be "a legally binding requirement". Though, perhaps useful for some purposes, this type of connotation is not as useful as it could be for the conceptual development of a performanced-based regulatory system.

For the purposes of conceptual development, a commitment could be viewed as the licensees' recognition of the obligation to satisfy the regulations and should stop with that. In that construct, a licensee would commit to meet each applicable regulation and offer an explanation as to how compliance would be accomplished. The staff would review the proposal and agree or disagree in each case that the requirement had been satisfied. This interaction would, no doubt, be iterative, but eventually the licensee would have staff agreement that the regulations were satisfied - thus, all commitments would be fulfilled. The license would be obligated to maintain the means for achieving compliance into the future. If an alternative means presented itself, and the licensee could support the substitution with the assurance that the alternative was no less effective in meeting the commitment, then the means could be changed, presumably without prior staff review and approval. At some point the licensee would report the change, probably in conjunction with the FSAR update.

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Inspection and enforcement would remain the independent mechanism for the agency to assure that commitments were maintained. However, the inspection process would not (contrary to the present situation) provide a pathway for introduction of alternatives or additional means for satisfying commitments. A legitimate area for review, however, would be changes in implementation as discussed above.

Unfortunately, this simple model is not entirely practicable because not all applicable, and indeed, necessary, safety requirements are contained within the regulations. (e.g., BWR Mark I hardened vents, or the entire array of TMI backfits). Again, unfortunately, the impetus for this quandary has come as much, or more, from the industry than it has from the regulator. So, commitments come in different "flavors", and there has not existed, until the review groups' attempt, a clear expression of what a commitment should be and what it should not. Consequently, for the foreseeable future, the concept of commitment must include requirements and "requests" as explained in the report.

Additionally, the definition must include, as has been proposed, some element of staff acknowledgement that the commitment is an integral part of a safety decision. This provides the potential for "line drawing" with respect to the nature of a safety decision. At present, every intended action described by a licensee is potentially viewed as an unalterable commitment. The distinction between the commitment to meet a requirement and the means to implement that obligation need to be distinguished from one another. The first discriminator should be whether the associated decision which the staff must make based on the information provided by the licensee specifically relates to the satisfaction of a requirement (or the present set of "requests" that should have been requirements). The second discriminator should be whether the decision being weighed by the Staff relates specifically and solely to the acceptability of the means to satisfy the requirement. If the decision involves staff preferences or comparisons between means employed by different licensees, or a matter other than a specific requirement, the licensee proposal or intended action should not be considered a commitment over which the staff exerts approval authority. The two tier commitment proposed in the report appears to be a reasonable approach. Alternatively, a separate term could be coined for this category.

The Review Group Report attempts this differentiation on Page 9 "...where [if] a commitment is not significant enough to be elevated to a regulatory status..." licensee controls should govern the change process. The Group's proposal in this area has merit and deserves careful evaluation, discussion, and debate to facilitate incorporation of a final definition into the regulations.

<u>Recommendation</u> - Offer the draft definition for commitment as a proposed addition to 50.54 and obtain additional

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public comment on the issue. Integrate the resulting definition and concept into the "current licensing basis" definition in Part 54 of 10CFR.

- Consider revision of 10 CFR 50.82 and Regulatory Guide 1.86 to address inconsistencies and define the possession only license (p. 10) (Details Section 2.3.3).

These proposed revision activities are both long overdue and very important. The information in Section 2.3.2 deals with the type of license which NRC must issue for decommissioning. Since there is a potential for hearings in any of the three scenarios discussed (i.e., amendment, new license, and renewed license) the type may not be significant to a licensee. The amended, new, or renewed license would be issued in concert with approval of the Decommissioning Plan and issuance of the Decommissioning Order. More importantly the type of hearing process that is used deserves careful consideration.

Given that the risks to the public health and safety during decommissioning are substantially less than those associated with power generation, and given that it is in the best interest of NRC and the public for licensees to maintain a possession only license until the plant is decommissioned, any hearings on a Decommissioning Plan should be far less formal than those of the full adjudicatory process. The legislative type hearings conducted by EPA, who typically (in one outing) listens to comments from all parties and then renders a decision, appears much more preferable to a protracted adversarial adjudication. The Office of General Counsel, in SECY-92-382, has also suggested other types of "hearings" to be used for the approval of Decommissioning Plans.

Recommendation - Revise and update the regulations and the Reg Guide to incorporate the following provisions for a Possession Only License (POL):

1. Specific acknowledgement that the following regulations do not apply to a POL holder.

   50.44, Combustible gas control system
   50.46, ECCS
   50.48, Fire protection
   50.49, EQ
   50.55a(g), Inservice inspection
   50.60, Fracture prevention
   50.61, Fracture toughness
   50.62, ATWS
   50.63, Loss of alternating current power
   50.65, Maintenance
   50.71, Maintenance of records, making of

reports
50.120, Training and qualification of NPP personnel
Part 50, Appendix G, Fracture toughness
Part 50, Appendix H, Materials surveillance program
Part 50, Appendix J, Containment leak testing
Part 50, Appendix K, ECCS
Part 50, Appendix R, Fire protection
73.21, Protection of safeguards information
171.15, Annual fees

2. Make specific revisions to the following regulations:

50.47 Emergency Plans
(identification of the individual sections that continue to apply - Change conclusion in Vol. II, App. R, P.134)

50.54 Conditions of License
(identification of specific sections that continue to apply - Add to list in Vol. II, App. A, p. 149)

50.59 Changes, Tests and Experiments
(acknowledgement that this regulation is applicable to the POL situation - Identified in Vol. II, App. A, p. 167)

Part 140 Financial protection requirements and Indemnity Agreements
(explicit reduction of primary liability coverage and exemption from secondary liability coverage)

In addition, the NRC should adopt a legislative hearing format for the POL process in 10CFR2.

- Address design basis testing and test frequency based on risk techniques (p. 10) (Section 2.3.7).

It is not entirely clear what is intended by the recomm[illegible]on. Certainly, the efforts by the staff to facilitate uses b[illegible]icensees of updated code editions and addenda has been very worth while. As the report points out, safety has benefitted by the work. Certainly, any refinement of testing frequency based on risk techniques should have a net benefit. We believe that this will also provide a benefit to safety, and we encourage and support efforts in this direction.

Implementation of In-Service Testing (IST) at plants has been very costly and has resulted, in some cases, in unanticipated system transients due to the unusual configurations required. As revisions to codes continue and more systems are added to the scope of testing, plants will be required to develop more testing procedures for systems not originally designed to support testing of this nature. Plants usually perform these tests with valving configurations (in lieu of modifying the system). In some cases these configurations can introduce more risks to equipment than benefits from the test results. Continued extension of testing in this regard could lead to an incident of equipment damage at a facility.

This issue seems to center on the suggestion that more testing should reflect accident conditions rather than normal conditions (usually such testing is conducted with the plant shutdown and system conditions are not bounding ones). Implementation of simulated accident condition testing at operating plants will be tenuous without, in many cases, major system modifications, for which the cost/benefit appears uncertain, at best. The industry has accumulated much actual experience, at a great deal of expense, during the implementation of the MOV testing program. As the industry continues to try to test MOVs under design basis conditions, either through in-situ testing or by means of diagnostic testing, it is becoming apparent that in-situ design basis testing should not, in general, be performed on an operating plant.

The original design verification test program conducted during construction and startup may offer the best opportunity for some boundary condition testing using plant systems. In the ideal, verification of component capability to perform under design basis conditions should have been performed by the equipment manufacture as part of the procurement process. Once these conditions have been demonstrated, during this initial test, testing of the component at the operating facility should be restricted to normally expected operating ranges.

An example of this ideal condition would be as follows: A new pump and motor purchased from the manufacture would be required by the procurement specification to be tested under all aspects of design basis conditions. This testing would verify that the pump and motor will perform to specification. The data obtained from the testing (i.e., pump performance curve) could then be used at the plant to verify that it is still performing within original design conditions but the data need not be taken under boundary conditions.

There may be critical components where this process was not followed or not followed rigorously enough. In those cases, great effort should be expended to obtain test data in test facilities. Expanding ISI testing to address this shortcoming is not a preferred approach.

Recommendation - Approach the imposition of design basis testing carefully such that test conditions requiring bounding design are not imposed into power station In-Service Test (IST) programs.

- Add a definition for "alteration" to 50.2 to clarify 10 CFR 50.23, 50.45, 50.56, and 50.92(a)(p.12) (Section 2.3.10).

The suggestion that the term "alteration" be made synonymous with the "design basis" is entirely inappropriate. The term "alteration" generally refers to "material alteration" which has its origin in the Atomic Energy Act and means a conceptual change to an already licensed facility. There have only been two material alterations since the beginning of licensee history: (1) Redesign of a zero power research reactor to a 50 MW design and (2) redesign of the West Valley fuel reprocessing facility to triple its output (the Construction Permit proceeding begun to facilitate this alteration culminated in the demise of the facility).

Since a material alternation, by statute, requires the issuance of a new Construction Permit (and the attendant adjudicatory hearing process), there should be no connection whatsoever between the term "alteration" and the term "design basis". Design basis is, and will remain, a detailed description of the engineering implementation of the licensing requirements. It will continue to change over the life of a facility for numerous reasons such as vendor product line changes. None of these types of changes even remotely approach the realm of a material alteration in the context of the AEA.

Recommendation - This proposal should be dropped.

- Policy statements should be deleted (11 cited) or revised into rulemaking proposals (6 cited) (p.13) (Section 2.3.11)

This "housecleaning" should be pursued on a time available basis, with the exception of the statement on "Availability and Adequacy of Design Basis Information at Nuclear Power Plants." This policy statement contains all the trappings of a full-blown requirement complete with inspection and enforcement provisions. We objected vigorously when this statement was proposed in draft form (58FR15885) on the basis that the response it required imposed a significant burden on licensees and would result in providing information generally available to the staff but in a different format. Additionally, we maintained that the request had not been justified to warrant the burden (Enclosure 4). We agree with the Review Group's assessment that the entire regulatory program such as was envisioned by this policy statement should have more substantitative regulatory underpinnings (i.e., receive the careful scrutiny and public debate associated with a rule). We disagree strongly, however, that this particular statement should become a rule. The industry initiative

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to undertake design basis reconstitution programs in conformance to the NUMARC guidelines for this activity obviate the need for any NRC requirement or guidance in this area.

Recommendation - Delete the Policy Statement on "Availability and Adequacy of Design Basis Information at Nuclear Power Plants" as duplicative of existing industry programs.

- The training and retraining in behavioral observation for aberrant behavior should not focus solely on substance abuse induced behavior (p. 13) (Section 2.3.5).

In our comments on the staff's proposed reduction in the random testing rate we highlighted the value of supervisor behavior observation programs in terms of detection and deterrence. (Enclosure 5). This recommendation is consistent with our belief that these programs deserve acknowledgement as beneficial in assuring a highly reliable work force. (As noted in our comments, the proposed high rate of random testing (50% for employees and 100% for contractors) does not accomplish this end.) The notion that aberrant behavior can only be substance related is (as noted in Section 2.3.5) indeed, unfortunate and needs to be corrected. However, we agree with the Review Group's assessment that <u>no further requirements</u> are necessary, but merely a clarification in intent via generic correspondence.

Recommendation - Issue the proposed Information Notice

- Provide a discussion of the Regulatory Research prioritization system in the preface of the Regulatory Agenda (p. 13) (Section 2.3.17).

The whole subject of prioritization methods deserves consideration. The methods employed by the agency to set priorities for regulatory focus should be readily understood by both employees and licensees. Describing the method in a readily accessible public document is a minimal step. The method itself deserves evaluation from the standpoints of ease of use, ease of understanding, comprehensiveness of weighing factors (health and safety as well as utilization of agency and licensee resources), comparative importance between issues and ease of relative ranking. The tool of risk analysis can facilitate some of these comparisons.

Prioritization methods used by some licensees may provide useful input in this regard. The point is to be sure that the prioritization method is relevant and useful and then that it actually is used.

Recommendation - Review the prioritization method for rulemakings for consistency with agency-wide methods.

- Drop any rulemaking for which staff resources are not anticipated to be available (p. 14) (Section 2.3.17).

These intended rule changes should probably not be dropped unilaterally. The intended improvement to the regulatory agenda (p. 13) should permit explicit discussion of the subject rulemakings. Public input should then be sought on the advisability of discontinuing these proceedings. Should the industry, or other members of the public, feel that any of these warrant continued effort, the mechanism discussed on Page 8 of the Review Group's report, industry development of a rule change package, should be considered.

Recommendation - Avoid any unilateral dropping of intended rulemaking activities.

ATTACHMENT #3

ISSUES RAISED WHICH WARRANT CRITICISM

- Section 2.3.4 of the report appears overly critical of the industry's efforts, or lack thereof, to "take full advantage of the available options for enhanced flexibility..." (P.20). In the opinion of our fire protection experts, the NRC Regional Offices often do not appear to have qualified fire protection engineers on staff who can understand and interpret the myriad of very complex NRC regulations and guidelines and the fire protection codes/standards applicable to a specific licensee. As a consequence, the guidelines and the regulations are sometimes inconsistently interpreted and applied. This practice often reinforces the perception that, as a matter of common practices, licensees will be penalized via inspection for any program reductions.

  One additional recommendation that might stem from this comment is that qualified fire protection specialists be readily available to each of the NRC regional offices.

- Section 2.3.5 endorses the Staff's proposed rule change for a reduction of random testing from 100% to 50% only for utility employees. We have strongly disagreed with this reduction urging a change for all nuclear plant workers to 10% on the basis of the effectiveness of behavior monitoring programs. (Enclosure 5).

- Section 2.3.10 contends that the staff has interpreted the terms alteration or material alteration to mean a modification of the design basis. We are unaware of any instances when the synonymous definition has been used and, if it has, we believe that it is erroneous. Material alteration is an extensive and fundamental modification to a facility such that it is so different that the licensing process must be repeated beginning with a new Construction Permit.

- Section 2.3.13 states "Manufacturers apparently do not find that an Appendix B QA program is in their economic interest for this relatively small market." (p. 91) because ".... a broad base group of vendors and suppliers with Appendix B programs does not exist." (p. 95).

  These statements appear to be a bit too absolute. For there are currently 300 "nuclear" vendors with Appendix B QA programs being audited on behalf of NUPIC. Many of these vendors are original equipment manufacturers (OEMs) and other vendors who remain committed to the nuclear industry and 10 CFR 50, Appendix B. In addition, Yankee utilizes another 40 to 50 "nuclear" vendors whose programs include commitments to 10 CFR 50, Appendix B; we are probably typical of most utilities. We do agree, however, that large numbers of vendors have fled the "nuclear market" for what they perceive as high hassle and high cost to them for what is a small segment of their overall customer base.

Efforts, now, to restore reason and remove intrusive hassle may be too little too late but, undoubtedly, should be pursued.

- Section 2.3.17 attempts to define the term "marginal" for use in a threshold to safety context (P. 131). It proposes a definition of marginal as a 10% increase in calculated risk. The basis for this proposal is the potential negative perception that could attach itself to a change which is larger in absolute terms (e.g., 50%) even though such a change may be insignificant in actuality. Perhaps a fully developed plant PRA analysis which includes a complete uncertainty analysis would be able to accommodate evaluations at a 10% level in most instances. However, as the report explains, most PRA analyses cannot accommodate making distinctions at the 10% level. An attempt to draw an arbitrary "line" in this fashion probably will not yield a beneficial result.

  The report concludes, rightfully, that the marginal to safety effort should not focus on a specific risk number below which effort is deemed "marginal to safety" but should focus on qualitative and performance-based criteria. We agree that this seems like a much better approach although, in some cases, quantitative PRA results provide valuable input.

  The more significant issue of this section is the utter inappropriateness of NRC staff to set a level of acceptability based upon their expectation about public perception. As a matter of policy, the NRC staff should be constrained to base decision criteria and all decisions on sound technical bases only.

- Section 4.4.5 describes graded QA implementation as "defining different categories of implementation for the SSC's commensurate with their relative importance." It explains a hypothetical diagram of the graded approach process for classifying relatively important and relatively non-important SSCs. It ends with the phrase "any application should not violate the defense-in-depth philosophy." Apparently defense-in-depth is perceived as not being an element of the significance weighting which determines "relative importance." Such a perception is entirely invalid. Further, and most significantly, this line appears to virtually invalidate the entire discussion, indeed, the entire concept by permitting an escape.

  Introduction of an "escape clause" such as this into the approach (technically invalid though it may be) undermines the entire thrust of Volume IV. If the intent of the report is to establish a graded approach to determination of adequacy of safety measures based upon relative risk significance, then the report should project that concept as clearly as possible. To do less, as in this case, severely wounds the initiative!

Enclosure 1

Recommendations for Improvement of the NRC Inspection Program

Introduction

The purpose of this Enclosure is to offer recommendations to improve the NRC's program for the inspection of power reactor licensees. We believe that licensees generally recognize that the inspection program is a necessary element of the regulatory environment; however, there are a number of potential reforms of this program which would reduce the burden imposed on licensees and provide for fairer and more effective utilization of the NRC's resources.

The choice of Vermont Yankee and Maine Yankee for parts of this evaluation was dictated solely by the availability of detailed inspection information on these two particular plants. Aside from the fact that both are single units and located in the same region, these are essentially random selections.

Recommendation Number 1: The NRC Should Establish More Realistic Criteria and Controls for Major Team Inspection

NUREG-1395, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities," reported the results of a survey conducted by the NRC staff in the Fall of 1989 at 13 utilities. The stated purpose of the survey was to obtain feedback from utility personnel on the potential safety impact that NRC licensing and inspection activities were having on plant operations. One of the many problems identified in this report was the significant time and resource burden imposed by NRC team inspections.(1)

At the 1991 NRC Regulatory Information Conference, it was indicated that as a result of the impact survey, one of the major areas for improvement would be the "scheduling and control of inspections, especially team inspections."(2) Subsequently, this policy change was formalized in the following addition to the guidance in the NRC's Inspection Manual for coordination of NRC visits to reactor sites:

> "In light of the potentially significant impact to licensees caused by major inspections and non-inspection activities the regional office shall ensure that no more than four major activities are conducted during any SALP cycle without approval of the Deputy Regional Administrator following coordination with the Associate Director for Projects, NRR. Major activities are

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(1)NUREG-1395, (March 1990), pp. A42-A43.

(2)NRC Regulatory Information Conference, (May 7-8, 1991), Abstracts, p. 65.

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defined as visits to commercial reactor sites by a group of four or more personnel for five or more days for the purpose of inspection, conducting research information visits, conducting licensing audits, or other activities requiring significant licensee input or interactions. This restriction does not apply to operator licensing or requalification examination visits, emergency preparedness visits, or reactive team inspections. In addition, visits by senior NRC management officials shall not be subject to these restrictions."(3)

Although the promised improvements in scheduling and control of inspections could contribute significantly to reducing the regulatory burden, there is evidence to suggest that implementation of this policy has not been particularly effective. There appear to be at least two elements contributing to this lack of effective implementation: first, the accuracy of NRC management's data collection and analysis is suspect, and second, some inspectors appear to be circumventing both the spirit and the letter of the stated policy.

Clearly, if NRC management is to provide proper oversight, they must have a reasonably accurate understanding of the activities of their personnel. In the last three Regulatory Information Conferences, slides were presented which were intended to show the trend in the number of team inspections. (See Attachments I, II, and III.) When one compares the information presented in these slides it is immediately obvious that the NRC's own data are not consistent from one year to the next. Note the variation in the number of team inspections for the period April 1990 through September 1990 reported at the 1991 Regulatory Information Conference and the 1992 Conference and likewise, the variation in the number of team inspections for the period October 1991 through March 1992 reported at the 1992 Conference and the 1993 Conference. In addition to this lack of consistency in the NRC data, the total number of reported team inspections appears to be implausibly low.

In order to assess whether the NRC's numbers are reasonable, we have undertaken a detailed review of the recent inspection histories of two plants, Maine Yankee and Vermont Yankee(4). The results of this review are provided in Attachments IV and V. Note that the indicated team size does not include participation by the resident inspectors at these sites. Attachment VI provides a comparison of the number of team inspections reported by the NRC staff for the total population of plants i.e., 115, to the number of team inspections for the two sample plants during the same reporting periods. In

(3)NRC Inspection Manual, Chapter 0301, Subsection 06.04 (9/30/92).

(4)See Introduction.

assessing the number of major team inspections we referred to the criteria stated in the NRC Inspection Manual. Although the Inspection Manual indicated that operator licensing or requalification examination visits and emergency preparedness visits are not to be counted in the major team inspection category, they have been counted in the summary of all team inspections for Maine Yankee and Vermont Yankee. Also, reactive team inspections were not included in any of the team inspection summaries for Maine Yankee and Vermont Yankee presented in Attachment VI.

If we look at the team inspection data for the total population of plants for the last three six month periods presented in Attachment II, i.e., October 1990 through March 1992, there were 180 team inspections of which 79 were categorized as large (or major). During this same period, Maine Yankee and Vermont Yankee had 20 team inspections of which 5 were large (or major) inspections. If NRC's data is correct, then this analysis indicates that 1.7 percent of the total plant population has borne the burden of 6.3 percent of all large team inspections and 11.1 percent of all team inspections of every category. Likewise, during their most recently completed SALP cycles both Vermont Yankee and Maine Yankee were the subject of four major team inspections, the upper limit specified by the NRC Inspection Manual guidance. If one assumes the average SALP cycle duration is approximately 18 months and uses NRC's inspection data for the last 18 months, on average one would expect less than one major team inspection per SALP cycle, i.e., 79/115 ≈0.69. We can find no reasonable explanation for why Maine Yankee and Vermont Yankee both should be such apparent outliers with respect to the NRC team inspection burdens.

In addition to the questionable NRC data concerning trends in team inspection burden, certain NRC staff practices have the effect of imposing a major team inspection burden without being acknowledged as such. For example, during Maine Yankee's last SALP cycle a team of five inspectors were on-site for four days for a safeguards inspection. Because this does not meet the NRC Inspection Manual minimum criteria of four inspectors for five days, it does not count as a major team inspection. However, in terms of licensee burden, there clearly is no substantive difference between dealing with five inspectors for four days rather than four inspectors for five days. To cite another example, one inspector appeared at the Rowe site for seven straight weeks for the stated purpose of closing out issues raised in the maintenance team inspection for this unit. In our judgement, the activities by this inspector certainly had the same impact on plant resources as a major team inspection. (It should also be noted that no deficiencies in Rowe's maintenance program were ever identified).

In recent industry meetings, other utility representatives have expressed similar concerns about NRC team inspection practices. These practices have included characterizing multi-inspector activities as "audits" or "follow-ups" in denying that the activity can be described as a major team inspection. Also, there is a sense that the use of three-person teams is more prevalent to avoid the Inspection Manual's criteria for major inspection activity.

To address the issues discussed above we offer the following specific changes:

1. The criteria for determining what constitutes a major team inspection should be redefined to more clearly identify the real impact of a multi-inspector activity. We suggest that a major activity be defined as two or more inspectors on-site for 160 or more person-hrs. In Recommendation Number 6 of this paper, we recommend that each inspection report include a summary of the total NRC resources utilized in the inspection. This summary would be the logical place for the NRC to also indicate if it is believed that the subject inspection is, in fact, a major activity.

2. The NRC should establish criteria for a minimum elapsed time between major team inspections. Current NRC practices seem to concentrate major team inspections in the final months of the SALP review cycle. (See Attachments IV and V.) Overlapping team inspections or team inspections occurring in rapid succession almost invariably impose a disproportionate burden on licensees. Better utilization of the NRC's Master Inspection Planning System as recommended later would be another means of dealing with this problem.

<u>Recommendation Number 2</u>: The NRC Should Perform Regular Reviews of the Effectiveness of Special Issue Major Team Inspections

Because of the very significant impact of major team inspection programs on the resources of both the NRC and licensees, we believe that the results of these inspections should be reviewed on a regular basis to assure that there is a safety benefit commensurate with the resource expenditure.

We are aware that the staff is currently planning a program effectiveness review of the Service Water System Operational Performance Inspections (SWSOPIs) and applaud this undertaking. However, we are concerned that the timing of this projected review is such that its value will be minimized. It was indicated at the May 1993 Regulatory Information Conference, that the SWSOPI review would occur when approximately one half of the planned inspections have been completed. By the time this review is completed and any recommendations implemented, only a minority of the licensee population will be able to benefit from the review.

In the future, we recommend that the staff undertake the effectiveness review much earlier in the program. We believe that a sample of ten to fifteen inspections should provide sufficient data to determine if continuing the program would be productive and, if the program is to be continued, what changes in scope are appropriate. This would also allow the NRC to quickly communicate significant findings to the remaining plant population. If after this first review it is determined that continuation of the program is warranted, we recommend that another review take place after the next ten plants are inspected. With a data base of twenty to twenty-five inspections, it should be possible to identify all generic issues and evaluate the program ffectiveness with a high degree of confidence.

We offer the following specific changes to increase the effectiveness and the efficiency of this program review:

1. The staff should be encouraged to expedite issuance of the team inspection reports. Our experience is that issuance of the team inspection reports takes approximately two months. The licensee who has been inspected can quickly obtain an indication of any potential problems during the inspection exit meeting. However, other interested parties e.g., INPO, NUMARC, other licensees, and other NRC staff, typically ely on the written report for communication of important inspection results.

2. Licensees and the public should be encouraged to comment on the results of the program effectiveness review. Providing a comment opportunity need not delay implementation of any program changes recommended as a result of the effectiveness review.

3. The staff should be encouraged to expedite the pilot program for auditing selected licensee assessments as a substitute for a direct NRC team inspection for special inspections such as the SWOPIs. We strongly endorse this initiative and suggest that the report of the results of the program effectiveness review could include specific guidance for scoping licensee self-assessments to assure that the goals of the direct team inspection are met.

Recommendation Number 3: The NRC Should Improve Management Oversight of Inspection Activities to Provide Assurance That Individual Inspectors Do Not Impose Inappropriate Regulatory Actions or Positions

The present NRC inspection process offers NRC Inspectors considerable latitude and opportunities for individual interpretations of NRC regulations. As noted in Recommendation Number 5, even the NRC's Inspection Manual which is to "ensure quality, uniformity and effectiveness of inspections and to present a well-defined base from which evaluations of licensee activities and overall performance can be made," is not subject to review and approval by the NRC's Committee for Review of Generic Requirements. As a result, there are numerous opportunities in the implementation of the Inspection Program for the imposition of inappropriate regulatory actions or positions.

The Backfit Rule, 10CFR50.109, provides a means for licensees to formally appeal such actions. However, since the adoption of the Backfit Rule in 1985, NRC management has recognized that staff retaliation for a licensee's raising a backfitting concern could be a problem. See, e.g NRC Backfit Workshop, Regions IV and V, April 29, 1986, at Tr. 136-139 (statement of Mr. Eisenhut). Concerns about perceived retaliation were also expressed during the Regulatory Impact Survey.

We believe that formal backfit appeals by affected licensees should not be the sole means of redressing such situations; the NRC should provide a proactive oversight function to identify and minimize such inappropriate aspects of the inspection process. We believe that this could best be accomplished by the Office of the Inspector General preparing an annual report assessing the compliance of the Commission's Offices and employees with the provisions of 10CFR50.109. This report, which would be available to the public, would be expected to clearly document the scope of the audit activities and the bases for concluding whether or not full compliance with 10CFR50.109 is being achieved. Also, it would be expected that the report would recommend corrective actions and program enhancements to improve compliance, where appropriate.

Recommendation Number 4: There Should Be a Strong Correlation Between Licensee Performance and the Inspection Burden

Given that available inspection resources are finite, one would expect that those units that are performing better than average would be subject to less inspection than those units with perceived problems. However, a review of NRC SALP scores and the distribution of inspection hours suggests that no such correlation exists in present inspection practices. Furthermore, there are significant regional differences that are difficult to understand.

In Attachment VII we provide a comparison of the NRC inspection effort by reactor unit for the period January 1992 to December 1992 and the units' SALP scores for the preceding assessment periods most closely matching this time frame. The inspection hours information is taken from the Proceedings of the May 1993 Regulatory Information Conference. Only data for operating units are included.

We note that Diablo Canyon 1 and 2, units which are identified by the NRC as having a "sustained high level of safety performance," were inspected more intensively in 1992 than Dresden 2 and 3, units included on the NRC's "problem plant list." Likewise, Grand Gulf and Summer, two more of the plants identified as superior performers, had greater inspection burdens than "problem plant" units at Dresden and South Texas. These are some of the more obvious inequities.

We believe that the lack of correlation between performance and inspection burden on an overall basis is shown in Attachment VIII and Figure 1. From the information provided in Attachment VII, we have calculated the mean inspection burden for peer plants (i.e., the same SALP ratings) and listed the highest and lowest inspection burdens for each peer group. Although the mean inspection burden does show an upward trend as performance decreases, the range of inspection burden in each peer group (excepting the two St. Lucie units with a SALP 1 average) is extreme. We further note that the distribution of total inspection hours about the mean for each peer group is very broad and the indicated highs and lows for each peer group are definitely not isolated anomalies. We also believe that the extreme differences in inspection burden for similar performing plants are not an isolated, one year phenomena. We reviewed the inspection burden data presented at the previous Regulatory Information Conference[5] and found that the extremes within the peer groups persist, i.e., the difference in burden for individual peer plants did not tend to average out.

Another curious anomaly is the variation in the average number of inspection hours per plant when compared on a regional basis. From Attachment VII it can be seen that in 1992, the average inspection burden for a Region II plant was 3670.4 hours, whereas for a Region IV plant the average burden was 5066.2 hours. Even if Regions IV and V are excluded from this comparison because of the fewer number of units and the geographical scatter, the average burdens for Region I and Region III plants were respectively 22% and 18% higher than the average Region II plant burden. The Region II average plant inspection burden was also consistently lower than the other regional averages for the preceding year.

In order to provide for a more intelligible and equitable distribution of the inspection burden we recommend that the NRC undertake a conscious effort to link licensee performance and inspection burden and implement this policy through better utilization of the NRC's Master Inspection Planning System (MIPS). We believe that the staff should be able to develop a "target inspection budget" for each refueling cycle of a plant based on licensee

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(5) NRC Regulatory Information Conference, (July 21 and 22, 1992), Abstracts, p.19

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performance. The "target inspection budget" concept would be one means of attempting to minimize the present significant differences in burden between performance peers and the above noted regional variations. MIPS, in turn, would serve as the detailed planning and scheduling tool for implementing the target budget. We acknowledge that scheduling as much as a refueling cycle in advance could create a burden on the NRC due to the need to respond to unanticipated events and issues. However, we believe that the benefits of planning on a longer term basis outweigh any perceived disadvantages and should be attempted.

An important element of any management system is the close tracking of actual performance against planned budgets and schedules. We would suggest that MIPS be updated every three to four months and reviewed with the licensee along with an evaluation of how closely the actual inspection hours are tracking the target inspection budget. (We believe that Recommendation Number 6 would greatly assist in this analysis.)

We fully recognize that the NRC can inspect wherever and whenever it chooses. The system we are proposing is simply intended to assist both the NRC and the licensee in planning their respective activities and to gain a better understanding of how the inspection process is functioning.

Recommendation Number 5: Proposed Rulemaking Packages and Changes in Regulatory Guidance Should Also Include the Related Additions or Changes to the NRC Inspection Manual

The information provided by the staff to the NRC's Committee for Review of Generic Requirements (CRGR) to explain and justify proposed rules and new guidance (e.g., Generic Letters, Regulatory Guides, NUREGs, etc.) should include the related proposed changes to the NRC Inspection Manual. NRC Inspection Manual modules often include significant direction, guidance, and interpretations of the NRC rules, Generic Letters, Regulatory Guides, etc. However, at the present time these modules are not subject to CRGR review thereby creating the possibility of inconsistencies and inappropriate expansion of the scope of the approved regulatory action. Inclusion of the proposed inspection module(s) in the information provided to CRGR in support of new rules and guidance should contribute to greater coherence and consistency in the implementation of the changes.

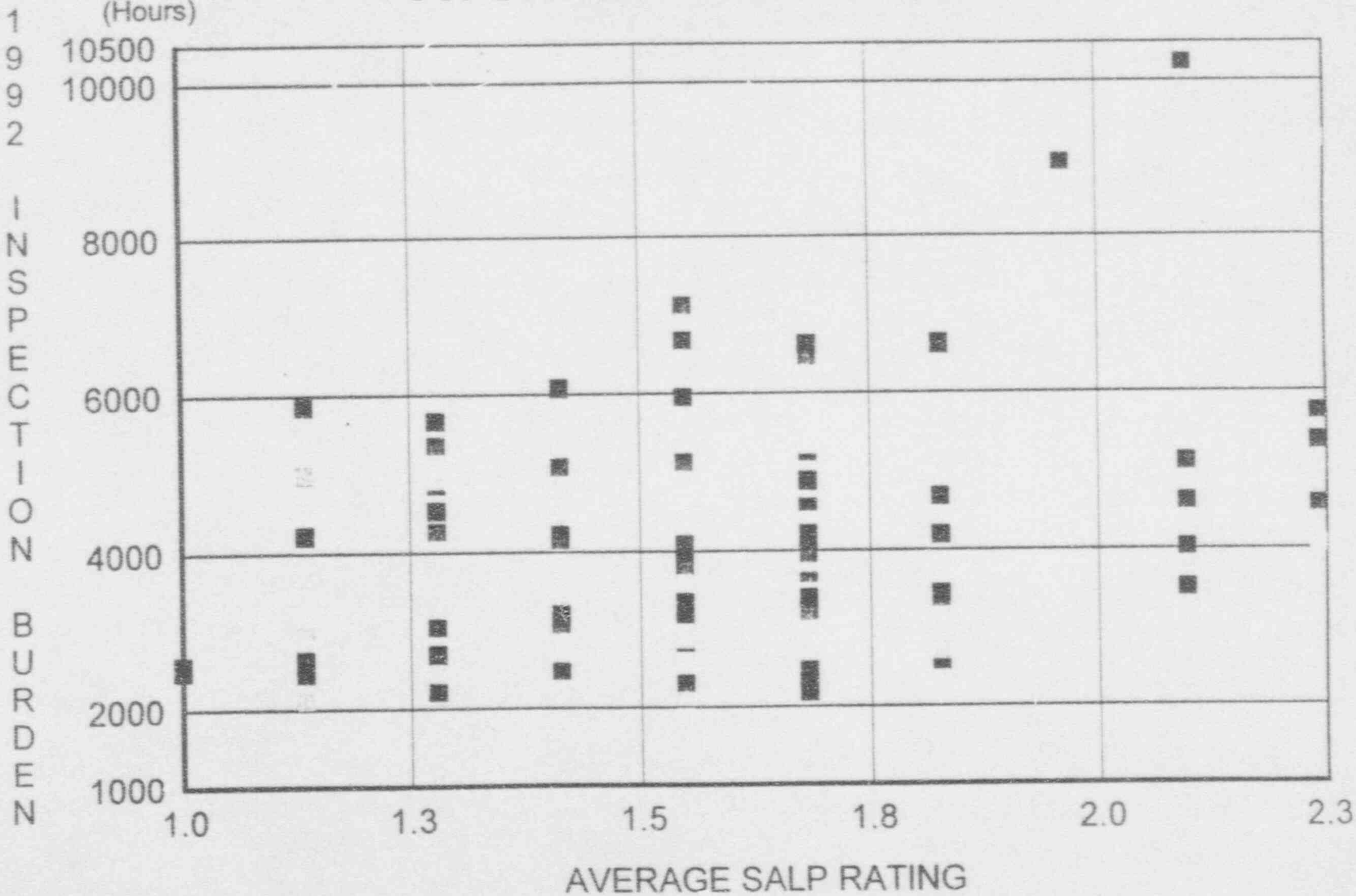
Recommendation Number 6: Resource Expenditures for Inspections Should be Clearly Documented in Each Inspection Report

We recommend that each inspection report include an accurate summary of the total NRC resources utilized in the inspection. We envision this summary including both the on-site hours and any hours expended off-site, e.g., management review, in the preparation of the report. A distribution of hours by SALP assessment category would also be informative. An excerpt from a NRC inspection report which is provided in Attachment IX to this pa[illegible] indicates that some inspection reports do attempt to quantify the expendit[illegible] of resources. However, this example is the exceptic[illegible] rather than t[illegible] rule, and such information is not provided on a consistent basis.

The purpose of this recommendation is to assist both li[illegible] and the NRC management in establishing a comprehensible correlation between resource expenditures and product. The number of inspection hours expended neither confirms or questions the quality or validity of an inspection. However, if one is able to readily identify anomalies such as significant differences in the hours required for a specific type of inspection, significant differences in the number of hours devoted to the different SALP assessment areas, wide regional variations, etc., potential inequities and misapplication of resources can be identified, evaluated and, where appropriate, corrected in a more timely manner.

FIGURE 1

1992 NRC INSPECTION BURDEN VS. AVERAGE SALP RATING



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TREND IN NUMBER OF TEAM INSPECTIONS

	APRIL 1990 THRU SEPTEMBER 1990	OCTOBER 1990 THRU MARCH 1991
LARGE TEAM INSPECTIONS	56	29
ALL TEAM INSPECTIONS	115	54

TREND IN NUMBER OF TEAM INSPECTIONS

	Apr 1990 thru Sep 1990	Oct 1990 thru Mar 1991	Apr 1991 thru Sep 1991	Oct 1991 thru Mar 1992
Large Team Inspections	53	29	25	25
All Team Inspections	109	54	60	66

TREND IN NUMBER OF TEAM INSPECTIONS

	Apr 1990 thru Sep 1990	Oct 1990 thru Mar 1991	Apr 1991 thru Sep 1991	Oct 1991 thru Mar 1992	Apr 1992 thru Sep 1992	Oct 1992 thru Mar 1993
Major Team Inspections (NRR Developed or led)	53	29	25	32	21	18
All Team Inspections (As defined in MC 2901)	109	54	60	53	48	40

ATTACHMENT III
NRC Regulatory Information Conference 1993

Attachment IV

Maine Yankee - Recent Team Inspection History

Date	Team Size	Days on Site	Comments/Notes
11/5/90	2	5	Safety Issues (Bulletin 88-04)
11/13/90	3	4	Safeguards (One team member characterized as observer)
1/28/91	2	4	Low Level Waste Transport
1/28/91	2	5	Eddy Current Testing
3/1/91	-	-	Start of Most Recent SALP Period
3/4/91	4	5	Operator Requalification
4/9/91	2	2	Special Human Factors Review
4/29/91			AIT for Main Transformer FIRS
5/27/91	3	5	Post AIT to Observe Plant Restart
6/10/91	4	10	Partial EDSFI
9/16/91	2	5	Operator Requalification
10/28/91	5	4	Safeguards (Including US Army Special Forces)
4/6/92	5	10	EDSFI
4/6/92	2	5	Effluent Environmental Monitoring
4/13/92	4	5	Engineering/Technical Support
6/22/92	7	4	Operational/Safeguards Response Evaluation (Includes US Army Special Forces Personnel)
6/22/92	3	4	Emergency Preparedness
6/27/92	-	-	End of Most Recent SALP Period

Attachment V

Vermont Yankee - Recent Team Inspection History

Date	Team Size	Days on Site	Comments/Notes
10/1/90	2	5	Radiological Controls
12/12/90	2	3	Fitness for Duty
1/28/91	3	5	Security
2/25/91	2	5	Rad. Protection
3/11/91	2	5	RG 1.97
3/17/92	-	-	Start of Most Recent SALP Period
4/23/91	6		AIT for Loss of Normal Power Event
5/20/91	4	5	Generic Letter 89-10
9/3/91	2	4	Radwaste Program
9/23/91	2	5	Radwaste Systems (Mobile NRC Chem Lab)
10/8/91	6	4	Operational Safeguards Response Evaluation (Included U.S. Army Special Forces Personnel)
10/21/91	6	5	Training Audit
2/24/92	5	5	Emergency Preparedness
6/1/92	3	5	Engineering/Technical Support
6/29/92	3	4	Emergency Preparedness
7/6/92	2	3	Pre-EDSFI Information Gathering
7/6/92	2	5	Effluent/Environmental Monitoring
7/22/92	8	10	EDSFI
8/15/92	-	-	End of Most Recent SALP Period

Attachment VI

Comparison of Team Inspection Data

	NRC DATA FOR TOTAL PLANT POPULATION	MAINE YANKEE	VERMONT YANKEE
October 1990 - March 1991			
Large Team Inspection	29	0	0
All Team Inspection	54	5	5
April 1991 - September 1991			
Large Team Inspection	25	1	1
All Team Inspection	60	3	3
October 1991 - March 1992			
Large Team Inspection	25	1	2
All Team Inspection	66	1	3
April 1992 - August 1992			
Large Team Inspection	*	3	1
All Team Inspection	*	5	4

* Data not provided

REGION I

PLANT	PERIOD	SALP AVG	1992 INSPECTION EFFORT (HRS)
Beaver Valley-1	1/91-6/92	1.43	3169.0
Beaver Valley-2	1/91-6/92	1.43	2471.4
Calvert Cliffs-1	4/91-3/92	1.43	4153.8
Calvert Cliffs-2	4/91-3/92	1.43	3442.3
Connecticut Yankee	2/90-7/91	1.14	5844.6
Fitzpatrick	2/91-4/92	2.29	9237.0
Ginna	10/90-1/92	1.57	4089.6
Hope Creek	8/90-12/91	1.29	4850.3
Indian Point-2	5/91-9/92	1.43	4296.0
Indian Point-3	6/91-8/92	2.00	8949.8
Limerick-1	10/90-3/92	1.14	4224.3
Limerick-2	10/90-3/92	1.14	2635.0
Maine Yankee	3/91-6/92	1.71	4782.6
Millstone-1	12/90-2/92	2.14	4023.2
Millstone-2	12/90-2/92	2.14	4310.1
Millstone-3	12/90-2/92	2.14	3504.5
Nine Mile Point-1	4/91-5/92	1.71	3639.2
Nine Mile Point-2	4/91-5/92	1.71	3784.1
Oyster Creek	4/91-7/92	1.86	5540.7
Peach Bottom-2	6/90-8/91	1.71	5240.9
Peach Bottom-3	6/90-8/91	1.71	4627.3
Pilgrim	8/90-9/91	1.57	6660.3
Salem-1	8/90-12/91	1.71	2335.9
Salem-2	8/90-12/91	1.71	2705.9
Seabrook	11/90-2/92	1.71	4887.5
Susquehanna-1	12/90-4/92	1.14	2941.3
Susquehanna-2	12/90-4/92	1.14	2586.0
TMI-1	5/90-11/91	1.14	4211.8
Vermont Yankee	3/91-8/92	1.71	6458.6
Regional Average		1.60	4469.1

REGION II

PLANT	PERIOD	SALP AVG	1992 INSPECTION EFFORT (HRS)
Browns Ferry-2	5/91-5/92	1.57	5123.3
Brunswick-1	10/90-11/91	2.29	5732.4
Brunswick-2	10/90-11/91	2.29	5363.4
Catawba-1	2/91-3/92	1.57	4400.0
Catawba-2	2/91-3/92	1.57	3818.5
Crystal River-3	4/91-8/92	1.57	7109.5
Farley-1	1/91-5/92	1.71	3954.5
Farley-2	1/91-5/92	1.71	3842.3
Grand Gulf-1	2/91-8/92	1.14	5002.5
Harris	6/91-9/92	1.29	5660.7
Hatch-1	3/91-5/92	1.29	2447.0
Hatch-2	3/91-5/92	1.29	2706.3
McGuire-1	2/91-8/92	1.29	2215.5
McGuire-2	2/91-8/92	1.29	2310.2
North Anna-1	9/90-11/91	1.29	3044.0
North Anna-2	9/90-11/91	1.29	2689.6
Oconee-1	8/90-2/92	1.57	2307.6
Oconee-2	8/90-2/92	1.57	3183.1
Oconee-3	8/90-2/92	1.57	1881.2
Robinson-2	3/91-6/92	1.57	3977.7
Sequoyah-1	6/91-8/92	1.71	4119.2
Sequoyah-2	6/91-8/92	1.71	3787.2
St. Lucie-1	11/90-5/92	1.00	2487.5
St. Lucie-2	11/90-5/92	1.00	2575.7
Summer	5/90-8/91	1.29	4920.7
Surrey-1	[illegible]/91-4/92	1.43	3165.0
Surrey-2	[illegible]/91-4/92	1.43	2721.3
Turkey Point-3	8/90-9/91	1.43	3349.0
Turkey Point-4	8/90-9/91	1.43	3068.2
Vogtle-1	10/90-9/91	1.86	3378.3
Vogtle-2	10/90-9/91	1.86	3439.8
Regional Average		1.51	3670.4

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REGION III

PLANT	PERIOD	SALP AVG	1992 INSPECTION EFFORT (HRS)
Duane Arnold	4/91-8/92	1.86	6582.5
Big Rock Point	5/90-9/91	1.86	6608.1
Braidwood-1	5/91-9/92	1.71	2584.1
Braidwood-2	5/91-9/92	1.71	2180.3
Byron-1	4/90-8/91	1.14	2112.0
Byron-2	4/90-8/91	1.14	2449.7
Callaway	2/90-1/92	1.14	4988.1
Clinton	2/91-4/92	1.71	4493.5
Cook-1	9/90-12/91	1.71	3488.5
Cook-2	9/90-12/91	1.71	3382.5
Davis-Besse	7/90-11/91	1.71	4865.1
Dresden-2	8/91-7/92	2.29	4561.0
Dresden-3	8/91-7/92	2.29	4063.8
Fermi-2	3/91-6/92	1.43	6065.5
Kewaunee	12/90-2/92	1.57	5947.3
LaSalle-1	10/90-12/91	1.57	3203.6
LaSalle-2	10/90-12/91	1.57	3348.0
Monticello	7/90-11/91	1.43	5071.8
Palisades	1/91-3/92	1.57	7036.3
Perry	8/90-10/91	1.71	6609.2
Point Beach-1	9/90-1/92	1.86	2713.0
Point Beach-2	9/90-1/92	1.86	2573.5
Prairie Island-1	5/89-4/91	1.29	3620.3
Prairie Island-2	5/89-4/91	1.29	4268.6
Quad Cities-1	3/91-5/92	1.71	4780.5
Quad Cities-2	3/91-5/92	1.71	4211.7
Zion-1	11/90-10/91	2.14	5106.1
Zion-2	11/90-10/91	2.14	4603.0
Regional Average		1.67	4339.9

REGION IV

PLANT	PERIOD	SALP AVG	1992 INSPECTION EFFORT (HRS)
Arkansas Nuclear-1	12/90-2/92	1.57	2879.1
Arkansas Nuclear-2	12/90-2/92	1.57	3683.7
Commanche Peak-1	2/91-1/92	1.29	5356.7
Cooper	7/90-1/92	1.71	5338.0
Fort Calhoun	5/90-7/91	1.43	6081.1
River Bend	4/91-9/92	1.86	6619.3
South Texas-1	6/91-8/92	1.86	4671.5
South Texas-2	6/91-8/92	1.86	4191.2
Waterford	5/91-8/92	1.29	4528.4
Wolf Creek	10/91-10/92	1.86	7314.7
Regional Avg.		1.63	5066.4

Region V

PLANT	PERIOD	SALP AVG	1992 INSPECTION EFFORT (HRS)
Diablo Canyon-1	1/90-6/91	1.43	4458.5
Diablo Canyon-2	1/90-6/91	1.43	4319.4
Palo Verde-1	12/90-2/92	1.71	4600.4
Palo Verde-2	12/90-2/92	1.71	3213.3
Palo Verde-3	12/90-2/92	1.71	3706.0
San Onofre-2	2/90-7/91	1.57	2847.2
San Onofre-3	2/90-7/91	1.57	2835.2
Trojan	4/91-5/92	1.71	NA
WNP-2	9/90-12/91	2.14	10231.1
Regional Avg.		1.66	4526.4

Attachment VIII

INSPECTION BURDEN DATA FOR 1992

SALP AVG	NUMBER OF UNITS	AVERAGE INSPECTION HRS	MAXIMUM INSPECTION HRS	MINIMUM INSPECTION HRS	MAX/MIN DIFFERENTIAL
1.00	2	2531.6	2575.7	2487.5	88.2
1.14	10	3699.5	5844.6	2112.0	3732.6
1.29	13	3739.9	5660.7	2215.5	3445.2
1.43	14	3988.0	6081.1	2471.3	3609.8
1.57	18	4129.5	7109.5	1881.2	5228.3
1.71	26	4139.2	6609.2	2180.3	4428.9
1.86	11	4875.7	7314.7	2573.5	4741.2
2.00	1	8949.8			
2.14	6	5296.3	10231.1	3504.5	6726.6
2.29	5	5791.5	9237.0	4063.8	5173.2

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Other instances of a faulted approach to instrumentation were manifest during this inspection period: the operators assumed an alarm condition of the service water RMS was due to erroneous instrumentation without having a solid basis; and the reactor engineer assumed core differential pressure indications were due to instrument failure.

The above instances indicate that some Maine Yankee personnel may be predisposed to question the accuracy of instrument indications when the indication is not normal. Although this approach does not appear prevalent, it indicates a partial lack of a sufficiently questioning attitude. This consideration warrants immediate management attention.

7. ADMINISTRATIVE

7.1 Person Contacted

During this report period, interviews and discussions were conducted with various licensee personnel, including plant operators, maintenance technicians and the licensee's management staff.

7.2 Summary of Facility Activities

The plant operated at full power from the beginning of the inspection period until December 17, when a tube failure in Steam Generator SG-1 forced a shutdown for repair. On January 7, 1991, inspection and repairs in all three steam generators were completed, and operators began a plant heatup. Criticality was achieved on January 8; on January 9, at 10:30 p.m., the plant was shut down from 19% power to replace MS-70, a failed main steam non-return bypass valve.

7.3 Interface with the State of Maine

Periodically, the resident inspectors and the onsite representative of the State of Maine discussed findings and activities of their corresponding organizations. Issues discussed included the status of state rule-making regarding Maine Yankee, the technical issues related to the tube leak in the Steam Generator SG-1 and the associated Emergency Technical Specification Change. The State Nuclear Safety Inspector also accompanied the Senior Resident Inspector on a tour of the expanded radiological control area. A summary of the findings based on that tour are included in Detail 2.2.

7.4 Exit Meeting

Meetings were periodically held with senior facility management to discuss the inspection scope and findings. A summary of findings for the report period was also discussed at the conclusion of the inspection.

The inspection involved 252 inspection hours, including 20 backshift and 28 deep backshift hours.

ENCLOSURE 2

YANKEE ATOMIC ELECTRIC COMPANY

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YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

July 8, 1993
SPS 93-057

Dr. Frank T. Gillespie
Chairman, Regulatory Review Group
Office of the Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Gillespie:

I appreciate having had the opportunity to meet with you and discuss the activities of your review group during my visit to Washington last May. As a result of a review of the draft report here at Yankee, we have been able to assemble what appear to be a meaningful set of comments for your consideration. We intend to devote some time reviewing them in the near future to be sure their intent is clear prior to forwarding a final written response.

In anticipation of that submittal, however, I would like to highlight a matter to you which may be of particular significance to the report because it represents a large potential savings to licensees if administered properly by the NRC staff. Additionally, it is not currently addressed in the report but will be included in our comments on it. This issue concerns a very significant reduction in the perception of seismic hazard that has evolved over the last decade but with no commensurate reduction in the NRC requirements for seismic "review" programs.

Over the last 15 years the Lawrence Livermore National Laboratory (LLNL), under sponsorship of the USNRC, has developed probabilistic models to assess the likelihood of exceeding the seismic design basis at existing nuclear facilities in the eastern United States (EUS). The earliest use of the LLNL results was in conjunction with the evaluation of the Systematic Evaluation (SEP) Plants (1979-1981).

Comparison of these early results to the seismic design basis at older plants (pre - 10 CFR 100, Appendix A) indicated tha the probability of exceeding the design basis was disturbingly .igh (possibly as high as 1 in 100 per year at some plants). Comparison of these results to post Appendix A plants indicated that the probability of exceeding modern plant design spectra was on the order of 10^{-3} to 10^{-4} per year, which the NRC Staff determined would be acceptable providing no "weak links" existed that could significantly reduce the capability of successfully coping with a large earthquake.

These early LLNL results, combined with the 'Charleston Issue', spawned several years of heightened NRC sensitivity concerning seismic issues. Arguably, it was this heightened NRC sensitivity that was the genesis of the two major seismic review programs currently in progress, namely, Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants" and the seismic Individual Plant Examination for External Events (IPE/EE).

During the eighties, with encouragement from the staff, the Electric Power Research Institute (EPRI) developed a state-of-the-art seismic hazard methodology. Comparisons of the EPRI and LLNL results for 1985 at typical EUS sites showed dramatically different perceptions of seismic hazard, with the EPRI results much less distressing. Comparison of the 1989 results showed some convergence but the differences were still apparent. However, as a result of continued discussion between scientists, as well as evolutions in the state-of-the-art, today's LLNL results (1992) compare favorably to the EPRI results at most EUS sites.

Significantly, while the plant-specific hazard associated with seismic events computed by LLNL and, thus, the perception of seismic hazard that it connotes has been steadily evolving in a direction of reduced hazard, the staff has continued to press for detailed seismic reviews of all plants (the seismic IPE), with the older vintage plants required to perform, not one, but two seismic review programs (IPE/EE and A-46).

It is important that you appreciate just how much the perception of hazard has been reduced. Figure 1 shows the progressive reduction in hazard for an eastern U. S. site. From these data it is readily apparent that the concern for seismic hazard in the late seventies and early eighties is now no longer substantiated by the research results from the staff's own consultants. From a cost vs. risk viewpoint, we believe that both the A-46 and seismic IPE programs warrant some measure of review as a consequence of this latest result from the NRC funded research. Both of these are discussed further below.

Regardless of these research results, several plants have recently received letters from the staff challenging the adequacy of that plant's Design Basis Earthquake (DBE) as the appropriate review level earthquake for resolving USI A-46. Some of these plants have committed to the use of a greater than design earthquake, others are considering a Backfit appeal.

I believe that it is imperative that the staff consult these latest LLNL hazard curve results for the insight they may provide regarding the adequacy of current DBEs. It is almost a certainty that these new results will substantiate the seismic design adequacy of the plants that have been questioned by the staff. At issue is a substantial investment, both in review and potential upgrade costs, if these plants are forced to increase their seismic designs based on earlier, but now clearly invalid, perceptions of hazard.

Further, with regard to Seismic IPE's, the level of effort for each plant intending to conduct a Seismic IPE has been established by a complex process that included some consideration of both LLNL and EPRI hazard results. Simply put, the greater the perceived hazard at a site the larger the "screening" earthquake and the more effort would be required in the review process. There is a small savings (in relay reviews) afforded plants that exhibited significant margin in the probability of exceeding their design bases (reduced scope plants).

The respective screening earthquakes to be used in searching for low cost opportunities for significantly improved seismic ruggedness were not intended to be incredible; low probability perhaps, but not incredible. Based upon the latest LLNL results, in the frequency range of interest for structures and equipment (2-10hz), at many sites, the **screening earthquakes (eg. 0.3g NUREG/CR-0098)** have such a low probability that were they "tornados", *they would be dismissed by existing staff guidance documents as incredible* and would be considered out-of-scope for the plant's IPE/EE.

At such sites, the level of effort for the seismic IPE should be sharply curtailed. It should be limited to a plant walkdown of critical equipment and structures by qualified seismic damage experts. Seismic IPE/EEs (or PRAs) are estimated by both the staff and industry to cost on the order of $1M, less upgrades. The potential exists for significant savings for many plants.

In summary, I believe that this topic deserves special note in the report. As can be seen from the forgoing, there apparently has been a transition in beliefs regarding seismically induced design basis accidents. It was perceived as a low probability event (pre-1980), then considered as perhaps highly likely (early 80's), and finally a consensus of LLNL and EPRI results that this design basis accident is, in fact, a very low probability event.

In the mean time, significant sums of money have been spent and are continuing to be spent without even a potential for a concomitant reduction of actual risk. Indeed, some of the greatest excesses in regulation of nuclear power plants may have occurred in the seismic area. The industry must be more diligent in seeking-out and recommending relief from unwarranted expenditures such as may be involved in this area. It seems appropriate that the Review Group Report should focus on this issue.

Very truly yours,

D. W. Edwards

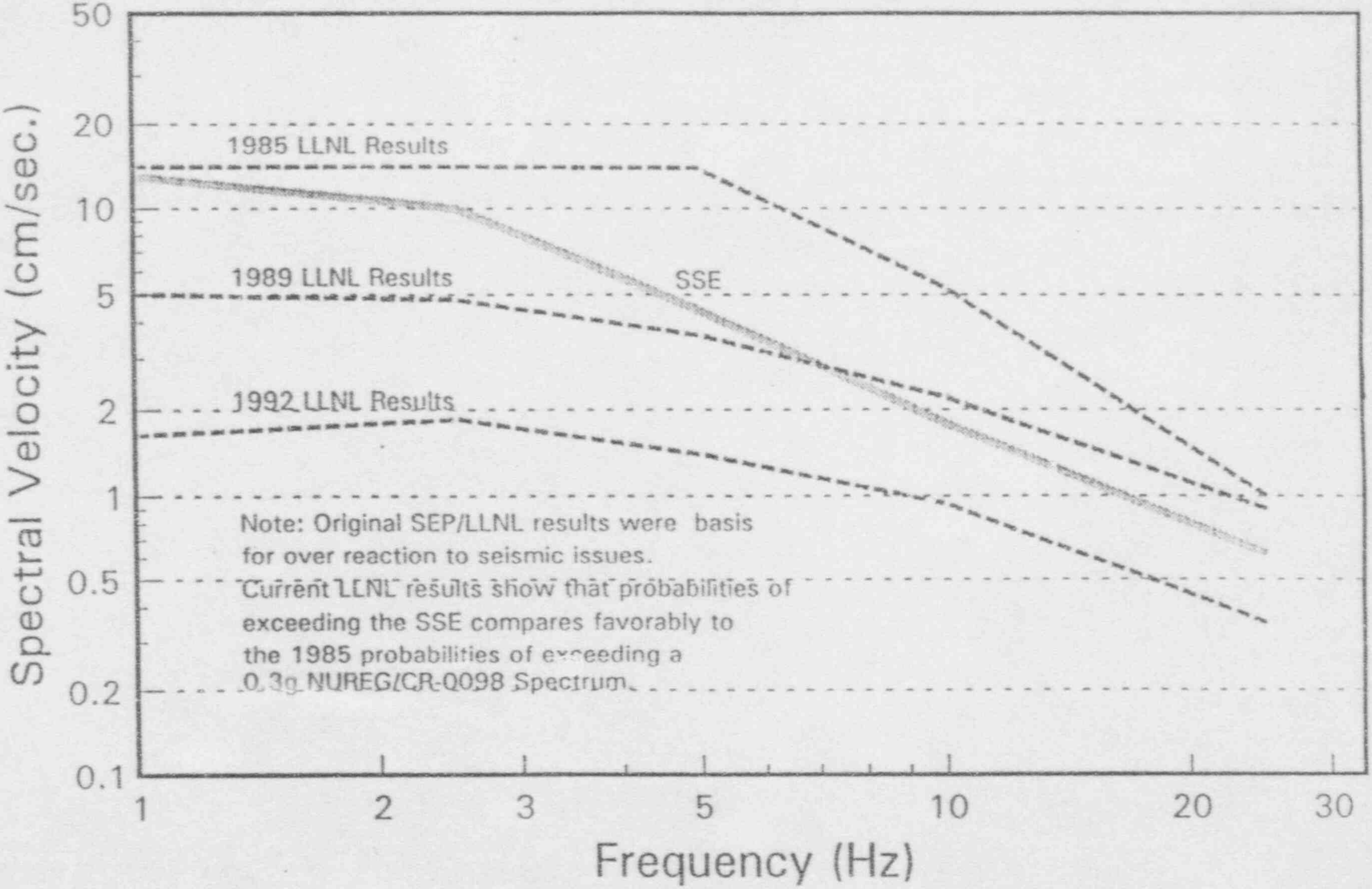
D. W. Edwards
Director, Industry Affairs

Attachment

c: S. Floyd (NUMARC)
T. Marsh (U.S. NRC)
R. Vollmer (U.S. NRC)

Comparison of 1.0E-3 Uniform Hazard Spectra

at a New England Site



ATTACHMENT

Enclosure 3

A Process to Expedite Granting of Relief Requests Generically

Volume One, page 5 of the Review Group's draft report poses the question of why licensees are not taking advantage of "...existing opportunities offered by the Commission via the Technical Specification Improvement Program line-item improvements." The Review group notes their agreement with the industry observation that "...amendment requests dealing with economic relief fall to the bottom of the priority list..." and further notes "...it is the staff view that the need for each licensee to customize requests eliminates the opportunity for rapid action." We would note that there is another deterrent to licensee's exercising the line-item improvement option: our experience has been that the simplest possible change to a plant's Technical Specifications costs at least $20,000.

The purpose of this discussion is to propose a cost-effective process to simultaneously remove an unnecessary requirement common to the licenses of a number of licensees. At present, the process for implementing generic issue license amendments functions as follows:

1. The NRC issues for public comment a notice of a proposed generic communication which identifies the specific Technical Specification improvement opportunity. Recent examples include: i) relocation of the Technical Specification tables on instrument response time limits (58FR18118), ii) line-item Technical Specification Improvements to reduce testing during power operations (58FR16881), and iii) modification of the Technical Specification administrative control requirements for Emergency and Security plans (58FR17293).

2. Comments on the proposed generic communication are resolved and the resulting Generic Letter is formally issued to licensees. The Generic Letter typically provides detailed guidance to assist licensees in preparing a license amendment request to implement the improvements offered in the letter.

3. Licensees who choose to act on the opportunity afforded by the Generic Letter submit their plant specific license amendment request pursuant to 10 CFR 50.30 and 10 CFR 50.90.

4. The NRC Project Manager for the plant is tasked with verifying that the license amendment request conforms to the guidance provided in the Generic Letter. Technical assistance is typically sought from the appropriate NRC technical review

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branch, or the regional office. The additional required steps in the review and approval process are detailed in Section 3.3 of NUREG/BR-0073, Rev. 1, the NRC Project Manager's Handbook and 10 CFR 50.91 and 50.92.

Several details of the process described above merit additional comment because of the manner in which they impact the overall efficacy of the process. Planning an effective, efficient work effort requires, among other factors, a reasonable estimate of the expected volume of required work and a knowledge of when information or data which is required to perform the work effort will be available. The current process denies this information to the NRC staff and thereby creates a significant resource allocation problem. When an improvement opportunity is formally noticed in a Generic Letter, the NRC has no way of knowing either how many licensees will choose to take advantage of the opportunity or when the license amendment requests will be filed. Although most licensees attempt to keep their Program Managers apprised of anticipated submittals of license amendment requests, to the best of our knowledge, this information is not utilized for any centralized staff planning and resource allocation effort.

Once a license amendment request is filed, the subsequent review and approval schedule is determined primarily by the licensee's Project Manager and the availability of whatever technical support is required to complete the review. What began as a generic treatment of a generic issue has at this point become a relatively fragmented activity. Each Program Manager deals with the license amendment request from a different perspective depending on factors such as his total workload, his personal knowledge of the issue, his assessment of what priority he believes the request merits, and his difficulty (real and perceived) in obtaining whatever additional support, (e.g., technical review resources) might be required to process the submittal.

An alternative approach for dealing with the "generic" Technical Specification or license improvements could be readily constructed from the elements identified above. The first recommended change in the existing process would be that when a draft Generic Letter offering a specific improvement opportunity is issued for public comment, the NRC should also request a preliminary expression of interest from those licensees who believe that they will act on the proposed improvement. The draft G.L. would necessarily contain a generic safety evaluation which contains reasonable alternative solutions which would envelope expected submittals. The responding licensees should at the same time be requested to estimate how long it will take to prepare their plant specific submittals (e.g., confirmation that they are bounded by the generic analysis, plant-specific no significant hazard analysis, and Tech. Spec. markup, if different from that proposed in the G.L.

When the Generic Letter is formally issued, licensees should be offered an opportunity for expedited processing of their amendment request if they submit by a specific deadline specified in the letter. It is expected that an appropriate deadline can be determined in large measure from the information provided by those licensees who have responded to the comment opportunity for the draft letter. Licensees would not be prevented from submitting after the specified deadline but it would be understood that "late" submittals would not be included in the generic "batch".

All license amendment requests submitted by the specified deadline should be assigned to an "Issue Manager" who would bear the primary responsibility for planning and implementing the review and approval process, i.e., for this group of amendment requests the Issue Manager assumes the responsibilities normally assigned to the individual Project Managers (PMs). Requests after the deadline would still go to the Issue Manager but they probably would be batched for later processing.

The Issue Manager would be expected to take maximum advantage of the expected generic safety analysis and common elements of the submittals and minimize unnecessary duplication of effort. This would include, as an example, the formulation of the staff's proposed "no significant hazards" determination which would envelope the submittals and be the basis for a single notice for public comment. The notice would specifically identify all of the plants which had made timely filings. At the conclusion of the public notice and comment period, the change would be issued to the plants identified. Since these types of actions generally fall in the post notice category (i.e., the staff analysis has already determined a marginal or non-existent connection to safety), requests for hearings could be satisfied without delaying the issuance of the change.

It is our belief that nothing in the "Sholly" amendment to Section 189a of the Atomic Energy Act precludes the treatment of more than one plant at a time in the steps required to satisfy the legal requirement of the Act.

We suggest that the staff consider applying the proposed process on a trial basis. Specifically, we suggest that removal of the requirements for the Post-Accident Sampling System be utilized as the test case. The NRC has already developed the necessary technical justification in NUREG/CR 4330 and we believe that relief on this issue will afford many licensees significant cost savings with respect to on-going training and maintenance requirements.

ENCLOSURE 4
YANKEE ATOMIC ELECTRIC COMPANY
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YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

April 23, 1993
FYC 93-009
SPS 93-036

Mr. David L. Meyer
Chief, Rules and Directives Review Branch
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Meyer:

These comments on NRC draft generic letter, "Availability and Adequacy of Design Bases Information," (reference FR 15885, March 24, 1993) are submitted on behalf of Yankee Atomic Electric Company (Yankee). Yankee owns the nuclear power plant in Rowe, Massachusetts and Yankee's Nuclear Services Division provides services to other nuclear power plants in the northeast.

Yankee is a member of the Nuclear Management and Resources Council (NUMARC) and we fully endorse NUMARC's comments concerning this draft generic letter. The NUMARC comment letter notes that most, if not all of the requested information has already been provided to the staff through a variety of channels. We can confirm that this is the case for the plants with which we are most familiar. Even more to the point, if the staff's own characterization of its knowledge of licensees' design document reconstitution programs as reported in SECY-91-364 is factual, there can be no reasonable justification for still another blanket survey. Repetitive solicitations for this information are clearly wasteful of both NRC and licensee resources and can have no credible safety benefit.

We have reviewed in detail the relevant supporting documentation for this proposal. We can find no evidence to substantiate a generic safety concern which would give rise to issuance of this draft generic letter. The supporting information contains only vague allusions to design basis documentation maintenance deficiencies at some unspecified number of facilities. This does not constitute, in our opinion, the kind of rigorous analysis which should warrant endorsement of this proposal by CRGR. In fact, other staff documents such as SECY 92-193 argue strongly to the contrary. Likewise, we could find no evidence of an intended inspection plan which then would be "prioritized" using the requested information.

C76\317

The draft letter's assertion regarding the voluntary nature of a licensee's response appears to have been included for the sole purpose of mooting a regulatory analysis and backfit evaluation. Without absolute assurance that there will be no retaliation should a licensee choose not to "volunteer" information, there clearly is no voluntary aspect to this solicitation. Both the subsequent "prioritization" of the apparently, yet-to-be devised inspection program, and the more traditional "adjustment" of SALP scores, serve as strong external compulsions. A voluntary program is characterized by the absence of external influences.

The Federal Register notice promises that "the NRC's final evaluation will include a review of the technical position and, when appropriate, an analysis of the value/impact on licensees." However, nothing provided in the proposed letter nor in the supporting documents can reasonably be characterized as a "technical position." We are confident that a rigorous value/impact analysis, such as should have accompanied this proposal in the first place, would demonstrate that the proposed generic letter is not justifiable.

Sincerely yours,

D.W. Edwards

D. W. Edwards
Director, Industry Affairs

DWE/dhm

C76\317

ENCLOSURE 5

YANKEE ATOMIC ELECTRIC COMPANY

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YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

June 21, 1993
FYC 93-015
SPS 93-061

Mr. Samuel J. Chilk
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Docketing and Services Branch

Subject: Proposed Rule: "Modifications to Fitness-For-Duty Program Requirements (58FR15810)

Dear Mr. Chilk:

Yankee Atomic Electric Company (YAEC) appreciates the opportunity to comment on the subject proposed rule. YAEC is the owner of the Yankee Nuclear Power Station in Rowe, Massachusetts and provides engineering and licensing services to nuclear power plants in New England. These comments are filed on behalf of Maine Yankee Atomic Power Company and Vermont Yankee Nuclear Power Corporation as well.

Background and Actual Experience

In 1988, the Regulatory Analysis supporting the then proposed Fitness for Duty (FFD) rule, offered that the reason for initiating such a rulemaking was the need to assure that impaired performance of personnel did not decrease "... the effectiveness of the response to an accident." The aralysis was grounded on the presumption that "... given the pervasiveness of the problem in our society, it seems reasonable to assume that alcohol and drug abuse, as well as other emotional and psychological factors, are also prevalent in the nuclear industry ...." (2/9/88, Regulatory Analysis, FFD rule). Although, perhaps true in an absolute sense, data from the testing programs instituted as a result of the regulation (10 CFR 26) have demonstrated that the assumed deleterious factors are not "prevalent" in the nuclear industry. In fact, the three-year average rate of random positives is 0.33% (see Enclosure 1).

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Indeed, data suggest that the behavior of personnel comprising the nuclear industry should never have been assumed to reflect the trends of the general population with regard to alcohol and drug abuse. Such an assumption ignored the array of screening and ongoing monitoring measures to which each individual is subjected. These requirements applied to all nuclear plant workers (contractors as well as utility employees) and necessarily differentiated them from the general population. Specifically they included:

- Background Investigations (credit, education, court records, references)
- Pre-employment Physical, Drug/Alcohol Screening
- Psychological Evaluation (which may include a clinical assessment)
- Inclusion in Behavioral Observation Program
- Pre-Access Drug and Alcohol Screening

At this point, as the industry completes three plus years of 100% random testing, industry experience has made it abundantly clear that the performance objectives of 10 CFR 26.10 have been satisfied:

- there is reasonable assurance that personnel are not under the influence of any substance or physically impaired,

- reasonable measures to assure early detections are in place, and

- the elements necessary to achieve the goal of a drug free workplace are also in place.

Based upon the data and relative to the general population, the nuclear industry represents a drug free population for all practical purposes. The first full year of data showed 0.28% positives for employees and 0.56% positives for contractors. The third year of testing yielded 0.20% and 0.45%, respectively. As a matter of interest, these rates of positives are 10 to 20 times LOWER THAN data presented for the U.S. Navy in the Statement of Considerations accompanying the 1988 proposed rule (54FR24474). Given these results, it is clear that adjustments to the requirements and, particularly, reduction of the random testing rate is absolutely warranted. NRC committed to this action in 1989 "... based on positive experience in the industry" (54FR24474). In a 1991 study, "Fitness for Duty in the Nuclear Power Industry" (NUREG/CR-5784), it was concluded:

> "... that results [of the study] indicate that the majority of continuous users will have been eliminated after <u>two years</u> under the 100% Standard Approach. Because licensees will have been testing on a random basis for at least two years prior to any change in testing rates that the Commission may consider, we would expect that all continuous users (except for a relatively few

coming into the workforce or becoming continuous users after the first tw years of the program) will have been detected before the testing rates are changed. If that is indeed the case, there would b[illegible]tle reason to consider efficiency in the detection of continuous users when reviewing the merits of alternative random testing rates" (p C-19) (emphasis added).

Conclusions from the 1991 study are supported by the enclosed analysis actual industry data provided in the Federal Register (58FR15812) and NUREG/CR-5758. To highlight these results, some simple calculations are presented in Enclosure 1. In our analysis, it is assumed that there are two populations of 1,000 individuals each. In the first, there are assumed to be 2.3 chronic users (representing employees) and in the second 5.2 (representing contractors). Application of the Binomial Distribution to these populations is justified because samples are returned to the population to potentially be chosen again in the next sample. Based on these calculations, several conclusions can be drawn:

1. Three years of 100% sampling has essentially eliminated chronic users from the two groups. The calculated undetected chronic users are less than 0.26 per thousand for contractors and 0.11 per thousand for employees.

2. The future sampling rate merely determines the rate at which the residual chronic user is detected. To effectively detect residual chronic users, sa[illegible]ing at 300% to 500% is required. Testing at these prohibitively hig[illegible]es is neither cost-effective nor warranted.

3. The predicted positive rate does not compare to the actual values observed. Thus, factors other than chronic addiction are involved in this population. The Federal Register notice initiating this rulemaking identified the occasional user and the user of short term detectable drugs as additional elements in the population. This analysis supports their existence. NUREG/CR-5784 documents the ineffectiveness of 100% or even 300% random sampling to detect these individuals.

Basically, the true value of random testing does not lie in detection nearly as much as it does in the creation of an environment of deterrence. However, in terms of both detection and deterrence, the most effective tool is behavioral observation by supervisory personnel. As shown by NUREG/CR-5758, the rate of positives for observed behavior by supervisors is almost 26%. This positive test result percentage far exceeds that for random testing.

C74\332

Shortcomings Regarding the Proposal

The proposal presented in the subject Federal Register Notice is a random testing rate reduction to 50% for employees. Random testing for contractors is to be retained at the 100% rate because of the "higher incidents" of positives. Although higher, the absolute value of the positive rate for contractors is very low and only looks "high" because of the extraordinarily low results for utility employees. In both cases, the residual of the chronic substance abuser after three years of 100% sampling is very low and the occasional or event-driven user is not effectively detected by random testing (NUREG/CR-5784).

At issue is the merit of the relatively excessive random testing rates (50% and 100%) proposed in light of industry experience to date and the efficacy of random testing itself as a primary means of detection. It is also important to consider recent conclusions from the General Accounting Office (GAO) in this regard. In their recent assessment of random test programs administered by 59 federal agencies, the GAO concluded:

> "... the percentage of positive test results identified through random drug testing does not vary significantly among agencies, regardless of whether the agencies test at a lower level, such as ten percent, or a higher level, such as fifty percent. ... [and] ... testing frequencies, whether lower or higher, do not have a direct impact on the deterrent value of testing." (GAO/GGD-93-13, Employee Drug Testing)

This GAO report has indicated that a reduction in random testing rates, where appropriate, will result in significant cost savings without impacting the deterrent aspect of the program.

A further deficiency in the subject proposal is the two-group testing program it contains. Ironically, the NRC, when re-examining their internal random sampling rate, rejected just that alternative because "... two distinct NRC random testing groups...result in administrative inefficiencies [and] limit cost saving/avoidance" (SECY 92-176, "NRC's Drug Testing Program, p. 5). The NRC has chosen a single reduced random sampling rate for their own program, but plans to impose this cumbersome administrative burden with its unnecessary costs on licensees.

Also contained within the subject proposal is a request for comments on the exclusion of "... certain positions critical to safe operation...such as reactor operators" from the testing rate reduction. The assertion that continued random testing of this group at a 100% rate will be an effective detection mechanism is no more valid in this special case than for the more general ones examined above. The proposed segmentation is completely unwarranted and could negatively impact the morale, self image and motivation of this group of highly trained and dedicated specialists.

Alternative Proposal

Deterrence is assured through the synergism of all elements of licensee Fitness for Duty Programs (i.e., pre-access screening, random testing, for-cause testing, follow-up testing, and, most importantly, behavioral observation). It should be clear that the random testing rate is one of the least significant factors in the entire set of deterrent elements and two-group random testing is an unnecessary and inappropriate complication.

As an alternative to the proposal presented in the subject Federal Register notice and, in recognition of the information presented herein, we would propose the following:

1. Establish a single random testing rate for all nuclear industry workers having unescorted access (employees and contractors).

2. Monitor the test results from this group as a total population - i.e., do not base the regulation in any way on individual plant statistics. Namely, do not regulate to the lowest common denominator.

3. Lower the required random testing rate to 10% with the caveat that a statistical doubling of the population's presently very low positive rate in any year from the then current industry average would be grounds for immediate regulatory review. This rate will yield over 15,000 samples per year which is more than ample to monitor the population. G[illegible]n this size sample, and given that 0.33% is the true population parame[illegible] for drug abusers, the likelihood of obtaining [illegible] estimate that is more th[illegible] double the 0.33% estimate without bei[illegible] detected is essentially zero. It should be noted th[illegible] with a smaller sample size, th[illegible] 95% confidence level [illegible] change with a variance going from .05% to .18%. [illegible] still very low levels (see Enclosure 1).

[illegible] to pro[illegible]

4. [illegible]havioral observation [illegible] for managers and supervisors in the area of

C761333

Mr. Samuel J. Chilk
June 21, 1993
Page 6

The annual cost for program implementation is nearly $69M in 1990 dollars. The general breakdown is:[1]

($ in 000's)

	Industry	@ Per Unit[3]
Direct Cost of Sampling	29,500	210
Retraining and Suitable Inquiry[2]	33,000	300
Lost Productivity	6,300	55
Total	68,800	565

If the random sampling is reduced to the appropriate level of 10%, the potential annual savings for the industry <u>without degradation in the overall program</u> is in excess of $30M.

<u>Summary</u>

Steps should be taken, now, to implement a change to the regulation which takes into account the following:

1. The proposed random testing rate reductions are inconsistent with actual experience - the rate should be reduced to 10% for the entire population of nuclear industry workers having unescorted access.

2. The GAO has concluded from its study of 59 Federal Agencies that the percentage of positive test results does not change significantly whether testing is conducted at 10% or 50%. They concluded, further, that testing frequency does not have a direct effect on the deterrent value of testing.

3. The value of random testing is the degree to which it contributes as a deterrent to drug/alcohol use as part of the overall Fitness for Duty Program.

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(1) Based upon a NUMARC survey of licensees reported to the NRC (T. E. Tipton to B. K. Grimes, September 20, 1991).

(2) The cost of retraining and suitable inquiry are not a function of sampling rate and are assumed to continue.

(3) Average values may not be representative of individual plant costs.

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4. A dual population test program such as has been proposed is unnecessarily burdensome, administratively cumbersome, and is unjustified based on actual experience. Also, a dual population test program is detrimental to morale, teamwork, and overall safe operation.

5. Conclusions about the behavior of the population must be founded on data representing the entire population and not impacted by statistical variances at individual units. Should individual units not meet industry norms, appropriate action should be taken against them, not the entire nuclear industry.

6. Perhaps the most powerful element of the Fitness for Duty program is behavioral observation and emphasis in this area will have the most value.

We urge the NRC to aggressively pursue this alternative to the changes currently proposed to the Fitness for Duty regulation. The industry has spent tens of millions of dollars to implement programs that actual experience (within the industry itself and throughout the government) has shown to be excessive. Failure to act expeditiously to fix this situation is contrary to the regulators' obligation not to impose any more burden on the regulated community than is absolutely necessary.

Sincerely yours,

D. W. Edwards

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Director, Industry Affairs

DWE/dhm

Enclosure

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Analysis of Industry Data

1. Positive Test Results (years 90, 91, and 92)

		EMPLOYEES			CONTRACTORS								
					SHORT TERM			LONG TERM			ALL		
Year	Source	#Tests	#Pos	%	#Tests	#Pos	%	#Tests	#Pos	%	#Tests	#Pos	%
90 & 91	FR	201,278	497	0.25	84,873	496	0.58	16,410	67	0.41	101,283	563	0.56
92	NUREG	98,611	199	0.20	50,242	233	0.46	7,877	29	0.37	58,119	262	0.45
All	-	299,889	696	0.23	135,115	729	0.54	24,287	96	0.39	159,412	825	0.52
All Workers		Total Tests - 459,301				Total Positives - 1,521					% (3 yr results) 0.33		

2. 95% Confidence Level Limits for 10% and 100% random testing rates assuming a Current Population Estimate of .33%.

Random Testing Rate	# of tests per year	Lower Confidence Limit	Current Population Estimate (3-yr results)	Upper Confidence Limit
10%	15,000	.0025	.0033	.0043
100%	100,000	.0030	.0033	.0036

Analysis of Industry Data (continued) Enclosure 1

3. The following example is used to examine the effectiveness of random testing.

Assumptions: 2 Populations of 1000 each representing a closed system
Pop #1 - 2.3 Chronic Users
Pop #2 - 5.2 Chronic Users
100% sample size for 3 years
10% sample size thereafter

Pop #1 - Employees			Pop #2 - Contractors	
Year	P (not detected)	Expected User Remaining	P (not detected)	Expected User Remaining
1	.3677	0.846	Same as for Pop #1	1.912
2	.1352	0.311		0.703
3	.0497	0.114		0.258
4 at 100%	.0183	0.0420		0.0952
4 at 50%	.0301	0.0692		0.156
4 at 10%	.0449	0.103		0.233
.	.	.		.
.	.	.		.
10 at 10%	0.0246	0.0319		0.128

4. The binomial distribution is the appropriate one to use for sampling from a population with return of the sample to the population (i.e., it could be sampled again)

$$P\ (\text{not detected}) = \frac{N!}{X!\ (N-X)}\ P^X\ Q^{(N-X)}$$

Where: $P = 1/N$
$Q = 1-P$
N = Total # picks
X = # times picked

The specific solution of this general expression for a determination of the probability of not being detected is:

$$P\ (\text{not detected}) = \frac{(1000)!}{0!\ (1000-0)!}\ P^0\ Q^{(N-0)} \text{ or } Q^N$$

Where: $N_n = \sum_{i=1}^{n} N_i$ for n samples

5. Considering Example Population 1

- Third year actual rate is 0.00276 Positives/Test
  Third year predicted rate is 0.000114 Positives/Test

- Actual experience is that the rate of positives is 0.00265 more than predicted or 2.65 positives/1000

- This demonstrates that other factors are involved which initiate substance abuse in the population. In other words, the problem appears to be the occasional or event driven user rather than the chronic user.

- Random testing is not effective in detecting this type of user. It is effective only to the degree that random testing can support a different mechanism for user detection, or better, reinforcement of avoidance behavior.

- Similar observations apply to example Population 2 with equal validity

- There is no fundamental difference in the behavior of either example population with regard to chronic users.