Elimination of Marginal Safety Requirements

Task 2
Application of Methodology and Development of Recommendations

Letter Report

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Elimination of Marginal Safety Requirements

I. Introduction and Summary Tables

Introduction

The Nuclear Regulatory Commission's annual "Policy and Planning Guidance 1987" states that existing nuclear regulatory requirements should be reviewed to see if some requirements could be eliminated without compromising safety, safeguards, or environmental protection. In support of this objective, the Office of Nuclear Regulatory Research contracted Battelle Columbus Laboratory (BCL) to identify regulatory requirements that would be potential candidates for modification or deletion. The results of the study are reported in the BCL report, "Effectiveness of LWR Regulations in Limiting Risk-Task 24," May 26, 1989, which identifies potential actions for forty-four Regulatory Positions.

In May 1990, the Office of Nuclear Regulatory Research contracted with SCIENTECH, Inc. to develop (Task 1) and apply (Task 2) a method to evaluate the potential actions identified by BCL and to select those that would reduce the regulatory burden on industry but have a marginal impact on safety. The method was developed and submitted to the Office of Nuclear Regulatory Research in the final Task 1 letter report dated December 10, 1990. This report presents the results of applying the method to the forty-four regulatory positions in the BCL report.

Sixty-five potential actions were identified for the forty-four BCL Regulatory Positions. Fifty-one of these potential actions, listed in Table 1, were eliminated from further consideration by the screening process defined in the Task 1 report. The remaining fourteen potential actions were fully evaluated resulting in the attribute values shown in Table 2. The values in Table 2 represent the consensus of a team of four experienced engineers.

In performing the analyses described in this report, generic considerations were encountered that appear to have relevance to many of the current NRC activities relating to improvement of regulatory requirements. These considerations related to: the differing character and purpose of the regulatory requirements; the range of plant designs to which specific requirements may be applied; inherent obstacles to prospective assessment of the effects of regulatory change; conflict between the desire for flexibility and the desire for stability in the licensing process, and the role of conservatism in providing reasonable assurance of adequate safety. Section II discusses these issues and others related to the types of potential actions listed in the BCL report.

Section III presents the evaluation reports for each of the Regulatory Positions in the BCL report. These have been numbered to reflect the sequence of their presentation in the BCL report. To assist the reader, the BCL analysis of the regulatory position is reproduced in full at the top of each evaluation report.

TABLE 1

POTENTIAL ACTIONS ELIMINATED FROM CONSIDERATION

RP No.	Potential Action	Step 1 *		Step 3 *
		Scope	NRC Activity	Importance to Safety
1	Clarify Reporting Requirements		X	
1	Eliminate Duplicate Reporting Requirements		X	
2	Eliminate 10CFR50.33(f) and Appendix C			X
4	Eliminate SRP Chapter 15			X
5	Eliminate/Clarify 10CFR50.34(f)	X		
6	Eliminate 10CFR50.34(g)			X
6	Replace 10CFR50.34(g)	X		
8	Expand10CFR50.46	X		
9	Eliminate 10CFR.49(b)(2) and (b)(3)			X
9	Update RegGuide 1.89	X		
9	Update 10CFR50.49 Schedule	X		
10	Update 10CFR50.55a Emphasis on System Performance	X		
10	Expand 10CFR50.55a to Include I&C Systems	X		
11	Update 10CFR50.55a(h)	X		
13	Replace 10CFR50.62	X		
14	Eliminate SEP Language	X		
14	Eliminate 10CFR50.71, Subparagraph (e)			X
14	Eliminate FSAR Submittal to NRC			X
15	Replace 10CFR50.73	X		
15	Expand 10CFR50.73	X		
16	Eliminate 10CFR50 Appendix A	X		
16	Clarify 10CFR50 Appendix A	X		
17	Update Load Combinations for Dynamic Design			X
18	Review and Update RegGuide 1.78	X		
18	Clarify Control Room Boundaries		X	
19	Expand General Design Criteria	X		
19	Eliminate Inconsistancy in 10CFR50.13 and Appendix A	X		
20	Expand General Operating Criteria	X		
21	Expand 10CFR50 Appendix B	X	X	
22	Update 10CFR50, Appendix B, SRP & R.G.s	X		
24	Update Appendix R	X		
25	Clarify 10CFR50, Appendix R, Para. III.L.3-7	X		
26	Revise 10CFR100 - Decouple OBE and SSE		X	
26	Convert 10CFR100, Appendix A to a RegGuide		X	
26	Eliminate Reference to TID 14844		X	
28	Update SRP 13.2.1, Reactor Operator Training		X	
29	Update Physical Security Requirements	X		
30	Clarify SRP 16.0, OBE		X	
31	Update RegGuides 1.3 & 1.4, Source Terms		X	
32	Update RegGuide 1.60, Seismic Response Spectra		X	
33	Update RegGuides 1.61 & 1.92		X	
35	Update RegGuides 1.92 & 1.122		X	
36	Update RegGuides 1.9 & 1.108		X	
37	Update RegGuide 1.109, Annual Dose to Man	X		
38	Eliminate RegGuide 1.115, Turbine Missiles	X		
39	Expand RegGuide 1.152, Software Standards	X		
40	Expand Guidance on Human Performance	X		
41	Eliminate Reactor Vessel Level Indication System			X
42	Expand Initial Conditions for Safety Analyses	X		
44	Expand Regs To Include Sunset Provision	X		

Step 1: Scope : Eliminated because the potential action would increase the net licensee and NRC effort.
NRC Activity: Eliminated because the NRC is currently reviewing the issue and is near resolution.
Step 3: Importance to Safety: The potential action could have a significant negative impact on safety.

TABLE 2

EVALUATION OF POTENTIAL ACTIONS

Potential Actions Evaluated		Short Term Burden		Long Term Benefit	
RP #	Potential Action	NRC	Licensee	NRC	Licensee
2	Conversion of 10CFR50, Appendix C	LOW	LOW	LOW	LOW
2	Relocation of 10CFR50.33(f) and Appendix C	LOW	LOW	LOW	LOW
3	Elimination of 10CFR50.33(a) and Appendix L	HIGH	LOW	LOW	LOW
3	Allow one submittal for several plants	LOW	LOW	LOW	LOW
3	Relocate within 10CFR	LOW	LOW	LOW	LOW
7	Convert 10CFR50.44 To RegGuide or SRP Sectio	MEDIUM	LOW	MEDIUM	MEDIUM
10	Revise 10CFR50.55a	MEDIUM	LOW	LOW	LOW
12	Clarify 10CFR50.59	MEDIUM	MEDIUM	LOW	HIGH
18	Increase Control Room Exposure Limit	MEDIUM	LOW	LOW	LOW
23	Revise Containment Leakage Test Requirements	MEDIUM	LOW	LOW	LOW
23	Convert Appendix J to a Regulatory Guide	MEDIUM	LOW	LOW	LOW
24	Convert Appendix R to a RegGuide	HIGH	LOW	HIGH	HIGH
27	Eliminate Requirement for MSIV LCS	MEDIUM	LOW	LOW	MEDIUM
34	Update RegGuide 1.76, Design Basis Tornado	MEDIUM	LOW	LOW	MEDIUM
43	Clarify "Important to Safety"	HIGH	HIGH	MEDIUM	HIGH

NOTE: Many of these values are conditional - see evaluation reports for details.

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II. Generic Issues and Considerations

In performing the analyses described in this report, the review team encountered a number of generic considerations. These considerations related to: the differing character and purpose of the existing rules; the range of plant designs for which the rules are applicable; inherent obstacles to prospective assessment of the effects of regulatory change; conflict between the desire for flexibility and the desire for stability in the licensing process; and the role of conservatism in providing reasonable assurance of adequate safety. These generic considerations are relevant to many of the current NRC activities relating to improvement of regulatory requirements. This section discusses these considerations. Section III then utilizes some of the ideas developed here in evaluating the actions listed in the BCL report.

Evolution of Regulatory Requirements

In the period leading to the early 1960s, there were fewer than a dozen AEC staff members with responsibility for safety review of nuclear power reactors; the ACRS was the center of regulatory technical expertise, and the applicants and their architect/engineers had primary responsibility for design and construction of safe nuclear power plants. The licensing review was carried out in a collegial atmosphere of mutual professional respect. As the number of applicants increased during the 1960s, some incorporated the latest technology and licensing positions into their designs, while others tended to rely on approaches they had successfully used in the design of conventional power plants. To achieve higher and more uniform safety standards within the industry, the regulatory staff began to develop guidance in the form of technical reports, general design criteria and regulatory guides (originally called safety guides). In essence, this process continued over the next two decades with an ever-increasing level of technical detail.

Regulatory Guides offered applicants assurance of approval if the defined methods were adopted; consequently, these Guides encouraged greater uniformity in applicant responses to licensing requirements. In developing the Guides, the staff generally sought acceptable approaches with the broadest possible applicability within the industry. It was not necessary that the staff position in the Guide be the best way for an individual plant to comply with the regulation, because the opportunity for the applicant to propose alternatives always existed, at least in theory. In practice, the additional licensing time required to justify alternative approaches discouraged the development of alternatives by applicants. The effect was to discourage innovation and was not in the best overall interest of safety.

The Standard Review Plan (SRP) was first issued in 1975 for staff use in reviewing license applications for compliance with NRC regulations. A major revision (NUREG-0800) was issued in 1981. There have been revisions of individual parts since 1980. The SRP incorporates all of the licensing criteria to be applied by the staff in its licensing reviews, including reference to regulatory guides, industrial standards and various SRP appendices that present Branch Technical Positions.

There was little motivation for the staff to develop licensing guidance in areas where the existing industrial practice was uniform and non-controversial. Likewise, there was little motivation for codification when the staff guidance in the SRP or regulatory guides was non-controversial. However, when the staff's licensing positions were not readily accepted by the licensees, or when public intervention resulted in repeated litigation, rulemaking was the preferred remedial course. Generally, the inclusion of a requirement in 10CFR50 was aimed at one or both of the following objectives:

1. to establish the basic procedural, administrative and information requirements necessary for the Commission to implement its licensing responsibilities, and

2. to resolve controversy with respect to the standards of design, construction or operation necessary to provide the NRC with the reasonable assurance of safety required for issuance of a license,

Examples of requirements related to the first objective are the procedures and information requirements for construction permit and operating license applications and their associated safety analysis reports. These requirements are basic to the fulfillment of the Commission's responsibilities and are not usually considered as candidates for deletion, although it is difficult to demonstrate their effect on safety. Possible modifications to these requirements that would have marginal effect on safety and some positive benefits include one-step licensing and other such fundamental changes that are or have been the subject of staff review. The Commission is currently involved in a decisionmaking process concerning the level of detail required to support licensing applications under 10CFR52.

Examples of requirements related to the second objective are the specific design requirements for emergency core cooling in Appendix K, which followed extensive review and adjudication of various technical issues in hearings and in the rulemaking process, and the detailed fire protection requirements of Appendix R, which was issued when a large segment of the industry balked at the requirements imposed by the staff in Appendix A to Branch Technical Position 9.5-1.

This codification in response to need is what has produced the so-called "patchwork" quality of 10CFR50. Some provisions of 10CFR50 have outlived their usefulness and have no current or future applicability. 10CFR50.34(f) relating to TMI requirements is a good example, along with other provisions related to deadlines long past. Elimination of these provisions is a housekeeping action without effect on safety or licensee burden.

Likewise, some outdated regulatory guides exist for which change or withdrawal has not been initiated because the benefits are minimal. It is easier to ignore the guide and review the other basis for licensing that the licensee proposes.

For operating plants, it might simplistically be argued that the applicable requirements are "codified" in the individual licensing conditions and FSAR commitments. Then 10CFR50.59 controls any changes, tests or experiments relative to these licensing conditions. Other existing rules might be useful insofar as they provide guidance with respect to safety review of changes, tests or experiments made or proposed by the licensee in matters not covered by their licensing conditions. New rules may require amendment of the license or FSAR; however, some rules affecting design are "grandfathered" in whole or in part for application to operating plants.[1]

Changes in subordinate documents,such as the SRP or regulatory guides, might be motivated by improved approaches that were developed or identified since the existing guidance was promulgated or by the resolution of a generic safety issue. However, changes in these subordinate documents are not automatically binding upon the licensees because licensing commitments generally reference a specific edition or revision of a subordinate document. Hence, there is no direct or immediate benefit to be derived from revising these documents unless:

a. the change reflects an NRC policy that licenses should be amended accordingly, or

b. there is a belief that the change will be of value in connection with future applications or licensee proposed amendments

In the case of a. above, rulemaking would likely occur only if difficulties were encountered in getting licensees to cooperate in making the desired amendments or it was specifically required as a matter of NRC policy.

<u>Existing Plants and Future Plants</u>

Because of the way 10CFR50 requirements evolved, they tend to be specific to light water reactors of current design.[2] Many of the detailed technical requirements are likely to be irrelevant for designs that depart significantly from

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[1]An exception of sorts is the requirement that licensees justify any deviation from compliance with updated references to ASME Section XI incorporated in 10CFR50.55a(b)

[2]Requirements for advanced plants will likely go through a similar evolution, beginning with general requirements and evolving toward more detailed requirements as the need develops. Detailed requirements are not equivalent to prescriptive requirements; this point was clearly made at the Commission's "Briefing on Nonprescriptive Nuclear Safety Regulation," October 30, 1990.

the existing pattern. Consequently, an assessment of the effects of changes in those requirements must be restricted to existing plants or new plants of similar design. Where the effects of a change would be felt largely or exclusively by new plants, the assessment of total effect is highly uncertain due to uncertainty in the number of new plants that will have designs sufficiently similar to existing plants for the existing rules to apply.

With regard to 10CFR50 revisions that would reduce licensee burden, some fairly obvious general observations can be made. Changes in construction or preoperational testing requirements will affect new plants directly and existing plants little or not at all. Changes in design requirements will affect new plants directly and existing plants with respect to replacements and renovation; they might also result in design margin changes that would be relevant to license renewal for existing plants. Changes in operating requirements would likely have similar effects for new and existing plants.

Quantitative Evaluation of Benefit

Probabilistic methods have been successfully applied in prospective evaluations of the effects of changes in design or operation of nuclear power plants. Generally, the assessment is done for typical plant designs and extrapolated to other plants where similar changes would be expected to have similar effects. The success of the approach is due to the improved understanding that has been developed since the mid-1970s of the relationship between the design of operational features and the events involved in accident sequences.

To apply probabilistic methods to changes in regulation requires an understanding of the relationship between the regulatory changes and the consequent changes in design or operation of the regulated facility. In other words, to evaluate the effect of a regulatory change on system reliability or public risk, it is first necessary to translate the regulatory change into a change in the siting, design, construction, or operation of one or more nuclear power plants. Existing methods of probabilistic assessment are applicable only to situations where performance and reliability can be reasonably estimated; the methods cannot be directly applied to regulations, and there are some situations where probabilities cannot be reasonably estimated.

Past efforts to evaluate the effect of a regulatory change on risk or reliability have postulated causal relationships that are dependent on a large number of assumptions, many of which are implicit. This makes it difficult to generalize the results. However, it is possible to use probabilistic assessment methods to evaluate the effect of a regulatory change if the consequential change in performance at representative nuclear power plants is within a narrow range. An example is a change in the level of redundancy required for a particular safety system. In such a case, a comparative probabilistic assessment for one or more typical plants will provide the desired quantitative measure of effect on system reliability. On the other hand, if a regulatory change could have a broad range of possible industry responses, it would be necessary to await industry responses, and then to perform differential risk assessments for each type of response to assess the effect of the regulatory change. (Elimination of such uncertainty is one factor that makes standard plants so attractive from a regulatory viewpoint.)

The same type of uncertainty arises when elimination of a requirement is considered. For example, if all NRC fire protection requirements were eliminated, it is not realistic to expect that utilities would eliminate fire protection systems. Some licensees might weaken their systems, some might leave them alone, and some might find more cost effective configurations, possibly strengthening the system as a result. Here, clearly, the effect of such a regulatory change could not be assessed until the the industry actions in response became evident.

There can be further complications. First, it must be remembered that, while the assessment methods provide a valid and useful *index* of risk, there are limitations on the ability to account for all factors that affect risk. For example, current assessment methods would be unable to evaluate the effect on risk of the requirement that an updated FSAR be periodically submitted to the NRC, although such an effect probably exists. Second is the fact that some regulatory requirements have direct objectives other than the reduction of risk. For example, the requirement for an FSAR update is aimed primarily at procedural benefits, e.g., a clearly defined safety basis for the plant to be used by the licensee in controlling changes to the plant, improved staff ability to review licensee safety evaluations and proposed changes, and improved inspection and enforcement capabilities.

The Role of Conservatism

The role of conservatism in the regulatory process is extremely complex, notwithstanding that it can be simply defined as *compensation for uncertainty*. The use of conservatism to compensate for uncertainty in various specific areas of safety review can result in cumulative, possibly excessive conservatism in dealing with overall plant risk. Also, over application of conservatism has sometimes masked realism making it impossible to understand plant response to upset conditions. To assess the role and effects of conservatism requires deliberate and cautious analysis of how, where and to what specific purpose it has been introduced into the licensing process.

Reasonable assurance that a nuclear power plant provides adequate safety basically requires two things, a concept of what constitutes adequate safety (a safety criterion) and an acceptable means of demonstrating that the plant meets that criterion (an assessment method). The safety criterion may be explicit or implicit, and it may be quantitative or qualitative. For years the NRC's safety criterion was implicit and qualitative; today, the Commission's safety goal is explicit and qualitative, and it has been interpreted quantitatively. The relationship between the assessment method and the safety criterion also may be explicit or implicit. A risk assessment would be explicitly related to the Commission's safety goal, whereas the regulatory requirements in 10CFR are implicitly related to the goal.

The expression of safety criteria is hierarchical. The safety criteria at one level may be disaggregated into lower level safety criteria. (See also Enclosure 1 to SECY-89-102.)

Conservatism can be introduced into a safety criterion at any level to compensate for uncertainty in its functional relationship to a higher level criterion. This type of conservatism has been implicitly introduced into the Commission's quantitative safety objectives. Other examples are the use of conservative dose limits to compensate for uncertainty in the health effects caused by low doses and the use of load factors to ensure conservatism in design.

Assessment methods usually involve a model representing the relationship between the characteristics of a plant structure, system or component and its safety performance relative to some explicit or implicit safety criterion. Uncertainties can arise with respect to the completeness, validity or accuracy of the model, or the associated input data. These uncertainties can be compensated by introducing conservatism into the safety criterion, the model or the data. Sometimes several sources of uncertainty will be accommodated by the introduction of conservatism in one element; for example using test "outliers" as input data to compensate for both measurement and model uncertainties. Sometimes, when the criterion, the model and the data are developed separately, redundant conservatism can be introduced. Consequently, when assessing whether a regulatory requirement is excessively conservative, it is important to know all sources of uncertainty to which the conservatism is addressed.

An illustrative example is the Commission's judgment that acceptable plant safety requires an Emergency Core Cooling System (ECCS). This deterministic safety criterion is expressed in GDC 35 and in 10CFR50.46(a). Lower level safety criteria that define the acceptability of an ECCS are found also in GDC 35 and in 50.46(b). The single failure criterion in GDC 35 represent conservatism to compensate for uncertainty in the reliability of the system. The quantitative criteria in 50.46(b) also include an undefined level of conservatism to compensate for uncertainty in the technology related to the specific issues addressed.

The evaluation of an ECCS against the criteria in 50.46(b) is done using evaluation models of the performance of the system. 50.46(a) requires either that the uncertainty in the calculated results be estimated and used to show a high probability that the criteria will be satisfied, or that models meeting the conservative requirements of 10CFR50 Appendix K be used. It has been claimed that Appendix K guidelines result in conservatively high calculated temperatures compared to more realistic models. One solution that has been used is to apply a more realistic model for which the uncertainties have been estimated in accordance with 50.46(a).

The role and effect of conservatism is less evident in other cases, such as the largely deterministic fire protection criteria in Appendix R, or the requirement for "defense-in-depth." A somewhat complicated example is the BCL proposal to change either the containment design leakage rate or the leakage testing requirements. The design leakage rate affects the calculation of offsite doses and could not be changed without considering its role in determining the level of conservatism in various other, related safety criteria. Such considerations may lead to larger allowable leakage rates due to new source term knowledge. The effect of changing the test requirements, with or without a change in design leakage rate, is difficult to quantify even though such tests serve to periodically contribute to NRC confidence in the leakage performance of the containment.

One way of looking at conservatism is that it doesn't change the required level of safety, but rather increases confidence that the level is being reached. While true, this does not change the fact that removal of conservatism increases risk, although not necessarily to an unacceptable level. Consequently, unless the the level of safety achieved can be expressed explicitly in terms of some safety criterion, the determination of the required level of conservatism is largely judgmental.[3]

Generic Evaluation of Conversion

Conversion, as used by BCL, means to remove a requirement from 10CFR50 but retain the substance and details of the requirement in the SRP or a regulatory guide. Conversion potentially would be most appropriate for highly detailed rules, like those covering ECCS performance (50.46), emergency planning (50.47), fire protection (50.48), electrical equipment qualification (50.49), and their associated appendices. These detailed technical requirements originally were codified primarily to resolve licensing controversies which have since faded as industry practice has become stabilized and technology has matured.

Converting a codified requirement to an SRP section or a regulatory guide would provide both the staff and the licensee or applicant with an opportunity to deviate from the detailed requirements with less formality and effort than are required for exemptions to 10CFR. It also would allow the licensee to determine, pursuant to 10CFR50.59, whether subsequent changes in the plant require prior NRC approval. Conversion would have minimal impact on safety provided the technical criteria for NRC approval of deviations do not change and the licensee evaluations under 10CFR50.59 are adequate. The benefits of conversion are a decrease in staff and licensee effort, provided adverse effects

[3]There are some cases where increased confidence is bought at the expense of a potential increase in risk. For example, where excessive testing is undertaken to confirm operability and results in wearing out the equipment.

do not occur because of errors in 50.59 evaluations at a rate that cancels out the benefits.

The possible adverse effects of conversion are: a) a number of licensee proposals for deviations that overwhelms the staff's capability to perform timely, adequate safety analyses, b) a significant increase in unacceptable licensee interpretations of 10CFR50.59, and c) licensee proposals and/or staff actions that stimulate public concern and intervention in the form of petitions (for operating plants) or hearing issues (for new applications). Each of these could result in increased regulatory effort, reducing the net savings associated with the presumably lower cost of the deviations. (An indirect negative effect on safety could also occur by distracting people from more important work.)

The level of effort involved in a rulemaking action to convert a regulation is uncertain. If the requirements are to remain essentially intact, the technical effort might be minimal. However, some members of the public might not have the same perception of the change (BRC is a case in point) and could see the action as an attempt to weaken the requirements. The possibility exists to deal with this concern efficiently by a rulemaking action to convert several of the detailed rules at the same time. The concern might be ameliorated or at least brought to light by a policy statement issued in advance.

<u>Generic Evaluation of Update/Replace</u>

Actions to update of replace regulations would be taken if changes in technology or operating procedures have made current requirements inappropriate or too conservative. Some of these views receive widespread acceptance and agreement, while others are subject to controversy. Evaluating the effect of these actions on either risk or licensee effort is subject to the limitations described in the section on "Quantitative Evaluation of Regulations," above.

<u>Generic Evaluation of Relocate/Clarify</u>

Actions to relocate or clarify regulations are generally motivated by a desire to achieve a simpler, more logical regulation. The reasons for the existing structure and content of 10CR50 are discussed above under "Evolution of Regulatory Requirements." Moving or modifying a few individual parts of the rule, even the perceived worst offenders, is not likely to result in a level of improvement that would justify the effort required. A wholesale restructuring of the rule, including some major conversions to lower level documents (e.g., changing several sections and appendices into Regulatory Guides) and some rewriting of the more abstruse sections, could have benefits that would justify the effort required. This would depend significantly on the scope of the changes made, the relevance of the changes to license renewal requirements and the expected number of new plants to which the present requirements of 10CFR50 would be applicable.

Conclusions

Changes to regulatory requirements, including additions, deletions and modifications, should be evaluated in relation to the intended purpose of the requirements. They may be intended to provide the commission with the basic information needed to review and decide upon license applications, to establish acceptable safety standards for the siting, design, construction or operation of a facility, or to achieve or enhance confidence in the safety analyses of the site, design, construction or operation of a facility.

Probabilistic assessment methods suitable for comparative analysis of changes in the site, design and operation of a nuclear facility are not well suited for analysis of regulatory change except in cases where the range of licensee responses is predictably narrow. Even in those cases, probabilistic assessment is likely to be useful only in evaluating the effect of changes in safety standards. Probabilistic assessment does not have anything to do with NRC requirements pertaining to the submission of information nor does it relate to the need for conservatism. On the other hand, uncertainty analyses associated with probabilistic assessment sometimes are useful in assessing the need for conservatism.

There is inherent conflict between the desire for requirements to be non-prescriptive and non-codified and the desire for licensing stability and simplicity. Codified regulatory requirements limit the licensees' flexibility, but also limit the range of staff interpretation and intervenor litigation. If the process of regulatory change is well managed, rulemaking enhances licensing stability. Prescriptive requirements also limit the licensees' flexibility but enhance licensing simplicity by providing pre-approved methods of compliance. The example of an innovative design for a standardized plant illustrates both points. The development of regulatory positions during the design process is likely to be a complex and taxing process of iterative test and analysis. However, the existence of a prescribed standardized design should provide a stable basis for simplified licensing.

The character of regulatory requirements is determined by a combination of policy, administrative, legislative, technical and economic considerations. Evaluating a change in requirements should take into account the relevant considerations. The evaluations reported in Section III, below, identified some promising candidates for additional study. However, a broader and more detailed analysis will be required to define and evaluate more specific and appropriate changes.

Regulatory Position No.1

10CFR21
10CFR50.9
10CFR50.55(e)
10CFR50.72
10CFR50.73

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate/Clarify*

BCL Comments: Reporting requirements permeate 10CFR and are particularly prevalent in Part 50. Part 50.8 in fact specifies more than thirty sections of Part 50 which contain information requirements approved by the Office of Management and Budget under the Paperwork Reduction Act of 1980.

Among the requirements are those associated with reporting discovered defects. In issuing an Advanced Notice of Proposed Rulemaking (SECY-89-xxx), NRC is considering amending the regulations on the reporting of safety defects found during the design, construction, and operation of nuclear facilities. The proposed amendments would eliminate duplicate evaluation and reporting, establish a uniform threshold for defects that need to be reported and a uniform content for safety defect reporting, and establish consistent time limits for evaluation and reporting of defects.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The potential action is within the scope of this project. However, BCL correctly notes that these actions are already underway. The staff began a rulemaking action in 1985 involving changes to the reporting requirements of 10CFR21 and 10CFR50. The proposed changes have the following significant objectives:

- to make it easier to identify reportable defects by reflecting the experience with defects discovered to date in the definitions of reportable problems in the NRC's reporting requirements and guidance,
- to reduce duplicative evaluation and reporting requirements,
- to establish uniform time frames for reporting (e.g., by using 48 hours and 30 days as standard intervals for the various reporting requirements,

- to reduce the number of marginally useful reports by raising the threshold for 50.55(e) reports to the level of the 10CFR21 threshold and making the content requirements similar, and

- to set a time limit for the transfer of responsibility for the safety evaluation required by 10CFR21.51(b)(2).

Several versions of the staff proposal have been prepared in response to various comments by the Commission and further revisions of the current version are currently being considered.

Therefore, no further consideration will be given in this report to these potential actions identified by BCL. (A separate potential action for 10CFR50.73 is addressed in the BCL report and in this report under Regulatory Position No. 15.)

A recent report by the Office of the Inspector General (OIG 89A-25) offered comments and recommendations relating to the NRC management of reporting requirements under 10CFR21. Those recommendations are currently under consideration by the staff.

Discussion of the requirements is provided in the section below.

STEP 2: Background Description

Rulemaking Motivation

1C FR21, "Reporting Of Defects And Noncompliance," was issued June 6, 1977 to implement Section 206 of the Energy Reorganization Act of 1974. Among other things, Section 206 required that responsible officers in a firm or organization involved in activities regulated by the Atomic Energy Act of 1954, must report known safety-related defects to the Nuclear Regulatory Commission (NRC). In response to Section 206, Part 21 was created to ensure that NRC licensees and other firms established internal procedures to assure that safety-related defects and noncompliance were brought to the attention of responsible company officers. These individuals, in turn, were required to notify the NRC.

10CFR50.9, "Completeness and accuracy of information," was issued December 31, 1987 to codify the obligations of licensees and applicants to provide the Commission with complete, timely, and accurate information. In addition, this rule codified the requirement for disclosure of information identified as having a significant implication for public health and safety or common defense and security.

10CFR50.55(e), "Conditions of construction permits," was issued January 19, 1956 to establish uniform reporting requirements regarding significant deficiencies identified during nuclear power plant design and construction.

10CFR50.72, "Immediate notification requirements," was issued February 29, 1980 to require the immediate reporting by telephone of significant events. After the accident at Three Mile Island, the NRC staff acted to ensure the timely and accurate flow of information from licensees following significant events. The NRC's Office of Inspection and Enforcement (OIE) issued bulletins requesting licensees to review their prompt reporting procedures. These procedures were intended to make certain that each licensee notified the NRC within one hour of the time a reactor was found not to be in a controlled condition. However, these bulletins were not requirements and did not describe in sufficient detail the specific types of significant events that were of concern to the NRC. In several instances licensees had not immediately report events that were deemed by the NRC to be significant. Part 72 described these events in detail and codified the reporting requirement.

10CFR50.73, "Licensee event report system," was issued July 26, 1983. Although a Licensee Event Reporting (LER) system was in existence prior to this rule, the Nuclear Regulatory Commission recognized that the LER system needed revision to make reporting more consistent among licensees, stop the reporting of unimportant events, and provide better data on significant events. Part 50.73 established a system that would provide information necessary for engineering studies and trend analysis of significant events. This information would then be used to identify and resolve threats to public safety and aid in the identification of accident precursors. Part 50.73 codified existing LER reporting requirements, established a single set of reporting requirements that would apply to all operating nuclear power plants, and provided consistency with 10 CFR 50.72.

Requirements for Licensees

10 CFR 21 requires responsible officers of organizations building, operating, or owning NRC-licensed facilities or conducting NRC-licensed activities, to report failures to comply with regulatory requirements and defects in components which may result in a substantial safety hazard. Substantial safety hazard is defined as "...a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety...." Part 21 requires initial notification within two days of receipt of the information. If the initial notification is not written (e.g., by telephone), a written report must be submitted within 5 days after the information is obtained. Initial notification must be made to the Office of Nuclear Reactor Regulation, or Office of Nuclear Material Safety and Safeguards, as appropriate, or to the Administrator of the appropriate Regional Office. Written notification is made to the Office of Nuclear Reactor Regulation or Office of Nuclear Material Safety and Safeguards.

10 CFR 50.9 places two requirements upon an applicant or licensee. First, information that is provided to the NRC and is also required to be maintained by the applicant or licensee must be complete and accurate. Second, the NRC must be notified by the applicant or licensee of any information having a significant implication for public health and safety or common defense and security. The appropriate Regional Office must be notified of such information

within two working days. Part 50.9 also states that this requirement is not applicable to information already required to be submitted by other requirements.

10 CFR 50.55(e) requires the holder of a construction permit to report all significant deficiencies. A significant deficiency is defined as a deficiency that could adversely affect the safety of operations of the nuclear power plant at any time throughout its life and which indicates:

(a) a significant breakdown in the quality assurance program;

(b) a significant deficiency in final design such that it does not conform to the criteria and bases stated in the safety analysis report or construction permit;

(c) a significant deficiency in construction of or damage to the plant that will require extensive efforts to meet the criteria and bases stated in the safety analysis report or construction permit; or

(d) a significant deviation from performance specifications which will require extensive efforts to meet the criteria and bases stated in the safety analysis report or construction permit.

10 CFR 50.55(e) also requires that the appropriate Regional Office receive initial notification of each significant deficiency within 24 hours. In addition, a written report of the deficiency is required to be submitted in accordance with Part 50.4 to the Commission within 30 days. Part 50.4 requires the original report to be sent to the NRC Document Control Desk with copies to the appropriate Regional Office and Resident Inspector.

Part 50.72 consists of two primary requirements. First, it requires that the NRC be notified of any declared Emergency Class (as listed in Appendix E of Part 50). This notification must be made within one hour of declaring the emergency. Second, it requires that the NRC be notified of certain non-emergency events. Initially, Part 50.72 defined 12 types of events that should be reported within the first hour. However, a 1983 amendment changed the reporting requirements based on the significance of the event. This change was made to lessen the impact of reporting requirements on the individuals responsible for operating the plant. Six types of events must now be reported within the first hour of their occurrence. These events include degradation to principal safety barriers, conditions that place the plant outside its design basis, and conditions that result or should result in the initiation of the Emergency Core Cooling System. The remaining six events must be reported within four hours of their occurrence and include any event that results in actuation of a Engineered Safety Feature, any event that could have precluded the fulfillment of a safety function, or any event requiring the transportation of a contaminated person to an off-site medical facility for treatment. All notifications made under Part 50.72 are made to the NRC Operations Center via the Emergency Notification System.

Part 50.73 requires licensees to provide detailed descriptions of safety-significant events. The descriptions of significant events and planned corrective actions provide the basis for more detailed study of serious events that might be precursors to serious accidents. Licensees are required to prepare an LER for events that meet one or more of the criteria listed in Part 50.73. The criteria are based on the nature, course, and consequences of the event. The LER report must be submitted within 30 days after the discovery of the event and is required to be submitted in accordance with Part 50.4.

<u>Intent of the Requirement</u>

The primary goal of the reporting requirements described above has been to ensure that safety-related information is reported to the NRC in a complete, accurate, and timely manner. This goal has remained consistent since each of the requirements was issued.

Regulatory Position No.2

10CFR50.33(f)
Appendix C, Financial qualifications

DRAFT BCL ANALYSIS

BCL Potential Action: *Convert/Relocate/Eliminate*

BCL Comments: Part 50.33 specifies the general information an applicant for a construction permit or an operating license must submit as part of the application. It includes an unusual mix of basic data, such as the name, address and business of the applicant, as well as more expansive information such as financial qualifications and emergency response plans. Paragraph (f) specifically addresses "information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out...the activities for which the permit or license is sought."

Appendix C to Part 50, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits," amplifies on this requirement by citing more specifically what types and forms of information the Commission expects to receive. Both the regulation and the Appendix distinguish between applicants which are "established organizations" and those which "newly-formed" for the purpose of constructing or operating the licensed facility. The former are permitted to rely more on historical data such as financial statements; while the latter are expected to specify funding sources, assets, liabilities and the like. This same type of information would be required under 10CFR50.80 if an organization or individual wished to receive a license by transfer from another party.

An interesting aspect of Appendix C is its explicit expression as being a "guide" which is "not intended to be a rigid and absolute requirement." By common practice, such admission would more aptly define a Regulatory Guide than a regulation. Thus, one potential action is to convert Appendix C to a Regulatory Guide. Another potential action is to relocate these regulations to another part of 10CFR which is more procedurally oriented, perhaps combined with the antitrust review of 50.33a and Appendix L.

The more fundamental issue is the extent to which the required information is an appropriate indicator of the safe operation of the licensed facility. The extent to which a causal relationship between financial qualifications and safety can be defined and defended should dictate whether the elimination of these requirements should be considered.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action regarding 10CFR50.33(f) is within the scope of this project. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of the action is appropriate.

STEP 2: Background Description

Section 182 of the Atomic Energy Act requires applicants for a license to provide such information as the Commission determines is necessary to decide on the technical or financial qualifications of the applicant. The Commission's determination is reflected in 10CFR50.33(f). On August 18, 1981 the Nuclear Regulatory Commission (NRC) issued a notice of proposed rulemaking to amend its regulations on financial qualifications review. The two major presumptions underlying the proposed rule were that regulated utilities (or those able to set their own rates) would be able to recover the costs for safe construction and operation, and that the more direct means of ensuring safety - inspection and enforcement - would be reasonably effective in deterring any "corner cutting" and in remedying safety problems. The proposed amendment would eliminate certain of the requirements for financial qualifications review and findings for electric utilities applying for licenses for production or utilization facilities, as follows:

(1) Eliminate entirely these requirements for construction permit applicants; and either

(2)(i) Eliminate entirely these requirements for operating license applicants; or

(2)(ii) Retain these requirements for operating license applicants to the extent they require submission of information concerning the costs of permanently shutting down the facility and maintaining it in a safe condition (i.e. decommissioning costs).

The proposal also included a requirement for power reactor licensees to maintain the maximum amount of commercially available on-site property insurance, from the time that the Commission first permits ownership, possession, and storage of special nuclear material at the site of the nuclear reactor.

In March 1981, after consideration of public comments, the Commission issued a final rule which incorporated option (1) and option (2)(i), above, and retained the requirement for on-site property damage insurance, or equivalent protection, adequate to cover reasonable decontamination and cleanup costs associated with the property damage resulting from an accident at the licensed facility. The coverage was required only after an operating license was issued.

As a result of a petition by the New England Coalition on Nuclear Pollution and others, the U.S. Court of Appeals for the D.C. Circuit remanded the rule in February, 1984, for clarification of the Commission's statement of basis and purpose. The court considered that the Commission's reasons for dispensing with the financial qualifications review for electric utilities would, if supported by the facts, apply generally to all license applicants. The Commission, upon reconsideration, noted that the financial difficulties and cancellations experienced by some plants suggested that eliminating financial qualification reviews at the construction permit stage should be given further study, and that the lack of any pending construction permit applications made such deferral have little practical consequence. A final rule, effective October 12, 1984, reinstated the financial qualification review for all construction permit applicants, but retained the exemption of electric utilities at the operating license stage.

Requirements for Licensees

The specific requirements for information are given in 10CFR50, Appendix C. Each applicant, with the exceptions noted above, is required to submit an estimate of construction costs and a statement on the source of the construction funds. In addition, annual financial statements are required to be submitted at the time they are issued. The rule distinguishes between applicants that are established organizations, for which a financial history may be sufficient, and applicants that are newly formed entities, for which more detailed information would usually be required.

Intent of the Requirement

The required information is intended to allow the Commission to assess the financial qualification of the applicant to construct and/or operate a production or utilization facility, i.e., a nuclear power plant, in a manner that provides adequate protection to the health and safety of the public.

Step 3: Determination of Importance to Safety

The following is a representative sampling of various relevant remarks the Commission has made in the Federal Register concerning the importance of the financial qualification reviews to safety.

46FR41786 (8/18/81): "The Commission believes that its existing financial qualifications review has done little to identify substantial health and safety concerns at nuclear power plants. However, there are matters important to safety which may be affected by financial considerations."

47FR13751 (3/31/82): ". . . the Commission in its *Seabrook* decision indicated its support for the substance of the proposed rule - elimination of the financial qualifications review because of any demonstrable link between public health and safety concerns and a utility's ability to make the requisite financial showing."

49FR13046 (4/2/84): "The Commission's experience leads it to question whether pre-licensing reviews of applicants' future ability to pay for the cost of safety measures provide any significant additional assurance of safety beyond the assurance provided by the pre-licensing review of facility structures , systems and components, operating and materials handling procedures, and technical qualifications, and by the Commission's inspection and enforcement program. However, the Commission has not conducted any detailed study to determine whether there exists any significant correlation between its financial qualifications reviews and later safe operation and use of nuclear materials. Therefore, the Commission does not propose such a rule at this time but might consider doing so later if there is adequate support."

49FR13046 (4/2/84), Commissioner Bernthal's Additional Views: "I urge special public attention and comment on the Commission's alternative proposal, i.e. that the Commission completely eliminate financial qualifications review for all license or permit applicants, including but not limited to electric utilities, not only on the grounds that no link has been shown between financial qualification review and assurance of safety, but because even having carried out such a review, the Commission is powerless to insure continued financial qualification of an applicant, or to predict what financial resources the public utility commission of jurisdiction might place at applicant's disposal."

49FR35748 (9/12/84): "Despite the long standing nature of the financial qualification reviews under the original rule, their safety rationale seems never to have been clearly set out. A financial disability is not a safety hazard *per se* because the licensee can, and under the Commission's regulations would be obliged to, simply cease operations if necessary funds to operate safely were not available."

49FR35750 (9/12/84), Additional Views of Commissioner Asselstine: "Although the NRC should not return to performing the same types of financial qualification reviews required by the old rule, the majority [of Commissioners] has gone too far in excluding virtually all consideration of the utility applicant's financial qualification in nuclear power plant operating license proceedings. Such a sweeping exclusion is contrary to the requirements of the Atomic Energy Act, is unsupported by the facts and is unjustified on the basis of this rulemaking record."

49FR35752 (9/12/84), Separate Statement of Chairman Palladino: "Commissioner Asselstine's criticism of the Commission's approach is not justified by either the facts or the law in this rulemaking. First, as the Court of Appeals observed in its decision remanding the Commission's March 1982 rule, even if the Atomic Energy Act of 1954 were interpreted as requiring financial qualification reviews, it would not preclude appropriate generalized criteria [such as the proposal] to eliminate financial qualifications reviews on the generic conclusion that the rate process assures for [utilities] the funds needed for safe operation of a nuclear power facility."

Clearly, the assessment of the importance to safety of the financial qualification reviews is judgmental and the Commission is uniquely authorized by the Atomic Energy Act to make that judgment. However, the Commission itself has noted there has not been any detailed study to determine whether there exists any significant correlation between its financial qualifications reviews and later safe operation and use of nuclear materials (49FR13046). The Commission deferred rulemaking that would extend the exemption from, or eliminate entirely, financial qualification reviews, pending the development of additional support for such an action. SCIENTECH considers that, in view of these considerations and the record summarized above, the question of the importance to safety of these reviews is moot and the proposed elimination of financial qualification reviews will not receive further consideration.

Conversion of Appendix C to a Regulatory Guide, or relocation of 10CFR50.33(f) and Appendix C to another part of the Commission's regulations would have minimal impact on safety provided the criteria for review are not changed thereby. Therefore, these proposed actions will receive further consideration below.

Step 4: Impact Analysis

Conversion of 10CFR50, Appendix C

Conversion of Appendix C to a Regulatory Guide offers both less risk and less benefit than the potential conversion of detailed technical requirements, such as Appendix K and Appendix R. Less risk because the linkage between safety and the financial qualifications of the licensee during operation is tenuous, and the Commission receives annual financial statements from the licensee pursuant to Appendix C. Less benefit because, although the preparation of the required information is a moderate burden on the licensee, deviations from the requirements are infrequent so that the more flexible and informal deviation procedures available under a Regulatory Guide would represent a marginal advantage, scarcely sufficient to exceed the associated rulemaking effort. These conclusions relative to the conversion of Appendix C can be summarized by assigning values to the attributes defined in the Task 1 report, as follows:

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Low	Low	Low	Low

Relocation of 10CFR50.33(f) and Appendix C

The rulemaking effort to relocate Appendix C to elsewhere in 10CFR would be relatively modest because the requirements for financial qualifications review have little, if any, relationship to other reouirements and are concisely stated in only a few locations. For similar reasons, the benefits of relocating these requirements from 10CFR50 to other parts of CFR would be minor in the absence of a wholesale effort to restructure the format of the regulations. These conclusions are summarized as follows:

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Low	Low	Low	Low

Regulatory Po... ion No.3

10 CFR 50 33a & Appendix L
Information for Antitrust Review

DRAFT BCL ANALYSIS

BCL Potential Actions: *Eliminate/Relocate*

BCL Comments: Section 105c of the Atomic Energy Act of 1954, as amended, directs the U. S. Attorney General to review antitrust aspects of the commercial nuclear power industry. The requirement for construction permit applicants to provide information for this review is specified in 10CFR50.33a. The information to be provided is specified in Appendix L. The purpose of the antitrust review is to assure that trade and commerce are protected from unlawful restraints and monopolies or unfair business practices.

The regulation divides applicants into groups based on their total electrical generating capacity: more than 1400 MW(e); 200 - 1400 MW(e); and less than 200 MW(e). The larger the applicant, the more information the applicant is required to provide. Further, the applicant must provide the required information as a separate document at least nine months but no more than 36 months in advance of any part of the application for construction permit. Separate documentation must be submitted for each application, regardless of prior similar submittals or reviews.

Appendix L describes 20 categories of information required in order to perform the antitrust review. Examples include data on loads and load growth, reserve capacity, alternative sources of generating capacity, transmission systems, neighboring utility systems, cost of power, corporate mergers with other electricity suppliers, and rates charged for power. A typical submittal for antitrust review is twenty copies of a 50-page (?) (sic) *document.*

The potential action is to eliminate this requirement from NRC's jurisdiction since it has no bearing on safety. An alternative action is to limit the required submittal to one time unless some significant change having antitrust implications has occurred since the last submittal. Finally another potential is to relocate the regulation and the Appendix out of Part 50, which is already complex enough, and into a more procedurally oriented part of Title 10.

It has been ten years since any organization applied for a construction permit. [How long? Who applied and for which plant?] (sic) *During that time the electric generating industry has undergone significant changes, such as mergers of small generators into larger ones and the legislated requirement for large utilities to purchase power from small independent producers. The concept of companies whose sole function is to operate nuclear power plants (as opposed to transmitting and distributing electricity, for example) is receiving significant attention. Also the implementation of 10CFR52, an alternative process for licensing commercial nuclear power plants, is now in effect. Potential action on* *the antitrust review requirements would need to be viewed from these perspectives.*

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed actions regarding 10CFR50.33a are within the scope of this project in that they might reduce the burden to licensees and applicants. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of the action is appropriate.

STEP 2: Background Description

Rulemaking Motivation

10CFR50.33a implements the requirements of Section 105c of the Atomic Energy Act by requiring submission of "such information as the Attorney General determines to be appropriate in regard to the finding to be made by the Commission as to whether the activities to be licensed would create or maintain a situation inconsistent with the antitrust laws specified in section 105a of the Act." This information is specified in 10CFR50 Appendix L.

In order to allow the antitrust review to be performed before initiation of the licensing safety review, 10CFR50.33a was amended in 1974 to require applicants for class 103 construction permits to file the required document "Information Requested by the Attorney General for Antitrust Review" at least nine months, but not more than 36 months prior to the date that any other part of the construction permit application is filed (except for construction permit applications submitted within nine-months after the effective date of the amendments).

In April 1978, the Nuclear Regulatory Commission began considering amending the regulations to reduce or eliminate the requirements for submission of antitrust information in certain "de minimis" instances and to clarify requirements for antitrust review of applications for class 103 facilities (commercial facilities) other than power reactors. After considering the comments and information developed during the rulemaking process, the Commission concluded that participants whose generating capacity at the time of application is 200 MW or less are not required to submit the information specified in Appendix L of part 50, unless specifically requested by the Commission to do so. Under these circumstances, smaller systems could also be required to submit the information if possible antitrust problems become apparent. The Commission also concluded that participants whose generating capacity at the time of application is more than 200 MW(e) and less than 1400(e) MW are required to respond only to question nine in Appendix L of Part 50. These proposed changes would reduce the burden of preparing antitrust-related data on applicants with small generating capacity, while at the same time maintaining an adequate standard of antitrust review.

On the basis of experience indicating that 25 copies were not necessary, in October 1978, the NRC eliminated the requirement for licensee submittal of 25 copies of the document titled "Information Requested by the Attorney General for Antitrust Review".

Complex business arrangements are sometimes entered into to support the construction of a nuclear power plant. An applicant may be one of several utilities that have come together as a group for this purpose. The same applicant may enter into different arrangements for different plants. The 10CFR50.33a requirement for separate submittals for each plant provides a timely certification by the applicant as to the accuracy of the information for each plant.

Requirements for Licensees

The requirements are adequately described in the preceding section.

Intent of the Requirement

The basic purpose of the NRC antitrust regulations is to comply with Section 105c of the Atomic Energy Act and assure that trade and commerce are protected from unlawful restraints and monopolies or unfair business practices. No safety issues are involved.

Step 3: Determination of Importance to Safety

Antitrust regulations of the NRC are not directly related to safety, although there is a potential for antitrust rules leading to unwanted partners and strained business relationships that could detract from management attention to safety.

Step 4: Impact Analysis

Elimination

It is likely that most, if not all, nuclear power plant construction permit applications submitted under 10CFR50 will be by applicants having a generating capacity greater than 1400 MW(e). Consequently, elimination of the requirement would eliminate the administrative burden of complying with the full requirements of Appendix L. However, this action cannot be accomplished through a rulemaking action; it would require legislative action. With no compelling argument in favor of such legislation, the effort to accomplish the change would likely be high.

The language of the legislation leaves it to the Attorney General to determine what information is required. Thus, it appears that the Attorney General could waive the requirement or establish conditions under which no information would be required. This course of action would place the major burden on the Department of Justice and probably would not involve a major effort by the staff and licensees

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
High	Low	Low	Low

Allow one submittal for several plants

The effort to amend the rule would be relatively small. However, the benefit would also be small because of the nature of the information being submitted. If the information does not change between submittals, the effort to compile and submit it a second time would be minimal. If there are significant changes, the effort required to modify portions by reference, addenda, or other means is likely to approximate the effort required to compile a complete modified submittal. The Attorney General's acceptance of this action would be necessary.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Low	Low	Low	Low

Relocate within 10CFR

Both the short term burden and the long term benefits would be small. There would be no change in the requirements and, consequently, no change in the staff or applicant effort to justify the rulemaking effort.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Low	Low	Low	Low

Regulatory Position No.

10 CFR.50.34(a)(4) & (b)(4)
Contents of applications
SRP Chapter 15 – Accident Analysis

DRAFT BCL ANALYSIS

BCL Potential Actions: *Eliminate*

BCL Comments: FSARs must contain assessments which demonstrate that for certain postulated design basis accidents, the public would not be exposed to radiation doses in excess of 10CFR100 limits. Several related RegGuides provide further guidance on assumptions to be used in performing the calculations. The results (i.e. the calculated doses) are also used to establish equipment specifications for certain engineered safety features.

There are suggestions that many of the Chapter 15 calculations (e.g. rod drop and rod ejection events) usually have no meaningful risk significance, are costly to perform and review, and have little or no impact on plant design. In the few cases where dose criteria are calculated to be exceeded, exemptions are granted based on conservatisms in fuel failure assumptions. Thus the calculations are unproductive by any real measure and could be eliminated as a requirement.

The counterpoint is that the results of calculations for fuel handling accidents (SRP 15.7.4) and spent fuel cask drop accidents (SRP 15.7.5) can affect features of containment isolation systems. A different example is that for at least one case involving high burnup fuel, a calculated 20% increase in thyroid dose would cause some plants to exceed 10CFR100 limits. The required Chapter 15 calculations enabled this to become recognized. There is also the perception that licensees have well established analytical codes and procedures for performing Chapter 15 analyses and that the NRC staff spends little time reviewing the results. Thus the burden is low.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The BCL potential action is to eliminate certain Standard Review Plan Chapter 15 (SRP 15) Subsection requirements for calculations of radiation doses to the public resulting from postulated design basis accidents. This action requires no change to 10 CFR 50.34(a)(4) or (b)(4). The proposed action is within the scope of this project because it would result in a reduction of effort by the staff and the applicant. Additionally, there is no present or planned NRC staff action in this area. Therefore, further consideration of the action is appropriate.

STEP 2: Background Description

Rulemaking Motivation

50.34(a)(4) and (b)(4) require an "analysis and evaluation of the design and performance of structures, systems and components of the facility with the objective of assessing the risk to public health and safety...." The calculations required by SRP 15 provide a measure of performance that is directly related to public risk, as well as confirmation that the exposure limits of 10CFR100 are met.

Requirements for Licensees

For each of the topical areas of the SRP 15 Subsections, the applicant must perform a calculation of safety systems performance. Radiological consequences to the public are required to be calculated for the SRP 15 events. As noted by BCL, the methods and procedures for the calculations are well developed and routine.

Intent of the Requirement

The intent of the SRP 15 requirements is to demonstrate adequate performance of safety systems and to confirm com; ance with the public exposure limits of 10CFR100.

STEP 3: Determination of Importance to Safety

The analyses of design basis events required by SRP 15 establish the safety envelope of the plant. The initial conditions, boundary conditions, and equipment performance assumptions form the technical basis for the Technical Specifications contained in the license. If the analyses do not demonstrate that the plant conforms to the safety envelope, there must be changes in the plant or its operation to bring it into conformance. Elimination of the analyses required by SRP 15 could have a significant negative impact on safety and will not receive further consideration.

Regulatory Position No. 5

10CFR50.34(f)
Contents of Applications; technical Additional TMI requirements

DRAFT BCL ANALYSIS

BCL Potential Actions: *Eliminate/Clarify*

BCL Comments: This regulation, commonly referred to as the "CP rule" was adopted in 1980 in the aftermath of TMI. It specifies requirements applicants must satisfy for an LWR construction permit or manufacturing license whose application was pending as of February 16, 1982. There are at least 50 major requirements imposed and most are quite specific. They include a plant/site specific PRA, various accident and reliability analyses, operability studies, improved simulation capability, improved operating procedures, control room design review, safety parameter displays, hydrogen control systems, valve qualification programs, QA program requirements, dedicated containment penetrations and many more.

The applicants to which this rule applies are mentioned by name in the text. As of January, 1989, all such plants have been cancelled. Thus, there is no immediate need for such a regulation. The continuing need for those requirements is based on the assumption that applications for new plants will be received and will have to comply with those requirements. Using the current text, it is unclear which specific requirements and design standards would apply to new plants. The process for licensing new plants is addressed in a new 10CFR Part 52, which should reference a modified 10CFR50.34(f) or other suitable regulations.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

10CFR34(f) was intended to ensure that the information contained in the construction permit and manufacturing license applications pending in early 1982 would be sufficient to assure the NRC that these applicants had given appropriate attention to TMI-related requirements, many of which were in the process of being introduced into the regulations and imposed on OL applicants and operating plants.

The Commission's July 30, 1985 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants affirms its belief that a new nuclear power plant design can be shown to be acceptable for severe accident concerns if the applicant demonstrates compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the so-called CP

Rule (10CFR50.34(f)). The reference to the rule was clarified in NUREG-1070 by staff responses to comments regarding the potential irrelevance of the CP Rule to future designs:

Comment: *...We also believe, however, that should the probabilistic risk assessment show any of the requirements of the CP Rule not to be cost effective, they should not need to be incorporated in the design.*

Response: *The Commission realizes that the CP Rule is moot because all pending CP applications have been cancelled. However, the rule is a useful compendium of the specific requirements flowing from TMI. Some of these requirements might be shown to be unnecessary (e.g., saving space for a filtered vent) in light of the conclusions that could be justified with a PRA and severe accident judgments in a rulemaking to certify a new reference design.*

This response suggests that the requirements of 10CFR50.34(f) were to be applied to future designs as a matter of policy rather than rule. However, in response to a different comment:

Comment: *...In order to be consistent and conform with the overall philosophy of the policy statement, it is suggested that [it] be modified to indicate that the applicant must adhere to the requirements set forth in the CP Rule unless it can be demonstrated that specific requirements of the CP Rule are not cost-effective.*

Response:*...A requirement to meet the CP Rule would not be different from the requirement to meet other Commission regulations. Specific exemptions can be granted, if justified. (See also the preceding response regarding mootness of the CP Rule.)*

This response seems to imply that the CP Rule would be applicable to new designs, in spite of the specific language of the rule limiting its applicability.

The Commission has had ample opportunity to modify the language of the rule and has not. Thus, the record to indicates the rule is not binding on new LWR applications but serves as a compendium of requirements to be applied in the implementation of the Commission's policy on severe reactor accidents in a manner consistent with the first staff response quoted above. In that context, elimination of the requirements from the policy statement would have a negative impact on safety, and elimination of the rule by relocating the compendium of requirements, perhaps by incorporation in the policy statement, would provide no benefit. Consequently, this potential action will receive no further consideration.

Regulatory Position No.6

10CFR50.34(g)
Conformance with the SRP

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate/Replace*

BCL Comments: Part 50.34 Paragraph (g) requires all applications for operating licenses, construction permits, manufacturing licenses, and design approvals for standard plants docketed after May 17, 1892 (i.e , all future reactors) to contain "an evaluation of the facility against [the then current] Standard Review Plan." The regulation requires the applicant to identify, describe and justify any differences in design features, analytical techniques and procedures from those included in the acceptance criteria of the Standard Review Plan. It goes on to assert that "the SRP is not a substitute for the regulations, and compliance is not a requirement."

While the intent of the regulation is to enable flexibility without sacrificing assurance of safety, the realized effect is to discourage innovation. The disincentives arise from having to justify a departure from a previously accepted design feature or method, even though the designer and operator may believe the innovation represents a net improvement in safety, and the attendant uncertainty associated with its ultimate approval. The potential action would eliminate this part of the regulations or perhaps replace it with a more clearly explained statement of policy.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Elimination of 10CFR50.34(g)

This action is within the scope of this project and no staff action on this issue is underway or planned. Therefore, further consideration is appropriate.

Replacement with a Clear Statement of Policy

SCIENTECH considers 10CFR50.34(g) to be a clear statement of policy. Replacing it with an alternative, equally clear statement of policy would not change requirements in any way. This action is not within the scope of this project.

STEP 2: Background Description

Rulemaking Motivation

The Standard Review Plan (SRP) is a guide to the staff, first issued in 1975, for use in reviewing license applications for compliance with 10CFR. In Office Letter No. 2 issued August 12, 1975, the Director of the Office of Nuclear Reactor Regulation authorized use of the SRP to assure a consistent evaluation of license applications.

The SRP is organized to parallel the format and content of the FSARs submitted by applicants in accordance with Regulatory Guide 1.70. It incorporates all of the licensing criteria that are applied by the staff in its licensing reviews, including reference to Regulatory Guides, industrial standards and various SRP appendices that present Branch Technical Positions. Changes are currently made to the SRP in a controlled manner, after a review within the staff and issuance of a draft for public comment.

Over the first few years after issuance of the SRP, there was a gradual transfer of responsibility, from the NRC staff to the license applicants, for identifying and justifying deviations from the SRP. After the accident at Three Mile Island, the SRP became an important source of requirements for most NRC reactor licensing activities and responsibility for justifying deviations completely shifted from the staff to the licensees. In 1982, to ensure uniform practice, the NRC codified in 10CFR50.34(g) the requirement that future license applications include the applicant's evaluation of the facility against the SRP and justify any deviations.

Requirements for Licensees

License applications are required to include an evaluation of the facility against the SRP and an explanation of how any differences in design features, analytical techniques and procedural measures relative to the SRP provide an acceptable method of complying with NRC rules and regulations.

Intent of the Requirement

10CFR50.34(g) is intended to document the licensees' rationale and basis for approaches to regulatory compliance that are not consistent with previously accepted criteria. This provides the staff with a basis for evaluating the depth and scope of the licensees' safety considerations, as well as the adequacy of the alternative approach. This regulation was, in part, motivated by concern that NRC had allowed some plants, e.g., TMI-2, to be less safe than others by not imposing the same requirements on all plants.

STEP 3: Determination of Importance to Safety

The lack of an adequate explanation of how the design features, analytical techniques and procedural measures established by the licensee meet NRC regulations could, and would, have a potentially significant negative effect on safety.

To avoid this effect, which would most certainly occur in the absence of a requirement that the licensee provide such an explanation, the staff would need to develop sufficient information in the course of its licensing review to justify deviations from the SRP. It is unlikely that NRC would be able to secure additional resources for such work. Inability to secure more resources would decrease the staff's ability to evaluate the depth and scope of the licensee's safety considerations, and make it more difficult for the staff to reach an overall safety conclusion. The net effect could be less than adequate safety review or delays in safety reviews. Also, requiring the NRC staff to justify deviations from established safety requirements would tend to shift responsibility for safety from the licensees to the NRC. For these reasons, this action will receive no further consideration.

Regulatory Position No.7

10CFR50.44
Standards for combustible gas control

DRAFT BCL ANALYSIS

BCL Potential Actions: Convert

BCL Comments: This regulation, commonly referred to as the "hydrogen rule" was adopted in 1978 as a response to TMI and modified subsequently. It requires that every operating LWR be provided with means to manage combustible gas, primarily hydrogen, that can be generated during an accident. It set standards for operating licenses and construction permits. The regulation is highly specific, comparable to those covering ECCS performance (50.46), emergency planning (50.47), fire protection (50.48), electrical equipment qualification (50.49), and their associated appendices. Its implementation has caused many changes in operating reactors including containment inerting, recombiners, high point vents, and reanalysis of containment response. A major issue here has been the regulation's assumption of 75% metal-water reaction and the implication that such extensive reactions can only occur beyond the plant's design basis.

The possible action is directed toward retaining the intent of the regulation but permitting licensees greater flexibility in satisfying the need. One possibility is to incorporate that intent into one or more of the General Design Criteria and relegating the more specific aspects of 50.44 to the Standard Review Plan or a RegGuide.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Although the BCL analysis is in error in several respects, the proposed action is within the scope of this project and there is no present or planned staff action in this area. Therefore, further consideration of the action is appropriate.

STEP 2: Background Description

Rulemaking Motivation

In 1971, 10CFR50 Appendix A included general requirements for the control of combustible gases inside containment in General Design Criterion 41 "Containment Atmosphere Cleanup" and General Design Criterion 50 "Containment Design Basis." Additional guidance was provided in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

In October 1978, the Nuclear Regulatory Commission (NRC) amended General Design Criterion 50 and established a new section, §50.44, to establish specific standards for the control of hydrogen, including a method and basis for calculating the amount of hydrogen generated after a loss-of-coolant accident. This rule led to the inerting of Mark I containments.

The accident at Three Mile Island, Unit-2, (TMI-2) showed that significant metal-water reactions could occur for loss-of-coolant accidents beyond the design basis. In October 1980, the Commission committed to a long-term rulemaking related to degraded-core and core-melt accidents beyond the design basis, and proposed to amend its current regulations to introduce prudent interim safety measures. The hydrogen generated by a loss-of-coolant accident, such as the one that occurred at TMI-2, received special attention by the Commission.

In February 1981, Unresolved Safety Issue (USI) A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," was initiated and provided a focus for the NRC's rulemaking and technical research efforts associated with hydrogen control. USI A-48 dealt with all containment types for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) with the exception of large, dry PWR containments. Large, dry PWR containments, such as the one at TMI-2, were excluded because preliminary analysis indicated that essential equipment would function during and after a large hydrogen deflagration and that the containment structure was able to withstand pressures significantly greater than design pressures.

A new regulation requiring inerting of BWR Mark II containments for hydrogen control was published in December 1981. A rule for BWRs with Mark III containments and PWRs with ice condenser containments was published in January 1985; it required a means for controlling the quantity of hydrogen produced by a 75% fuel-cladding metal-water reaction, but did not specify the method of control.

The research on hydrogen control has been completed and reviewed by the National Research Council Committee on Hydrogen Combustion. Based on the research results, the NRC has concluded that the interim rules established in 1981 and 1985 provide adequate protection for the public health and safety, and that no additional regulatory requirements are needed.

Requirements for Licensees

The requirements for hydrogen control are numerous and specific. A summary of these requirements is provided below.

- Capabilities must be provided to monitor and control combustible gas concentrations in the containment following a postulated loss-of-coolant accident.
- It must be shown that an uncontrolled hydrogen-oxygen recombination will not take place in the containment or that the plant could withstand the consequences of such a recombination. If these conditions can not be

demonstrated, the atmosphere inside the containment must be made inert. (Atmospheres inside BWR Mk I and II type containments must be made inert regardless.)

- High point vents must be provided for the reactor coolant system, the reactor vessel head, and other systems required to maintain adequate core cooling.
- Specific requirements are provided regarding the amount of hydrogen that must be considered during postulated loss-of-coolant accidents.

The rule also requires equipment necessary for safe shutdown and containment integrity to be qualified for the environmental conditions resulting from hydrogen deflagration or detonation, as appropriate.

The BWR Mark I and Mark II containments are required to operate with an inerted atmosphere (by addition of an inert gas, such as nitrogen), which effectively precludes combustion of any hydrogen generated. Thus, the recommendations of USI A-48 have been fully implemented at BWR plants with Mark I and Mark II containments.

Intent of the Requirement

The current requirements in 10CFR50.44 are intended to assure adequately conservative hydrogen control in operating nuclear power plants. More specifically, the rule imposes an inerted atmosphere on BWR Mark I and Mark II plants as the only acceptable means of achieving adequate control. For BWR Mark III and PWR ice condenser plants, the rule does not specify a method, but does specify 75% metal-water reaction as a design basis.

The rule is also intended to assure that attention is given to appropriate qualification of equipment that is required to function during or after hydrogen deflagration or detonation.

STEP 3: Determination of Importance to Safety

When the NRC published NUREG-1370 to resolve USI A-48 in September 1989, it concluded that the current regulatory requirements were necessary and sufficient to provide adequate protection for the public health and safety. No new safety information since 1989 calls this conclusion into question. Consequently, any conversion should ensure that all current requirements remain intact. This could be accomplished by the proposal to retain a general regulation and place the more specific requirements in a Regulatory Guide or in the SRP.

The determination of whether such a conversion would be of marginal importance to safety is conditional. The conversion would provide both the staff and the licensee or applicant with an opportunity to deviate from the detailed requirements specified in the Regulatory Guide (or SRP) with much less formality and effort than is required for exemptions from the rule. It also would allow the licensee to determine, pursuant to 10CFR50.59, whether a plant change requires prior NRC approval. Conversion would have marginal importance to safety provided that deviations are reviewed and approved on the basis of the same technical considerations that would apply to exemptions from 10CFR50.44, and that licensees correctly implement 10CFR50.59 for any changes made to systems affected by 10CFR50.44.

STEP 4: Impact Analysis

The hydrogen control requirements of 10CFR50.44 are applicable to all current and future "boiling or pressurized light-water nuclear power reactor[s] fueled with oxide pellets within cylindrical zircaloy cladding." Current and "evolutionary" PWR designs utilizing large dry containments require no particular hydrogen control methods under current requirements. For other current and evolutionary designs, the conversion of 10CFR50.44 to a Regulatory Guide (or a position in the SRP) would provide both the staff and the licensee or applicant with an opportunity to deviate from the detailed requirements specified in the Regulatory Guide (or SRP) with much less formality and effort than is required for requirements in 10CFR. It also would allow the licensee to determine, pursuant to 10CFR50.59, whether a change requires prior NRC approval. For any deviation from the specific requirements, including changes subject to 10CFR50.59, there would be a decrease in staff and licensee effort. The magnitude of the net benefit would increase with an increase in the number of deviations. SCIENTECH considers that the number of future deviation proposals from operating reactors will not be large.

Advanced reactor applicants make various claims about containment and severe accidents (involving hydrogen generation) that will require special attention by NRC apart from present hydrogen control requirements for LWRs.

It is possible that conversion could have adverse effects, such as: a) a number of licensee proposals for deviations that overwhelms the staff's capability to perform timely, adequate safety analyses, b) an increase in enforcement problems related to unacceptable licensee interpretations of 10CFR50.59, and c) licensee proposals and/or staff actions that stimulate public concern and intervention in the form of petitions (for operating plants) or hearing issues (for new applications). Each of these possibilities would result in increased effort, reducing the net savings associated with the presumably lower cost of the deviations.

The inerting of BWR Mark I and Mark II containments encountered significant opposition from BWR owners who felt the requirement was unnecessary, dangerous for operations personnel and too constraining of efficient operations. If the rules on inerting are converted to Regulatory Guides, it should be expected that this controversy will be rekindled at potentially high cost to NRC staff resources. In addition, some members of the public may see the conversion action as an attempt to weaken the requirements. As discussed in Section ii of this report, this public concern may be preempted, or at least dealt with efficiently, by a rulemaking action, possibly preceded by a policy statement, to convert several detailed rules at the same time.

In view of the complex potential interactions involved, the assessment of burden and benefit presented in the table below is highly uncertain.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Medium	Medium

Regulatory Position No.8

10CFR50.46
Acceptance Criteria for ECCS

10CFR50 Appendix K
ECCS Evaluation Models

DRAFT BCL ANALYSIS

BCL Potential Actions: *Expand*

BCL Comments: *As currently constituted, these regulatory positions address only fuels composed of uranium dioxide clad with Zircaloy. The potential action is to expand the scope of these and related regulatory positions by adding guidance for reload cores, some of which include cladding other than Zircaloy and some of which entail significantly higher burnup than was originally envisioned.*

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action is not within the scope of this proje because it would increase the licensing effort by current and future licens and the NRC.

Expansion or addition of a regulatory requirement can reduce the burden on licensees when it provides a generic resolution of a burdensome issue. There is no evidence to suggest that this criterion is met by the expansion proposed for this regulatory position.

[Note: Although not proposed by BCL, the conversion of Appendix K to a Regulatory Guide, with appropriate modifications of other sections of Part 50, would have essentially the same value and impact as those described in the Introduction and in the evaluation of the proposed conversion of Appendix R.]

Regulatory Position No.9

10 CFR 50.49, Environmental Qualification
SRP 3.11
Regulatory Guide 1.89

DRAFT BCL ANALYSIS

BCL Potential Actions: *Eliminate/Update*

BCL Comments: This regulation deals with the ability of a broad array of equipment to survive the environments postulated to be associated with accidents in operating reactors. This rule was originally adopted in 1983, largely in response to findings of confirmatory research on the operability of electrical equipment in various thermal, radiation and humidity environments. The rule imposed a major burden on the licensees and NRC staff. It requires licensees to develop and execute an extensive program for qualifying their plants' electrical equipment, to document and report the results, to plan replacement of nonqualifying equipment, and to satisfy listed schedules for all of the above. At least three major controversies accompany this rule: the scope of equipment it covers, the environmental test conditions, and the implementation schedules.

The rule explicitly applies to "safety-related electrical equipment," "nonsafety-related electrical equipment" whose failure under the proposed environments would prevent satisfactory performance of the former, and "certain post-accident monitoring equipment." One potential action would be to reduce the scope of the rule to only that equipment whose malfunction directly impairs satisfactory achievement of a safety function, i.e., only safety-related electrical equipment. This reduction could be achieved by eliminating paragraphs (b)(2) and (b)(3) of the rule.

The rule specifies the environmental parameters (e.g., temperature, pressure, radiation, aging, etc.) which must be included in the qualification program, as well as the need to consider synergistic effects and margins. RegGuide 1.89 is the more specific guidance on accident environments and test conditions. In essence it specifies environmental conditions attendant to design basis accidents, including radiological source terms characteristic of RegGuides 1.3 and 1.4. The potential action would be to modify the RegGuide to represent more realistically the environmental conditions suggested by more modern thought. Thus potential action on this matter is tied to potential action on the source term "megaissue."

All deadlines specified in the rule have passed. In some instances NRC has granted extensions for "good cause" or "sound reasons." Some licensees have still not demonstrated compliance. The rule contains no explicit provisions for applicability to operating licenses granted after November 30, 1985. Therefore it is unclear what requirements must be satisfied by future reactor designs.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

Eliminate 10CFR50.49(b)(2) and (b)(3)

The proposed action regarding 10 CFR 50.49 is within the scope of this project because it would reduce applicant and NRC effort. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of this action is appropriate.

Update Regulatory Guide 1.89

Modification of Regulatory Guide 1.89 to reflect new knowledge concerning accident and post-accident environments would provide greater confidence in the ability of the equipment to function effectively in those environments. However, SCIENTECH considers it more likely that such an update would result in a net increase in NRC and licensse effort. Consequently, this action is not within the scope of the project and will be given no further consideration.

Update Schedule

BCL's inference that, "The rule contains no explicit provisions for applicability to operating licenses granted after November 30, 1985" is an incorrect interpretation of the many requirements in the rule that address deadlines for compliance and licensee justification for continued operation pending completion of compliance. The technical requirements in the rule apply to each holder of or applicant for a license to operate a nuclear power plant, regardless of docketing dates. The potential action is unnecessary and inappropriate. Therefore, it is outside the scope of this project and will receive no further consideration.

Step 2: Background Description

Rulemaking Motivation

Early in the development of nuclear power, electrical components were expected to be of high industrial quality, but no exacting requirements were specified. As more information became available, the NRC began to rely on more specific standards. General Design Criterion 4 of 10CFR50 Appendix A, issued in Feburary 1971, required that equipment important to safety be designed to accommodate environmental conditions associated with postulated accidents, among other things. For nuclear plants licensed after 1971, qualification of electrical components was judged on the basis of the consensus national standard, IEEE 323-1971. Plants with Safety Evaluation Reports issued after July 1, 1974 were evaluated using Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Light-Water-Cooled Nuclear Power Plants," which endorses IEEE 323-1974.

In the late 1970s, some NRC-sponsored research raised concerns with regard to the level of assurance that safety equipment would continue to operate in the harsh environments that might follow an accident. In December 1979, the staff issued NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Its purpose was to provide guidance to NRC staff on improved qualification techniques, primarily testing for new and near-term OL plants, and to aid a more orderly and systematic implementation of qualification programs by industry. On May 23 1980, in the midst of considerable controversy and petitions to shutdown operating reactors because of this issue, the NRC issued a Memorandum and Order directing that all operating plants complete the qualification of safety-related electric equipment in accordance with specific guidance (Division of Operating Reactors Guidelines and NUREG-0588) no later than June 30, 1982.

In January 1982, the NRC sought to close the controversy in this important safety area by modifying its regulations at 10CFR50.49 to issue additional requirements and to formalize the changes first issued as NUREG-0588. Until that time, qualification methods contained in national standards, regulatory guides and other NRC publications had been given different interpretations and did not have the force of agency regulation. In June 1982 an interim Part 50.49 was adopted and in January 1983 the final rule was promulgated by the NRC. In June l984, the NRC issued its latest revision to Regulatory Guide l.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants."

Requirements for Licensees

Pursuant to 10CFR50.49, licensees or applicants must have a program for qualifying electric equipment important to safety. Electric equipment important to safety includes 1) safety-related electric equipment, 2) nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions, and 3) certain post-accident monitoring equipment. The electric equipment qualification program must include and be based upon:

- Temperature and Pressure,
- Humidity,
- Chemical Effects,
- Radiation,
- Aging,
- Submergence,
- Synergistic Effects, and
- Margins to account for uncertainty.

Qualification methods are the following:

- Testing of an identical item under identical or similar conditions with supporting analysis,
- Testing of a similar item with supporting analysis,
- Experience with identical or similar equipment under similar conditions with supporting analysis, and
- Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

Recordkeeping requirements are specified. Additionally, a large portion of the rule discusses deadlines for compliance and requirements for licensee justification for continued operation pending completion of equipment qualification in accordance with the requirements. Because of the complex technical issues involved, implementation of these environmental qualification requirements by licensees took longer than originally anticipated.

<u>Intent of the Requirement</u>

10CFR50.49 is intended to ensure that electric equipment important to safety is capable of performing its safety functions during and after exposure to environmental conditions associated with normal operations and design basis accidents.

STEP 3: Determination of Importance to Safety

<u>Eliminate 10CFR50.49(b)(2) and (b)(3)</u>

10CFR50.49(b)(2) requires environmental qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions. Eliminating the requirement would open the possibility that equipment could fail to adequately perform its safety function in an accident. SCIENTECH considers this to be a significant negative impact on safety. No further consideration will be given to this action.

10CFR50.49(b)(3) requires environmental qualification of certain post-accident monitoring equipment. This was an important lesson learned from the TMI-2 accident where some important monitoring equipment was not qualified for the post-accident environment and failed, leaving the operators unaware their actions were contributing to worsening plant conditions. SCIENTECH considers that a requirement for a post-accident monitoring capability implies a requirement that the equipment be qualified to survive the accident environment and function satisfactorily in the post-accident environment. This view has been shared by the NRC, the industry and the Congress for a number of years, and there is no new technical information to change it. No further consideration will be given to this action.

Regulatory Position No.10

10CFR50.55a, Codes and standards

DRAFT BCL ANALYSIS

BCL Potential Action: *Update/Clarify/Expand*

BCL Comments: This Part was originally adopted in 1971 and has been revised nearly every year since then. It amplifies General Design Criteria 36, 37, 39, and 40, which address inspection and testing of safety systems. The original focus of 50.55a was on requirements for In-Service Inspection (ISI). The regulatory philosophy was to endorse applicable current industry standards (e.g., ASME standards). It is one of the few (the only?) parts of the regulations which requires a licensee to modernize procedures and equipment in accordance with the latest approved industry standards.

In 1984, NRC modified the rule to include In-Service Testing (IST) primarily in response to the Davis-Besse loss of feedwater incident. The rule change required all 105 operating reactors to develop and document IST programs. As of late 1988, there were 12 such programs submitted, reviewed, and approved via a Safety Evaluation Report. The rest are in progress and are usually characterized by very many requests for exceptions.

The potential modification represents a package of changes which will collectively reduce regulatory burden and enhance safety. The more important specific changes suggested are:

- *Segregate the text addressing IST from that addressing ISI and add supplemental clarifying text where appropriate*
- *Eliminate the requirement for NRC to review and approve every change to a licensee's ISI and IST programs*
- *Encourage (require?) a process whereby deviations from ASME Codes are justified by licensees rather than preapproved by NRC*
- *Develop a Reg Guide which documents generally approved deviations from the ASME Code (e.g., Section XI incorporating OM-6 and OM-10)*
- *Require licensees to maintain all relevant information on-site and available for inspection*
- *Place more emphasis on system performance relative to component testing*
- *Impose a one-time mandatory update to the ISI and IST programs of all operating reactors in the 1992 time frame; then allow all subsequent updates to be voluntary except for ASME Code revisions*
- *Develop/introduce standards and criteria for testing the instrumentation and control portion of safety systems*
- *Permit more flexibility in scheduling "ten-year inspections" at multiple unit sites.*

The potential actions are motivated by a need to improve the in-service programs themselves as well as the process by which they are reviewed and implemented. The suggested changes would represent a major impact on licensees with relatively weak in-service programs and a significantly lesser impact on licensees that currently have stronger programs.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Although the BCL report presents the potential action as "a package of changes," two of the listed items are excluded from further analysis based on the following considerations:

- Place more emphasis on system performance relative to component testing

System performance is not usually addressed by the ASME Code so this change would simply add the emphasis on system performance to the Code requirements. This potential action is outside the scope of this project and will receive no further consideration.

- Develop/introduce standards and criteria for testing the instrumentation and control portion of safety systems

Such standards and criteria may be needed, but they would represent an increase in staff and licensee effort and are thus outside the scope of this project and will receive no further consideration.

The remaining actions are derived from, or related to, the proposals made by the NRC Director of Nuclear Reactor Regulation (NRR) in his May 12, 1988 memorandum to the Director of Nuclear Regulatory Research, "Proposed Amendment to 10 CFR 50.55a - Codes and Standards." The motivation for the NRR proposals was an assessment that the staff resources required for timely review of relief request submittals, program revisions and program updates pursuant to 50.55a were "extremely heavy and virtually insupportable." The burden on licensee staff resources is known also to be high. It is clear that the revised procedures for handling deviations from the ASME Code and the additional guidance for implementation of 50.55a that are outlined in this memorandum would reduce the administrative burden of the licensees as well as the requirements for staff review. However, it should be noted that NRR proposed a larger body of regulatory guidance than has been listed by BCL, which would be likely to increase the detailed technical requirements associated with 10CFR50.55a.

There appears to be a potential for significant net benefits arising from a thorough review and revision of 10CFR50.55a. A potential action, which will be characterized below as "Major Revision of 10CFR50.55a, is within the scope of this project and further consideration is appropriate.

Step 2: Background Description

<u>Rulemaking Motivation</u>

The purpose of 10CFR50.55a is to formally endorse specific parts and editions of the ASME Boiler and Pressure Vessel Code, with certain specified limitations. The edition of the ASME Code referenced in the rule is updated every year or two to ensure that the latest engineering technology is incorporated into the safety requirements imposed on the licensees.

<u>Requirements for Licensees</u>

Licensees are required to conform to, or justify deviations from, the latest referenced version of the ASME Code. Problems arise from lack of uniform understanding of the applicable scope of the rule and from the schedular requirements. It appears that there are differences between the language of the rule, the staff interpretation of the rule, and the implementation of the rule that arise out of the need to find ways of overcoming the practical difficulties encountered.

The effect of subsections (g)(4) and (g)(5) is to require all licensees to develop revised In-Service Inspection and In-Service Test programs that conform to recently updated ASME standards on a schedule that allows the identification of needed changes to the Technical Specifications within six months and of any necessary deviations from the updated standards within twelve months. The schedule is conditioned upon the schedular requirements for requests to the NRC for Technical Specification changes and for relief from the implementation of updated standards that the licensee considers to be impractical.

The implementation of these requirements results in the flood of relief requests that NRR's May, 1988 memorandum calls "extremely heavy and virtually unsupportable."

<u>Intent of the Requirements</u>

The intent is to assure, in light of current technology, the integrity of the reactor coolant system and associated engineered safety features. The integrity of theae systems and features is an important prerequisite to the underlying assumptions in many of the NRC's safety analyses.

Step 3: Determination of Importance to Safety

Any major revision of 10CFR50.55a will require the underlying basis for the requirements to be analyzed carefully. Clearly, the NRC endorsement of a later version of the ASME Code does not change the level of safety offered by an operating plant. Thus, the requirement that plants update their programs to conform to later editions must be directed either to a desired increase in the level of safety or change (up or down) in the level of conservatism. As is pointed out elsewhere in this report, the introduction of regulatory conservatism is not always explicitly done or explained, which makes analysis too difficult to be accomplished within the scope of this project. For this project, SCIENTECH assumes that the rulemaking process attendant upon any major revision of 10CFR50.55a would successfully guard against any significant negative effect on safety.

Step 4: Impact Analysis

The proposed changes represent a potentially complicated rulemaking that would probably last for several years, coupled with a demanding program of guidance development. The outcome probably would increase the efficiency with which the objectives of 50.55a are achieved. However, the large rulemaking effort and the additional technical burden on the licensees make the possible net reduction of effort highly uncertain.

The central technical problem is one of determining what plant-specific actions are appropriate in response to changes in generic industry standards. The current rule resolves this regulatory problem by staff review of the licensees' plans. Relieving the staff of this burden would require an acceptable degree of assurance that the licensees' approaches will be sound in the absence of staff review. The NRR memorandum recognizes this by coupling the reduction of review with the development of additional guidance and audit of the licensees' records. NRC guidance would relieve the problem to the extent that less generic, or more detailed derivative standards can be developed. It is not clear at present whether and to what extent this can be done.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Low

Regulatory Position No.11

10CFR50.55a(h)
Codes and standards
Protection systems

DRAFT BCL ANALYSIS

BCL Potential Actions: *Update*

BCL Comments: The paragraph currently codifies IEEE 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations." An industry working group is currently revising IEEE Std 279-1971.

The rule should be updated to codify IEEE Std 603-1980 "Criteria for Safety Systems for Nuclear Power Generating Stations," a newer, more comprehensive standard. The requirements and recommendations of IEEE Std 603-1980 on the power, instrumentation and control portions of safety systems incorporate the requirements and recommendations of IEEE Std 279-1971, which is limited to protection systems only. Currently operating reactors should be grandfathered.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

IEEE 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE 603-1980 "Criteria for Safety Systems for Nuclear Power Generating Stations" have different purposes. IEEE 603 provides general functional criteria for the sensing and signal processing parts of safety systems (i.e., protection systems), while IEEE 279 provides design criteria for the same protection systems. Consequently, one cannot replace the other. Adding IEEE 603 would increase the licensing effort by current and future licensees and the NRC, therefore, the proposed action is not within the scope of this project.

Regulatory Position No.12

10CFR50.59
Changes, tests and experiments

DRAFT BCL ANALYSIS

BCL Potential Action: *Clarify*

BCL Comments: Under Part 50.59, a licensee may make a change, test or experiment without prior approval by the NRC unless the change, test or experiment involves a change in a technical specification or an "unreviewed safety question." The rule then proceeds to define what the licensee must do in the latter case in terms of recordkeeping, safety analysis, and reporting.

The crux of this matter is in determining if a change involves an unreviewed safety issue, the so called "50.59 evaluation." The rule provides descriptive language to help define an unreviewed safety issue, the basic intent of which is to assure that the plant remains within the envelope represented by the safety analysis report. The evaluation focuses on design basis events only, since these are the events of interest in the plant's safety analysis report. Further, if the licensee performs such an evaluation and concludes that a technical specification change or an unreviewed safety issue is involved, such change, test or experiment has to be achieved through an amendment to the operating license pursuant to Part 50.90, a substantially involved process which licensees usually seek to avoid.

The potential action is to clarify 50.59. Most commenters understand the need for the regulation, but many find its language ambiguous, vague or confusing. The result is perceived to be too many evaluations submitted which are not really significant from a safety standpoint and probably some evaluations which are significant but are not submitted for review by the NRC. An industry task force (NUMARC?) is currently reviewing this matter at the request of NRC's management (5/27/86). Their task is to develop review criteria and guidelines for licensees conducting 50.59 evaluations. They expect to issue their findings and recommendations in (when?).

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Improved criteria and guidelines for 50.59 evaluations have the potential for reducing NRC and licensee effort by reducing the number of unnecessary submissions to the NRC. Such guidance could also improve safety by improving the effectiveness of the licensees' implementation of this requirement. The proposed action is within the scope of this project.

STEP 2: Background Description

Rulemaking Motivation

The AEC amended 10 CFR Part 50.59 in 1974 to simplify the separate procedures governing the handling of "amendments" and "changes" then in effect. The rule resolves the question of what licensee changes, tests and experiments at an operating nuclear power plant require prior NRC approval in the form of a license amendment. The rule also addresses the issue of maintaining records and submitting reports to the NRC with respect to other changes, tests and experiments.

Requirements for Licensees

For any change in the facility or in the procedures described in the safety analysis report, and for any test or experiment not described in the safety analysis report, the license must perform a safety analysis and determine whether the change, test or experiment involves a change in the technical specifications or an unreviewed safety question.

For any of the actions described above that involved an unreviewed safety question or for any change in the technical specifications, the licensee must submit an application for a license amendment.

For any of the actions described above that do not involve an unreviewed safety question or a change to the technical specifications, the licensee must maintain records of the change, test or procedure and of the licensee's evaluation pursuant to 10CFR50.59. In addition, the licensee must furnish an annual report to the Commission containing a brief description and a summary of the safety evaluation of each such change, test, or experiment.

Intent of the Requirement

10CFR50.59 is intended to ensure that any changes, tests, or experiments performed at licensed plants are controlled and receive adequate licensee review for safety hazards and considerations and appropriate licensing review by the NRC for any change to a plant's technical specifications or any changes, tests, or experiments which would involve an unreviewed safety question, thereby preserving the licensing basis for the safety of the plant.

STEP 3: Determination of Importance to Safety

Clarifying the meaning of a rule could result in improved implementation and a consequential improvement in safety. However, the issue of clarity may lie less in the language of 10CFR50.59 than in the lack of a uniform understanding of the actions that are appropriate to determine whether the criteria are met. The NUMARC review mentioned by the BCL report resulted in the June 1989 publication of NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations."

Although slightly less limiting than a literal interpretation of 10CFR50.59 would be, the practical guidance provided by NSAC-125 should significantly improve licensees' ability to identify those things that should be evaluated under 10CFR50.59 and to perform the evaluations. This improvement should result in some, as yet unquantifiable, reduction in risk.

STEP 4: Impact Analysis

The long term benefit of clarification to achieve a more uniform understanding of the scope of 10CFR50.59 would be a more effective and efficient licensee screening procedure and a reduction in the number of inappropriate 10CFR50.59 submissions to the NRC. These benefits might be reduced to some degree by the recognition of USQs that would not have been identified without the added guidance. However, the recently published industry guidance in NSAC-125 appears to represent the maximum level of clarification for which a consensus could be developed at that time. Consequently, it appears that a large NRC effort would be required to significantly improve and extend the existing guidance. Because of the industry strong interest in the outcome of this effort, there would be a relatively high degree of involvement by licensees in the development of the additional guidance.

The long term benefit of more detailed guidance could be quite high for the licensees because the screening and evaluation of potential 50.59 items would be simpler and more readily reduced to routine. On the other hand, the NRC effort would not change drastically if, as is expected, the elimination of unnecessary submissions is balanced by an increase in the scope of items judged to be subject to 50.59.

Both the burden of developing additional guidance and the resulting benefit could increase sharply if other regulatory changes were also made. For example, the scope of 50.59 coverage would increase if the detailed requirements now in 10CFR (e.g., Sections 50.46, 50.47, 50.48 and 50.49) were converted to Regulatory Guides or positions in the SRP. Similarly, if the Technical Specification Improvement Program results in less detailed Technical Specifications more evaluations under 50.59 would be required. The SCIENTECH evaluation of burden and benefit was done with the assumption that no other major changes would be made. Although both burden and benefit would increase if these other changes are made, it is probable that the increase in benefit would be proportionately greater.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Medium	Low	High

Regulatory Position No.13

10 CFR 50.62, ATWS

DRAFT BCL ANALYSIS

BCL Potential Actions: *Replace*

BCL Comments: After many years of technical analysis and debate, the NRC adopted Part 50.62 in its final form in 1984. The regulation applies to all commercial light water reactors. Within it there are specific additional elements which apply to particular types of reactors only. Boiling water reactors receive particular attention. The requirements relate to hardware "fixes" and include a schedule for final implementation.

Once the currently operating reactors have modified to comply with the regulation, the only continuing purpose for the regulation is enforcement. Some suggest that for future reactors, ATWS will be "designed away" and thus can be accommodated within the General Design Criteria. The potential action is to limit the current 50.62 to currently operating reactors and for future reactor designs to replace it with some as yet unspecified alternative documented position.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Limiting the current requirements in 10CFR50.62 to operating reactors, without a specific alternative for new reactors provides no basis for confidence that the ATWS issue would be effectively resolved and must be viewed as a negative impact on safety. The lack of some specific alternative for specifying scram reliability and ATWS mitigation capability, i.e., "designing away" ATWS, in future reactors provides no basis for assessing the potential benefit of such a change. The future approach could conceivably require more NRC and licensee effort and expense than the current approach. Consequently, the proposed action is not within the scope of this project and no further consideration will be given to this action. Additional information is provided in Step 2 to support this conclusion.

STEP 2: Background Description

Rulemaking Motivation

Anticipated Transients Without Scram (ATWS) are expected operational transients (such as a loss of feedwater, condenser vacuum, or offsite power) accompanied by a failure of the reactor trip system to shut down the reactor. Under certain postulated conditions, severe core damage and release of radioactivity could result from an ATWS event. Backup safety systems and procedures for shutting down the reactor given the failure of the reactor protection system are therefore important to safety.

There was a partial failure of the scram system at a Browns Ferry reactor on June 28, 1980. In November 1981, the NRC proposed three rule alternatives for dealing with ATWS events. One alternative had been developed by the industry, one by the NRC staff, and one by a former NRC Chairman. On February 25, 1983, two limited ATWS events occurred at the Salem 1 nuclear generating station. However, no core damage or release of radioactivity occurred. (The reactor was scrammed manually about 30 seconds after failure of the automatic trip system.)

Following extensive public comment on the proposed rules, and consideration of lessons learned from the Salem event, the NRC issued a final rule in 1984 requiring improvements in reactor designs to reduce the probability and mitigate the likely consequences of an ATWS event.

Requirements for Licensees

10CFR50.62 specifies system design requirements for each main reactor design type. Requirements include diversity of scram systems for equipment; diversity of auxiliary feedwater initiation and turbine trip mechanisms for pressurized water reactors (PWRs); increased capability for the standby liquid control system (SLCS) of boiling-water reactors (BWRs); and an automatic recirculation pump trip mechanism for BWRs.

The requirements of the ATWS rule have been implemented differently at different plants. Radiation exposure and other considerations necessitated differences in the manner in which some older BWRs implemented the requirements for SLCS. BWR plants that had been granted a construction permit prior to July 26, 1984, and had not been designed and built to include the required SLCS features, were not required to install new equipment. The Commission found these plants to be sufficiently safe, given other hardware requirements and the implementation of new guidelines for emergency operating procedures.

Intent of the Requirement

10CFR50.62 is intended to reduce the likelihood of failure of the reactor protection systems to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event, should one occur.

Regulatory Position No.14

10 CFR 50.71(e)
Maintenance of records
Making of reports.
FSAR Update

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate/Replace*

BCL Comments: *Part 50.71 addresses the retention of records and reports required by the regulations, technical specifications and conditions of the license. Paragraph (e) of this part specifies further that all operating licensees must periodically update the Final Safety Analysis Report and submit such updated information to the NRC. Examples of required updates includes the effects of all physical or procedural changes to the facility as described in the FSAR, all safety evaluations supporting license amendments or Part 50.59 evaluations, and all analyses of new safety issues. The rule provides a timetable for updating FSARs and adds an information item specific to operating reactors which were included in the NRC's Systematic Evaluation Program (SEP) at the time.*

Paragraph (e) became effective in July 1980 in response to difficulties the NRC staff was encountering in assuring current information on the design and operation of plants needed to evaluate safety. It was also a time when significant changes were being implemented in response to post-TMI requirements, and there was a perceived need to assure that such wholesale changes in the plants were recorded in an orderly manner.

There are several potential actions possible. The first is to eliminate that text which distinguishes among plants that were and were not in the SEP. That language has outlived its usefulness. The second is to maintain the requirement for regularly updating the FSAR but replace that portion which makes submittal of such updates to the NRC mandatory. The licensee could be required to maintain and produce on demand a current FSAR at the plant site or other similarly appropriate location. The final option is to eliminate Paragraph (e) all together on the grounds that changing the FSAR has no direct impact on the plant's safety and represents only a burdensome exercise in generating and moving paper.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Eliminate SEP Language

As noted by BCL, the language of 10CFR50.71(e)(3)(ii) relating to plants within the Systematic Evaluation Program has no current or future applicability. Elimination would have no effect on safety, nor on the level of effort of the NRC and licensees. The only benefit to the NRC would be the shortening of the regulations by eliminating obsolete portions of the rule. This benefit is more than offset by the short-term NRC burden to accomplish the rule change. Consequently, this action is not within the scope of this project and will not be given further consideration.

Eliminate 10 CFR 50.71, Subparagraph (e)
Eliminate Updated FSAR Submittal to NRC

These potential actions are within the scope of this project. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of the actions is appropriate.

STEP 2: Background Description

Rulemaking Motivation

In 1980, the NRC issued a rule requiring the periodic updating of Final Safety Analysis Reports (FSARs). The rule stemmed from NRC and licensee concerns that the safety analysis, which guides the safe operation of the plant, be kept current with changes in the plant. There was a precedent for updating SARs in 10 CFR 50.30(c)(2), which required applicants for construction permits to update their applications. These applications included safety-related information such as the Preliminary Safety Analysis Report and preliminary emergency plans. Updating the application ensured that superseded information was removed, and the updated application provided an index of the changes made prior to the public hearing on the application. However, the NRC noted that no corresponding regulation existed for an applicant or holder of an operating license to update the FSAR. As a result of this discrepancy, 10 CFR 50.71(e) was formulated. The Commission made the following statement regarding the need to update the FSAR on a regular basis.

> Revision of the FSAR to reflect the current status of a facility's safety related structures, systems, and components would be of value to provide a reference document for recurring safety analyses performed by the applicant or licensee and the Commission. Maintenance of the FSAR in this manner will remove the need for repeated review of outdated portions of the FSAR and succeeding documents related to the outdated portions document.

The updated FSAR provides a means for the NRC and the licensee to arrive at a timely common understanding of the current safety design configuration of the plant.

Those licensees who participated in the Systematic Evaluation Program (SEP) were not required to comply with 50.71(e) because they already provided much of the information required by this rule. However, when a licensee was notified that the SEP was complete, a complete and updated FSAR had to be submitted within 24 months, with yearly revisions thereafter.

Requirements for Licensees

Each licensee must update the FSAR annually to assure that it includes the latest material developed for the facility. The updated FSAR must include the effects of:

(a) changes made in the facility or procedures described in the FSAR;

(b) safety evaluations performed by the licensee in support of license amendments, or in support of conclusions that changes did not involve an unreviewed safety question; and

(c) analyses of new safety issues performed by or on behalf of the licensee at Commission request.

The updates must include all changes made more than 6 months prior to the date of filing and may include more recent changes.

The rule requiring FSAR updates has undergone two minor amendments since it went into effect. In 1987, the NRC deleted a requirement of the rule specifying the number of copies that had to be sent to the NRC. This change resulted from a new rule that consolidated copy requirements and mailing instructions for Part 50 and 51 submittals. In 1988, the NRC added a requirement for licensees to retain the updated FSAR until the Commission terminated their licenses. This requirement was added to comply with a requirement of the Office of Management and Budget, which called for a retention period to be specified for each record requirement imposed by Federal regulations.

Intent of the Requirement

The intent of 50.71(e) is to ensure that an updated reference document exists for use by the NRC and the licensee in performing safety analyses. The updated FSAR is currently the only document required by and routinely available to the NRC that describes the current (within the limits of the submission schedule) safety-related configuration of the plant. It is a valuable reference document used by the licensee as the basis for all safety analyses and determinations involving unreviewed safety questions. It is also used by the NRC as a tool in its review of the licensee's safety analyses.

STEP 3: Determination of Importance to Safety

Eliminate 10 CFR 50.71, Subparagraph (e)

The primary purpose of 50.71(e) has not been affected by the passage of time or by technical innovation. Elimination of this rule would allow licensees to make a variety of modifications without incorporating those changes into the FSAR. As these changes accumulate, the FSAR would no longer reflect actual plant facilities, procedures, etc. This would degrade the design basis documentation for the plant and invalidate the use of the FSAR as a reference document for safety evaluations or during emergency situations. Loss of this resource would militate against the NRC and licensee arriving at a timely common understanding of the safety configuration of the plant. It could also lead to mistakes on the part of both the licensee and the NRC if incorrect or outdated information is used during safety analyses or in reviews of license amendment applications.

Elimination of 50.71(e) would have a negative impact on safety that is potentially significant. This finding completes the review of this item. No further consideration will be given to this action.

Eliminate Updated FSAR Submittal to NRC

The NRC conducts its licensing reviews and inspections against the regulations and the FSAR. If an updated FSAR is not available to the NRC, it would likely result in erroneous licensing and inspection actions. Therefore, elimination of the requirement to submit the updated FSAR to the NRC would have the same effects as are described above, except that the licensees' unilateral evaluations and decisions would be less affected because updated information could be available at the plant. SCIENTECH considers that this exception does not significantly reduce the potential negative impact on safety. Consequently, no further consideration will be given to this action.

Regulatory Position No.15

10CFR50.73 Licensee event report system

DRAFT BCL ANALYSIS

BCL Potential Action: *Replace/Expand*

BCL Comments: *This part defines reportable events and reporting procedures. The broad issue this part has traditionally raised is the scope of reportable events and whether such reports should be required versus voluntary. A specific instance has been raised wherein paragraph (a)(2)(vi) would allow the failure of an important component, such as an emergency diesel generator, to remain unreported by a licensee if a redundant component performed successfully. Left unreported, this could lead to an underestimation in the rate of diesel generator failures at a time when the use of failure rate data for PRAs is increasing. It could also allow a series of seemingly unrelated failures at many sites to continue unattended.*

If they so elect, licensees may report such failures to INPO's NPRDS or maintain their own records. The potential action is to delete 10CFR 50.73(a)(2)(vi) and introduce clarifying or replacement language which adds reporting requirements for important components. An alternative is to require licensees to keep plant-specific records of such failures on-site without having to report them to NRC.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action is not within the scope of this project because it would increase the NRC and licensee effort, therefore, further consideration of this action is inappropriate.

Regulatory Position No.16

10 CFR 50 Appendix A
General Design Criteria
Definitions and Explanations

DRAFT BCL ANALYSIS

BCL Potential Actions: *Eliminate/Clarify*

BCL Comments: Adopted in 1971, this section contains two footnotes intended at the time to provide clarifying information. Footnote 1 addresses loss of coolant accidents and says "Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development." There appears to be no continuing need for this footnote, so it can be eliminated.

Footnote 2 is intended to clarify the definition of single failure. The definition refers to active and passive components of fluid and electrical systems. The footnote says "Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development." Traditionally, in electrical systems no distinction is made between active (e.g. switches) and passive (e.g. wires) components in terms of system failure, thus the footnote does nothing to amplify the basic definition. There also appears to be no apparent need for the reference to passive components in fluid systems.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The development of additional criteria referred to in the footnotes of this section has occurred through the implementation of the Standard Review Plan (NUREG-0800) and establishment of licensing practice within the agency. Additionally, changes have been made to Criterion 4, Environmental and Dynamic Effects Design Bases, to incorporate consideration of leak-before-break considerations for high energy piping. The extensive leak-before-break rulemaking action included technical consideration related to the subject of both footnotes of this section. At present, both footnotes are obsolete and have no safety significance.

The sole benefit of the potential action would be the shortening of the regulations and elimination of potentially confusing language. The benefit is negligible and offset by the short-term burden to accomplish the rule change. No further consideration will be given to this action.

Regulatory Position No.17

10 CFR 50 Appendix A
General Design Criteria 2
General Design Criteria 4
SRP 3.9.3

DRAFT BCL ANALYSIS

BCL Potential Actions: *Update*

BCL Comments: One or more of the above should be updated to allow decoupling of design basis accident (e.g. pipe rupture) loads from seismic loads as they pertain to the design of mechanical equipment. Revisions are justified by the clearly distinguishable timing and duration of such loads which are now required to be combined. Such revisions would apply only to mechanical equipment and would primarily affect BWRs. (Most PWRs can qualify for leak-before-break and most BWRs do not.) Some staff believe that even for some piping runs in PWRs (such as main steam lines), leak-before-break does not apply and the existing regulatory positions should be retained.

Updated SRP 3.9.3 has ramifications for SRP 3.8.1 through 3.8.4, which address structural aspects (including load combinations) for containments. Any of the above revisions, if made, should retain the coupling of accident and external event loads for structures.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action regarding GDC 2 and GDC 4 is within the scope of this project. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of this action is appropriate.

STEP 2: Background Description

Rulemaking Motivation

Until the middle of the 1970s the dynamic qualification of mechanical equipment consisted of qualification for seismic loads because no other dynamic loads significant to design had been identified before that time. In 1975, dynamic loads resulting from asymmetric pressurization of the annulus just outside the reactor vessel were identified as having design significance. The NRC subsequently required these loads to be combined with seismic loads in the design of various structural and mechanical components and in the dynamic qualification of equipment. Although annulus pressurization loads had much less significance for BWR plants than for PWR plants, the 1970s saw the evolution of regulatory requirements related to several new dynamic loads associated with LOCA discharge into the suppression pools at BWRs with pressure suppression containment. These loads also were required to be combined with seismic loads in the design and qualification of equipment.

In a two stage rulemaking, culminating in an amendment to GDC 4 in October 1987, the Commission allowed the dynamic effects of pipe rupture to be excluded from the design basis where analysis showed that detectable leaks would provide adequate time for shutdown prior to a large break. This effectively eliminated LOCA loads from the dynamic qualification of equipment for PWRs that met the leak-before-break criteria. However, the pool dynamic loads at plants with pressure suppression containment depended only on the size of the LOCA, not on its location. Consequently, the combination of LOCA and seismic loads for dynamic qualification of equipment in those plants is unchanged.

The broader question, raised implicitly by the BCL report, is whether it is appropriate to combine seismic and LOCA loads when the probability of their simultaneous occurrence is considered to be exceedingly low. The Piping Review Committee noted in their report (NUREG-1061, August 1984) that "There has never been a well-developed rational basis for considering concurrent earthquake and large loss-of-coolant-accident (LOCA) loads in the design basis." The Committee suggested that the NRC continue work then underway to learn whether the probability of concurrent occurrence was low enough to exclude the event combination from the design basis.

In 1986, the staff considered decoupling seismic and LOCA loads for the mechanical design of components and their supports by revising SRP 3.9.3. the NRC Executive Legal Director expressed an opinion that such a change would require similar revisions to SRP 3.8.1 through 3.8.4, decoupling seismic and LOCA loads for structures (including containments), or a rulemaking action to justify the decoupling for components while retaining the load combination for structures. The staff chose to do nothing with regard to rulemaking and SRP revisions and to handle the relatively few licensing actions involving decoupling as deviations from SRP 3.9.3.

Requirements for Licensees

Licensees are required to qualify equipment using dynamic inputs consisting of appropriate combinations of LOCA and seismic motions.

Intent of the Requirement

The NRC intent is to ensure conservative safety margins by requiring combined seismic and LOCA loads in the design of mechanical equipment although there is broad agreement that the probability that a seismic event would occur simultaneously with a postulated design basis accident is extremely low.

STEP 3: Determination of Importance to Safety

Ever since the LOCA dynamic loads were identified in the 1970s and were required to be combined with seismic loads for design and qualification purposes, the industry has argued that the combination was excessively conservative. The NRC has consistently rejected these arguments. It is worth noting that the need for the combination is not found in the GDCs, as they only require "appropriate" combinations of natural events and accidents. The specific load combinations required are licensing positions developed by the staff.

It is generally agreed that the requirement to combine seismic and LOCA loads for design and qualification of equipment is conservative. If it were removed, large margins would exist in the design of many structures systems and components of existing plants. These margins would be missing or much reduced in future plants not designed for this load combination. The importance to safety of the proposed action is measured by the need for the imposed conservatism. This need is a judgment call, based on the perceived uncertainties in the probability of concurrent occurrence and in the "fragility" of the structures systems and components (i.e., the range between the design basis and the dynamic levels that will cause degradation or failure).

Although SCIENTECH considers that the combination of LOCA and seismic events is conservative, it also concludes that its removal could have a significant impact on safety, not a marginal one. Whether the resulting level of safety would remain acceptable is not within the scope of this study to determine. Although no further consideration need be given to this action, a general discussion of potential benefits will be provided in Step 4.

STEP 4: Impact Analysis

The burden associated with the proposed action would be that of changing the staff position and making appropriate changes to the FSARs. No changes in equipment would be required as a result of the revision. However, the long term benefit could be substantial for existing plants, if they were to take advantage of the change when procuring replacement components. The long term benefits for new plants would be large.

These estimated benefits are for a simple decoupling of the seismic and LOCA events. It is quite possible that decoupling could be accompanied by other changes in load definition and acceptance criteria to accommodate the staff's concerns with design conservatism. SCIENTECH views this as a complex issue, which will only be resolved by a substantial, long term NRC action.

Regulatory Position No.18

10 CFR 50 Appendix A, GDC 19
Control Room
Regulatory Guide 1.78
Regulatory Guide 1.114

DRAFT BCL ANALYSIS

BCL Potential Actions: *Replace/Clarify/Update*

BCL Comments: *GDC 19 requires licensees to provide a control room from which actions can be taken to operate the plant safely and to maintain it in a safe condition under accident conditions. It also requires the room be adequately protected from radiation such that personnel in the control room during an accident receive no more than 5 rem whole body or its equivalent for the duration of the accident. This dose limit often is the determining factor driving design considerations for control room habitability.*

One potential action is to replace the current 5 rem limit with a new limit of 25 rem. The rationale is that emergency workers are permitted to receive up to 25 rem during an accident, and that during an accident control room staff would be considered emergency workers. [From where does the alleged 25 rem dose limit derive?]

Another potential action is to review and update the current regulatory positions associated with habitability of control rooms, particularly in reference to hazardous chemical releases. The current RegGuide 1.78 was issued in 1974 and has never been revised.

A related potential action is to clarify what constitutes the physical boundaries of the control room as that term is used in 10CFR50.54(m). Such action is underway through a revision of RegGuide 1.114 in progress.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

<u>Increase Exposure Limit to 25 rem</u>

This action is within the scope of the project and no NRC action is currently underway or planned. Therefore, further consideration will be given to this potential action.

Review and Update Regulatory Guide 1.78

Regulatory Guide 1.78 identifies chemicals which, if present in sufficient quantities, could result in the control room becoming uninhabitable and general design considerations relative to the capability of the control room to withstand releases occurring either on the site or within the surrounding area. The guidance remains generally valid. Although it is impossible to predict the outcome of a staff effort to update the Guide, it seems more probable than not that the result would be an increase in the scope of review and in the licensee effort involved. Thus, this action is not within the scope of the project and further consideration of this action is inappropriate.

Clarify Control Room Boundaries

As noted by BCL, the staff has this action underway in the form of a revision of Regulatory Guide 1.114. Therefore, further consideration of this action is inappropriate.

Step 2: Background Description

Rulemaking Motivation

The General Design Criteria were developed during the late 1960s to provide applicants with general guidance concerning the standards for acceptable design of safey-related structures, systems and components.

Requirements for Licensees

The control room and its supporting structures, systems and components (e.g., ventilation and biological shielding) must be designed so that worker exposure does not exceed the 5 rem limit for all design basis accidents in which workers follow prescribed procedures.

Intent of the Requirement

The requirements related to control room habitability, including the 5 rem radiation dose limit, are intended to assure that workers will be able to stay in the control room and manage the accident without undue risk of health effects from radiation exposure.

The 1989 EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents provides a general limit of 5 rem for emergency workers, with the condition that doses should be held as low as practicable. For activities involving life saving or preventing high risk to populations, a 25 rem limit applies when lower doses are not practicable. The Guide notes that persons undertaking any emergency operation in which the dose will exceed 25 rem to the whole body should do so only on a voluntary basis and with full awareness of the risks involved.

STEP 3: Importance to Safety

It seems clear that the control room activities during some postulated accidents would come reasonably within the category of "lifesaving or preventing high risk to populations," which is the PAG criterion for a limit of 25 rem. Since there is little difference in risk between a one time exposure of 5 rem and 25 rem, and since radiation workers are well trained and understand these matters, it is likely that the change in exposure limit would have little effect on the willingness of operators to carry out their tasks. Consequently, this action is considered marginal to safety.

STEP 4: Impact Analysis

The proposed actio would impose no burden on the licensees because a relaxation of the requirements would not require any action. For operating plants, the short term benefit would be small because there is little motivation to change the design of existing systems (there might be a benefit when replacements or renovations are made); accident management procedures might be changed in ways that reduce the risk of damage to the plant, but this is highly speculative. For new plants, the long term benefit could be large by allowing less stringent design criteria.

For the NRC, there would be little or no benefit in the short or long term. The short term burden associated with the rule change is likely to be large, due to the probable interest by workers and their unions in the motivation for the change, even if its effects are perceived to be small.

The current ongoing effort directed at redefinition of the accident source term could affect details of design to meet the exposure limits for control room workers. In addition, there could be changes in the way doses are calculated as a result of the recent revisions in 10CFR20. The potential effect of these activities should not significantly affect the issue of increasing the dose limits from 5 rem to 25 rem, but will need to be taken into account during the rulemaking action.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Low

Regulatory Position No.19

10CFR50.13
10CFR50 Appendix A

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand/Eliminate*

BCL Comments: As currently formulated, the General Design Criteria contain no specific provisions for designing against sabotage. Recent reviews of new standard designs rely on NUREG-0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans" August 1982 for regulatory guidance. The suggested action is to synthesize into a new General Design Criterion the accumulated wisdom gathered from past experience.

Interestingly, 10CFR50.13 explicitly advises applicants that they are "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 person, or (b) use or deployment of weapons incident to U. S. defense activities." Adopted in 1967, the language could be interpreted to apply only or primarily to acts of war. A more rigorous interpretation could suggest that sabotage need not be a design consideration in any sense. This part would imply conflict with the potential expansion of Appendix A, such that any inconsistency between 10CFR50.13 and an expanded Appendix A would need to be resolved. Part 50.13 also exemplifies a common instance where the text of the regulations is used to define what is not required, rather than what is required.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action is not within the scope of this project because it would increase the NRC and licensee effort. Further consideration of this action is inappropriate here.

It should be noted that production and utilization facilities are required to provide protection for special nuclear material in accordance with 10 CFR 73, "Physical Protection of Plants and Materials." The requirements of Part 73 include the organization of a security organization, installation of physical barriers, and establishment of contingency and response plans.

Regulatory Position No.20

10 CFR 50 Appendix A
General Operating Criteria

DRAFT BCL ANALYSIS

BCL Potential Actions: *Expand*

BCL Comments: The General Design Criteria were originally adopted in 1971 to "establish minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." Despite explicit expectations of change, the General Design Criteria have remained relatively stable over the years since their adoption, though the voluminous subsidiary implementing regulatory positions have swelled.

The potential action is to develop a comparable set of "General Operating Criteria" to help regulate plant operations analogous to the way the GDC establish basic principles for plant design. The focus of reactor regulation has shifted from design/construction in the sixties and seventies to operations in the eighties and beyond. It is apparent that the regulations have not kept pace with this shift. One example of a "deficiency" is that nowhere do the regulations require an ongoing training program for plant maintenance staff. Nor have requirements for emergency procedures, control room shift staffing, and safety parameter display systems, which were implemented via Generic Letters after TMI, ever been integrated effectively into the regulations.

There had been an attempt to develop a few such GOC soon after TMI, but the effort withered. The Institute of Nuclear Power Operations has developed criteria to guide their evaluations of the nuclear reactor industry, but under the current institutional relationships, those criteria are voluntary.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed expansion of the regulations to establish plant operating criteria would increase NRC and licensee effort and, therefore, is not within the scope of this project. Therefore, further consideration of this action is not appropriate here.

Expansion or addition of a regulatory requirement can reduce the burden on licensees when it provides a generic resolution of a burdensome issue. There is no evidence to suggest that this criterion is met by the expansion proposed for this regulatory position.

Regulatory Position No.21

10CFR50 Appendix B
Quality Assurance Criteria
Regulatory Guide 1.28
QA Program Requirements
(Design and Construction)

DRAFT BCL ANALYSIS

BCL Potential Actions: *Expand*

BCL Comments: Appendix B and RegGuide 1.28 establish quality assurance controls for procuring products and services. Both positions enable licensees to enhance their ability to detect defective workmanship, but neither were specifically developed with fraudulent products in mind. Recent discoveries of such products in operating reactors leads to the recommendation for modifying the regulations to lessen the associated risk.

The action proposed is to "strengthen" Appendix B. NRC has issued Generic Letter 89-02, which addresses this matter in part by conditionally endorsing industry efforts (EPRI NP-5652 "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications"). The letter describes modifications to the EPRI Guidelines which will "satisfy existing requirements of Appendix B" and "reduce the likelihood of the introduction of counterfeit or fraudulent products." The staff has also issued an Advanced Notice of Proposed Rulemaking (SECY-89-010) soliciting public comment on the matter.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The proposed action is not within the scope of this project because it would increase the effort of the licensee and the NRC. In addition, the staff has an ongoing rulemaking action underway on this issue. Comments on the Advanced Notice of Proposed Rulemaking are currently under review by the staff.

Regulatory Position No.22

10 CFR 50 Appendix B
SRP 14.2
SRP 17.1
SRP 17.2
RegGuide 1.28
RegGuide 1.33

DRAFT BCL ANALYSIS

BCL Potential Action: *Update*

BCL Comments: *Appendix B contains the basic requirements for quality assurance. Beyond Appendix B are many different versions and interpretations of the same basic requirements. Contributing to the confusion is the increased tendency to deal with immediate safety concerns through the mechanism of Generic Letters and then failing to follow through in timely fashion with revisions to appropriate regulations, SRP sections, and RegGuides.*

A case in point is the matter of post-trip review. Generic Letter 83-28 identified required actions based on the generic implications of the Salem ATWS event. Two subsequent Generic Letters (85-09 and 85-10) added additional technical specifications. Yet the relevant SRP Section 4.3 Nuclear Design and RegGuide still contain no such requirements. Nor do recent industry standards (e.g., ANS 3.2-1988) appear to be current.

The potential action is to make a concerted effort to review and update the quality assurance aspects of all current regulatory positions. This would benefit current licensees and future reactors.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Technological advances inevitably render technical guidance, particularly detailed technical guidance, obsolete. This does not appear to have happened yet to Appendix B, which offers guidance sufficiently general to be almost universally applicable. The question of updating Appendix B resolves to whether it should be extended to provide guidance for the "managerial and administrative controls used to assure safe operation" that are mentioned in the Introduction to Appendix B. Such extension would increase the effort of the NRC and the licensee.

With regard to the lower level documents, the SRP, Regulatory Guides and industry standards, it is true, as noted in the BCL report, that they represent a mosaic of requirements and guidance that are of varying age and relevance. Quality assurance standards and practices continue to evolve. The documentation of new standards and practices is being rewritten, replaced and withdrawn at a pace that reflects the need for care and consensus among the many organizations and groups involved. It is SCIENTECH's opinion that an attempt at a comprehensive coordinated revision of the relevant requirements and standards would not result in a significantly increased rate of change nor in improved long term stability of the requirements and standards.

Licensees draw from the existing mosaic of standards and requirements to develop quality assurance programs that a) reflect the individual licensee's managerial philosophy and style, and b) are acceptable to the NRC staff. Although it is clear that the replacement of this mosaic with a coherent, state-of-the-art body of requirements and guidance would make the development and review of quality assurance programs simpler, it is by no means clear that the implementation of the new requirements would mean a net decrease in effort for the NRC or the licensees. In general, quality assurance evolution has been attended by a trend toward greater licensee effort and attention being required for quality assurance programs.

For the reasons discussed above, this potential action is not considered to be within the scope of the project and will receive no further consideration.

Regulatory Position No.23

10 CFR 50 Appendix J
Primary Reactor Containment Leakage
Testing Water-Cooled Power Reactors
SRP 6.2.6
Containment Leakage Testing

DRAFT BCL ANALYSIS

BCL Potential Actions: *Convert/Update*

BCL Comments: *Appendix J was originally adopted in 1973 to support 10CFR50.54(o) and General Design Criteria 50 through 57. Licensees must demonstrate that their plants satisfy the containment leakage testing requirements of Appendix J as a condition of the operating license. Three types of tests are defined:*

Type A - an integrated leak test at not less than design basi pressure
Type B - tests for local leaks around containment penetrations
Type C - tests for local leaks in containment isolation systems.

The Appendix codifies the relevant ANSI Standard N45.4-1972, specifies other acceptable test methods, sets requirements for scheduling tests, specifies test parameters, establishes acceptance criteria, and defines procedures for validating and reporting test results.

The principal thrust of the potential actions is to bring practices for containment leak testing more in line with recent insights from probabilistic risk assessments. A secondary consideration is that the significant amount of detail in this Appendix makes it better suited as a RegGuide than a regulation. There is also continuing concern within the industry about the significant costs of containment leak testing.

Appendix J has been recognized as a candidate for major revision since results of early risk assessments showed risk to be dominated by accidents involving core damage and major breaches of containment (as opposed to the relatively intact containment integrity typified by Appendix J). A prior detailed review of this subject (NUREG/CR-4330) done according to NRC value-impact guidelines concluded that a 100-fold increase in allowable leak rates could be permitted without significant adverse effects on public safety. No rule change resulted from this finding. The NRC did modify Appendix J in 1988 to permit use of the exemptions. Still the major issues of test type, test frequency, and allowable leak rates remain unchanged.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

Revise Containment Leakage Test Requirements
Convert Appendix J to a Regulatory Guide

The potential actions are within the scope of this project because they have the potential for reducing NRC and licensee effort. A rulemaking action to amend Appendix J has been underway for several years and the staff plans to submit a final rule to the Commission in early 1991. The primary focus of the amendment is to provide for, and allow, improved technology in the implementation of the tests. In addition, the periodic retest schedule would be redefined to permit tests during alternate refueling shutdowns at intervals not to exceed two years. These changes do not address the full scope of the potential actions identified by BCL, therefore, further consideration of the actions is appropriate.

STEP 2: Background Description

Rulemaking Motivation

10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors", effective March 16, 1973, was issued to provide uniform requirements for containment leakage testing. Prior to the issuance of Appendix J, containment leakage testing requirements were specified on an individual basis in the technical specifications for each power reactor.

Requirements for Licensees

Appendix J requires three different types of containment leakage tests.

1. Type A measurement of the containment integrated leakage rate is required three times during each ten year period during the operating life of the plant.

2. Type B measurement of the leakage across each pressure-containing or leakage-limiting boundary for various primary reactor containment penetrations is required at intervals not to exceed two years, except that air locks are tested every six months.

3. Type C measurement of the containment isolation valve leakage rates is required at intervals not to exceed two years.

The American National Standards Institute standard ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors" was incorporated by reference into Appendix J, with modifications and exceptions. Appendix J provides test frequencies, pretest requirements, test methods, and acceptance criteria for each of the tests described above. Appendix J also describes the situations in which special test requirements are required and the reporting requirements for the test results. In addition, an Information Notice was issued in 1985 (IEN 85-71) that provided additional guidance on the implementation of Appendix J.

Intent of the Requirement

The periodic tests required by Appendix J are intended to assure that the containment will continue to perform its function throughout the life of the plant. In addition, the testing assures that leakage through penetration systems and components does not exceed allowable leakage rates and that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

STEP 3: Determination of Importance to Safety

Revise Containment Leakage Test Requirements

The containment leakage tests are confirmatory and do not directly reduce risk, provided the tests disclose no structural change to the containment or no open valves or other penetrations. Such results seldom occur. Rather, the tests serve to identify needed actions to restore and maintain design basis leakage rates. They establish and maintain confidence that the assessed level of risk is and remains adequately low.

During consideration of the Appendix J amendment now in the final stages of rulemaking, the staff studied the possibility of further changes to the nature and frequency of the Type A tests and concluded that there was not an adequate basis for such changes. SCIENTECH believes that a study of the full spectrum of objectives and benefits of containment leak rate testing would identify possible changes that would be marginal to safety. This view is supported further by studies that have shown increasing containment design leakage by several orders of magnitude has a relatively minor effect on overall plant risk. NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements," summarizes a number of such studies.

The insensitivity of risk to changes in the containment leakage rate, suggests that increasing the design leakage rate might be a feasible approach to easing the leakage test requirements. However, the containment design leakage rate is specified in the technical specifications or other design bases and is not determined by Appendix J. One obstacle to increasing the design leakage rate is that it could result in the calculated doses at the boundaries of the exclusion area and low population zone exceeding the dose limits set by the Commission in 10CFR100. However, the staff is currently working on changes to 10CFR100 and 10CFR50 that could eliminate the calculation of doses for the exclusion area and low population zone. (See Regulatory Position No. 26). SCIENTECH considers that changes in the design leakage rate would be marginal to safety and should be considered in connection with the current effort to revise 10CFR50 and 10CFR100, but that they are outside the scope of this project.

The impact of revising containment leakage test requirements without a change in the design basis leakage rate is analyzed below.

Convert Appendix J to a Regulatory Guide

The determination of whether conversion would be of marginal importance to safety is conditional. The conversion would provide both the staff and the licensee or applicant with an opportunity to deviate from the detailed requirements specified in the Regulatory Guide (or SRP) with much less formality and effort than is required for exemptions from the rule. It also allows the licensee to determine, pursuant to 10CFR50 59, whether a change, interpreted by the licensee to be within the limits of the Regulatory Guide requirements, requires prior NRC approval. Conversion would have marginal importance to safety provided that deviations are reviewed and approved on the basis of the same technical considerations that would apply to exemptions from Appendix J, and that licensees implement 10CFR50.59 in a manner that correctly reflects the safety significance of any changes made in the required tests.

STEP 4: Impact Analysis

Revise Containment Leakage Test Requirements

NUREG/CR-4330 estimates the cost of a Type A test at $1.2 to $2.6 million. Three Type A tests are required during each ten year operating period. Most of the cost is associated with the cost of replacement power. It should be noted that the test requirements in Appendix J, although prescriptive, are not highly detailed. Exemption requests are generally motivated by the desire to minimize the down time required for the tests.

It is clear that any change in the character or frequency of the tests that reduced average down time could result in large benefits, for example, a change to a five year interval would save one third of the costs. Other possibilities are an increase in the containment design leakage rate, or replacement of some or all of the Type A tests with other tests or inspections to detect degradation of containment capability. However, without some specific characterization of the changes that might be technically appropriate, it is not possible to determine whether and to what degree benefits might be accrued.

NUREG/CR-4330 estimates the combined cost of Type B and C tests to be about $16,000. SCIENTECH believes this estimate is low by as much as an order of magnitude. These tests are required at intervals not to exceed two years, except that air lock tests are required every six months. A wide variety of changes in these tests might be possible. (The ongoing rulemaking action introduces some changes related to advances in technology.) As with the Type

A tests, SCIENTECH cannot determine whether and to what degree benefits might be accrued, without a more specific characterization of the changes that might be appropriate.

The short term burden involved in the rulemaking action to change or convert Appendix J is uncertain. For changes in the character of the tests, the technical effort could be large; for changes in the frequency, the technical effort could be relatively small; the overall rulemaking effort should be typical.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Low

Convert Appendix J to a Regulatory Guide

The conversion of Appendix J to a Regulatory Guide would provide both the staff and the licensee or applicant with an opportunity to deviate from the detailed requirements with much less formality and effort than is required for exemptions to 10CFR. It also allows the licensee to determine, pursuant to 10CFR50.59, whether a change requires prior NRC approval. For any deviation from the specific requirements, including changes subject to 10CFR50.59, there would be a decrease in staff and licensee effort to arrive at a decision. In addition, it can be expected that the deviations proposed by the licensees will represent a significant savings in the cost of testing. The magnitude of the long term benefit would depend upon the number and character of the proposed and approved deviations. In view of the relatively large number of exemption requests that have been submitted with respect to Appendix J, it seems probable that the number of future deviation proposals would also be substantial.

It is also possible that the conversion could have adverse effects, such as: a) a number of licensee proposals for deviations that overwhelms the staff's capability to perform timely, adequate safety analyses, b) an increase in enforcement problems related to unacceptable licensee interpretations of requirements, and c) licensee proposals and/or staff actions that stimulate public concern and intervention in the form of petitions (for operating plants) or hearing issues (for new applications). Each of these possibilities would result in increased effort, reducing the net savings associated with the presumably lower cost of the deviations. The probability of significant adverse effects is small but not negligible for this conversion.

For conversion, the technical and overall rulemaking effort should be minimal, unless members of the public see the action as an attempt to weaken the requirements. This potential public concern may be preempted, or at least dealt with efficiently, by a rulemaking action, possibly preceded by a policy statement, to convert several detailed rules at the same time. This possibility is discussed in Section II of this report.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Low

Regulatory Position No.24

10CFR50 Appendix R
Fire Protection Program
For Nuclear Power Facilities
Operating Prior to January 1, 1979

DRAFT BCL ANALYSIS

BCL Potential Actions: *Convert/Update*

BCL Comments: General Design Criterion 3 Fire Protection requires licensees to design and locate structures, systems and components important to safety to minimize effects of fires and explosions and to provide appropriate systems for detecting and fighting fires. In 10CFR50.48 the NRC requires licensees to have a fire protection plan which expands on the physical considerations of GDC 3 and adds administrative and personnel considerations. That part proceeds to reference Appendix R, set schedules for compliance, and establish procedures for review and approval.

Adopted originally in November 1980 as a response to the Brown's Ferry fire, Appendix R "establishes fire protection features required to satisfy" GDC 3. The Appendix consists of nearly eight pages of "general requirements" and "specific requirements" related to hazards analysis, equipment, barriers, safe shutdown capability, water supplies, detection, fire brigade, training, records, and other related matters. The NRC staff has issued four related Generic Letters:

81-12	*Safe shutdown capability after fires*
83-33	*NRC positions on Appendix R*
86-10	*Implementation of fire protection requirements*
88-12	*Move fire protection program from technical specifications to FSAR.*

There was an apparent lack of referencable industry standards when Appendix R was being developed. That situation appears to persist today.

The highly detailed nature of these requirements leads some to suggest that Appendix R would be more appropriate as a RegGuide than a regulation. Most existing reactors have complied or received exemptions. Much of Appendix R has been incorporated into SRP 9.5.1, so duplication exists. Retaining Appendix R in 10CFR50 assures the regulatory staff of a continuing source of enforcement authority.

New plants are expected to comply with Appendix R without exception, though the regulation does not mention this explicitly. There is some concern that for new plants Appendix R is necessary but insufficient, thus meriting some additional modification (e.g. increased separation distances). Resolution of this matter is achieved during the review process for new plants.

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The potential action regarding the conversion of Appendix R to a Regulatory Guide is within the scope of this project. Additionally, there is no present or planned NRC rulemaking in this area. Therefore, further consideration of the action is appropriate.

Updating Appendix R to modify requirements and include new requirements would increase licensee and NRC effort. This potential action is not within the scope of this project and will receive no further consideration here.

STEP 2: Background Description

Rulemaking Motivation

In 1975, a fire occurred at the Browns Ferry nuclear power plant. The fire caused extensive damage to electrical control cables and components. As a result, many of the systems normally relied upon for safe shutdown and cooldown of the reactor were not available. At the time of this event, Criterion 3 of 10CFR50, Appendix A (General Design Criteria) was the governing requirement for fire protection. Fire protection safety evaluations based on Criterion 3 were the basis for NRC acceptance of fire protection programs implemented by the licensees. However, these evaluations were not detailed and did not focus on safe shutdown capabilities following a fire. They were, like other industrial fire protection measures, designed to protect the plant.

After the Browns Ferry fire, a special review group was commissioned to evaluate the fire and its consequences. The group recommended that NRC should develop additional specific guidance for implementation of Criterion 3 and should make a detailed review of the fire protection program at each operating plant comparing it to that guidance.

In response to the first recommendation, NRR developed Branch Technical Position Auxiliary Power Conversion Systems Branch 9.5-1 (BTP 9.5-1), "Guidelines for Fire Protection for Nuclear Power Plants" and Appendix A to BTP 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976."

In response to the second recommendation, the NRC requested every operating plant to (1) compare its fire protection program with the above guidelines and (2) analyze the consequences of fire in each plant area. The NRC then reviewed the licensee's analysis against the guidance contained in Appendix A to BTP 9.5-1 and visited each plant to examine the relationship of the structures, systems and components important to safety with both in situ and transient fire hazards, the potential consequences of fire, and the associated fire protection features. (45 FR 36082)

As a result of this plant-specific effort, most licensees accepted the NRC interpretations and positions of Appendix A to Branch Technical Position (BTP) 9.5-1. However, by the late 1970s there were still 17 generic issues in the fire protection safety analysis reports for 32 plants where agreement had not been reached between the licensees and the NRC (45 FR 36083). To establish a definitive resolution of these contested subjects in a manner consistent with the general guidelines of Appendix A, and to ensure timely compliance by licensees, the NRC found it necessary to issue a proposed fire protection rule, 10 CFR 50.48, and Appendix R to 10 CFR Part 50.

Section 50.48 required the creation of fire protection plans, and Appendix R provided the more specific minimum fire protection requirements for each issue. Section 50.48 and Appendix R were issued in final form in November 1980, and have not been substantively amended since then.

Requirements for Licensees

The NRC's stated purpose of Appendix R is to provide generic requirements that must be incorporated into fire protection plans for those nuclear power plants licensed to operate prior to January 1, 1979. Appendix R consists of both general and specific requirements.

The general requirements in this section of Appendix R state the need for a comprehensive fire protection program at each nuclear power plant. In general terms, the requirements call for:

- establishment of a fire protection program;
- performance of a fire hazards analysis;
- establishment of fire prevention features for those areas containing or presenting a fire hazard to structures, systems, or components important to safety; and
- alternative or dedicated safe shutdown capability in areas where fire protection features cannot ensure safe shutdown capability.

In addition to the general and specific requirements of Appendix R, various documents related to the implementation of Appendix R have also been issued. The following is a synopsis of those issuances.

GL 81-12: Clarified information required by the NRC to complete reviews of alternative safe shutdown capabilities.

GL 83-33: Provided additional information related to NRC interpretations of Appendix R.

IEN 83-41: Discussed safety-related equipment rendered inoperable by actuation of the fire suppression system.

IEN 83-69: Discussed improperly installed fire dampers.

IEN 84-09: Provided lessons learned from NRC inspections to evaluate compliance with Appendix R.

GL 85-01: Provided a report by the Fire Protection Policy Steering Committee. The Steering Committee had been formed to make recommendations to expedite compliance with Appendix R at older plants and to assure consistent levels of fire protection safety at all plants. This letter also provided staff positions on commonly asked questions related to Appendix R.

GL 86-10: Provided a copy of Interpretations of Appendix R (a handout provided to participants in the regional fire protection workshops sponsored by the NRC). This letter also notified licensees that Paragraph 50.48(c)(6), which contained scheduled exemptions for Appendix R, was no longer valid and that fire protection programs approved by the NRC were to be incorporated into the Final Safety Analysis Report.

GL 88-10: Provided guidance to licensees for preparing a license amendment to remove fire protection requirements from Technical Specifications.

For those plants operating prior to January 1, 1979, the guidance documents listed serve as the basis for licensing reviews for fire protection and subsequent safety evaluation reports. For those plants not operating prior to January 1, 1979, Standard Review Plan (SRP) 9.5-1 (formerly BTP 9.5-1) applies to plants whose applications for construction permits were docketed after July 1, 1976, and Appendix A to BTP 9.5-1 applies to plants whose applications for construction permits were docketed prior to July 1, 1976. With few exceptions, SRP 9.5-1 and Appendix A to BTP 9.5-1 contain the same information found in Appendix R.

Appendix R has been implemented at all currently operating nuclear power plants. This implementation may have taken the form of backfits to operating plants, a determination that applicant plants meet the requirements of BTP 9.5-1, or exemptions to the specific requirements based on alternative approaches which achieve the requisite level of safety.

Intent of the Requirement

Appendix R was issued to resolve disputes and to ensure a consistent resolution of specific fire protection issues at all nuclear power plants.

STEP 3: Determination of Importance to Safety

Because Appendix R applies to nuclear power plants licensed to operate prior to January 1, 1979, future license applications for light-water reactors will be reviewed against the SRP. This has no technical significance because SRP 9.5-1 contains essentially the same requirements as Appendix R. The proposed conversion action is equivalent to eliminating Appendix R and making all plants conform to the guidance in SRP 9.5-1, except plants whose applications for construction permits were docketed prior to July 1, 1976. The major difference will be that changes in the fire protection systems of plants subject to Appendix R have to apply for exemption from the rule, while plants not subject to Appendix R may apply for staff approval without the formality and effort involved in the exemption procedure. This action has marginal significance for safety.

STEP 4: Impact Analysis

NRC and licensee effort would be reduced by the conversion/elimination of Appendix R. Less administrative effort is necessary to document compliance with or exceptions from specifications in the SRP than is necessary for regulations.

Thousands of exemptions have been processed for meeting the intent of 10CFR50.48 by using alternatives to the methods specified in Appendix R . It is difficult to predict whether licensees will make major system improvements or modifications in the future that will require many more exemptions.

The effort involved in the rulemaking and subsequent implementation would not be large because the actual requirements for fire protection would remain unchanged. Public intervention is unlikely because the replacement requirements in the SRP are currently applied to existing plants. If the licensees greet the conversion/elimination of Appendix R as an opportunity to submit a larger number of exceptions and modifications, the benefits of the decreased administrative effort described above could be significantly diminished; however, the likelihood of this occurring appears low.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
High	Low	High	High

Regulatory Position No.25

10CFR50 Appendix R
Fire Protection
Paragraphs III.L.3-7
Alternative and dedicated shutdown capability

DRAFT BCL ANALYSIS

BCL Potential Actions: *Clarify*

BCL Comments: *These paragraphs relate primarily to the need for on-site vs. off-site power to electrify equipment for cold shutdown. (More to follow.)*

SCIENTECH ANALYSIS

STEP 1: Procedural Screen

The BCL report does not provide sufficient information to allow analysis of the proposed action.

Regulatory Position No.26

10CFR100 Reactor Site Criteria
10CFR100 Appendix A
Seismic and Geologic Siting Criteria

DRAFT BCL ANALYSIS

BCL Potential Action: *Update/Convert/Eliminate*

BCL Comments: Part 100 constitutes the basic requirement regarding factors important in determining site suitability for power reactors. Appendix A contains more detailed requirements on the seismic and geologic aspects of siting. These regulations were adopted "as an interim guide" in 1962 and have remained essentially unchanged since the 1970's. Substanti updates are proposed in two areas; radiological source terms and geolo science.

The thrust of the potential action is to use more effectively th obabilistic insights gained in the last decade in time to benefit the licensi reviews of new reactors. Possible modifications related to the radiological source term would affect Part 100.11, which deals with determining exclusion area, low population zone, and population center distance. One suggestion is to eliminate the footnote referencing Technical Information Document 14844 (March 23, 1962) for "further guidance". Doing so would make more apparent that alternatives to strict compliance with TID 14844 may be acceptable. The effects such elimination would have on subsidiary regulatory positions (e.g. Regulatory Guides 1.3 and 1.4) would need to be evaluated carefully.

The second area for modification is Appendix A. Technical updates would address the definitions of the Safe Shutdown Earthquake and the Operating Basis Earthquake as well as the relationship between the two. The SSE is defined by the response spectra corresponding to the maximum vibratory ground motion associated with faults in the vicinity. Certain safety systems must remain functional during SSE. The OBE is that earthquake producing "vibratory growth (sic) motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional." Traditionally the SSE is twice the amplitude of the OBE. While this traditional approach generates a conservative stance regarding safety, some believe that the associated OBE is unnecessarily conservative and engenders significant cost burdens in terms of structures and equipment. One suggestion is to allow the applicant to determine the OBE, thus increasing flexibility, while the NRC staff retains authority to determine the SSE. An alternative approach is to convert Appendix A into a RegGuide, thus theoretically introducing more flexibility into the process.

A significant overhaul of Part 100 has major implications in terms of staff effort, industry effort, and reactor licensing. There are many subsidiary regulatory positions potentially affected by changes to Part 100. Some believe the long term benefits are important enough to merit priority over revisions to Part 50.

SCIENTECH ANALYSIS

STEP 1: Procedural Screening

<u>Eliminate Reference to TID 14844</u>
<u>Decoupling OBE and SSE</u>
<u>Convert 10CFR100, Appendix A to a Regulatory Guide</u>

The potential actions regarding 10 CFR Part 100 and its Appendix A are within the scope of this project. However, the staff has rulemaking actions underway that address the substance of all of the potential actions, therefore further consideration of the action is inappropriate.

In October 1990, the staff submitted proposals (SECY 90-341) to the Commission related to the establishment of site suitability criteria and the definition of exclusion areas and low population zones that would be independent of the safety design of nuclear power plants. A separate action is underway to remove the engineering considerations from 10CFR100, Appendix A and focus the seismic requirements of Appendix A on the characterization of the Safe Shutdown Earthquake (SSE). Also under consideration are: decoupling the OBE from the SSE, revising the definition of the seismic input for design analysis, and changing the method for determining the postulated location of the design basis earthquake.

Regulatory Position No. 27

SRP 6.7
SRP 15.6.5D
RegGuide 1.96

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate*

BCL Comments: General Design Criterion 54 "Piping Systems Penetrating Containment" requires, in part, that piping systems penetrating containment be provided with leak detection, isolation and containment capabilities having redundancy, reliability and performance capabilities that reflect the importance to safety of isolating these piping systems. Operating experience in the early 1970's showed degradation of BWR MSIV's. This led to supplemental design features to control and contain the leakage of radioactive material from MSIV's as described in RegGuide 1 6 and SRP 6.7. SRP 15.6.5D describes acceptable means for calculating the release of fission products and their contribution to off-site doses following a large break LOCA.

The principal thrust of the potential action is to bring practices and designs related to MSIVLCS's more in line with recent insights from probabilistic risk assessments. There is a range of potential elimination as well. One could opt to eliminate only the Chapter 15 calculations on radiological consequences. More aggressively, one could opt to eliminate aspects of the design, inspection or testing of the MSIVLCS. One possibility is to downgrade MSIVLCS to a nonsafety system. In the extreme, one could opt to eliminate or shut down totally the MSIVLCS.

This subject has been recognized as a candidate for elimination for several years. A prior detailed review of the matter (NUREG/CR-4330) using NRC valueimpact guidelines concluded that, if treated as a new requirement for operating reactors, the MSIVLCS would not be justified as a backfit. Going further, the elimination of MSIVLCS would generate substantial savings without significantly increasing risk. Despite this evidence the NRC staff has held its position on MSIVLCS, in part because there are dose calculations for at least one plant (Nine Mile Point) for which accidents involving MSIV leakage was a limiting case.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The elimination of the requirement for a leak control system (LCS) for main steam isolation valves (MSIVs) is within the scope of this project. Currently, the staff is reviewing an effort by the Boiling Water Reactor Owners Group to develop a generic justification for eliminating both the MSIV testing requirements and the MSIV LCS. It appears at the present time that the BWROG will be successful in justifying such changes. Further consideration of the potential action within this project is appropriate.

Step 2: Background Description

The background of the requirement is adequately described in the BCL report on this regulatory position, as reproduced above.

Step 3: Determination of Importance to Safety

The comparative risk assessments reported in NUREG/CR-4330 and NUREG-0933 indicate that the Main Steam Isolation Valve (MSIV) Leak Control Systems (LCS) do not significantly reduce overall plant risk. NUREG/CR-4330 estimates for a typical plant (Grand Gulf) the total risk for all sequences associated with MSIV leakage was 2.14 person-rem/year without LCS and 1.86 person-rem/year with LCS. The dominant sequence contributed 1.6 person-rem/year, with or without LCS. The reduction in risk due to LCS was 0.3 person-rem/year. The reduction of risk for 25 plants over a 30 year life is 225 person-rem which compares with the estimate of 1000 person rem developed in the analysis of Generic Safety Issue C-8 in NUREG-0933.

NUREG-0933 notes that the uncertainties associated with the calculations are judged to be large, although they were not quantified. NUREG/CR-4330 does not discuss uncertainties, nor does it provide a total plant risk against which the 0.3 person-rem/year reduction could be assessed. NUREG 1150 provides a range of risk for Grand Gulf of 2 to 89 person-rem/year for dose to the population within 50 miles. If the 0.3 person-rem/year risk reduction is compared to the median of the range, it represents a reduction of 0.7%.

NUREG/CR-4330 estimates that 10 man-weeks per year are required for maintenance and surveillance of the LCS, and that one man-week would be required to disable the LCS. Although there are no data readily available on the dose to workers associated with maintenance of the LCS, it is likely that the worker exposure exceeds 0.3 person-rem/year. These data indicate that there would be no net negative impact on safety resulting from the elimination of the requirement to install MSIV LCS.

Step 4: Impact Analysis

Cost estimates in NUREG-0933 and NUREG/CR-4330 are similar and are used for this impact analysis. The estimated costs for LCS installation is $500,000 and for maintenance is $20,000 per reactor-year, which results in a cost ratio of over $100,000 per person-rem based on the risk calculations given above.

The NRC studies and analyses necessary to support the elimination of the requirements for LCS are estimated to cost about $500,000. Additional NRC costs of about $11,000 per plant would be associated with the need to make changes to the Technical Specifications at some or all plants. The applicant's cost to disable the existing LCS and make any necessary changes to the Technical Specifications is estimated in NUREG/CR-4330 to be about $12,000 per plant. (SCIENTECH believes this estimate to be low by as much as an order of magnitude.)

Assuming 50 BWRs with remaining plant life of 30 years (25 plants with and 25 plants without an existing LCS), the benefit of not requiring installation of systems in 25 plants is:

(25 X $500,000) + (25 X 30 X $20,000) = $27.5 Million

The benefit of permitting 25 existing LCS to be disabled is:

(25 X 30 X $20,000) - (25 X $12,000) = $14.7 Million

The NRC cost to support the change in requirements is:

$500,000 + (25 X $11,000) = $275,000

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Medium

Regulatory Position No. 28

SRP 13.2.1
Reactor Operator Training

DRAFT BCL ANALYSIS

BCL Potential Action: *Update/Replace*

BCL Comments: SRP 13.2.1 was originally issued in July 1981 as one mechanism for codifying the TMI Action plan requirements related to reactor operators. It specifies what the Preliminary and Final Safety Analysis Reports must contain regarding training programs for reactor operators and senior reactor operators in order for an applicant to receive a license. The Section references the relevant parts of the regulations (10CFR Part 50.54 (i) (m); Part 50 Appendix A; Part 50.55); RegGuides (1.8, 1.149); and other regulatory positions (NUREG-0094, NUREG-0718, and NUREG-0737).

The potential action is to update SRP 13.2.1 to reflect more current regulatory guidance. One particular action mentioned frequently is to replace NUREG-0094 "NRC Operator Licensing Guide" with NUREG-1021 "Operator Licensing Examiner Standards" October 1983. Referenced RegGuides have been revised (e.g., RegGuide 1.8 Qualifications and Training of Personnel for Nuclear Power Plants endorses ANSI/ANS 3.1-1981) but remain to be incorporated into the SRP. An updated SRP 13.2.1 should also reflect the Commission's 1985 and 1988 policy statements on training as well as the current IOCFR55. For example, in at least four places, SRP 13.2.1 references IOCFR55.22, a paragraph which does not exist in Part 55.

Another point of view suggests that any training program accredited by the Institute of Nuclear Power Operations could serve as an adequate replacement for the specific positions defined in SRP 13.2.1.

[For further information see K. Perkins and the following Generic Letters:

88-13	*Operator licensing exams*
88-09	*Pilot testing of fundamentals exam*
87-14	*Operator licensing exams*
87-07	*Transmittal of final rulemaking for revisions to operator licensing - 1OCFR55 and conforming amendments]*

SCIENTECH ANALYSIS

Step 1: Procedural Screen

For operating plants, the documents referenced in the FSAR or license conditions are the ones to which the licensee is committed; updating the SRP would have no effect on these commitments. For new applications, the staff would be expected to implement the latest revision of any regulation or regulatory guide referenced in the SRP. Thus, updating the SRP with respect to these documents would represent a housekeeping action with little effect on practice and no significant reduction of NRC or licensee effort. On these grounds, it would be reasonable to consider this action not within the scope of this project.

In addition, the staff is embarking on a rulemaking action that could have a direct and significant effect on the nature of training requirements. On November 26, 1990, the Supreme Court upheld a federal appeals court decision requiring the NRC to issue regulations and establish instructional requirements for the training and qualifications of nuclear plant operators, supervisors and technicians. Although the rule is expected to be a broad outline of training requirements, it could affect some of the detailed standards and requirements in the SRP and other subordinate documents.

The potential action will receive no further consideration in this project

Regulatory Position No. 29

SRP 13.6 Physical Security
All Division 5 RegGuides related to security, e.g., RegGuide 5.12 Use of locks for protection and control RegGuide 5.44 Perimeter intrusion alarm systems

DRAFT BCL ANALYSIS

BCL Potential Action: *Update/Expand*

BCL Comments: All existing RegGuides on security have been rendered obsolete by newer security technology. They do not apply to such advances as microprocessor controlled security systems and thus would be of little value in licensing new replacement equipment for operating reactors or for new plants.

NUREG-0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans," August 1982 presents a more modern perspective and is current practice for regulatory reviews, but is still not current with modern security technology.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

When existing regulatory guides become outdated, the staff uses later information and technology to review the licensee proposals. If there is no disagreement between the staff and the licensees on the appropriate measures, there is little motivation for the development of new guidance, as explained in Section II of this report. Because security technology has been advancing rapidly in comparison with the time required for development of formal regulatory guidance, it is unlikely that an action to update the existing Regulatory Guides would result in a net reduction of NRC or licensee effort. Consequently, this potential action is not within the scope of this project and will receive no further consideration.

Regulatory Position No. 30

SRP 16.0 Technical Specifications

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand/Clarify*

BCL Comments: Current requirements do not clearly address exceedance of the Operating Basis Earthquake. Matters such as what defines the exceedance of the OBE, the associated reporting requirements, and the criteria for restart are not codified except by precedence. Industry working groups are preparing draft positions on the matter. NRR is also considering a new focused rule addressing instrumentation for seismically-induced shutdown.

[For further information see Memorandum Treby to Bagchi, "Interpretation of Part 100, Appendix A regarding: Proposed guidelines for determining when Operating Basis Earthquake is exceeded," May 3, 1988.]

SCIENTECH ANALYSIS

Step 1: Procedural Screen

This potential action is within the scope of this project. The staff is currently working (RM i47) on changes to 10CFR50 and 10CFR100 that would address these issues. As currently envisioned, 10CFR100 would be revised to focus on site investigations, for example, to determine and characterize the Safe Shutdown Earthquake (SSE). Associated changes would be made to 10CFR100 and 10CFR50 with regard to the engineering applications of the seismic input. Under consideration are: 1) decoupling OBE and SSE; 2) how the design basis earthquake should be defined (e.g., response spectra or energy density functions), and 3) specification of the control point (i.e., the location of the EQ). It is difficult to estimate how close this issue is to resolution; however, it is not likely that an independent effort would result in quicker resolution. No further consideration will be given here to this potential action.

Regulatory Position No. 31

RegGuide 1.3 BWR source terms
RegGuide 1.4 PWR source terms

DRAFT BCL ANALYSIS

BCL Potential Action: *Update*

BCL Comments *RegGuides 1.3 and 1.4 provide acceptable assumptions for use in calculating potential radiological consequences from postulated loss of coolant accidents in BWRs and PWRs respectively. As two of the oldest RegGuides, their basic assumptions about radiological source terms have influenced many subsequent regulatory positions. The basic perceived problem with these positions is that they overstate the release of radioactive material during an accident and as a result contribute to unnecessarily conservative design.*

One example of this is Paragraph C.I.f of RegGuide 1.3 which states in its entirety "No credit shall be given for retention of iodine in the suppression pool." Based on research results, the NRC staff has already modified Sections [which ones?] of the Standard Review Plan to permit credit. The potential action would extend this modification to the RegGuide. Other examples of behavior in which research has changed perceptions of conservatism involve the fate of iodine isotopes, the nature of nonvolatile radioactive species, and the distribution of energy sources.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The staff is developing proposed regulatory changes that would decouple siting criteria from safety design requirements, e.g., to define the low population zone and exclusion area in terms not related to the calculated radiological consequences of a postulated accident. In addition, there is work underway to redefine the accident source term. The final resolution of these issues will determine whether Regulatory Guides 1.3 and 1.4 are needed and, if so, what changes would be appropriate. The proposed action would be premature at this time and would therefore be unlikely to result in any net reduction of NRC and licensee effort. Meanwhile, as noted in the BCL report, the staff's review of licensee proposals reflects a conservative application of current knowledge and understanding.

Further consideration of this potential action is not appropriate here.

Regulatory Position No. 32

RegGuide 1.60
Seismic Design Response Spectra

DRAFT BCL ANALYSIS

BCL Potential Action: Update

BCL Comments: Update to reflect newer data on Eastern earthquakes. (More to follow)

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The BCL draft report provides only a sketchy definition of the potential action. SCIENTECH believes that a revision of Regulatory Guide 1.60 to incorporate recent data on eastern earthquakes would likely mean different response spectra for eastern and western sites. It further seems probable, but highly uncertain, that the new spectra would represent a decrease in design conservatism for eastern sites. However, any such decrease in conservatism could be negated by changes in other requirements.

The uncertainty associated with seismic design inputs for eastern sites has been the source of serious concern for many individuals inside and outside the NRC. The cumulative conservatism embodied in the various regulatory requirements has resulted in reasonable assurance of safety. Any change that decreases seismic design conservatism in one area is likely to generate pressure to review other areas to ensure that eastern seismic uncertainty is adequately compensated.

Taken alone, these observations would suggest the potential action be fully evaluated within this project. However, the rulemaking action described under Regulatory Position No. 30 has direct relevance for this potential action, as well as that under Regulatory Position No. 33. That rulemaking will address revisions of 10CFR100 and 10CFR50 with regard to: 1)investigating the seismic characteristics of the site; 2) decoupling OBE and SSE; 3) defining seismic input for design basis (e.g., response spectra or energy density functions), and 4) specification of the control point (i.e. the location of the earthquake). Thus, response spectra may not be used to define seismic inputs in the future. These considerations suggest the proposed action would be premature at this time, and make it difficult, if not impossible, to evaluate its importance to safety and potential benefits. Consequently, futher consideration of this action in not appropriate.

Regulatory Position No. 33

RegGuide 1.61 Seismic Design Damping Values
RegGuide 1.92 Seismic Response Analysis Modal Responses

DRAFT BCL ANALYSIS

BCL Potential Action: *Update/Replace*

BCL Comments: *Endorse ASME Code Case N411 and resolution of Unresolved Safety Issue A-40. This was done in RegGuide 1.84 specifically for piping. (more to follow)*

SCIENTECH ANALYSIS

Step 1: Procedural Screen

This action is within the scope of this project as it would result in a reduction of design stresses for structures and components. However, the rulemaking action described under Regulatory Position No. 30 has direct relevance for this action, as well for Regulatory Position No. 32. That rulemaking will consider various approaches to defining the design basis seismic input (e.g., response spectra or energy density functions) and could significantly change the nature of the Regulatory Guides required. These considerations suggest the proposed action would be premature at this time, and that it would be difficult, if not impossible, to evaluate its importance to safety and potential benefits. Consequently, futher consideration of this action in not appropriate.

(It should be noted that the staff has been applying Code Case N411 and accepting higher damping values on a case-by-case basis for several years. The staff's approach is consistent with the recommendations of the NRC's Piping Review Committee, which were developed in the early 1980s and reported in NUREG 1061.)

Regulatory Position No. 34

RegGuide 1.76 Design Basis Tornado

DRAFT BCL ANALYSIS

BCL Potential Action: *Update*

BCL Comments: *Revise to reflect modern knowledge. Could relax or endorse national standard (more to follow).*

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The proposed action is within the scope of this project and there is no present or planned Staff action to revise Regulatory Guide 1.76. Therefore, further consideration is appropriate.

Step 2: Background Description

Regulatory Guide 1.76 has been used since 1974 by industry and the staff to determine the design basis tornado (DBT) for each of the geographical regions defined in the Guide. Due to the fact that very little area specific data on the damage areas and tornado intensity was available, generalized conservative estimates were used in the development of the DBTs in the Guide.

Pacific Northwest Laboratories (PNL) conducted an NRC-sponsored study using data for the 30,000 tornadoes during the period 1954 – 1983 and published the results in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1986. PNL found that the 10^{-7} annual probability wind speed ranged from 153 mph to 332 mph and concluded that it would appear to be reasonable to use DBT wind speeds of 200 mph west of the Rocky Mountains and 300 mph east of the Rocky Mountains. The staff agreed with PNL's proposed revisions to the methodology, but considered that the uncertainties in the data base and analyses required the use of a conservative strike probability. Using the PNL upper 90% confidence level for the 10^{-7} probability of occurrence, the staff developed DBT parameters for each of four geographic regions of the contiguous United States. These DBTs were issued as an interim position applicable to the Advanced Light Water Reactor standard design in the form of a "Safety Evaluation by the Office of Nuclear Reactor Regulation of Recommended Modification to the R.G. 1.76 Tornado Design Basis for the ALWR."

The interim position reduced the maximum wind speed and pressure drop for DBTs for significant areas of the United States. However, the maximum wind speed for the central region was 330 mph, which, of course, would have to be the DBT for a standard design intended to be used anywhere in the United

States. In response, EPRI noted that a 10^{-7} probability of occurrence at the 90% confidence level is conservative when compared with that used to evaluate other external events. EPRI, referencing ANSI/ANS 2.3-1983, proposed a DBT with a maximum wind speed of 260 mph as representing "an upper limit" at a probability of occurrence of 10^{-6}.

Step 3: Determination of Importance to Safety

Clearly, a less conservative DBT will reduce design loads and consequently reduce the margin of safety, unless there are some, as yet unidentfied, indirect negative effects on safety resulting from the current design requirements. (EPRI mentions that the elimination of certain provisions, such as HVAC inlet and exhaust labyrinths, will enhance maintenance.) Conservative application of the existing data on tornadoes will allow the development of appropriate DBTs without an unacceptable reduction of safety.

Step 4: Impact Analysis

The interim position established by the staff for application to the ALWR is, of course, also applicable to any other reactor in the contiguous U.S. Thus, the impact of the potential action depends on whether the revised Regulatory Guide would contain D[illegible] less restrictive than those now accepted by the staff. In issuing the interim position, the staff stated that it "constitutes a conservative reduction of the design basis winds which can be used by EPRI and standardized plant designers until a revised R.G. 1.76 is available." This statement suggests, and past experience would confirm, that the staff would be willing to accept a less conservative position within the more formal process of revising a Regulatory Guide than it has established for its interim position.

The effect of the DBT on design is significant. However, some relief is already available in the form of the staff's interim position. The assignment of values to the attributes in the table below is based on the potential for further reduction.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
Medium	Low	Low	Medium

Regulatory Position No. 35

RegGuide 1.92 Seismic Response Analysis Modal Responses
RegGuide 1.122 Seismic Response Spectra Floor Design

DRAFT BCL ANALYSIS

BCL Potential Action: *Undate*

BCL Comments: *These RegGuides address acceptable methods for calculating the stiffness of concrete structures other than containment. Research results generated recently at Los Alamos and Taiwan suggest that the current methods may overestimate stiffness and thus should be considered for revision.*

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The stiffness of structures is an input to some of the calculations addressed in these Regulatory Guides; however, the calculation of the stiffness is done prior to addressing the issues covered by the Guides. Consequently, revising the Guides will not resolve the issue raised in the BCL report. No further consideration will be given to the potential action.

Regulatory Position No. 36

RegGuide 1.108 Emergency Diesel Generator Periodic Testing
RegGuide 1.9 Emergency Diesel Generator Qualification

DRAFT BCL ANALYSIS

BCL Potential Action: *Replace/Update*

BCL Comments *RegGuide 1.108 specifies testing for emergency diesel generators and includes provisions for cold start. The cold start is expected to confirm reliable start of the emergency power supply in the event of a large break LOCA (?) coincident with loss of normal power supply. Unfortunately cold starts increase the wear and tear on diesels and could actually decrease their reliability over the long term. Furthermore, the significance of large break LOCA as an initiating event is less than originally believed (relative to accidents which develop more slowly).*

The suggested action is to withdraw RegGuide 1.108 and incorporate updated guidance into RegGuide 1.9. The updated guidance would reduce the frequency of cold start test to once per six months and would permit some warm start-up as expressed in Generic Letter 85-14. This action is in progress along with several related changes/updates on calculating loss of AC power duration and diesel generator reliability (RegGuide 1.155 Station Blackout).

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The potential action is within the scope of this project. However, the staff has been actively addressing this issue for several years. The problem is to find an approach that both promotes the reliability of the emergency diesel generators and provides adequate confidence in their reliability. Tests, while providing evidence of the reliability and performance of the diesel generators, contribute to wear that may decrease the long term reliability of the diesel engines.

The station blackout rule sharpened the regulatory focus on the reliability of emergency diesel generators. In response to the station blackout rule, the industry established several initiatives, one of which was to "reduce or eliminate cold fast starts of the emergency diesel generators through changes in the Technical Specifications or other appropriate means." Consistent with this initiative, the staff proposed a revision to Regulatory Guide 1.9 together with the withdrawal of Regulatory Guide 1.108. This proposal has been held up by controversy concerning the use of 10CFR50.54 to impose otherwise uncodified requirements on the utilities (e.g., to monitor diesel generator reliability).

This issue is currently under active consideration by the staff and the Commission and further consideration within this project is not appropriate.

Regulatory Position No. 37

RegGuide 1.109 Calculation of Annual Doses to Man

DRAFT BCL ANALYSIS

BCL Potential Action: *Update*

BCL Comments *RegGuide 1.109 provides the equations by which the NRC staff estimates radiation exposures for maximum individuals and population within 50 miles of the plant site. These equations yield the dose rates to various organs from various exposure pathways. Revised last in 1977, it is used primarily to demonstrate compliance with IOCFR50 Appendix I (ALARA). However, it also serves to provide acceptable methods for calculating doses to control room staff during postulated accidents (SRP 6.4) and off-site doses for various postulated accidents (RegGuides 1.3, 1.4, 1.25, and Safety Guide 5.)*

The potential action would update RegGuide, particularly the dose conversion factors and whole body/organ dose equivalents, with current information from ICRP 26. [Such revision is either already underway or is scheduled to follow current revisions to IOCFR20. See H. Petersen.]

SCIENTECH ANALYSIS

Step 1: Procedural Screen

Current information from ICRP 26 has been incorporated in the revision of 10CFR20 that was recently approved as a final rule. The staff action to bring appropriate Regulatory Guides into conformity with the revised 10CFR20 has focused primarily on those that are necessary to understand the implementation of the revised 10CFR20. The Regulatory Guides and SRP section listed in the BCL report do not meet this criterion and, consequently, were not included in the staff action.

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" is used, as its title indicates, to confirm compliance with the design objectives for effluents resulting from routine operation. 10CFR50, Appendix I established a quantification of ALARA for nuclear power plant effluents. The Appendix I requirements are based on the radwaste treatment technology available to reduce radioactive effluents and on the general standard that additional effluent treatment equipment is justified if the cost is less than $1000 per man-rem of dose reduction.

With the effluent treatment capabilities currently in place as a result of licensing requirements, the cost/benefit ratio of additional treatment using available technology is far in excess of the $1000 per man-rem standard. SCIENTECH expects the change in calculated dose that would result from an update of

Regulatory Guide 1.109 would have no significant effect on the cost/benefit ratio for additional treatment.

The other requirements listed in the BCL report relate to the calculation of exposures resulting from reactor accidents. Each presents conservative formulas for the calculation of atmospheric dispersion of the radionuclides released and for the approximation of whole body and thyroid dose received by persons in the path of the plume. Revision of these Guides to conform with 10CFR20 would not significantly affect the calculated doses.

In addition, with respect to Regulatory Guides 1.3 and 1.4, which are addressed also under Regulatory Position No. 31, the changes to the source term and to the definition of the exclusion areas and low population zones now under consideration would have a far greater effect on those Guides than any difference in dose calculations that would result from conformance with the revised 10CFR20.

SCIENTECH concludes that the proposed potential action would have no benefit and is therefore not within the scope of this project.

Regulatory Position No. 38

RegGuide 1.115 Protection against Low-Trajectory Turbine Missiles

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate*

BCL Comments *General Design Criterion 4 "Environmental and Missile Design Bases" protection of structures, systems and components important to safety against the effects of missiles that might result from equipment failures. The "equipment failures" of principal interest at the time were overspeed failures of main turbine-generator sets. The operating experience available at the time suggested that protection of important components against missiles from turbine failure was an appropriate safety consideration, particularly since many early plants had turbines oriented tangentially to the containment.*

RegGuide 1.115 describes acceptable methods for showing that the risk from turbine missiles is acceptably small, either through spatial orientation or physical protection. The RegGuide was last revised in 1977. Since then, newer plants have been designed with the turbines oriented radially to the containment. In addition there have been substantial improvements in turbine materials, turbine monitoring and overspeed protection which appear to have substantially reduced the risk of catastrophic failure.

The potential action is to eliminate RegGuide 1.115. The NRC staff no longer uses it, preferring instead to focus on the procedures and schedules for turbine inspection. On the other hand, the RegGuide imposes no apparent significant burden on anyone and remains sound design guidance.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

As noted in the BCL report, Regulatory Guide 1.115 remains sound guidance for plants that have tangentially oriented turbines. Most newer plants have, and future plants are expected to have, radially oriented turbines. The potential action is not within the scope of this project as elimination of the Regulatory Guide would require staff effort with no compensating reduction of NRC or licensee effort.

Regulatory Position No. 39

RegGuide 1.152 Criteria for Digital Computer Software

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand*

BCL Comments *RegGuide 1.152 describes acceptable methods for complying with GDC 21 "Protection system reliability and testability" as applied to safety related systems using programmable digital computer systems. The method applies to designing software, verifying software, implementing software, and validating computer systems. The RegGuide endorses ANSI/IEEE-ANS-7-4.3.2-1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations" as a method acceptable for designing software, verifying software, implementing software, and validating computer systems used in safety-related systems. That standard complements several others which also relate to safety systems:*

IEEE Std 279-1971 Criteria for Protection Systems
IEEE Std 467-1980 [Title?]
IEEE Std 603-1980 Standard Criteria for Safety Systems (hardware)
ANSI/ASME NQA-1-1979 [Title]

The nature (the existence?) of an issue here is somewhat unclear. In the late 1970's when a digital protection system was proposed for Arkansas Nuclear One, there was considerable consternation and effort to achieve reasonable assurance that the system would perform satisfactorily. Concerned about an error in the software, the NRC staff performed a line-by-line review. Though the system was ultimately approved, the experience deterred both licensees and staff from encouraging submittal of digitally-based designs.

In the interim, computer technology (e.g., microprocessors, fiber optics, PC's, etc) has progressed exuberantly in general and in safety-related applications in other industries. New standards have emerged. The advantages offered by digital systems over analog in terms of accuracy, reliability, versatility and cost have become generally recognized. On the other hand, design errors or failures in digital systems may be more subtle or difficult to detect. Yet there are no specific requirements and few documented regulatory positions to guide designers, licensees and reviewers. For example, NUREG-0700 "Review Criteria for Control Room Design Evaluations" does not meaningfully address new digital instrumentation, control and displays.

This is likely to remain an issue as licensees seek to refurbish operating reactors and as new designs are submitted for design certification. Current practice appears to sidestep the issue until forced by a particular event. For certification of new designs, the procedure appears to involve a "licensing basis *agreement" with designers which commits to the use of best available industry standards [L. Rubenstein?].*

SCIENTECH ANALYSIS

Step 1: Procedural Screen

Digital computer technologies continue to advance at a rapid rate. An expanded regulatory guide adopted under these conditions would be likely to lag behind, and consequently to discourage the use of, the best available industry standards in favor of methods already reviewed and accepted by the NRC Staff. To the extent that this occurs, the net impact upon safety would be negative and potentially significant.

To avoid this negative impact, the guidance would have to be written in a way that encouraged the use of the best available industry standards, which is, as noted in the BCI report, what is currently being achieved under the existing regulatory guidance. Thus, expanding the scope of the Regulatory Guide would, at best, require staff action without any reduction in effort by licensees.

The proposed action is not within the scope of this project and will receive no further consideration here.

Regulatory Position No. 40

Possible Regulatory Guide on credit for human performance

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand/Create*

BCL Comments *The evolution of reactor designs has been such that designers have placed specific reliance on the performance of reactor operators to terminate design basis events. Where such reliance exists, design guidance in the form of a current standard "20-minute rule" has been used frequently.*

With the advent of more investigation into events beyond design basis, there is a growing tendency to claim that plant operators and other plant personnel are capable and will take action necessary to manage an accident. What usually ensues is substantial discussion about why or why not such claims are valid.

The potential action would create a RegGuide or expand an existing one which provides guidance to licensees on an acceptable method for justifying credit for a specified operator action. An example is the action necessary to open a wetwell vent line manually during a station blackout sequence in a boiling water reactor. In order to receive credit for such action, the licensee could show the following:

-an analysis of the physical behavior of the plant under such conditions
-a procedure describing the specified action, including any tools or other assistance or personnel the procedure may entail
-certification that relevant training has been provided to those who must implement the procedure.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

When operator action is proposed as a method for coping with a particular event sequence, licensees already perform analyses, prepare procedures, and conduct training. Licensees justify the use of operator action to the staff when asked to do so. Because the appropriateness of assigning an operator to perform a particular task is a function of many factors, including the specific event involved, plant-specific design characteristics, training, operating philosophy, etc., the preparation of generic guidance would be difficult. The effort to develop such guidance would represent a large short term effort by NRC staff with no compensating long term reduction in effort by licensees .

The proposed action is not within the scope of this project and will receive no further consideration here.

Regulatory Position No. 41

Generic Letter 82-28
Reactor Vessel Level Indication System

DRAFT BCL ANALYSIS

BCL Potential Action: *Eliminate*

BCL Comments *The requirement for a means of unambiguously determining the water level in the reactor pressure vessel was imposed via generic letter as TMI Action Plan Item (NUREG-0730 II.F.2). The rationale was that had such an indication been apparent to the control room crew at TMI, they would have acted to restore inventory and flow through the core in the critical early hours of the accident.*

To date, [number] reactors have in them a Reactor Vessel Level Indication System. For boiling water reactors [some description of status]. For pressurized water reactors [some description of status]. Most licensees characterize the RVLIS as a required system whose benefit is unproven and thus whose cost is unjustified. Licensees propose that newly revised emergency operating procedures in use by trained operators assure identification of threats to core uncovery and thus satisfy the same safety function as would a backfitted RVLIS at considerably less cost. In those systems where RVLIS has been installed, operating performance and reliability has been "poor."

The potential action is to delete the requirement for RVLIS as it applies to currently operating reactors. For future reactors, it appears to be more practical to design a means for directly measuring water level in the pressure vessel. Incorporating RVLIS into the design should also lead to improved performance and reliability versus backfitted systems. Thus this requirement could be retained on the basis of lower cost for similar benefit.

[Despite repeated opportunities, no one on the staff defended RVLIS, but no one proposed to eliminate it either. Its inclusion here is primarily a carryover from the 1985 review of regulations. A similar situation existed for Safety Parameter Display Systems.]

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The proposed action is within the scope of this project. There is no staff action to address this issue; thus further consideration is appropriate.

Step 2: Background Description

The background for this Regulatory Position is adequately described in the BCL report.

Step 3: Determination of Importance to Safety

Operating plants have completed those measures necessary to provide unambiguous reactor pressure vessel water level indication. To remove the requirement for RVLIS would be to return to indirect and inferential indications rather than direct measurement of a parameter of critical safety concern (reactor vessel water inventory). SCIENTECH's judgement is that the level of safety afforded by RVLIS is commensurate with the cost of maintaining existing systems or installing an equivalent capability in future plants. The cost of implementing RVLIS in future plants is significantly smaller than the backfit of such a system, and the reliability of such systems can reasonably be expected to exceed that of current designs. To eliminate this requirement would have a potentially significant negative impact on safety and will not be given further consideration here.

Regulatory Position No. 42

General - Initial conditions

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand*

BCL Comments *For the most part, the regulations and associated regulatory positions assume the starting point for safety analysis to be operation at full power, an equilibrium condition, and all safety systems in compliance with technical specifications. In some instances, licensees have been required to perform safety evaluations from other conditions, such as 5% power. Operating experience and analyses [reference?] suggest that there are other operating modes in which the possibility of serious consequences may not be fully appreciated. For example during maintenance outages, loss of shutdown heat removal could lead to core damage with safety systems out of service and a relatively open containment.*

The potential action is to expand the regulations to address additional plant operating modes, such as extended shutdown, refueling, or other potential situations. The range and implications of such events should be explored, perhaps quantifying their risk. The results of this exploration should then be reviewed relative to 10CFR50 to see if changes to the regulations are appropriate. Having such regulations would increase assurance that all significant contributors to risk have been identified and controlled.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

One should not infer that regulatory requirements do not apply at conditions other than those assumed in safety analyses. The initial conditions for analysis are intended to bound the range of plant responses to an event, or at least to be representative of plant behavior. When new potential safety issues are identified, the staff requires licensees to evaluate whether they have design or operating significance. Sensitivity analyses may be employed to identify the parameters which render the analysis most conservative or most representative of expected plant behavior. Thus, at any given time, all initial conditions perceived to have safety significance are within the envelope of analyses being performed by the licensee.

NRR has work underway to develop a better appreciation of the risk associated with events that begin while the plant is shut down and safety systems are disabled (e.g., a leak in the reactor coolant system concurrent with loss of the one operating train of RHR while the vessel head and containment hatch are open). Expanding the regulations, based on the NRR or other work, would increase regulatory effort and is not within the scope of this project.

Regulatory Position No. 43

General - Safety terminology

DRAFT BCL ANALYSIS

BCL Potential Action: *Clarify*

BCL Comments *Various terms which include the word "safety" in them have been used liberally in the regulations and subsidiary positions since the beginning of regulation with the Atomic Energy Act of 1954. The more common ones are: "safety-related," "important to safety," "nonsafety-related," "safety function," "safety margin," and several others. There was considerable confusion surrounding these terms for many years, particularly when issues arose which might involve backfitting or design decisions. The confusion was such that in 1984, then-Director of Nuclear Reactor Regulation, H. Denton issued Generic Letter 84-01 to all licensees and applicants redefining the terms. That guidance appears to hold today as the last documented statement in that regard. Yet there still appears to be confusion within the reactor safety community, including the NRC staff.*

The potential action is somehow to clarify or reassert the definitions and appropriate uses of common terms containing the word "safety."

SCIENTECH ANALYSIS

Step 1: Procedural Screen

The proposed action is within the scope of this project and there is no present or planned Staff action in this area. Therefore, further consideration is appropriate.

Step 2: Background Description

Use of the terms "safety-related" and "important to safety" has not been consistent over time or between organizations, even those with common interests: the terms have been used differently by different licensees and by different organizations within the NRC Staff. As a result, contradictory positions on the meaning of these terms have emerged and have been documented by licensees, professional societies, and members of the NRC Staff.

The terms "safety-related" and "important to safety" were, at one point, generally understood to be equivalent. With the proposed Revision 1 to Regulatory Guide 1.89, Revision 2 to Regulatory Guide 1.105, and the draft Regulatory Guide on Instrument Sensing Lines (Task IC 126-5), the term "important to safety" was used in an expanded sense to include systems that were not safety systems or safety-related systems. To confirm the Staff's intention to use this expanded definition, the Director of the Office of Nuclear Reactor Regulation explicitly stated that the term "safety-related" described a subset of the category "important-to-safety." This was documented in an internal NRC memorandum dated November 20, 1981.

This change in regulatory terminology created significant concern in the industry. The Power Engineering Society of the Institute of Electrical and Electronics Engineers wrote, in a letter dated May 10, 1982, to the Director of the Office of Nuclear Regulatory Research, that

> "Broadening the usage of the term "important-to-safety" to encompass an undefined set of systems, in addition to safety-related or safety systems, increases confusion in the dialogue on current NRC requirements/guidance and creates an unworkable situation."

After the close of the public comment period for the proposed 10CFR50.49, language was inserted into the approved "EQ Rule" (10CFR50.49) that defined a class of non-safety equipment that was "important to safety." Licensees perceived this to be an imposition of a new requirement without appropriate review of its backfitting implications.

To bolster their position, the NRC issued Generic Letter 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related,' " which was referenced in the BCL report. In 1986, a position paper was drafted by the staff, proposing definitions and additional guidance to licensees. This was signed by the Executive Director of Operations but never published. According to Federal Register notice dated Monday April 24, 1989, the timetable for rulemaking is on hold based on a decision by the Commission.

Step 3: Determination of Importance to Safety

Clarifying the terminology of regulation could result in improved implementation by both regulators and licensees, which could be expected to enhance safety. Regulatory requirements that may not be necessary that have been applied to non-safety equipment could be removed, permitting better focus of attention on safety-related equipment. Additional requirements, if any, would be imposed more uniformly on other systems, structures, and components.

Step 4: Impact Analysis

The burden associated with the proposed action would be very large. The Staff would have to achieve internal consensus on the definition of these terms and respond effectively to a wide variety of industry and public comments. This is a complex issue because regulatory decisions have been based upon the interpretations of these terms by individual members of the staff. Areas of regulation and regulatory activity that might be inconsistent with a new definition would have to be reviewed and rectified. Licensees would have to bring their plants into conformance with the new regulatory language. There is a high probability that 10CFR50.109 would be invoked.

Benefits could be commensurately large for future licensing activities as an unequivocal definition would eliminate the significant licensing burden for the industry created by the inconsistency of interpretation. The current trend toward increasingly broad but inconsistent interpretations suggests that the net benefit would not be significantly diminished, even if the new definition of the term "important to safety" turned out to be broader in overall coverage than current interpretations.

Generic Impact Attributes

Short Term Burden		Long Term Benefit	
NRC	Licensees	NRC	Licensees
High	High	Medium	High

Regulatory Position No. 44

General - Currency of regulations

DRAFT BCL ANALYSIS

BCL Potential Action: *Expand*

BCL Comments *One perceived difficulty in ensuring effective, efficient regulation is the extent to which regulatory requirements have become highly intertwined and outdated. Despite efforts to keep regulation current and self-consistent, the body of regulatory positions has become too complex to manage. In many cases requirements have been imposed or negotiated through mechanisms (e.g., Generic Letters, Orders) other than formal rulemaking. In other cases the regulations have been changed in response to a specific incident (e.g., TMI, Brown's Ferry), focusing on existing reactors and imposing specific implementation schedules. In both cases, the intent is usually to harmonize all relevant regulatory positions at some later time, but in practice subsequent events combined with finite resources often delay selfconsistency for years or indefinitely.*

The potential action is to expand the regulations by adopting a "sunset" provision which forces periodic review of regulatory positions at some prespecified time after their initial adoption. The review would address the continuing need for the position and any recommended changes. The outcomes of such a review might be a reaffirmation of the continuing need, a proposed revision, or a cessation of effectiveness. To a major extent this process is practiced informally during the normal course of regulation. The suggestion here is to formalize this process more, thus motivating more regular, higher level attention.

SCIENTECH ANALYSIS

Step 1: Procedural Screen

A mandatory, formal, periodic review of regulatory requirements would result in elimination of requirements that are no longer relevant, rejustification of requirements that remain relevant, and replacement of requirements for which improved approaches can be defined. The first two results would have little benefit and would likely result in a net increase in NRC and licensee effort. The third result would certainly introduce change into the regulatory environment, and change has frequently been unfavorably equated with instability. Whether the net effort would increase or decrease is impossible to anticipate.

Further consideration of this potential action is not appropriate for operating reactors and future LWRs of evolutionary design; however, such an effort could be appropriately included in the NRC program to review regulations relative to the licensing of advanced reactors.