ASSESSMENT OF OPERATING LICENSE

SEABROOK STATION, UNIT NO. 1

U. S. Nuclear Regulatory Commission

Regulatory Review Group

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1. INTRODUCTION

1.1 Background

The Regulatory Review Group (referred to hereinafter as Review Group) charter calls for the assessment of operating licenses by selecting several licenses issued at various times, determining how the regulations and regulatory guidance were incorporated into the licenses, determining how much inherent flexibility the licensees have in making changes to their plants or operations and determining what in the regulatory process may be inhibiting the use of the inherent flexibility. In addition, the Review Group considered areas where enhanced flexibility could potentially be provided.

The following sections describe the selection of the plants whose operating licenses were assessed and the approach that was used to assess the licenses.

1.2 Selection of Plants (Licenses)

Four plants (licenses) were selected for the assessment. This number was based on the number judged necessary to accomplish the objectives of the Review Group's Charter and the number needed to be representative of a significant number of plants (licenses).

A substantial number of criteria were considered in the selection of the four plants. However, it was the view of the Review Group that the following criteria were the most important (listed in order of importance) for the purposes of this activity:

- Recent and early licenses
- BWR and PWR plants (licenses)
- Representativeness of significant number of plants
- Availability of PRA/IPE (for possible interface with the PRA Technology Subgroup)

Using the above criteria, Seabrook Station, Unit No. 1; Surry Power Station, Unit No. 1; Perry Nuclear Power Plant, Unit No. 1; and Peach Bottom Atomic Power Station, Unit No. 2 were selected from among all of the plants currently licensed to operate.

Seabrook was selected because it is one of the most-recently-licensed PWRs; it is a Westinghouse four-loop plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE that had been reviewed by the NRC.

Surry was selected because it is one of the earliest-licensed PWRs; it is a Westinghouse three-loop plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE whose review by the NRC is nearly complete. Surry 1 is also one of the plants evaluated in WASH-1400 and NUREG-1150.

Perry was selected because it is one of the most-recently-licensed BWRs; it is a General Electric BWR-6, Mark III containment plant and is, therefore, representative of a significant number of plants (licenses); and it has an IPE which is under review by the NRC.

Peach Bottom was selected because it is one of the earliest-licensed BWRs; it is a General Electric BWR-4, Mark I containment plant and is, therefore, representative of a significant number of plants (licenses); and, although the NRC has not completed its review of its IPE, it is one of the plants evaluated in WASH-1400 and NUREG-1150.

1.3 Assessment Approach

The assessment approach is summarized in Table 1. The approach involved the assessment of items of the operating license, either individually or collectively. For the purposes of this assessment, an item is defined as any license condition or Technical Specification definition, safety limit, limiting safety system setting, limiting condition for operation, design feature or administrative control that is designated alphanumerically in the license. Technical Specification bases were excluded since they are not part of the Technical Specifications and, hence, the license. Except for the applicability section, a Technical Specification limiting condition for operation and its associated surveillance requirement were counted as a single item.

A typical operating license contains several hundred items as discussed above. To facilitate the assessment and to ensure adequate consideration of all types of license requirements, the items were reviewed and assigned to one of the seven categories described in Table 2. Where an item could be assigned to more that one category, it was assigned to the most dominant category.

The categories were defined to optimize the assessment effort and to ensure adequate consideration of all types of license requirements. First, categories were established which would allow all of the items in as many categories as possible to be assessed collectively. This meant that all of the items in the category had to have similar characteristics. Secondly, where it was not possible to assess the items collectively and the items had to be assessed individually, the categories were established to allow the items to be representative of as many of the others in the same category as possible.

The items were reviewed to determine which categories contained items with similar enough characteristics to be assessed collectively. The items in the remaining categories were then considered to determine the percentage that could be assessed individually. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category.

The items which were to be assessed individually were selected from the remaining categories. The items were selected based on their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or because they were of special interest.

Although not every item of the license was assessed, the categorization of the items and the selection of a significant number of representative items for assessment from each category ensured adequate coverage of the license. The selection of items for assessment from subsequent license(s) will be based on validating the findings from the license(s) already assessed and expanding both the number and scope of the items assessed.

The items were assessed either collectively or individually as appropriate by considering the answers to specified questions presented in Table 3. The questions were designed to determine whether the item has a sound regulatory basis, is related to public health and safety, inherently allows the licensee flexibility in making changes to the plant or operations, or could be modified to provide increased flexibility to the licensee. The questions were written in such a manner that a "no" response would elicit additional review. The items were analyzed as necessary to ensure an adequate understanding of the items' regulatory bases, safety relevance, inherent flexibility and potential for enhanced flexibility.

Summaries of the assessments were prepared for each of the items. Each of the summaries contained overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. The results of each of the assessments were integrated and summarized, and the overall findings and recommendations were developed. Those items which inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Table 1

SUMMARY OF ASSESSMENT APPROACH

1. Review each operating license item and assign it to a category
2. Determine which categories contain items that are appropriate to be assessed collectively
3. Determine which items from the remaining categories will be assessed individually
4. Assess items in accordance with specified questions; research items as necessary
5. Prepare assessment summaries
6. Integrate results, and develop overall findings and recommendations

Table 2

CATEGORIES OF ITEMS

A. Technical Requirements - items which impose requirements based upon plant design, operational or other technical constraints (e.g., limiting conditions for operation).

B. Non-Technical License Conditions - items exclusive of the Technical Specifications which discuss broad management/issue considerations, generally of a non-engineering nature (e.g., financial conditions, organizational constraints).

C. License Conditions which Rely on Other Documents for Requirements - items which refer to other documents (e.g., physical security plan, NPDES permit) for the required actions or constraints.

D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements) - items in the Technical Specifications which impose non-technical organizational and programmatic requirements (e.g., station staff, committees, training), exclusive of specific reporting and recordkeeping provisions.

E. Reporting and Recordkeeping Requirements - items which discuss licensee reports and records, or impose related requirements (e.g., routine and annual reports, record retention and distribution).

F. Unique Plant Features - items which describe a design feature of the plant and its environs or define plant system/component configuration details (e.g., site characteristics, reactor and containment design parameters).

G. Other - items which impose conditions which are not covered by any of the other categories (e.g., legal provisions, exemptions, definitions, statements).

Table 3

ASSESSMENT QUESTIONS

1. Regulatory Bases

   A. Are the items supported by documented regulatory bases (e.g., regulatory guidance or requirements)?

   B. Are the regulatory bases supported by a legal requirement (e.g., Atomic Energy Act, Commission regulation or order)?

   C. If not legally required, have regulatory guidance and/or licensee commitments been appropriately used to impose the items?

2. Safety Relevance

   A. Are the items necessary to ensure public health and safety (e.g., are they needed for adequate protection, defense in depth, etc.)?

   B. Are the items in the group generally consistent, coherent and commensurate with safety significance?

   C. Are the items, as implemented, reasonably within their original intent?

   D. Are surrogate items (e.g., quantitative requirements) both necessary and appropriately used to meet the safety objective?

3. Inherent Flexibility

   A. Does an inherent flexibility exist which allows the licensee a tradeoff of items without a reduction in overall safety?

   B. Are other means, besides a license amendment, available to the licensee for revising the items?

   C. Can the change/revision be made without NRC pre-approval?

   D. If yes, can the change/revision be made without an NRC post-implementation review?

Table 3, (Cont'd.)

ASSESSMENT QUESTIONS

4. Enhanced Flexibility Potential

   A. If prescriptive language appears in the items, is it needed to convey the intended requirement?

   B. Would the use of performance-based criteria be inappropriate to add flexibility to item implementation?

   C. If specific factors which limit flexibility are identified, are all these factors beyond the control of the NRC?

   D. Would further NRC review of this area for enhanced flexibility be unproductive? (i.e., the licensee doesn't need or isn't likely to use any resulting initiatives)

   E. Are NRC programs currently ongoing or under evaluation for implementation which would provide enhance flexibility to the licensee?

2.0 ASSESSMENT OF SEABROOK OPERATING LICENSE

2.1 Seabrook License

The Seabrook Station, Unit No. 1 full-power operating license was issued on March 15, 1990. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the environmental protection plan, which is Appendix B to the license. The license as reviewed had been amended through Amendment 11, dated May 29, 1992.

2.2 Assessment of License

The Seabrook operating license contains 331 items. Each of the items was reviewed and assigned to one of the categories in Table 2. The numbers of items in the Seabrook operating license by category are shown in Table 4.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on the items' regulatory bases, safety relevance, inherent flexibility and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions;" Category F, "Unique Plant Features;" and Category G, "Other." These three categories encompassed 77 items or approximately 23 percent of the total number of items.

The number of items in the remaining categories that would be assessed individually was determined to be approximately ten percent or 25 of the 254 remaining items. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., ten percent, or five of the 50 items in Category D would be selected for further assessment. With the 77 items which would be assessed collectively, this meant that 102 or approximately 31 percent of the 331 total items would be assessed either collectively or individually.

The items which were to be assessed individually were selected based on their representativeness of a significant number of other items in the category, their enhanced flexibility potential or because they were of special interest. The items which were assessed are listed in Table 5.

Each of the items were assessed either collectively or individually as appropriate by considering the answers to specified questions presented in Table 3. The questions were designed to determine whether the item has a sound regulatory basis, is related to public health and safety, inherently allows the licensee flexibility in making changes to the plant or operations, or could be modified to provide increased flexibility to the licensee. The questions were written in such a manner that a "no" response would elicit additional review. The items were analyzed as necessary to ensure an adequate understanding of the items' regulatory bases, safety relevance, inherent flexibility and potential for enhanced flexibility.

Assessments summaries were prepared for each of the items. Each of the summaries contain overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. The results of each of the assessments were integrated and summarized, and the overall findings and recommendations were developed. Finally, those items which inherently allow licensees flexibility in making changes to their plants or operations were reviewed to determine what in the regulatory process may be inhibiting their use of this flexibility.

2.3 Results of Assessment

The summaries of the assessments of each of the items are provided in the Appendix. The summaries are presented in order of the categories into which each of the items were assigned. Within each category, the items are addressed in the order in which they appear, first, in the operating license (OL) itself, next, in the Technical Specifications (TS) and finally, in the environmental protection plan (EP).

Table 4

SEABROOK OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	136
L Non-Technical License Conditions	4
C. License Conditions which Rely on Other Documents for Requirements	32
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	50
E. Reporting and Recordkeeping Requirements	36
F. Unique Plant Features	10
G. Other	63
Total	331

Table 5

INDEX OF SEABROOK OPERATING LICENSE ITEMS WHICH WERE ASSESSED

Item	Subject	Page*
	Category A (13 of 136)**	
TS 2.1.2	Reactor coolant system pressure	26
TS 3.0.3	General limiting condition for operation	27
TS 3.1.2.7	Isolation of unborated water sources	28
TS 3.3.3.3	Seismic instrumentation	29
TS 3.4.6.2	Operational leakage	30
TS 3.5.4	Refueling water storage tank	31
TS 3.6.1.7	Containment ventilation system	32
TS 3.7.1.2	Auxiliary feedwater system	33
TS 3.7.4	Service water system	35
TS 3.8.2.1	D.C. electrical power system	37
TS 3.9.4	Containment building penetrations	38
TS 3.12.2	Land use census	39
TS 5.6.3	Spent fuel storage pool capacity	40
	Category B (4 of 4)***	
OL 2.B.7	Sale and leaseback condition	41
OL 2.H	Financial protection condition	41
OL 2.I	Marketing of energy condition	41
OL 2.J	Effective date and expiration condition	41
	Category C (3 of 32)**	
OL 2.E	Physical security condition	42
TS 3.4.10	Structural integrity	43
TS 6.2.2.e	Station staff working hours	45
	Category D (5 of 50)**	
TS 6.2.2.a	Minimum shift crew composition	46
TS 6.2.3.2	ISEG composition	47
TS 6.4.1.7	SORC responsibilities	48
TS 6.7.3	Temporary changes of procedures	50
EP 3.1	Changes in design and operation	52

Table 5 (Cont'd.)

INDEX OF SEABROOK OPERATING LICENSE ITEMS WHICH WERE ASSESSED

Item	Subject	Page*
	Category E (4 of 36)**	
OL 2.G	Violation reporting condition	53
TS 3.3.3.4	Meteorological instrumentation	54
TS 6.4.1.8	SORC records	56
TS 6.8.1.5	Monthly operating reports	57
	Category F (10 of 10)***	
OL 2.A	Applicability condition	58
TS 5.1.1	Exclusion area	58
TS 5.1.2	Low population zone	58
TS 5.1.3	Unrestricted areas	58
TS 5.2.1	Containment configuration	58
TS 5.2.2	Containment design pressure and temperature	58
TS 5.3.1	Reactor fuel assemblies	58
TS 5.3.2	Reactor control rod assemblies	58
TS 5.4.2	Reactor coolant system volume	58
TS 5.5.1	Meteorological tower location	58
	Category G (63 of 63)***	
OL 1.A	Finding - licensee	59
OL 1.B	Finding - construction completion	59
OL 1.C	Finding - conformance with requirements	59
OL 1.D	Finding - reasonable assurance	59
OL 1.E	Finding - technical qualification	59
OL 1.F	Finding - financial protection	59
OL 1.G	Finding - issuance of license	59
OL 1.H	Finding - satisfaction of requirements	59
OL 1.I	Finding - special nuclear material	59
OL 2.B.1	Authorization - possess, use and operate	59
OL 2.B.2	Authorization - possess	59
OL 2.D	Exemptions	59
TS 1.1	Definition - action	59
TS 1.2	Definition - actuation logic test	59
TS 1.3	Definition - analog channel operational test	59

Table 5 (Cont'd.)

INDEX OF SEABROOK OPERATING LICENSE ITEMS WHICH WERE ASSESSED

Table 5 (Cont'd.)

INDEX OF SEABROOK OPERATING LICENSE ITEMS WHICH WERE ASSESSED

Item	Subject	Page*
TS 1.38	Definition - thermal power	59
TS 1.39	Definition - trip actuating device operation	59
TS 1.40	Definition - unidentified leakage	59
TS 1.41	Definition - unrestricted area	59
TS 1.42	Definition - ventilation exhaust treatment system	59
TS 1.43	Definition - venting	59
EP 1.0	Objectives	59
EP 4.2.2	Terrestrial monitoring condition	59
EP 4.2.3	Noise monitoring condition	59

OL = Operating license condition
TS = Technical Specification
EP = Environmental Protection Plan condition

* Page number of assessment summary in the Appendix

** Numbers in parentheses indicate the number of the total number of items in the category which were assessed.

*** Items which were assessed collectively; all others were assessed individually

3. FINDINGS AND RECOMMENDATIONS

3.1 Introduction

The item assessment summaries were reviewed to determine which of the items appear to exceed the applicable regulatory requirements, given their safety significance and regulatory bases; which of the items should be considered for possible reduction in regulatory burden; which of the items provide at least some inherent flexibility, and why licensees may not be taking full advantage of that flexibility; and which of the items should be considered for enhanced flexibility. The items which have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Item 2E, the physical security operating license condition, appears in three groups. The item appears to have potential for reduction in regulatory burden; it has at least some inherent flexibility; and it appears to have potential for enhanced flexibility.

3.2 Findings and Recommendations

3.2.1 Items which appear to exceed applicable regulatory requirements

Findings: Seven of the items assessed appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the Seabrook operating license. It is recognized that 10 CFR 50.50 authorizes the Commission to include in licenses such conditions as it deems appropriate. The Review Group was not able to review the entire body of underlying regulatory guidance for all of these items. Therefore, although all of the items appear to prescribe conditions or require actions that exceed applicable regulatory requirements, there may indeed be additional regulatory bases for their presence as license conditions.

The items which appear to exceed the applicable regulatory requirements are as follows:

TS 3.1.2.7	Isolation of unborated water sources
TS 3.7.1.2	Auxiliary feedwater system
TS 3.7.4	Service water system
TS 3.8.2.1	D.C. electrical power system
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.8.1.5	Monthly operating reports

Technical Specification 3.1.2.7 exceeds the provisions of both the Standard and Improved Standard Technical Specifications in that they contain no provisions for isolation of unborated water sources in the shutdown modes.

Although generally similar in design to other Westinghouse four-loop plants, some of Seabrook's systems are unique, both in meeting applicable regulatory guidance and in providing component and system redundancy which exceeds regulatory requirements. Technical Specification 3.7.1.2 appears to elevate the interpretation of branch technical position guidance to the status of a general design criterion resulting in the imposition of additional requirements somewhat inconsistent with the original plant design. Technical Specifications 3.7.4 and 3.8.2.1 appear to ignore the extra redundancy afforded by the original systems' design and either impose additional provisions on the systems or require that the extra components receive the equivalent Technical Specification controls mandated for other Westinghouse four-loop plants without spare equipment. The licensee, in effect, appears to have been penalized for providing this additional redundancy and hence, increased safety margin, and for its attempt to use unique design applications.

The problems with Technical Specifications 3.1.2.7, 3.7.1.2, 3.7.4 and 3.8.2.1 appear to be in their implementation in the Seabrook operating license. Since the problems are plant-specific in nature, they can be pursued directly by the Seabrook licensee. However, these and similar types of Technical Specification provisions may exist at other plants. Therefore, consideration should be given to providing additional guidance for accommodating the governing criteria of systems with extra component redundancy and unique design applicability.

Technical Specifications 6.2.2.a, 6.2.2.e and 6.8.1.5 elevate provisions of Commission policy statements, regulatory guides and other non-requirements to the status of legal requirements. Technical Specification 6.8.1.5 elevates a Regulatory Guide reporting provision for which there is questionable safety justification to the status of a legal requirement.

Recommendation(s): Based on the foregoing, the Review Group recommends the following:

- o Reconsider the practice of elevating Commission policy statements, regulatory guides and other non-requirements to the status of legal requirements without following the disciplined rulemaking process.

- o Evaluate the adequacy of existing guidance for reviewing design features which exceed regulatory requirements or provide alternative means of compliance. Such guidance should encourage flexibility in the Technical Specifications for those design features for which the review concludes that increased safety margin is provided.

3.2.2 Items which should be considered for possible reduction in regulatory burden

Findings: Four of the items assessed appear to have the potential for possible reduction of regulatory burden. They are as follows:

OL 2.E	Physical security condition
TS 3.3.3.3	Seismic instrumentation

TS 3.3.3.4	Meteorological instrumentation
TS 6.8.1.5	Monthly operating reports

The physical security license condition, OL 2.E, essentially repeats the 10 CFR 50.54(p) requirement to obtain a license amendment to make changes to the physical security plans which decrease their safeguards effectiveness. Similar plans, e.g., the emergency response plan and the quality assurance plan, do not require a license amendment to make such changes. Although required by the regulations, this higher-level change process does not appear to be justified in terms of the physical security plans' safety significance relative to that of the other plans. Also, consideration should be given to providing enhanced flexibility in the implementation of the physical security plans. This aspect is addressed in Section 3.2.4 of this report.

Two items, Technical Specifications 3.3.3.3, seismic instrumentation, and 3.3.3.4, meteorological instrumentation, impose reporting requirements as surrogates for corrective actions. Further analysis, however, revealed that these Technical Specifications do not appear in the Improved Standard Technical Specifications and, therefore, can be considered for line-item elimination.

Technical Specification 6.8.1.5 imposes a Regulatory Guide reporting provision for licensees to submit monthly operating reports. This appears to be a significant burden for the licensees without a commensurate return in safety. Although the Committee to Review Generic Requirements (CRGR) and the Reporting Requirements Task Force have evaluated a number of specific reporting requirements, a broader approach which considers all the information needed by the NRC to satisfy its regulatory mandate may be appropriate.

Recommendation(s): Based on the foregoing, the Review Group recommends the following:

- Evaluate the efficacy of a consistent approach for accommodating changes to the physical security, emergency response and quality assurance plans within their proper regulatory and safety contexts; reconsider the current requirement for physical security plans to be included in a license condition.

- Conduct a comprehensive reevaluation of the information/data the NRC needs from nuclear power plant licensees in order to accomplish its mandate of protecting the health and safety of the public (recognizing the efforts of the CRGR and the Reporting Requirements Task Force); information/data requirements without a clear nexus to that mandate, and duplicative reporting requirements should be eliminated.

3.2.3 Items with inherent flexibility

Findings: Six of the items assessed were found to have at least some inherent flexibility. That an item has at least some inherent flexibility does not preclude it from consideration for enhanced flexibility or reduction in regulatory burden. The items with inherent flexibility are as follows:

OL 2.E	Physical security condition
TS 3.4.10	Structural integrity
TS 3.9.4	Containment building penetrations
TS 3.12.2	Land use census
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours

The nature of the inherent flexibility provided by these items varies from item to item. For example, the physical security and land use census items provide inherent flexibility by specifying the conditions under which changes to their respective programs can be made without prior NRC approval. The item governing the structural integrity of ASME Code components derives its flexibility not only from the ASME Code component classification process, but also from the relief request process used to exempt impractical Code requirements. Further flexibility has been provided by NRC guidance, such as Generic Letter 91-18, which is an example of a regulatory enhancement to flexibility with no adverse impact on safety.

The inherent flexibility of the containment building penetrations item is recognized in the options provided for compliance with the operability criteria. The minimum shift crew composition item specifies just that; licensees may exceed the minimums without NRC approval. The station staff working hours item provides the licensee essentially unlimited flexibility in setting the staff's working hours without NRC approval provided the appropriate procedures are followed.

Of the six items with inherent flexibility, one item--physical security--was also judged to have potential for reduction in regulatory burden and enhanced flexibility. These aspects are addressed in Sections 3.2.2 and 3.2.4 of this report. Another item--land use census--has been eliminated from the Improved Standard Technical Specifications and, therefore, could be considered by licensees for line-item elimination from their Technical Specifications. The remaining four items revealed no bases for further consideration.

Although licensees appear to be taking advantage of much of the inherent flexibility afforded them, a significant amount of that flexibility is not being exercised. Possible reasons include (1) the lack of awareness on the part of the licensees that the flexibility exists; (2) the flexibility afforded by an item is not needed; (3) the cost in time or resources to take advantage of the flexibility outweighs its benefits; (4) potential for public hearing if exercise of the flexibility requires a license amendment or prior NRC approval; (5) fear of second-guessing by NRC reviewers or inspectors if the change is subject to post-implementation scrutiny; (6) fear of ratcheting by NRC reviewers or inspectors during the change process; (7) negative perception of the licensee's actions by state regulatory bodies, the NRC or the public; (8) complacency on the part of the licensee; and (9) reluctance of a licensee to assume the lead in pursuing changes to license requirements, e.g., line-item improvements in accordance with the Technical Specification Improvement Program.

Recommendation(s): Based on the foregoing, the Review Group recommends the following:

- Invite the industry to provide the staff with candid insights on licensees' reasons for not taking more advantage of the inherent flexibility afforded them.

3.2.4 Items which should be considered for enhanced flexibility

Findings: Six of the items assessed appear to have enhanced flexibility potential. They are as follows:

OL 2.E	Physical security condition
TS 3.0.3	General limiting condition for operation
TS 3.6.1.7	Containment ventilation system
TS 6.2.3.2	ISEG composition
TS 6.4.1.7	SORC responsibilities
TS 6.7.3	Temporary changes of procedures

The physical security license condition, OL 2.E, provides flexibility in making changes to the physical security plans; however, additional flexibility could be provided in the implementation of the plans. For example, compensatory measures are generally prescriptive and may not always be in the best interest of overall plant security. Allowed outage times are not permitted as they are for safety-related equipment in the Technical Specifications. In addition, the baselines from which changes can be made without prior NRC approval are set by the provisions of the plans themselves, not by the regulations.

Technical Specification 3.0.3 may be unduly prescriptive in that it requires that the plant be shut down within specified completion times when the other Technical Specification limiting conditions for operation and their associated action statements are not met. It does not consider the risk of extending the completion times relative to that of shutting down the plant. This is an area that could be made more performance based and in which the application of risk assessment methodology could be considered.

Technical Specification 3.6.1.7 appears to be unduly prescriptive in ensuring the intended containment isolation requirement. More performance-based options for assuring that valves are "locked-closed" or "sealed-closed" are needed. In addition, flexibility in the surveillance requirements, especially for the smaller diameter penetrations, may be appropriate, particularly if properly coordinated with the provisions of 10 CFR 50, Appendix J. This is an area in which the application of risk assessment methodology could be considered.

Technical Specification 6.2.3.2 was initially identified for consideration for enhanced flexibility but it has been replaced in the Improved Standard Technical Specifications by a substantially more flexible requirement. Therefore, it may be considered for a line-item improvement.

Technical Specification 6.4.1.7 appears to be unduly prescriptive in that it requires the SORC to provide the same level of consideration to required procedures and all proposed changes to station systems or equipment that affect nuclear safety. A more performance-based or graded approach which takes into account the relative safety significance of the different areas and items under review would provide additional flexibility. Such implementation flexibility would likewise affect the conduct of Technical Specification 6.7.3 activities, as the need for controls over temporary procedure changes could be conditioned on the safety significance of the affected procedures. These are areas in which the application of risk assessment methodology could be considered.

Recommendation(s): Based on the foregoing, the Review Group recommends the following:

- o Consider providing additional flexibility in the implementation of the physical security plans, such as providing Technical-Specification-type allowed outage times.

- o Evaluate the feasibility of employing a graded approach to the applicability of the technical provisions of certain limiting conditions for operation and surveillance requirements, and in the implementation of specific review committee functions, e.g., SORC procedure and design change reviews. The appropriate application of risk assessment methodology could be valuable in establishing both the bounds and direction of such an approach.

3.2.5 Items considered or being considered in other programs

Findings: Seven of the items assessed have already been or are being considered in other programs. They are as follows:

TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 3.12.2	Land use census
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.2.3.2	ISEG composition
TS 6.8.1.5	Monthly operating reports

Technical Specifications 3.3.3.3, 3.3.3.4, 3.12.2, and 6.2.3.2 have already been considered and eliminated by the Technical Specification Improvement Program. Therefore, these items can be considered for possible elimination from plant-specific Technical Specifications as line-item improvements.

The subjects of minimum shift crew composition and station staff working hours are being considered for possible modification by the Office of Nuclear Reactor Regulation. This effort could result in changes to their underlying Commission policy statements and regulations, and, consequently, the Improved Standard Technical Specifications.

Technical Specification 6.8.1.5 is being considered by the Reporting Requirements Task Force.

Recommendation(s): Based on the foregoing, the Review Group has no recommendations.

3.2.6 Items for which no further consideration is warranted

Findings: Ninety-three of the items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is included here.

The items for which no further consideration is warranted are as follows:

OL 2.G	Violation reporting condition
TS 2.1.2	Reactor coolant system pressure
TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 3.4.6.2	Operational leakage
TS 3.4.10	Structural integrity
TS 3.5.4	Refueling water storage tank
TS 3.9.4	Containment building penetrations
TS 3.12.2	Land use census
TS 5.6.3	Spent fuel storage pool capacity
TS 6.2.2.a	Minimum shift crew composition
TS 6.2.2.e	Station staff working hours
TS 6.2.3.2	ISEG composition
TS 6.4.1.8	SORC records
TS 6.8.1.5	Monthly operating reports
EP 3.1	Changes in design and operation
Cat. B items	Non-technical license conditions (4 items)
Cat. F items	Unique plant features (10 items)
Cat. G items	Other (63 items)

Recommendation(s): Based on the foregoing, the Review Group has no recommendations.

REFERENCES

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- Code of Federal Regulations, Title 10 - Energy, revised as of January 1, 1992.

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- USNRC Generic Letter 82-02, "Nuclear Power Plant Staff Working Hours," February 8, 1982.

- USNRC Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours," June 15, 1982.

- USNRC Generic Letter 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements," June 4, 1987.

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- USNRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," November 7,1991.

- USNRC IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2, November 8, 1979.

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- USNRC Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1, April 1974.

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- NUREG-0020, "Licensed Operating Reactors Status Summary Report."
- NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Draft Revision 5, (referred to in the Assessment Report as "Standard Technical Specifications").
- NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.
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- NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," December 1992.
- NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," September 1992, (referred to in the Assessment Report as "Improved Standard Technical Specifications").
- NUREG/CR-5925, "Risk Based Technical Specifications: Development and Application of an Approach to the Generation of a Plant Specific Real-Time Risk Model," October 1992.
- USNRC Branch Technical Position, BTP ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," Revision 2, July 1981.
- USNRC Inspection Manual, IP 37001, "10 CFR 50.59 Safety Evaluation Program," December 29, 1992.

REFERENCES (CONT'D.)

- USNRC Task Force on NRC Staff Assessment of Reporting Requirements for Power Reactor Licensees.
- Seabrook Station Emergency Response Manual.
- Seabrook Station Radiological Emergency Plan.
- Seabrook Station Updated Final Safety Analysis Report, Revision 1.
- ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
- ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities of Nuclear Power Stations," 1976.
- ANSI 18.7(ANS3.2), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, 1976.
- ISA-S67.03, "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection."
- NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," June 1989.

APPENDIX

ITEM ASSESSMENT SUMMARIES

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 2.1.2

Seabrook Technical Specification 2.1.2, reactor coolant system pressure, requires that the reactor coolant system pressure not exceed 2,375 psig. This item was chosen because it is representative of the Seabrook Technical Specification safety limits.

The regulatory bases for this Technical Specification are 10 CFR 50.36 and 10 CFR 50.55a. The former requires that the Technical Specifications include " ... limits on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity," in this case the reactor coolant pressure boundary. The latter requires that pressurized reactor coolant pressure boundaries meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The Technical Specification is relevant to safety in that it is needed to ensure the integrity of the reactor coolant pressure boundary, one of the plant's multiple barriers against the release of reactivity.

The Technical Specification provides no inherent flexibility to the licensee; it prescribes the maximum limit for the reactor coolant system pressure. That degree of prescriptiveness is not inappropriate in view of its safety significance. There appears to be no enhanced flexibility potential for this requirement.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 3.0.3

Seabrook Technical Specification 3.0.3, general limiting condition for operation, specifies what action must be taken when other limiting conditions for operation action statements are not met. This item was chosen because of its potential for enhanced flexibility.

10 CFR 50.36(c)(2) requires that when a Technical Specification limiting condition for operation, the lowest functional capability or performance level required for safe operation, is not met, the licensee shall follow any remedial action permitted by the Technical Specifications or shut down the reactor until the condition can be met. Technical Specification 3.0.3 delineates the completion times for shutting down the reactor when the limiting conditions for operation and their associated action statements are not met.

The requirement is relevant to safety in that the Technical Specification limiting conditions for operation and their associated action statements cannot cover all possible situations. Such a requirement is needed to cover those circumstances in which the other requirements are not met. The Technical Specification provides no inherent flexibility to the licensee.

It is not clear that the Technical Specification could not be made more flexible. Since not all limiting conditions for operation have the same safety significance, the completion times allowed for achieving hot standby, hot shutdown and cold shutdown could possibly be made more performance oriented, e.g., by considering situation-specific factors. Further, it may not always be safer to change operational modes. For example, if there is reasonable assurance that the situation could be rectified within one hour after the completion time for changing modes expires, it might be safer to maintain the reactor in its present mode for that additional period of time than to change modes.

Based on the above considerations, it is concluded that the Technical Specification is appropriate; however, it may be unduly restrictive. Therefore, it is recommended that further consideration be given to this item for possible enhanced flexibility. This might be an area where risk assessment methodology could be applied to compare the relative risks of extending the completion times and shutting down the plant.

SUMMARY OF ASSESSMENT

Category: A Items(s): TS 3.1.2.7

Seabrook Technical Specification 3.1.2.7, isolation of unborated water sources, requires isolation of the reactor coolant system from unborated water sources in the shutdown modes. This limiting condition for operation (LCO) ensures that the boron dilution flow rates cannot exceed the value assumed in the plant transient analysis. This item was selected for review because it is representative of the requirements for reactivity control systems and also provides the opportunity to evaluate shutdown provisions.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A and has safety relevance in providing reactivity controls (i.e., precluding boron dilution) that ensure acceptable fuel design limits are not exceeded. While some flexibility is allowed by providing the licensee options on component manipulations, there appears to be little overall inherent flexibility in this item. It is prescriptive in the LCO provisions as well as the action requirements. This Technical Specification also might also be considered a surrogate item in that it requires non-safety-related systems to be maintained in an inoperable state as a means of ensuring that an acceptable shutdown margin is maintained, whereas the capability to provide adequate boration during shutdown modes is redundantly assured by other Technical Specification requirements.

It is noted that both the Standard and Improved Standard Technical Specifications do not specify a comparable requirement to this item for the isolation of unborated water sources during shutdown conditions. Also, an inconsistency between the LCO and the documented bases in the Seabrook Technical Specifications was identified in that the bases imply that the isolation provisions are needed in Mode 3 (i.e., hot standby) but the LCO as written is not applicable in Mode 3.

Based upon the above discussion, it is not clear whether either the prescriptive language of this item or the item itself is a needed Technical Specification requirement. The potential for delaying core alterations (e.g., refueling operations) if the LCO is not met exists. However, any change to enhance the flexibility of the item may not be worth the effort, because the overall requirements are not considered onerous.

This item appears to be unique to the Seabrook Technical Specifications. While having a regulatory based safety intent, this item is prescriptive and appears to go beyond the regulatory requirements which provide the equivalent assurance of acceptable reactivity controls for similar reactors. More review is required to determine whether revision or elimination of this item from the Seabrook Technical Specifications is warranted. This item appears to illustrate how prescriptive technical requirements may be added as license conditions without a clear and consistent rationale for either the prescriptiveness or the lack of equivalency.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.3.3.3

Seabrook Technical Specification 3.3.3.3, seismic instrumentation, requires that the seismic monitoring instrumentation, delineated as a specific listing of components, be operable at all times. This capability is deemed necessary to permit a comparison of the measured response to any earthquake to the design basis of the plant. Selection of this item for review was based upon the desire to evaluate a technical provision which prescribes the submittal of a report as the only action requirement.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 100, Appendix A, with reference to 10 CFR 50, Appendix A criteria and describes seismic instrumentation intended to meet the recommendations of Regulatory Guide 1.12. Safety relevance is established by the need for data to determine if the plant can continue to be operated safely following an earthquake. While there is no inherent flexibility in meeting the limiting condition for operation or the action and surveillance requirements, continued operation is permissible with the seismic instrumentation inoperable. The prescriptive language in the surveillance requirements appears warranted to meet the safety intent of maintaining operable instruments and of analyzing seismic data following an earthquake. However, the prescriptive action requirement to submit a special report to the NRC if one or more seismic instruments is inoperable for more than 30 days appears to represent an example of a report being substituted as a surrogate item to the actual goal, i.e., timely repair of the instrument.

Reduction in regulatory burden could be provided by the elimination of the surrogate special report. It is recommended that all Technical Specification action items which require only a report to the NRC be reviewed further for appropriate usage. If the reporting requirement is only a surrogate for corrective action, a more direct and flexibly worded action statement or the elimination of the item altogether may be better. It is noted that seismic monitoring instrumentation is not included in the Improved Standard Technical Specifications.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.4.6.2

Seabrook Technical Specification 3.4.6.2, operational leakage, states that the reactor coolant system leakage shall be limited to the following: no pressure boundary leakage, one gpm unidentified leakage, one gpm total reactor-to-secondary leakage through the steam generators and 500 gpd through any one steam generator, ten gpm identified leakage, 40 gpm controlled leakage and reactor coolant system pressure isolation valve leakages as prescribed by formula and the referenced table. This item was chosen because it is representative of a technical requirement that does not provide flexibility.

The legal requirement for this item is contained in 10 CFR 50, Appendix A, General Design Criteria 30 which states that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant system leakage. The guidance for achieving this requirement is contained in Regulatory Guide 1.45.

Maintaining the integrity of the reactor coolant system pressure boundary is a primary safety concern. Consistent with that philosophy, it is necessary to maintain the prescriptive requirements related to the leakage limits currently contained in the Technical Specifications. The only requirement where some flexibility may be permissible is related to the ten gpm identified leakage limit provided that it could be demonstrated that there would be no reduction in the margin of safety if this limit were increased (i.e., the sensitivity of the leakage detection system was not degraded).

There are many surrogate methods of detecting reactor coolant system leakage, however, most do not provide a quantitative measurement. Regulatory Guide 1.45 contains several acceptable alternate methods and the Instrument Society of America Standard ISA-S67.03 also identifies alternate methods of leakage detection. Although these surrogates are available, it is questionable that they would provide the sensitivity required to satisfy the primary requirement of this Technical Specification or if these alternatives would be any easier to operate or maintain.

Based on the above considerations, it is concluded that the current Technical Specification requirements are appropriate to assure primary reactor coolant system integrity.

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 3.5.4

Seabrook Technical Specification 3.5.4, refueling water storage tank, requires that the refueling water storage tank contain a minimum volume of borated water, a minimum boron concentration, and a minimum and maximum solution temperature. This item was chosen because it is an example of a Technical Specification requirement that has the potential to provide additional flexibility.

The legal bases for this requirement is contained in 10 CFR 50, Appendix A, General Design Criteria 27, which requires that the reactivity control systems be designed with the capability of adding poison to the reactor through the emergency core cooling system to assure that reactivity changes can be controlled under accident conditions. Standard Review Plan Section 4.3 provides the guidance related to this requirement.

This requirement is important to safety since it provides a second independent method of reactivity control during accident conditions. This requirement is also prescriptive and affords little flexibility. The poison injection systems for boiling water reactors can use different combinations of poison concentration and flow rates provided the solution in the tank is maintained at a temperature that assures the poison remains in solution. Since this approach has been found acceptable and used for boiling water reactors, it may also be applicable to pressurized water reactors. However, there may not be any significant benefit for PWRs since the minimum volume of borated water in the refueling water storage tank is dictated by emergency core cooling system considerations.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 3.6.1.7

Seabrook Technical Specification 3.6.1.7, containment ventilation system, requires that each containment purge supply and exhaust isolation valve be operable to ensure primary containment isolation capability. The large, 36-inch diameter containment purge isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. The selection of this item for review was based upon its representativeness of Technical Specifications where administrative controls (e.g., locking closed valves) are implemented to comply with the limiting conditions for operation (LCO).

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in the primary containment isolation criteria of 10 CFR 50, Appendix A, and the radiation dose criteria of 10 CFR 100. The surveillance requirements of this item are also related to 10 CFR 50, Appendix J, but are more prescriptive in their provisions. A clear and coherent safety relevance has been established in the LCO, action and surveillance requirements; however, no inherent flexibility exists within the item. This is evidenced by the fact that even with blind flanges installed in the shutdown purge and exhaust pipe lines, no relief from the routine valve surveillances is inherently available. The blind flanges were installed to meet the quantitative local leak rate criteria for the valves.

The prescriptive language of this item does not appear to be necessary to convey the primary containment isolation functional requirements. For example, an asterisked note regarding verification of valve position monthly could be interpreted to require visual checks upon containment entries, even though the circuit breakers for these fail-closed valves are locked open and valve position indication is available in the control room. The enhanced flexibility potential for this item is, therefore, great. However, a Technical Specification revision would be required to clarify the existing language and expand the licensee's options to comply with the intended requirement. As a result of NRC inspection activities regarding Technical Specification compliance in this area, the Seabrook licensee is currently working with the Office of Nuclear Reactor Regulation and Region I on the interpretation and possible revision of this item.

While this item has a sound regulatory basis and safety relevance, the overall language is prescriptive and precludes the use of flexibility to meet the intended containment isolation requirement. The use of standard convention (e.g., what options exist to maintain a valve "locked-closed" or "sealed-closed") may add flexibility. Risk assessment methodology could be used to further evaluate the prescriptive requirements applied to all valves which are used to isolate the containment atmosphere. The results may indicate that smaller diameter penetrations require less rigorous surveillance requirements or administrative controls.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.7.1.2

Seabrook Technical Specification 3.7.1.2, auxiliary feedwater (AFW) system, requires at least three independent steam generator auxiliary feedwater pumps and associated flow paths to be operable. This capability ensures that the reactor coolant system can be cooled down to the point when the residual heat removal system may be placed into operation, in the event of loss of offsite power. This item was selected for review because it represents a case where a Seabrook safety system, such as the AFW system, design differs from the Westinghouse standard design.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in several General Design Criteria of 10 CFR 50, Appendix A, and is required to meet Branch Technical Position ASB 10-1 regarding diverse power sources in the application of the Standard Review Plan to the acceptability of the AFW design. While this item is safety relevant, the Seabrook AFW system design is unique (i.e., one 100% electric motor-driven pump in one AFW train instead of two 50% pumps to go along with the steam turbine-driven pump). This unique design has resulted in the addition of the non-safety-related startup feedwater pump to the AFW system Technical Specification as a third pump capable of being powered by an emergency electrical power supply upon manual operator action. The treatment of the startup feedwater pump as a AFW system Technical Specification requirement appears to go beyond the regulations and be otherwise based on a conservative interpretation of Branch Technical Position ASB 10-1, along with the apparent intent that the Westinghouse Standard Technical Specifications, which requires three AFW pumps, be mimicked.

This item has little inherent flexibility. The action requirement for an inoperable startup feedwater pump is the same as for either of the other two safety-related emergency feedwater pumps. Only when two pumps are declared inoperable is the action time extended if one of the inoperable pumps happens to be the startup feedwater pump. Given that the startup feedwater pump is located in the turbine building (i.e., a non-safety, non-seismic structure) and is normally powered by non-Class 1E (i.e., non-safety electric power), it appears that enhanced flexibility could be provided to the Seabrook licensee by at least allowing for a greater outage time for the startup feedwater pump than would be justified for either of the other two safety-related emergency feedwater pumps.

While the prescriptive language in this item was found to be needed to clearly delineate the requirements, the technical basis for incorporating all the startup feedwater pump requirements into this Technical Specification is neither consistent, nor coherent. For example, two startup feedwater pump flow paths, via both the normal, non-safety main feedwater flow path and the emergency feedwater header, are required to be demonstrated operable; whereas each emergency feedwater pump requires only its normal flow path to the steam generators. This surveillance requirement, in effect, adds an additional requirement that would not have been imposed if a third emergency feedwater pump had been designed into the AFW system.

While the above discussion reveals a unique Seabrook AFW question, it may be an example of a more generic issue. Plants whose system designs meet the regulations, but differ from Standard Review Plan guidance or Standard Technical Specification format may be penalized for their unique applications. As a generic coherency question, this issue may warrant further review.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.7.4

Seabrook Technical Specification 3.7.4, service water system, requires at least two independent service water loops to be operable with three operable pumps in each loop. The operability of the service water system ensures that sufficient cooling capacity is available for the continued operation of safety-related equipment during normal and accident conditions. This item was selected for review because the limiting condition for operation restrictively dictates the number of pumps in each service water loop which must be operable.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A, and has safety relevance in its functional capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink. However, while General Design Criterion 44 requires that "suitable redundancy" in components shall be provided, assuming a single failure, this item goes beyond the regulation by prescribing action if any one of six 100% pumps (or any combination thereof) is inoperable. Furthermore, this item requires more prescriptive actions than specified in the Standard Technical Specifications. In effect, it appears that in this case, the Seabrook licensee is being penalized for having a spare pump installed in each service water loop.

There exists no inherent flexibility in this item. The safety-related cooling tower onsite is designed with two independent cooling loops and provides an adequate ultimate heat sink option to the normal service water bay cooling path. Additionally, with two 100% capacity pumps in each loop of the service water cooling path, the loss of one pump in each loop would still provide redundant cooling capability to the normal ultimate heat sink, i.e., the Atlantic Ocean. However, given the above scenario (i.e., cooling tower totally available and each service water path functional, but one pump in each loop out of service), the Seabrook plant is placed in a three-day action requirement to shutdown. By comparison, a plant upon which the Standard Technical Specification requirement was imposed would have to take similar action only if it was down to one operable service water loop (versus the four available Seabrook loops posed for the above scenario).

The foregoing discussion illustrates that an enhanced flexibility potential is great for items where the licensee has chosen to design "spare" components into the safety-related plant systems. This upfront conservatism could be viewed by risk assessment methodology and/or performance-based system criteria as an enhancement to system availability. However, if the Technical Specification requirements do not recognize the inherent redundancy of the installed "spare" components, both the flexibility and the consistent application of safety significance are diminished.

As a generic issue, plant designs which utilize installed "spare" components to increase system reliability should be encouraged and not penalized by the addition of prescriptive Technical Specification requirements. While such spare components (e.g., pumps) must be safety-related and should be governed by Technical Specification surveillance requirements, the NRC should evaluate the need for imposing shutdown actions on plants with fully functional and redundant loops available to perform the system safety function. The Seabrook licensee is currently reviewing this item, and other similar items whose system design employs spare equipment, and plans to submit Technical Specification revisions to address total loop, versus component operability.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.8.2.1

Seabrook Technical Specification 3.8.2.1, D.C. electrical power system, identifies the D.C. electrical power sources that are required to be operable and energized when the plant is not shut down. This item was selected because it is an example of a Technical Specification that appeared to exceed the applicable regulatory requirements.

The primary regulatory requirement for this item identified in Standard Review Plan Section 8.3.2 is General Design Criteria 17 which states that the D.C. power system must be capable of performing its safety function assuming a single failure. The acceptance criteria for this requirement is contained in various regulatory guides and IEEE Standards.

This requirement can be satisfied by having two independent D.C. battery banks, one on each independent electrical train (i.e., Train A and B). Seabrook Technical Specification 3.8.2.1 requires the licensee to have two operable 125-volt D.C. battery banks on each electrical train, which is twice the number required by the regulations. In addition, although the extra batteries are not required, the Technical Specifications contain an action statement that requires the plant to be shut down if one of the battery banks in one of the trains is inoperable for 30 days and requires the surveillances to be performed on these batteries to demonstrate operability. Other plants have installed backup battery banks and the NRC has required them to be included in the Technical Specifications because they are safety-grade systems that are used in place of the primary battery system. However, the NRC imposed no operability requirements on these backup battery systems. The surveillance requirements are only applicable to these batteries when they are used in place of the primary batteries and no plant shutdown requirements are imposed if the batteries are inoperable when not in use (performing the backup function). Although the licensees generally maintain these batteries in accordance with the surveillance requirements, they are not subject to Technical Specification violations. This affords the licensees flexibility that is not permitted in the Seabrook Technical Specifications.

Based on the above considerations, it is concluded that the Seabrook Technical Specification requirement related to D.C. battery sources goes beyond the regulatory requirements. Although this item reveals a plant-specific issue, it may be representative of a more generic concern. Therefore, it is recommended that the incorporation of requirements that go beyond the regulatory bases into plant-specific Technical Specifications be evaluated further.

SUMMARY OF ASSESSMENT

Category: A   Item(s): TS 3.9.4

Seabrook Technical Specification 3.9.4, containment building penetrations, requires all containment building penetrations to meet a specified status during core alteration activities such as refueling. These requirements ensure that a release of radioactive material within containment will be restricted from leakage to the environment. This item was selected for review because it is representative of the Technical Specifications governing refueling operations.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in both 10 CFR 50, Appendix A, and 10 CFR 100. It has safety relevance and the provisions appear commensurate with a postulated radioactive material release, i.e., a fuel element rupture with the containment is at atmospheric pressure. Inherent flexibility in both the limiting condition for operation and surveillance requirements exists, since options are provided for complying with the stated operability criteria. Additionally, the action statement is consistent with the safety intent by requiring only a suspension of core alterations or the movement of irradiated fuel in the containment building, which represent the only applicable ongoing activities which relate to the postulated fuel element rupture event.

While there exists a certain prescriptiveness in the Technical Specification, such language appear to be necessary to convey the intended technical details. Therefore, the enhancement flexibility potential for this item is considered low, particularly since the action statement is logical and not onerous. Further review of this area for enhanced flexibility is likely to be unproductive.

Overall, this item, even though limited in applicability to general refueling operations, appears to be technically sound and well directed to its safety intent, while at the same time allowing the licensee some flexibility of compliance activities. A direct correlation exists between the wording of this item and the language of the corresponding section of the Standard Technical Specifications. No additional review of this Technical Specification appears warranted.

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 3.12.2

Seabrook Technical Specification 3.12.2, land use census, requires that a land use census be conducted and identify within a distance of five miles in each of the meteorological sectors the location of the nearest milk animal, the nearest residence, and the nearest garden greater than 500 square feet producing broad-leaf vegetation. This item was chosen because it was representative of requirements contained in the radiological environmental monitoring section of the Technical Specifications

The legal requirement for this Technical Specification is contained in 10 CFR 50, Appendix I, and the regulatory bases for the implementation of Appendix I is contained in Regulatory Guide 1.109.

This requirement is relevant to safety in that it is necessary to protect the health and safety of the public. Maintaining doses as low as reasonably achievable is consistent with that philosophy. The land use census provides the information needed to identify a location that yields an exposure to the public from routine releases of plant radioactive effluent that is greater than at a location from which samples are currently being obtained.

This requirement has a great deal of inherent flexibility with regard to how and when this census is taken. Only the requirements that the survey be conducted at least once per 12 months during the growing season and the time limitations on incorporating new locations into the radiological monitoring program are prescriptive.

The one area where reduction might be possible is related to the frequency of the land use census, however, this would be dependent on the significance of the regulatory burden and if data were available to support a reduction in this requirement. It is noted that this item has been removed from the Improved Standard Technical Specifications and placed under the administrative control of the licensee. Therefore, this change could be considered by the licensee.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would probably be unproductive.

SUMMARY OF ASSESSMENT

Category: A Item(s): TS 5.6.3

Seabrook Technical Specification 5.6.3, spent fuel storage pool capacity, states that the spent fuel storage capacity is designed and shall be maintained with a capacity limited to no more than 1,236 fuel assemblies. This item was chosen because it is representative of a design feature Technical Specification.

There is no specific legal requirement for this item. The regulatory bases for this requirement is identified in SRP Section 9.1.2, Subsection III.1 which states that the minimum storage capacity in the spent fuel storage pool shall be in accordance with ANS 57.2 Paragraph 5.1.15 (equal to or exceed one full core discharge plus the maximum normal fuel discharge for a single unit facility). This requirement is important to safety in that General Design Criteria 17 states that the system shall be designed with the capability to permit periodic inspection and testing of components important to safety. Therefore, it is necessary to have the capability to off-load the core.

Although there is no flexibility in the spent fuel storage capacity, this limit can be changed by a Technical Specification amendment based on design considerations, e.g., criticality, rack size and heat load limitations.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would be unproductive.

SUMMARY OF ASSESSMENT

Category: B   Item(s): All

The Seabrook license contains four items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. They deal with sale and leaseback transactions, financial protection, marketing of energy from the plant and the effective and expiration dates of the license. Specifically, the Category B items are as follows:

OL 2.B.7	OL 2.H
OL 2.I	OL 2.J

The financial protection license condition is based on Section 170 of the Atomic Energy Act and 10 CFR 140. The effective and expiration dates license condition is required Section 103 of the Atomic Energy Act and 10 CFR 50.51. The other two license conditions, the sale and leaseback transaction and marketing of energy license conditions, are not regulatory requirements, but are authorized by 10 CFR 50.50 which provides that the license may contain such conditions as the Commission deems appropriate. Given Seabrook's unique financial and ownership situation, these conditions do not appear to be inappropriate.

None of the items are directly related to safety. Although the license conditions are prescriptive, they do not appear to be unduly restrictive. None of the items appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the non-technical license conditions are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: C Item(s): OL 2.E

Seabrook License Condition 2.E, physical security condition, requires the licensee to implement and maintain in effect all provisions of its approved physical security, guard training and qualification, and safeguards contingency plans, and all amendments and revisions to the plans made pursuant to 10 CFR 50.90 and 10 CFR 50.54(p). This item was chosen because it allows the plans to go beyond the requirements specified by the regulations, and, thereby provides opportunity for ratcheting. It also elevates the baseline from which changes can be made without prior NRC approval to that higher level. In addition, it is similar to a number of other plans, such as the emergency response plan, quality assurance plan and environmental protection plan, which are required by the regulations or the license.

The physical security plans are required by 10 CFR 50.34 and 10 CFR 73. Changes to the plans which do not decrease their safeguards effectiveness may be made without prior NRC approval in accordance with 10 CFR 50.54(p). Changes to the plans which decrease their safeguards effectiveness must receive prior NRC approval in accordance with 10 CFR 50.90. The plans are safety relevant in that they assure protection of the plant against radiological sabotage and the potential resulting release of radioactive materials.

The regulatory process provides flexibility in developing and revising the plans. However, additional flexibility could be provided in the implementation of the plans. For example, the generally assumed compensatory measure for loss of a plant perimeter alarm system is the immediate placement of guards within line of sight of each other around the perimeter. The placement of the guards around the perimeter could call unnecessary attention to the fact that the perimeter alarm system is not operable and, therefore, may not be in the best interest of overall plant security. No allowed outage times are permitted as they are for safety-related equipment in the Technical Specifications. Given the likelihood of a threat during relatively short periods of inoperability of the perimeter alarm system and the effectiveness of other security barriers, e.g., access to the plant buildings and vital areas, it seems that Technical-Specification-type allowed outage times would provide additional flexibility without reducing the overall safeguards effectiveness.

Based on the above considerations, it is concluded that the requirement is appropriate; however, it may be unduly restrictive. Therefore, it is recommended that further consideration be given to standardizing the change processes for these and similar plans and providing additional flexibility in their implementation, e.g., by providing Technical-Specification-type allowed outage times.

SUMMARY OF ASSESSMENT

Category: C Item(s): TS 3.4.10

Seabrook Technical Specification 3.4.10, structural integrity, requires that the structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with the inservice inspection (ISI) and inservice testing (IST) programs for the plant in accordance with the ASME Boiler and Pressure Vessel Code, Section XI. This item ensures that the structural integrity and operational readiness of the piping and pressure boundary components governed by the ASME Code are maintained at an acceptable level throughout the life of the plant. This item was selected because of its reliance on other documents (e.g., the ASME Code) for technical requirements. In addition to the requirements contained in Technical Specification 4.0.5, this Technical Specification contains specific surveillance provisions for the reactor coolant pump flywheel which reference Regulatory Guide 1.14 (Revision 1) related to flywheel in-service inspection.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.55a and 10 CFR 50, Appendix A. While Regulatory Guide 1.14 is not a legal requirement, it also has basis in 10 CFR 50, Appendix A, and the guidance that is referenced in Technical Specification 3.4.10 appears to be consistent with other ISI program requirements. Since the reactor coolant pump flywheel is not a pressure boundary component, this regulatory guidance provides technical details unavailable in the ASME Code.

This item has safety relevance and appropriately uses a graded approach to the action requirements, dependent upon the ASME Code Class of the affected component. Reliance upon a regulatory guide to provide the reactor coolant pump flywheel inspection details also is appropriate, given the missile impact hazard and the lack of other standard technical criteria. Inherent flexibility does exist, since this item refers to Technical Specification 4.0.5 which allows relief from the pertinent code requirements, if granted by the NRC, in accordance with 10 CFR 50.55a(f)(6)(i). Such relief requests are generally used to exempt code requirements that are impractical to a specific plant design or configuration. While the overall ISI/IST programs, which are submitted to the NRC for review and safety evaluation, may represent surrogate items to the intended goal (i.e., acceptable structural integrity of the pressure boundaries and associated components), the use of these surrogate items appears both technically sound and appropriate from a regulatory standpoint.

While prescriptive language is used in this Technical Specification and its referenced documents, i.e., the ASME Code and Regulatory Guide 1.14, such details are needed to provide the appropriate technical criteria. Performance-based criteria are already incorporated into the ASME Code, Section XI, requirements upon which the plant ISI/IST programs are based. Any attempt to use additional performance-based criteria, beyond the ASME Code provisions, would unnecessarily complicate this Technical Specification.

Further NRC review of this area for enhanced flexibility does not appear warranted from a regulatory standpoint. However, from a research and technical standpoint, continued NRC liaison with the ASME Code Section XI committees will continue to provide for progressive revisions and additional flexibility, if appropriate. It is noted that with Generic Letter 91-18, further flexibility in the form of NRC Inspection Manual Technical Guidance was provided in this area by allowing continued operation with nonconforming piping/support components until the next refueling outage, if certain referenced analytical criteria (e.g., Appendix F of Section III of the ASME Code, NRC Bulletins 79-02 and 79-14) are met. Given that such guidance for continued operation can be supported by quantitative analysis, this Technical Specification currently establishes reasonable and acceptable controls. While no further review of this item is warranted, the use of a Generic Letter 91-18 to add flexibility to this area appears to have been beneficial and could be explored further in other areas.

SUMMARY OF ASSESSMENT

Category: C   Item(s): TS 6.2.2.e

Seabrook Technical Specification 6.2.2.e, station staff working hours, requires that the licensee develop and implement administrative procedures which limit the working hours of station staff who perform safety related functions. The Technical Specification further requires that the amount of overtime worked by such personnel "... be limited in accordance with the NRC Policy Statement on Working Hours." This item was chosen because it is an example of a Commission policy statement which has become a defacto requirement by its incorporation by reference in the plant's Technical Specifications.

The Commission's original "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" was issued on February 18, 1982 (47 FR 7352) and was forwarded to applicants and licensees by Generic Letter 82-02, "Nuclear Power Plant Staff Working Hours." The policy statement itself contains a request for applicants and licensees to include in their Technical Specifications administrative procedures regarding working hour restrictions which conform to those in the policy statement. The policy statement was revised slightly on June 1, 1982 (47 FR 23836) and was forwarded to applicants and licensees by Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours." The requirement is also contained in the Improved Standard Technical Specifications.

The requirement is relevant to safety in that personnel working in a fatigued condition could have reduced mental alertness or decision-making ability. It is noted that limiting working hours is used as a surrogate for limiting fatigue. Other surrogates have been considered, but have been rejected.

The requirement has a great deal of inherent flexibility. Although there is no flexibility in the requirement for the license to have an administrative procedure, the policy statement and, hence, the Technical Specification, provides essentially no limit on the amount of overtime an individual can work. It only specifies that the overtime be given deliberate consideration and authorized in writing.

The Office of Nuclear Reactor Regulation is considering this issue for possible rulemaking and, to that end, has requested the Office of Nuclear Regulatory Research to proceed with the development of a rulemaking package.

Based on the above considerations, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive. Although the Commission clearly intended that this policy statement become a defacto requirement by its incorporation in plants' Technical Specifications, such is not the case for policy statements in general. Therefore, it is recommended that the elevation of non-requirements, such as policy statements, into requirements and the regulatory status of policy statements in general be pursued further.

SUMMARY OF ASSESSMENT

Category: D Item(s): TS 6.2.2.a

Seabrook Technical Specification 6.2.2.a, minimum shift crew composition, specifies the minimum on-duty shift crew size and composition for the various operational modes. This item was chosen because it not only repeats the minimum licensed operator shift staffing requirements of 10 CFR 50.54(m), but also adds minimum shift staffing requirements for auxiliary operators and the shift technical advisor.

10 CFR 50.54(m) specifies minimum licensed operator shift staffing requirements for the various operational modes. The Technical Specification is consistent with that regulation for licensed operators. The NRC has no minimum shift staffing requirements for auxiliary operators or the shift technical advisor. The shift technical advisor is the embodiment of the Commission's policy statement on engineering expertise on shift. The policy statement, not a legal requirement, provides that engineering expertise on shift may be provided by either a dedicated shift technical advisor or by a senior reactor operator serving in a dual role. Technical Specification 6.2.2.a also provides that flexibility. In summary, the Technical Specification repeats an existing legal requirement, and elevates a policy statement and non-requirement to defacto legal requirements. This requirement is also contained in the Improved Standard Technical Specifications.

The requirement is relevant to safety in that it prescribes the minimum shift staffing requirements for the plant. It is noted that the shift technical advisor is a surrogate for engineering expertise on shift.

The requirement, although prescriptive, offers inherent flexibility in that it only prescribes the minimum staffing requirements. The licensee is free to exceed these minimum requirements and, in practice, usually does. However, the Technical Specification appears to have little if any potential for enhanced flexibility.

The Office of Nuclear Reactor Regulation is reevaluating the Commission's policy statement on engineering expertise on shift, including the need for and use of shift technical advisors, and the broader issue of minimum shift staffing requirements.

Based on the above considerations, it is concluded that further consideration of this requirement for possible in regulatory burden or enhanced flexibility would prove unproductive. However, it is recommended that the elevation of non-requirements, such as policy statements, to the status of requirements and the regulatory status of policy statements in general be considered further.

SUMMARY OF ASSESSMENT

Category: D Item(s): TS 6.2.3.2

Seabrook Technical Specification 6.2.3.2, Independent Safety Engineering Group (ISEG) composition, states that the ISEG shall be composed of at least five, dedicated, full-time engineers located on site with a science or engineering degree and at least two years experience in the degreed field with one year of experience in the nuclear field. This item was chosen because of its very prescriptive with regard to manpower requirements.

This requirement is based on TMI Action Plan Item I.B.1.2 contained in NUREG-0737. This particular item was required of applicants for operating licenses only. The purpose of ISEG is to perform independent reviews and audits of plant activities and review other appropriate internal and external information available and to provide recommendations to management where useful improvements can be made. Other than the scope of issues that ISEG reviews the licensee has no control over the utilization of the five dedicated plant staff assigned to this function. In the Improved Standard Technical Specifications, ISEG has been replaced with an independent review and audit function that provides a relaxation of this requirement. This independent review and audit function permits more flexible methods of performing the ISEG function (i.e., by a standing or ad hoc committee, or assigning individuals capable of conducting these reviews and audits).

A survey performed on a limited number of plants licensed after TMI determined that some licensees have already requested and received license amendments that incorporate the provisions of the Improved Standard Technical Specifications into their Technical Specifications. Some of the older plants' Technical Specifications were also surveyed and it was determined that the Technical Specifications also have a review and audit function. In addition at least some of the older plants have adopted the ISEG approach and some have adopted the Improved Standard Technical Specification approach. The remaining older plants surveyed have incorporated variations of these approaches. It is not clear at this time why the older plants surveyed all have incorporated an independent review and audit function into their Technical Specifications since it was not required by NUREG-0737. However, except for perhaps a few isolated cases, it appears that it was included on a voluntary bases.

Based on the above considerations, it is concluded that the Seabrook Technical Specification requirement related to the composition of ISEG provides no flexibility. However, a Technical Specification change can be submitted adopting the Improved Standard Technical Specification approach; that would provide considerable flexibility in the implementation of this requirement.

Based on the viable alternative available, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would be unproductive.

SUMMARY OF ASSESSMENT

Category: D Item(s): TS 6.4.1.7

Seabrook Technical Specification 6.4.1.7, Station Operation Review Committee (SORC), requires the SORC to make specific written recommendations to the Station Manager, render written determinations whether certain items constitute unreviewed safety questions, and provide written notification of disagreements between SORC and the Station Manager. This administrative control implements a continuing monitoring activity which is considered to be an integral part of the routine supervisory function. This item was selected as a representative review activity of a committee required by the Technical Specifications.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.40(b) as it relates to the licensee being technically qualified to engage in licensed activities. The guidance provided by ANSI Standard N18.7 (ANS 3.2), as endorsed by Regulatory Guide 1.33, conveys additional regulatory criteria for the required review activities of an onsite operating organization. While the SORC monitoring activities have safety relevance in providing a timely oversight of routine and revised plant operations, the details of exactly what SORC is responsible to review, document and report in writing, have little basis in the regulation and relate more specifically to Standard Review Plan (NUREG-0800) provisions. The language in this item resembles the wording of the applicable section of the Improved Standard Technical Specifications.

While a certain degree of inherent flexibility exists for the implementation aspects of this item (e.g., telephone meetings, agenda, etc.), there is no inherent flexibility in what this Technical Specification requires SORC to accomplish (e.g., recommend approval or disapproval of changes to any procedures required by the Technical Specifications; reference Technical Specification 6.7). This prescriptiveness does not appear to be either consistent or commensurate with the intended safety impact, because not all of the referenced procedures carry the same safety significance. While the use of SORC subcommittees can add some additional flexibility in workload allocation, a rigid interpretation of many of the SORC requirements, e.g., recommend in writing approval or disapproval of "all proposed changes or modifications to station systems or equipment that affect nuclear safety" (emphasis added) appears onerous given the various levels of safety significance which are inherent in nuclear power plant system and component designs.

It should be noted that the SORC has only advisory authority in that it recommends and renders determinations; the Station Manager has the responsibility for the resolution of any disagreements on overall station operation. Thus, the language in this item to convey the administrative control of SORC requirements appears to be overly prescriptive and could be flexibly enhanced by the use of performance-based criteria or a graded approach to safety-significant review activities. Use of risk assessment methodology could provide valuable input into the prioritization of SORC efforts and the determination of where limited review time could be most effectively directed.

This Technical Specification is prescriptive, yet broadly scoped such that interpretation is required to define implementation details. Such a reliance on interpretation can lead to misapplication of this license condition in the inspection and enforcement arena. While the safety intent of the SORC as an overview and advisory authority is soundly based, achieving enhanced flexibility in the administrative control of the SORC functions would be a worthwhile initiative. The Improved Standard Technical Specifications while reducing the overall SORC review responsibilities, do not significantly alter the plant review function directed by this item. It is recommended that further review of this item beyond what is already in progress in the NSAC-125/10 CFR 50.59 area be conducted to evaluate not only the need for the current prescriptive language of Technical Specification 6.4.1.7, but also the prospects for enhanced flexibility by supporting more of a graded safety approach to the SORC review/recommendation functions.

SUMMARY OF ASSESSMENT

Category: D   Item(s): TS 6.7.3

Seabrook Technical Specification 6.7.3, temporary changes of procedures, allows temporary changes to the procedures required by other Technical Specifications if the change is accomplished in accordance with specified provisions. These provisions include the requirements that the "intent" of the original procedure not be altered and other approval conditions. This item was selected as a representative administrative control governing plant procedures and programs.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50.40(b) as it relates to the contribution of the administrative procedures to the technical qualification of the licensee; and also to 10 CFR 50.54(l) which requires that designated individuals be responsible for directing the licensed activities of plant operators. By reference, an association with Regulatory Guide 1.33 and the endorsed ANSI Standard N18.7 (ANS-3.2) also exists. Additionally, 10 CFR 50, Appendix B, delineates general quality assurance criteria for procedures and in conjunction with Regulatory Guide 1.33, provides regulatory measures governing safety-related procedural controls. The safety relevance of this item is clearly established by the above regulatory references and by the need for procedure changes to properly reflect the appropriate safety-related requirements.

Some inherent flexibility can be found in this item both in the plant management staff options for review and in the judgement allowed for the determination of whether an original procedure intent has been altered. However, once a temporary procedure change is determined to be appropriate, this Technical Specification is generally prescriptive as to the controls that are required prior to and after implementation. While the prescriptive language in this item may not be necessary, in that other review and approval processes could provide equivalent temporary procedure change controls, the existing requirements appear to not only incorporate standard industry guidelines, but also represent a sound practice which is not particularly burdensome.

One area where enhanced flexibility might be beneficial for this item is the possible reduction of the total number of procedures for which the full review and approval conditions must be applied. Since not all safety-related and Technical Specification required procedures carry the same safety significance, a "non-intent" temporary change to a procedure governing activities of lesser safety relevance may not need the full review dictated for temporary changes of greater impact. Performance-based criteria could be used to distinguish the safety significance of different levels of procedural controls. In turn, a graded approach to the review and approval process for procedural changes could thus be applied. However, development of such a hierarchical process of controls may not be worth the effort, especially if the simplicity and conservatism in the existing Technical Specifications provisions are not considered onerous by the licensee.

Overall, this item has a sound regulatory basis and is coherent in the application of a logical review process to the procedural controls of safety-related activities. While little inherent flexibility exists, initiatives to enhance flexibility may over-complicate the practice and not provide any tangible benefits. Also, since temporary procedure changes represent a contingency option to the formal procedure revision process, the need for additional flexibility may be neither great, nor practical. No further NRC review of the Technical Specification is recommended. However, the use of a graded approach to procedure safety significance as discussed in the Summary Assessment for Technical Specification 6.4.1.7 would likewise provide implementation flexibility in the controls of temporary procedural changes.

SUMMARY OF ASSESSMENT

Category: D Item(s): EP 3.1

Seabrook Environmental Protection Plan Section 3.1, changes in design and operation, specifies that before engaging in additional construction or operational requirements that may significantly affect the environment, the licensee shall prepare an environmental evaluation of such activity to determine if the activity involves an unreviewed environmental question. Section 3.1 also requires the licensee to provide a written evaluation of any activity that involves an unreviewed environmental question, and to obtain NRC approval and maintain records of the changes associated with these activities. This item was selected because it an example of an administrative control.

The legal bases for this requirement is contained in 10 CFR 50.36b which requires that conditions to protect the environment should be incorporated into an attachment to the license that is made a part of the license. The requirement provides protection to the health and safety of the public by assuring that changes to the plant design or operation that could significantly affect the environment are evaluated prior to implementation. This requirement provides limited flexibility for items that do not constitute an unreviewed environmental question.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: E Item(s): OL 2.G

Seabrook Operating License Condition 2.G, violation reporting condition, states that the licensee shall report any violations of the requirements contained in Section 2.C of the license initially via the Emergency Notification System and with written follow-up within 30 days in accordance with procedures described in 10 CFR Part 50.73(b). This item was chosen because it is representative of a license condition that contains reporting requirements.

There does not appear to be a legal requirement or a regulatory bases for this license condition. This reporting requirement was put in the operating license to provide assurance that the licensee was fulfilling all of its commitments identified under Section C of the license.

This reporting requirement does not have a great deal of flexibility and is judged to have little potential for any increased flexibility. There is, however, one aspect of this license condition that some licensees may be misinterpreting that results in increased reporting requirements. Section 2.G of the license, as currently written, does not clearly define the licensees responsibilities for reporting violations of the Technical Specifications identified in Section 2.C(2) of the license. The wording in Section 2.G can be interpreted as requiring additional reporting requirements beyond those specified within the Technical Specifications. The wording in Appendix A to the license specifically states that violations of the Technical Specifications will be reported in accordance with the requirements of 10 CFR 50.72 and 50.73. It, therefore, appears it was not the intent of the operating license to require reports that go beyond these requirements. In addition Section 2.G of some of the newer licenses specifically exclude Technical Specifications (Section 2.C(2) of the license) from the reporting requirements of Section 2.G.

A license amendment specifically excluding Section 2.C(2) of the license from this reporting requirement would eliminate any possible misinterpretation of the Technical Specification reporting requirement contained in Section 2.G. It is concluded that beyond a plant-specific license amendment, consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: E Item(s): TS 3.3.3.4

Seabrook Technical Specification 3.3.3.4, meteorological instrumentation, requires that the specified meteorological monitoring instrumentation be operable at all times. This requirement ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This item was selected as a representative Technical Specification where the only action is a reporting requirement.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in both 10 CFR 100.10(c)(2) and 10 CFR 50.36a(a)2. The detailed requirements provide a capability to evaluate the need for initiating protective measures under certain plant conditions to protect the health and safety of the public and are consistent with the recommendations of Regulatory Guide 1.23. The support to radiological dose assessment capabilities provided by the details of this Technical Specification is therefore also connected to 10 CFR 50, Appendix E, and 10 CFR 20.

The safety relevance of this item is clearly established by the significance that correct and timely meteorological information has in proper dose assessments and emergency planning decisions. However, the consistency and safety significance of the action requirement of this Technical Specification is not readily evident. Given the inoperability of certain meteorological monitoring instrumentation, the action statement requires the licensee to submit a Special Report to the NRC outlining the cause of the malfunction and the plans for restoration. Such a reporting requirement within a ten-day deadline after an allowable outage time of seven days appears to be inconsistent with the fact that, in accordance with the Seabrook Station Emergency Response Manual, an Unusual Event would have to be declared if certain categories of meteorological data (e.g., wind speed) became unavailable.

There is no inherent flexibility in the provision for the aforementioned report submittal when the conditions and timing triggers this requirement. The function of such a Special Report could be questioned, particularly if its purpose is only to encourage the licensee to take prompt corrective action. Such an intent would make the Special Report nothing more than a surrogate for timely restoration of the instrumentation. Given the existence of the Seabrook Station Radiological Emergency Plan, written in compliance with 10 CFR 50.34(b) and 10 CFR 50, Appendix E, and the potential for entrance into an Emergency Action Level (i.e., Unusual Event) upon loss of meteorological data, the need for such reporting appears even less consistent and significant. As discussed from a regulatory basis, the meteorological instrumentation has safety relevance. However, a more meaningful action, upon loss of some monitoring capability, would be an evaluation of the inoperable equipment in the context of any diminished capacity of the overall Emergency Response Plan.

This item is prescriptively worded and similar to the language in the Standard Technical Specifications. It is noted that the Improved Standard Technical Specifications do not include meteorological monitoring instrumentation. Therefore, enhanced flexibility could be provided by either eliminating the item or directing an action more consistent with the unique Seabrook Radiological Emergency Plan. This item also warrants further review to determine the function and utility of the special report currently directed by this Technical Specification action. It is recommended that this item, along with any other Technical Specifications which require reports as the only actions (see also Summary of Assessment for Technical Specification 3.3.3.3), be evaluated further for appropriateness and/or improved coordination with existing plant programs that already address corrective response measures.

SUMMARY OF ASSESSMENT

Category: E   Item(s): TS 6.4.1.8

Seabrook Technical Specification 6.4.1.8, Station Operation Review Committee (SORC) records, specifies the recordkeeping requirements of the committee. It requires that the SORC maintain written minutes of each meeting that document the results of all Technical Specification required SORC activities, and that the SORC provide copies of the minutes to the Executive Director Nuclear Production and the Nuclear Safety Audit Review Committee. This item was chosen because it is representative of a number of Technical Specification administrative controls. This item was chosen because it is representative of a number of the Technical Specification reporting requirements.

The stated regulatory requirement for this item is 10 CFR 50.40(b), which requires that the licensee be technically qualified to engage in the licenses activities.

The requirement is relevant to safety in that it ensures that the offsite review committee and the corporate-level individuals responsible for the safe operation of the plant are kept informed of the Technical Specification required activities of the SORC.

The requirement provides no inherent flexibility to the licensee; it prescribes minimum requirements for content and distribution of the report. That prescriptiveness does not appear to be inappropriate. In view of its nature and safety significance, there appears to be no enhanced flexibility potential for this requirement.

Based on the above considerations, it is concluded that the Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: E Item(s): TS 6.8.1.5

Seabrook Technical Specification 6.8.1.5, monthly operating reports, requires the licensees to submit routine reports of operating statistics and shutdown experience to the NRC on a monthly basis. The guidance for submitting these reports is contained in Regulatory Guide 1.16. This Technical Specification was chosen because it is an example of a reporting requirement that is inflexible and whose safety significance is questionable. In addition it appears that this is an example of NRC staff guidance that has been made a legal requirement.

Although the regulatory bases for this Technical Specification are contained in Regulatory Guide 1.16, there does not appear to be any direct regulatory requirement. The staff provides these reports to other agencies, e.g., Environmental Protection Agency, Department of the Interior, National Institute of Standards and Technology, pursuant to memoranda of understanding. In addition some of the data in these reports is used by AEOD to evaluate performance indicators, e.g., critical hours, and also by users outside the NRC, e.g., public utility commissions, intervenors, consultants. The information from these reports is also used in the preparation of NUREG-0020 (Gray Book) which may be is used by the industry to track the performance of other licensees.

This requirement provides no flexibility with regard to either reporting or the frequency of reporting. Since the usefulness of the information contained in these reports has not been determined, it is difficult to assess the merits of requiring the licensees continuing to provide these reports on a monthly basis, if it could be provided less frequently, or if it could be totally eliminated. The determination of the usefulness of the information provided should include an assessment of its need by other agencies, by the industry, public interest groups and the general public, in addition to the NRC.

The task force formed to evaluate reporting requirements for power reactors is also evaluating the need for this requirement.

Based on the above considerations, it is concluded that although this and other specific reporting requirement are currently being evaluated, a broader approach which determines all the information needed by the NRC to accomplish its safety mission may be appropriate and result in a possible reduction of regulatory burden.

SUMMARY OF ASSESSMENT

Category: F Item(s): All

The Seabrook operating license contains ten items in Category F, "Unique Plant Features." These items were deemed appropriate to be assessed collectively. They identify the plant and its location, and delineate the plant's major design features. Specifically, the Category F items are as follows:

OL 2.A	TS 5.1.1
TS 5.1.2	TS 5.1.3
TS 5.2.1	TS 5.2.2
TS 5.3.1	TS 5.3.2
TS 5.4.2	TS 5.5.1

These items are basically statements of facts. They generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items are directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the unique plant features items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF ASSESSMENT

Category: G   Item(s): All

The Seabrook operating license contains 62 items in Category G, "Other." These items were deemed appropriate be assessed collectively. They include legal provisions, including exemptions; definitions and statements of fact. Specifically, the Category G items are as follows:

OL 1.A	OL 1.B	OL 1.C	OL 1.D
OL 1.E	OL 1.F	OL 1.G	OL 1.H
OL 1.I	OL 2.B.1	OL 2.B.2	OL 2.D
TS 1.1	TS 1.2	TS 1.3	TS 1.4
TS 1.5	TS 1.6	TS 1.7	TS 1.8
TS 1.9	TS 1.10	TS 1.11	TS 1.12
TS 1.13	TS 1.14	TS 1.15	TS 1.16
TS 1.17.a	TS 1.17.b	TS 1.17.c	TS 1.18
TS 1.19	TS 1.20	TS 1.21	TS 1.22
TS 1.23	TS 1.24	TS 1.25	TS 1.26
TS 1.27	TS 1.28	TS 1.29	TS 1.30
TS 1.31.a	TS 1.31.b	TS 1.31.c	TS 1.32
TS 1.33	TS 1.34	TS 1.35	TS 1.36
TS 1.37.a	TS 1.37.b	TS 1.38	TS 1.39
TS 1.40	TS 1.41	TS 1.42	TS 1.43
EP 1.0	EP 4.2.2	EP 4.2.3	

These items are basically statements of facts. They generally appear to be required by the Atomic Energy Act or the Commission's regulations. None of the items are directly related to safety. Although the items are prescriptive, they do not appear to be unduly restrictive. None of the items appear to have enhanced flexibility potential.

Based on the above considerations, it is concluded that the unique plant features items are appropriate and not unduly restrictive. In addition, it is concluded that further consideration of these items for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SECTION 3

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SECTION 3

Use of PRA/PSA Information in Performance-Based Regulation

In 1975, the U.S. Nuclear Regulatory Commission completed the first quantitative study of the probabilities and consequences of severe reactor accidents in commercial nuclear power plants--the Reactor Safety Study, published as WASH-1400. This work for the first time used the techniques of probabilistic risk analysis (PRA) for the study of severe core damage accidents in two commercial nuclear power reactors. The product of probability and consequence - a measure of the risk associated with severe accidents, was estimated to be low, relative to other man-made and naturally occurring risks for the two plants analyzed.

Following the completion of WASH-1400, and similar efforts conducted in parallel in other countries (most notably, Phase A of the German Risk Study), research efforts were initiated in several countries to develop advanced methods for assessing accident frequencies, improved means for collecting and analyzing operational plant data were put in place, methods were initiated to improve the ability to quantify the effects of human errors, and studies to better predict the nature and effect of common cause failures were begun. Further, limited research was begun on those key severe accident physical processes identified in the Reactor Safety Study.

The 1979 accident at Three Mile Island substantially changed the character of the analysis of severe accidents world-wide. Based, at least in part, on the comments and recommendations of the major investigations of that accident, a substantial research program on severe accident phenomenology was planned and initiated with international sponsorship. This program has been the subject of many reviews and comments, and included both experimental and analytical studies. It was also recommended in the various Three Mile Island investigation reports that probabilistic risk analysis techniques be used to complement the traditional non-probabilistic methods of analyzing nuclear plant safety.

A large number of nuclear power plants have been or are being analyzed using probabilistic techniques throughout the world. Individual plant examinations (IPEs) are being or have been performed on all U.S. plants. At the present time, most nuclear power plants have been or are being analyzed, at least to determine the frequency of severe accidents, and important insights are being gained relative to the actions that might be taken to improve plant safety.

In 1984, a study was performed by the U. S. Nuclear Regulatory Commission to evaluate the state-of-the-art in risk analysis techniques, and a summary of PRA perspectives was published (NUREG-1050, Probabilistic Risk Assessment (PRA) Reference Document). Before commenting on the proper usage of PSA analyses at present, we shall revisit the general conclusions of that document relative to the current state of the art. This is done briefly below, followed by a discussion of possible uses of the results, recognizing both the strengths and weaknesses in the technology at present.

In the area of systems modeling, much of the basic methodology remains unchanged from that of the Reactor Safety Study. However, there is a wealth of experience in applying these methods, and improved computer codes now permit the efficient handling of the more complex models required to analyze the effects of fires and external events such as earthquakes. Much, if not all, of the analysis of internal events can now be performed on personal computers, substantially reducing the cost and improving the efficiency of studies performed today. Techniques are available to calculate importance measures of plant systems and components from a variety of viewpoints, in a form amenable for use in determining the relative importance of systems and components to plant safety. The decision of the detail to which systems are modeled, however, is generally left to the judgement of the analyst, usually based on a perception of what may be important relative to other components or subsystems. Little guidance is available in the literature in this regard. Thus, before the results can be used in a regulatory application, the boundary conditions and assumptions used in the analysis must be examined to ensure they are appropriate to the specific usage envisioned.

Considerable data have been acquired on initiating event frequencies and component reliability, although this may vary somewhat from plant to plant. Thus, while a comprehensive plant-specific analysis is within the current capabilities, it sometimes is not performed, because of the costs and resource allocations required. Thus, before a current analysis is relied upon to support plant-specific regulatory initiatives, the degree to which the PSA analysis is also plant-specific must be ascertained.

Detailed methods have been developed for evaluating the signifi... of dependent failures, which address not only the quantitative aspec... f the analysis, but, more importantly, the qualitative knowledge gained which can help prevent their occurrence. At the present time, we are limited more by the lack of readily accessible root cause data on dependent failures from operating and maintenance logs, rather than the methods for analyzing the data. (The raw data is generally available to the plant owner/operator, but in many cases it may is not in readily useable form to the PRA analyst or to the regulator.) Guidance on acceptable ways of analyzing the raw data for dependent failures have been developed jointly by EPRI and NRC. Methods for evaluating the reliability of solid-state control and protection devices are not yet available for routine application, particularly with respect to the adequacy of the software associated with the solid-state device. Some information is available from the aerospace and defense industries in this regard and this, when coupled with research efforts currently underway should do much to improve the situation. However, at the present time, quantitative results when software driven solid state devices are analyzed should be viewed with considerable caution.

In the area of human interactions, improved methods are available and additional data has been acquired that permits a more detailed analysis of the likelihood of failing to follow procedures for a number of situations. The state of the art is still relatively weak in the ability to address cognitive and comprehension errors, or to consider the pervasive effect of a poor safety attitude at a plant. Substantial work is underway in these areas in many countries, and some improvements are expected in the future. However, at the present time, the use of PSA information in a regulatory framework will be enhanced if such application avoids the direct use of absolute human error probabilities. Even when human errors are treated in a relative manner, however, care must be taken to ensure that dependencies and boundary condition changes are properly considered.

As identified above, a detailed and comprehensive research program is well underway in the areas of severe accident progression, containment response, and radionuclide transport. This effort has recently been re-evaluated to ensure it is directed to those elements necessary to reach regulatory closure on severe accident issues. The most recent assessment of the uncertainties in these portions of the analyses was contained in the USNRC-sponsored NUREG-1150, Severe Accident Risks, An Assessment for Five U.S. Nuclear Power Plants. Although much has been learned in the current research programs, the uncertainties associated with the ability to predict accident progression in detail, or to estimate the magnitudes of the releases of the various radionuclides to the environment are very large. While, in general, the central estimates (means, medians) of the distributions associated with these releases are lower in magnitude than those predicted in earlier studies such as WASH-1400, the uncertainty range remains large and will remain so even after current research is completed. In the area of consequence analysis, models have been substantially improved, and many sensitivity analyses are now available. However, comprehensive uncertainty analyses of the models are only now being performed. Some modification of the models currently available may be desirable, reflecting the new information regarding the biological effects of ionizing radiation now becoming available. Thus, to the extent possible, the use of probabilistic information in developing performance-based criteria may be more appropriate and robust when applied to the potential for severe core damage or to system availability under given conditions, rather than public risk. The inherent uncertainties in assessments of individual or societal risk make analyses of such parameters more amenable to comparisons with goals, rather than determination of compliance with criteria.

The ability to analyze the effect of fires, floods, and other external events has improved substantially. Major limitations still exist relative to the ability to estimate recurrence frequency for very rare catastrophic events (such as great earthquakes) and it does not appear that the uncertainties associated with such estimations will be narrowed substantially in the near future. Similarly, some of the subtle effects associated with certain other external events will require considerable more study before they can be quantified without considerable uncertainty (e.g., effects of smoke and soot during fires). These factors will limit the use of probabilistic-type approaches in these areas of regulation unless full consideration is given to the impact of the large uncertainties involved on the regulatory decision-making process.

The ability to perform comprehensive uncertainty analyses, including the development of structured approaches for incorporating the results of expert elicitation, has improved greatly. The most detailed study of this type is included in NUREG-1500. However, that method is extremely resource intensive and time consuming. Improved, more efficient methods are needed before such analyses can be performed on a routine basis for use in regulatory decisions.

As a word of caution, the process of performing a probabilistic study may appear to be deceptively simple. The analysis requires a highly competent and dedicated staff and a detailed knowledge of the plant, particularly the interactions between the various plant systems as well as between the plant and those who operate and maintain it. The analysts must have a clear message to obtain their best estimate, emphasizing neither conservatism nor optimism. They must be careful not to tailor results to meet established limits or goals. A certain discipline is also required from all using the final product to seek the underlying insights regarding plant performance, and to avoid degenerating to unnecessary "number exercises". They must fully appreciate the assumptions and boundary conditions that underlie the analysis, for alterations in these boundary conditions and assumptions can have profound effects on both the quantitative results and the derived insights.

Given these strengths and weaknesses, how can probabilistic results be utilized? A comprehensive discussion appears in Probabilistic Safety Assessment in Nuclear Power Plant Management, edited by N. J. Halloway and sponsored and published by Principal Working Group 5 (Risk Assessment), OECD/NEA. It evaluates the value of PSA as a general engineering analysis for assessing and managing the safety related operations of a nuclear power plant. The report draws the following conclusions:

* The application of PSA provides plant management with a general systems engineering tool which generates insights not readily available from the traditional deterministic safety and licensing analyses. While some of these insights derive from probabilistic evaluation, the majority do not, but simply arise from the systematic yet unprejudiced nature of the PSA procedures. Some of the most important new insights have been derived from the integrated model of plant system behavior and operator actions which PSA can create.

* The existence of a PSA capability within a plant operator's organization provides for a logical framework of regulatory discussion and negotiation to be created. Furthermore, this framework is plant-specific, and can thus be used for plant-specific evaluation and more logical resolution of generic safety issues.

* The benefits derived by plant operators are generally greatest when there is a full commitment to development and maintenance of an internal PSA capability, with minimal dependence on outside experts except for an initial technology transfer phase. Although such commitments are quite expensive, those who have undertaken them are generally of the opinion that the benefits more than compensate.

* The application of PSA to an existing plant has always resulted in the identification of effective ways of achieving plant safety, and has thus contributed to the overall effectiveness of plant operation.

Therefore, the report comes to the unambiguous conclusion that the implementation of PSA as an aid to nuclear power plant safety management is directly beneficial to those implementing it in support of their plant designs or operations, and to all those concerned to ensure nuclear plant safety.

It is in this vein that the U.S. has initiated the Individual Plant Examination (IPE) process, in which each plant is requested to conduct a risk-based search for vulnerabilities utilizing substantial involvement of the utility staff.

Probabilistic analysis techniques also are of interest to the regulator in a variety of ways, and most of the comments addressed to utility use in the report referenced above are applicable in this venue as well. These techniques provide a new perspective that permits an independent consideration of the body of regulatory requirements to ensure that potentially risk significant factors are properly considered and that regulatory resources are not needlessly expended on unimportant matters by either the regulated or the regulator. They can be used to identify those systems, trains, and components that are important to maintaining the likelihood of severe core damage at a low value, and, conversely, can also identify those items that have little influence on the likelihood of an accident. However, this must be done with a clear appreciation for the strengths and weaknesses discussed above, and a detailed understanding of the messages gained from operational data, particularly in those areas where we know the PSA methods are still developmental, such as those areas associated with operator cognitive and comprehension errors. Even here, however, the probabilistic techniques can be used to gain valuable insights through sensitivity and uncertainty analyses, and by examining relative comparisons which recognize the limitations and are performed conditional on the response of the items which are still developmental.

The results of PRA or PSA studies provide information useful in prioritizing the expenditure of resources for plant evaluations and future safety research. This is particularly so when detailed uncertainty analyses are performed, Obviously, such an activity is complex, and cannot rely solely on mathematical manipulations, such as rank regression analyses, but must focus on the underlying knowledge regarding the input data and an appreciation of the degree to which results might be plant specific.

In like manner, the models generated in a probabilistic study are useful in evaluating the significance of both plant-specific and generic issues. They are also useful when developing strategies to react to or manage a severe accident as it occurs. As before, this must be done with an appreciation of the boundary conditions and assumptions used in the original analyses. While items found risk-significant might warrant further analysis or regulatory attention, this will depend on the specifics of the situation, the degree to which existing regulatory instruments are met, and the potential for approaching or exceeding any safety goals which might be established. Similarly, items cannot be dismissed on the basis of low risk until it is clear the analysis is robust in the area of interest and that it adequately supports the decision.

In brief, the strongest insights gained from a probabilistic analysis derive from the integrated and comprehensive examination that analyses of these types entail, the attention devoted to interactions between systems, the operating staff and the plant systems, and the structured examination of operating experience. In general, those insights and importance rankings developed from the analysis of a system, or from analyses of groups of systems to assess the frequency of severe core damage are more robust than those which require an evaluation of overall risk, as discussed above. The weakest insights are those that derive primarily from the quantitative rankings alone, without considering the meaning of the results in an engineering context. While the quantitative results are important, they should be considered as most useful for a screening of the results to identify important accident sequences and plant features, and to give indication of areas with little importance in a probabilistic context.

Probabilistic analysis presents an additional tool, an additional source of information which can be used to focus regulatory decision-making in many areas, identifying features most important to plant safety. Used properly, with recognition of the its limitations and proper attention to the scope, boundary conditions, and assumptions of the analysis, it can be used to exploit the flexibility presently existing within the regulatory environment to improve plant safety while reducing undue regulatory burden. It can also be used to suggest areas where performance-based regulatory practices can be employed in the future. Techniques are now being developed and employed to improve plant configuration control and to optimize the required plant response to equipment outages and mode changes. These may also suggest future modifications to the regulations and the way they are implemented..

ASSESSMENT OF NRC-SPONSORED PROGRAMS

Probabilistic Risk Assessment (PRA) or Probabilistic Safety Assessment (PSA) applications having the potential to significantly reduce regulatory burden or provide ;more flexibility in the regulations and in the implementation of the regulations while maintaining safety are those that primarily address configuration control and quality assurance (QA) issues. Configuration control applications generally involve the utilization of PRA/PSA methods to optimize surveillance test intervals (STIs) and allowed outage times (AOTs). QA applications generally involve the utilization of PRA/PSA to support "graded" QA; that is, optimizing QA for those structures, systems or components that are safety significant based on PRA/PSA insights. Current NRC-sponsored programs were examined to identify those efforts that are utilizing PRA/PSA that could provide potential insights in these areas.

The use of PRA/PSA by the NRC has been both broad and narrow. The broad application is seen in the many various and diverse activities which have increased over time, particularly since the TMI accident. The utilization of PRA/PSA, however, has been narrow in that it has been limited to a small set of applications. These activities have been defined and summarized into several categories (as reported in the NRC PRA Working Group Report) as follows:

- *Licensing of reactors* which involves utilizing PRA/PSA in the review of analyses submitted as part of advanced reactor design certification applications, and plant-specific licensing actions such as technical specification modifications, justifications for continued operations, etc.

- *Regulation of reactors* which involves utilizing PRA/PSA in monitoring of operations (with risk-based inspections); screening of events for significance (including operational event screenings, generic safety issue screenings, and facility screening risk analyses): analyses of events and issues (including operational events analyses, component and system failure data analyses and trends,reliability monitoring now developing as a result of the maintenance rule, generic safety issue analyses, and severe accident research studies); facility analyses (both those performed by the staff such as NUREG-1150 and those performed by licensees in the individual plant examination process); and in regulatory analyses supporting regulatory actions such as backfits.

- *Licensing of fuel cycle and materials* which involves utilizing methods similar to risk analyses (called performance assessment methods) and are being used as part of the licensing of proposed high level waste repositories, focused in NMSS.

These activities are summarized below in Table 1.

Table 1
Summary of Staff PRA/PSA Uses

CATEGORY	APPLICATION
Licensing of Reactors	• Reviews of advanced reactors. • Reviews of plant-specific licensing actions.
Reguiations of Reactors	• Monitoring operations by inspection. • Issue screening of operational events, generic safety issues, and facility screening risk analyses. • Issue analyses of operational events analyses, operational data and trending analyses, maintenance rule regulatory guide, generic safety issues, and severe accident issues. • Facility analyses involving staff studies and individual plant examinations. • Regulatory actions including regulatory analyses.
Licensing of Fuel Cycle and Materials	• Reviews involving high level waste facilities.

As can be seen, these PRA/PSA efforts are relatively diverse; and although each NRC Office (i.e., AEOD, NRR and RES) is involved in programs utilizing PRA/PSA, current utilization of this type of integral analysis by the NRC is rather limited when focused on attempts to reduce regulatory burden or provide additional flexibility with the regulations and licenses. Current NRC-sponsored programs that can provide insights in support of this area primarily involve configuration control regarding technical specification optimization. No NRC-sponsored programs supporting graded QA based on PRA/PSA were identified.

These specific types of activities are summarized below for each NRC office.

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

The Office for Analysis and Evaluation of Operational Data (AEOD) utilizes PRA/PSA techniques and insights in the accomplishment of its mission. Although their ongoing PRA/PSA-related programs are not focused on determining ways to reduce regulatory burden and provide flexibility in licensing and regulatory actions, the Trends and Patterns Analysis and the Reactor Operations Analysis Branches within the Division of Safety Programs are involved in efforts that can ultimately provide the data requirements and insights for PRA/PSA-based programs supporting configuration control and graded QA (from a regulatory perspective).

The Trends and Patterns Analysis Branch have ongoing programs that analyzes operational data to identify and provide a quantitative content for new safety issues; evaluates the effectiveness of current regulations, regulatory actions and initiatives taken by licensees to resolve safety issues concerns; and helps guide and focus engineering evaluations. These programs support four major activities as follows:

- Hardware performance studies of risk-important components, systems, initiating events and accident sequences

- Safety and regulatory studies of trend performance for selected regulatory issues through an appropriate parameter related to the specific issue to determine effectiveness of implementation.

- Data base studies involving common cause failure event data and a human performance data base that trends human actions important to plant safety and risk.

- Risk assessment studies evaluating the risk implications of trending results from the hardware, safety issues and special data analyses.

The Reactor Operations Analysis Branch's ongoing Accident Sequence Precursor (ASP) Program also provides needed support for the PRA/PSA utilization in configuration control and graded QA optimization. The ASP program provides a safety significance perspective of nuclear plant operational experience. The program uses PRA/PSA techniques to provide estimates of operating event significance in terms of the potential for core damage; that is, accident sequence precursors are events that are important elements in core damage accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one of more postulated events, could result in a plant condition leading to severe core damage. The precursors are selected and evaluated using an evaluation process and significance quantification methodology. The types of events evaluated include initiators, degradations of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences.

OFFICE OF NUCLEAR REACTOR REGULATION

The Office of Nuclear Reactor Regulation (NRR) have current PRA/PSA efforts directly supporting licensing and regulatory activities that are providing regulatory burden reduction and flexibility in the implementation of the regulations. These efforts are being performed in the Operational Reactor Support and Systems Safety Analysis Divisions by the Technical Specifications and Probabilistic Safety Assessment Branches, respectively.

In 1987, the Commission issued its interim "Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" encouraging licensees to voluntarily implement a Technical Specification Improvement Program. As a result of this policy statement, five sets of improved Standard Technical Specifications (STS) were developed; one for each NSSS vendor (i.e., Westinghouse, Babcock and Wilcox, Combustion Engineering, General Electric BWR 4, and General Electric BWR 6). PRA/PSA was utilized in the development of these STS as follows:

- A number of completion times (i.e., allowed outage times, AOTs) and surveillance test intervals (STIs) were relaxed based on NRC staff-approved topical reports and on draft NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements." In their topical reports justifying the relaxations, the NSSS vendors based their conclusions on PRA/PSA insights. NUREG-1366 used qualitative rather than PRA/PSA insights to support such relaxations.

- Utilizing the Grand Gulf and Surry PRAs from NUREG-1150, the core damage frequencies were recalculated with the new STS changes to identify any potential concerns. No significant increase in core damage frequency was observed as a result of these changes.

A "lead" plant for each NSSS STS has been identified by industry.

As the implementation of the improved STS and development of line-item improvements proceeds, the staff's intends to utilize PRA/PSA along with deterministic bases to support its decisions. This utilization will primarily be based on evaluations of industry's proposals. The information from the programs currently in progress in RES will be used to support or validate, as appropriate, industry's risk-based proposals.

Currently the staff is evaluating risk-based changes to technical specifications proposed by the South Texas Nuclear Project. This effort is currently in progress in RES.

The Probabilistic Safety Assessment Branch activities that directly involve PRA/PSA efforts to improve plant operations and maintenance primarily include providing risk assessment of potentially safety significant issues and reviewing applications submitted by the licensees. The issues reviewed for their risk impact are a result of identified safety concerns. Recent examples include:

- Intersystem LOCA
- Shutdown Risk
- Alternative Tube Plugging Criteria

The applications submitted by the licensees are generally requests for exemptions (or waivers) from regulatory requirements. The justification for requesting and granting the exemption is primarily based on PRA/PSA insights. Recent examples include:

- Waiver to allow refurbishment of service water system

- Minor actions involving man-made hazards, tornado protection, containment penetrations, toxic gas detectors

OFFICE OF NUCLEAR REGULATORY RESEARCH

The Office of Nuclear Regulatory Research (RES) has several ongoing PRA/PSA efforts directly supporting licensing and regulatory activities. These programs are being performed in the System Research, Safety Issue Resolution and Engineering Divisions by the Human Factors Branch, the Severe Accident Issues and Probabilistic Risk Assessment Branches, and the Electrical and Mechanical Engineering Branch, respectively.

The PRA/PSA programs in the Human Factors Branch are currently those that have the greatest potential in assisting in the assessment of risk technology for providing regulatory burden reduction and flexibility while maintaining safety. These efforts are primarily focused on developing methods in direct support of technical specification improvements as follows:

- Risk impact in varying AOTs and STIs at power and during shutdown and considering the effects of test errors on optimum test intervals.

- Risk impact from action statements requiring shutdown, if equipment needed during shutdown (e.g., residual heat removal), fails.

- Risk implications of taking equipment out-of-service for maintenance looking at rolling maintenance schedules, optimizing the frequency of schedule maintenance, and integrating surveillance with preventive maintenance.

- Dependent failures examining improved methods for recognizing and preventing dependent failures.

- Configuration management considering a conceptual framework for risk-based configuration management.

The methods that are being developed are reliability-engineering tools that analyze technical specification requirements within the framework of a PRA/PSA and which can estimate the risk impact of changing the level of a particular requirement in technical specifications; and therefore, they can provide a risk perspective on the bases for these technical specification requirements and for related maintenance guidelines.

These applications share the strengths and weaknesses of PRA/PSA. They are useful to integrate and prioritize only those considerations that can be quantified in terms of reliability and availability; therefore, they are applicable to only a fraction of the requirements in technical specifications. In general, these methods are directly applicable to evaluating AOTs and STIs for active, front-line systems and support systems. The methods are only marginally applicable to instrumentation, and are not applicable to concerns not modeled in PRA, such as security and occupational health. These [illegible] not yet sufficiently refined to treat uncertainties. It is expected that consid[illegible] uncertainties will be incorporated with use of these methods. For example, where uncertainties appear to be important to the decision at hand, as for example when comparing alternative courses of action which differ greatly in uncertainty, the comparison can be made with mean values.

There are currently five ongoing programs that are developing these methods as described below.

Procedures for Evaluating Technical Specifications

In 1983, a task force established by the EDO provided recommendations to improve surveillance testing requirements in technical specifications. The resulting actions formed the Technical Specification Improvement Program. In 1987, a Commission Interim Policy Statement on Technical Specifications Improvement encouraged licensees to voluntarily implement a Technical Specification Improvement Program that included applying risk-analysis methods and human-factors principles to improve technical specifications. In support of this program, research began to develop methods for evaluating the risk impact of requirements in technical specifications, to explore alternative approaches, and to provide a technical basis for improvements.

This research, which is largely completed, has published methods to evaluate the risk impact of AOTs and STIs (including the impact of test errors). The work also outlined a conceptual approach for operational configuration control. The remaining work on this project, which is being completed in 1993, will provide a method to evaluate the risk impact of scheduled maintenance intervals. The approach analyzes the balance between beneficial and adverse effects of maintenance, and models three states: operable, degraded (i.e., ready for preventive maintenance), and failed. The method can use NPRDS data for incipient, degraded, and complete failures. The results of this research will allow analysis of the risk impact of issues such as not permitting certain preventive maintenances during power operation, and instead requiring that AOTs during power operation be used only for corrective maintenance.

One of the new STS's will be used as a testbed for a limited pilot application of the methods described in this report for evaluating requirements in technical specifications. This pilot application involves developing a strategy and criteria that will result in clear, simple statements of requirements that integrate risk and practical considerations to control risk efficiently. These criteria are intended to address:

- The scope and frequency of updating of the PRA/PSA and data base that form the basis for the licensee's risk analysis.

- What risks must be assessed to support technical specification changes and acceptable ways to model them (e.g., test intervals, test effectiveness, test errors, and aging effects).

- Prioritizing risk contributors in technical specifications.
- Acceptable changes in risk.
- Experience feedback, if appropriate, in updating technical specification requirements.

Technical specification Requirements During Shutdown

NRC is reevaluating regulatory requirements for nuclear power plants during shutdown. One aspect of this reevaluation is to consider how effectively technical specifications control risk during shutdown.

In support of this endeavor, this project was established to develop methods for evaluating the risk impact of plant configurations permitted and surveillance required by technical specifications during shutdown; to explore alternative approaches; and to provide a technical basis for improvements. These analysis methods use as a framework the low-power-and-shutdown PRA/PSAs (described elsewhere in this report).

These models and trial applications to a PWR and a BWR will be completed in late 1993.

Action Statements That Require Shutdown

As part of the program to improve technical specifications, action statements that require plant shutdown if an allowed outage time is exceeded are being developed.

The issue concerns a few systems, such as residual heat removal (RHR), standby service water (SSW), and auxiliary feedwater, that may be required to cool the plant during shutdown. Currently, action statements in technical specifications typically require that plants shut down when an AOT is exceeded, even though shutdown may require use of the system that is out-of-service for maintenance. The work has developed a decision-analysis method for comparing the risk impact of transferring the plant to shutdown versus the risk impact of continued power operation.

The method and trial application to RHR and SSW at a BWR-6 are being published this Spring. An equivalent method and trial application to a PWR will be completed in early 1994.

Technical Specification Defenses Against Dependent Failures

Technical specifications set surveillance requirements and AOTs in order to assure the availability of a plant's safety systems. These safety systems are designed to achieve high availability through redundancy. Redundancy, however, can be defeated by dependent (e.g., common-cause) failures. For example, the Davis-Besse loss of all feedwater in 1985 involved several valves stuck shut (dependent failures). Despite the importance of dependent failures, most technical specification requirements do not explicitly address and protect against dependent failures.

In support of this concern, a method and criteria are being developed for explicitly addressing dependent failures in setting STIs and AOTs. This method uses a NUREG-1150 PRA as the framework within which to model and evaluate the risk impact of postulated technical specification improvements. A recent AEOD analysis of industry-wide experience with dependent-failure events is used as a reality check to supplement the PRA/PSA. Possible improvements in technical specifications that might better defend against such dependent failures are being postulated.

The purpose is to determine whether simple changes in surveillance requirements and AOTs would substantially reduce the risk of operating reactors. The result will be an assessment of the effectiveness of this approach.

Method for Monitoring Dependent Failures

This effort is a related project that supports AEOD trends and analysis of operational data, This project has developed a method for analyzing failure data to estimate the fraction of failures that are dependent failures. The method compares the distribution of observed times-between-failures with the distribution expected if the failures were independent. The difference reflects dependent failures. The method estimates the fraction of dependent failures (e.g., a beta factor) and the actual safety-system unavailability with this degree of dependency.

The methods-development has been completed, and the report will be published in mid 1993. AEOD and RES are discussing whether additional work is warranted to make the software directly applicable to AEOD screening of data to help recognize dependent-failure events.

These five programs are focused on developing methods for technical specification optimization. The methods developed, given that the limitations, boundary conditions, assumptions, uncertainties, data, and human performance issues associated with PRA/PSA are properly addressed, can provide assistance in determining the ground rules or restrictions that would be necessary to maintain the current level of safety while providing additional flexibility in the implementation of the regulations. In addition, there are other ongoing programs within RES which also utilize PRA/PSA, will provide necessary insights, and will provide assistance in addressing the above mentioned concerns.

Technical Analysis of Proposed Changes to the South Texas Technical Specifications

Houston Lighting and Power, the licensee for the South Texas Nuclear Project (STNP), submitted a proposed amendment to its operating license. The Probabilistic Risk Analysis Branch is developing a framework for analysis and a technical basis for evaluating the proposed changes to AOTs and STIs for the STNP. The evaluation involves reviewing the system failure models and sequence level cut sets of the STNP PSA, establishing a systematic risk profile for the base case three-train configuration of the STNP, obtaining the overall risk impact of the proposed changes in AOTs and STIs, and developing a framework which will support the bases for approval of the proposed changes in AOTs and STIs based on risk arguments.

Although this effort is not a formal program to develop "generic" methods for evaluating proposed technical specification changes, insights can be used for generic applications.

Individual Plant Examination Data Base

On November 23, 1988, Generic Letter 88-20 was issued requesting licensees to perform an Individual Plant Examination (IPE) with the general purpose of each licensee "to develop an appreciation of severe accident behavior, to understand the most likely severe accident sequences that could occur at its plant, to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (if necessary) to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents."[1]

In support of this effort, an IPE Data Base has been developed which catalogs the information provided in each licensee's IPE submittal. The type of information being input to the data base for each IPE includes the following:

- Plant information (e.g., reactor and containment type)
- Initiating event information (e.g., initiating event and its associated frequency)
- Accident sequence information (e.g., accident sequence description and associated frequency)
- System and component dependency information

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[1]INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES - 10 CFR §50.54(f), Generic Letter No. 88-20.

- Core damage frequency information
- Plant damage state information

The data base will allow users to gather information both by plant and across plants. For example, the data base will identify those plants where a certain issue such as loss of offsite power is a concern; will identify concerns for a group of plants such as identifying the dominant contributors for 3-loop westinghouse plants; will identify those plants where a system concern may exist such as identifying plants where diesel generators are dependent on instrument air. These are a few examples of the IPE data base.

The information currently being entered into the data base only includes IPE data. As part of the IPE effort, licensees were only required to examine internal initiators and internal flooding. NUREG-1407 provides the guidelines for the IPE of external events. The data base will be expanded to include this information for each licensee.

Low Power and Shutdown PRA

PRA/PSAs have traditionally examined severe accidents only occurring at full power operation. Analyses have indicated that severe accident occurring at low power and shutdown could be significant. A major program has been in progress to assess the frequencies and risks of accidents initiated during low power and shutdown modes of operation for two nuclear power plants by performing detailed PRAs for the carious operational modes. This effort also involves the development of new methods and will compare the assessed risk with those of accident initiated during full power operation.

The work involves examining the accidents initiated by internal events (including flooding and fire) as well as external events (e.g., earthquakes). Ultimately a full PRA (core damage frequency, fission product releases and consequences) will be completed.

Plant Aging

[TO BE WRITTEN]

PRA Working Group

In 1991, the Executive Director for Operations formed a working group of staff management (i.e., PRA Working Group) to "consider what improvements in methods and data analysis are possible and needed, the role of uncertainty analysis in different staff uses of PRA, if improvements are needed in the allocation of existing PRA staff, and the need for recruitment of more staff (or for identifying other means for supplementing staff resources."[2]

The objectives of the PRA Working Group are to develop guidance on consistent and appropriate uses of PRA/PSA within the NRC; to identify skills and experience necessary for each category of staff use; and to identify improvements in PRA/PSA methods and associated data necessary for each category of staff use. In support of these objectives, the Group has defined the scope of its work as follows:

- Ascertain present uses of PRA/PSA by the staff; future PRA/PSA uses which are not now well defined (e.g., possible transition to risk-based reactor regulation) are not included in the Group's scope of work.

- Review of available or developing risk analysis documents and guides, and develop recommendations for improvement. Such improvements are the responsibility of the user organization, with oversight by the Working Group. It is not within the Group's scope to update or replace such guides, although the group may make recommendations to update them.

- Assess staff skills and experience needed to appropriately apply PRA/PSA, including staff organizational considerations, if appropriate. While the skills and experience assessment is within the scope of the Group's work, the development and implementation of plans to change staffing levels, staff training, or organizational arrangements are the principal responsibility of the Office of Personnel and the affected offices, as part of the overall development and implementation of the agency's Human Resources Strategic Plan.

- Assess needed improvements in PRA?PSA techniques and data to support appropriate staff use of risk analysis. This assessment focuses on improvements needed for particular uses, rather than a broad assessment of needed improvements in risk analysis methods, and uses state-of-the-art risk studies such as NUREG-1150 as reference and resource material. The performance of any such improvements is the responsibility of the appropriate staff organization, not the Working Group.

It must be ensured that the current level of safety is maintained when utilizing an integral analysis, such as PRA/PSA, to provide more flexibility in the regulations and in the implementation of the regulations. NRC-sponsored programs were inventoried in a first step to determine what types of general rules and restrictions would need to be imposed so that

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[2]Letter from James M. Taylor, Executive Director for Operations, NRC, to David A. Ward, Chairman, ACRS, October 1, 1991.

PRA/PSA can be used while maintaining the current level of safety. A summary of these PRA/PSA programs that could provide insights are provided in Table 2 below.

Table 2
Summary of NRC-Sponsored PRA/PSA Programs

RESPONSIBILITY	PROGRAMS	APPLICATION
AEOD/DSP/TPAB	Analysis of operational data to identify and provide quantitative content for safety issues	Data support to technical specification and graded QA optimization
AEOD/DSP/ROAB	Accident Sequence Precursor Program	Data support to technical specification and graded QA optimization
NRR/DORS/TSB	Technical Specification Improvement Program	Utilization of technical specification optimization
NRR/DSSA/PSAB	• Risk Evaluation of Safety Issues • Review of Licensee Requests for Exemption	Information support to technical specification and graded QA optimization
RES/DSR/HFB	• Procedures for Evaluating Technical Specifications • Technical Specification Requirements During Shutdown • Actions Statements That Require Shutdown • Technical Specifications Defenses Against Dependent Failures • Method for Monitoring Dependent Failures	Development of technical specification optimization methods
RES/DSIR/PRAB	Technical Analysis of Proposed Changes to the South Texas Technical Specification	Information support to technical specification and graded QA optimization
RES/DSIR/SAIB	Individual Plant Examination Data Base	Information support to technical specification and graded QA optimization
RES/DSIR/PRAB	Low Power and Shutdown PRA	Information support to technical specification and graded QA optimization
RES/DE/EMEB	Plant Aging	
RES/DSIR/PRAB	PRA Working Group	Information support to technical specification and graded QA optimization

ASSESSMENT OF NRC-SPONSORED PROGRAMS

Probabilistic Risk Assessment (PRA) or Probabilistic Safety Assessment (PSA) applications having the potential to significantly reduce regulatory burden or provide ;more flexibility in the regulations and in the implementation of the regulations while maintaining safety are those that primarily address configuration control and quality assurance (QA) issues. Configuration control applications generally involve the utilization of PRA/PSA methods to optimize surveillance test intervals (STIs) and allowed outage times (AOTs). QA applications generally involve the utilization of PRA/PSA to support "graded" QA; that is, optimizing QA for those structures, systems or components that are safety significant based on PRA/PSA insights. Current NRC-sponsored programs were examined to identify those efforts that are utilizing PRA/PSA that could provide potential insights in these areas.

The use of PRA/PSA by the NRC has been both broad and narrow. The broad application is seen in the many various and diverse activities which have increased over time, particularly since the TMI accident. The utilization of PRA/PSA, however, has been narrow in that it has been limited to a small set of applications. These activities have been defined and summarized into several categories (as reported in the NRC PRA Working Group Report) as follows:

- *Licensing of reactors* which involves utilizing PRA/PSA in the review of analyses submitted as part of advanced reactor design certification applications, and plant-specific licensing actions such as technical specification modifications, justifications for continued operations, etc.

- *Regulation of reactors* which involves utilizing PRA/PSA in monitoring of operations (with risk-based inspections); screening of events for significance (including operational event screenings, generic safety issue screenings, and facility screening risk analyses): analyses of events and issues (including operational events analyses, component and system failure data analyses and trends,reliability monitoring now developing as a result of the maintenance rule, generic safety issue analyses, and severe accident research studies); facility analyses (both those performed by the staff such as NUREG-1150 and those performed by licensees in the individual plant examination process); and in regulatory analyses supporting regulatory actions such as backfits.

- *Licensing of fuel cycle and materials* which involves utilizing methods similar to risk analyses (called performance assessment methods) and are being used as part of the licensing of proposed high level waste repositories, focused in NMSS.

These activities are summarized below in Table 1.

Table 1
Summary of Staff PRA/PSA Uses

CATEGORY	APPLICATION
Licensing of Reactors	• Reviews of advanced reactors. • Reviews of plant-specific licensing actions.
Regulations of Reactors	• Monitoring operations by inspection. • Issue screening of operational events, generic safety issues, and facility screening risk analyses. • Issue analyses of operational events analyses, operational data and trending analyses, maintenance rule regulatory guide, generic safety issues, and severe accident issues. • Facility analyses involving staff studies and individual plant examinations. • Regulatory actions including regulatory analyses.
Licensing of Fuel Cycle and Materials	• Reviews involving high level waste facilities.

As can be seen, these PRA/PSA efforts are relatively diverse; and although each NRC Office (i.e., AEOD, NRR and RES) is involved in programs utilizing PRA/PSA, current utilization of this type of integral analysis by the NRC is rather limited when focused on attempts to reduce regulatory burden or provide additional flexibility with the regulations and licenses. Current NRC-sponsored programs that can provide insights in support of this area primarily involve configuration control regarding technical specification optimization. No NRC-sponsored programs supporting graded QA based on PRA/PSA were identified.

These specific types of activities are summarized below for each NRC office.

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

The Office for Analysis and Evaluation of Operational Data (AEOD) utilizes PRA/PSA techniques and insights in the accomplishment of its mission. Although their ongoing PRA/PSA-related programs are not focused on determining ways to reduce regulatory burden and provide flexibility in licensing and regulatory actions, the Trends and Patterns Analysis and the Reactor Operations Analysis Branches within the Division of Safety Programs are involved in efforts that can ultimately provide the data requirements and insights for PRA/PSA-based programs supporting configuration control and graded QA (from a regulatory perspective).

The Trends and Patterns Analysis Branch have ongoing programs that analyzes operational data to identify and provide a quantitative content for new safety issues; evaluates the effectiveness of current regulations, regulatory actions and initiatives taken by licensees to resolve safety issues concerns; and helps guide and focus engineering evaluations. These programs support four major activities as follows:

- Hardware performance studies of risk-important components, systems, initiating events and accident sequences

- Safety and regulatory studies of trend performance for selected regulatory issues through an appropriate parameter related to the specific issue to determine effectiveness of implementation.

- Data base studies involving common cause failure event data and a human performance data base that trends human actions important to plant safety and risk.

- Risk assessment studies evaluating the risk implications of trending results from the hardware, safety issues and special data analyses.

The Reactor Operations Analysis Branch's ongoing Accident Sequence Precursor (ASP) Program also provides needed support for the PRA/PSA utilization in configuration control and graded QA optimization. The ASP program provides a safety significance perspective of nuclear plant operational experience. The program uses PRA/PSA techniques to provide estimates of operating event significance in terms of the potential for core damage; that is, accident sequence precursors are events that are important elements in core damage accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one of more postulated events, could result in a plant condition leading to severe core damage. The precursors are selected and evaluated using an evaluation process and significance quantification methodology. The types of events evaluated include initiators, degradations of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences.

OFFICE OF NUCLEAR REACTOR REGULATION

The Office of Nuclear Reactor Regulation (NRR) have current PRA/PSA efforts directly supporting licensing and regulatory activities that are providing regulatory burden reduction and flexibility in the implementation of the regulations. These efforts are being performed in the Operational Reactor Support and Systems Safety Analysis Divisions by the Technical Specifications and Probabilistic Safety Assessment Branches, respectively.

In 1987, the Commission issued its interim "Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" encouraging licensees to voluntarily implement a Technical Specification Improvement Program. As a result of this policy statement, five sets of improved Standard Technical Specifications (STS) were developed; one for each NSSS vendor (i.e., Westinghouse, Babcock and Wilcox, Combustion Engineering, General Electric BWR 4, and General Electric BWR 6). PRA/PSA was utilized in the development of these STS as follows:

- A number of completion times (i.e., allowed outage times, AOTs) and surveillance test intervals (STIs) were relaxed based on NRC staff-approved topical reports and on draft NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements." In their topical reports justifying the relaxations, the NSSS vendors based their conclusions on PRA/PSA insights. NUREG-1366 used qualitative rather than PRA/PSA insights to support such relaxations.

- Utilizing the Grand Gulf and Surry PRAs from NUREG-1150, the core damage frequencies were recalculated with the new STS changes to identify any potential concerns. No significant increase in core damage frequency was observed as a result of these changes.

A "lead" plant for each NSSS STS has been identified by industry.

As the implementation of the improved STS and development of line-item improvements proceeds, the staff's intends to utilize PRA/PSA along with deterministic bases to support its decisions. This utilization will primarily be based on evaluations of industry's proposals. The information from the programs currently in progress in RES will be used to support or validate, as appropriate, industry's risk-based proposals.

Currently the staff is evaluating risk-based changes to technical specifications proposed by the South Texas Nuclear Project. This effort is currently in progress in RES.

The Probabilistic Safety Assessment Branch activities that directly involve PRA/PSA efforts to improve plant operations and maintenance primarily include providing risk assessment of potentially safety significant issues and reviewing applications submitted by the licensees. The issues reviewed for their risk impact are a result of identified safety concerns. Recent examples include:

- Intersystem LOCA
- Shutdown Risk
- Alternative Tube Plugging Criteria

The applications submitted by the licensees are generally requests for exemptions (or waivers) from regulatory requirements. The justification for requesting and granting the exemption is primarily based on PRA/PSA insights. Recent examples include:

- Waiver to allow refurbishment of service water system

- Minor actions involving man-made hazards, tornado protection, containment penetrations, toxic gas detectors

POSSIBLE USE OF RISK-BASED TECHNICAL SPECIFICATIONS

The need for improvements in Technical Specifications has been recognized for some time, culminating in the NRC's Technical Specification Improvement Program and the Commission's 1987 Interim Policy Statement on Technical Specifications. As a direct result of this effort, new standard Technical Specifications (STS) for various types of plants were issued by the staff in 1992.

In parallel, the NRC has sponsored an active research program on the use of probabilistic concepts to minimize risk and optimize surveillance test intervals and allowed outage times at a given plant. Methods have also been developed to compare the risk of shutting down to the risk of remaining at power under certain circumstances. Qualitative insights from this research were used in developing the new Standard Technical Specifications. To a limited extent, quantitative analyses developed under the research program were also used in deriving values for specific applications in the new STS. In the main, however, the new STS can not be considered "risk-based".

To investigate the potential which might exist to incorporate "risk-based" concepts into the STS, we have surveyed the Westinghouse and General Electric BWR 6 Standard Technical Specifications to determine what fraction of the Completion Times and Surveillance Requirements might be easily amenable to probabilistic optimization. The results are attached. In general, we considered an item to be easily amenable to optimization if it e=were usually modeled in sufficient detail to do so in the PRAs which have been completed to date.

ADDITIONAL OBSERVATION

Based on our discussions with various industry representatives, we sense that there is not strong support to go further in applying risk-based optimization to the STS at the present time. Rather, we sense a commitment to evaluate the adequacy of the new STS, as they presently exist, and to gain practical experience with there use before progressing further.

We note there is considerable activity in exploring the concept of probabilistic-based controls on plant activities internationally. Most efforts of which we are aware are still research efforts, rather than applications. However, we understand techniques similar to those developed under the NRC's research program are being applied at the Laguna Verde plant in Mexico, and configuration controls have been established to differing degrees at two plants in the U.K. (Heysham-2 and Torness).

GENERAL ELECTRIC BWR 6 STANDARD TECHNICAL SPECIFICATIONS

COMPLETION TIMES AND SURVEILLANCE REQUIREMENTS
(preliminary estimate)

TOPIC	NUMBER CT	NUMBER SR	EASILY AMENABLE TO RISK-BASED OPTIMIZATION CT	EASILY AMENABLE TO RISK-BASED OPTIMIZATION SR
Reactivity Control Systems	45	27	7	13
Power Distribution Limits	8	8	0	0
Instrumentation	183	127	53	40
Reactor Coolant System	2	1	0	0
Emergency Core Cooling System and RCIC	26	15	26	15
Containment Systems	146	86	16	15
Plant Systems	35	20	5	9
Electrical Power Systems	77	37	30	17
Refueling Operations	(not considered)			
Special Operations	(not considered)			
TOTAL (excluding instrumentation)	339	194	84	69

NOTE:

- Values are approximate. Some double counting exists where the action statements are complex.
- Values are subjective. For example, if there are different surveillances on the same instrument (continuity, check and calibration) it is counted as 3; however, all instruments of the same type are counted as one (e.g., the calibration of 8 SG level detectors was counted as one surveillance.
- Although risk-based information could be used to modify many of the containment action times and surveillance intervals, little credit was given because it requires a Level II PRA and would require significant changes in Appendix J.

WESTINGHOUSE STANDADR TECHNICAL SPECIFICATIONS

COMPLETION TIMES AND SURVEILLANCE REQUIREMENTS
(preliminary estimate)

TOPIC	NUMBER CT	NUMBER SR	EASILY AMENABLE TO RISK-BASED OPTIMIZATION	
			CT	SR
Reactivity Control Systems	21	21	0	0
Power Distribution Limits	21	12	0	1
Instrumentation	258	368	194	204
Reactor Coolant System	44	54	17	16
Emergency Core Cooling System	20	19	20	7
Containment Systems	88	85	43	35
Plant Systems	75	46	32	21
Electrical Power Systems	65	48	46	28
Refueling Operations	(not considered)			
TOTAL (excluding instrumentatiom)	314	288	158	108

NOTE:

- Values are approximate. Some double counting exists where the action statements are complex.

- Values are subjective. For example, if there are different surveillances on the same instrument (continuity, check and calibration) it is counted as 3; however, all instruments of the same type are counted as one (e.g., the calibration of 8 SG level detectors was counted as one surveillance.

- Although risk-based information could be used to modify many of the containment action times and surveillance intervals, little credit was given because it requires a Level II PRA and would require significant changes in Appendix J.

POTENTIAL USE OF PROBABILISTIC METHODS
IN IMPLEMENTING
A GRADED QUALITY ASSURANCE PROGRAM

As noted elsewhere, a graded approach to quality assurance is consistent with 10CFR50 Appendix B. Probabilistic methods, supplemented with deterministic engineering analysis, provide a tool for a ranking of those systems, structures and components important to safety in terms of their relative safety importance.

In the normal performance of a probabilistic risk analysis, the analyst calculates an overall severe core damage frequency and can identify the principal accident sequences that contribute to that total. Similarly, for each accident sequence, the principal cutsets (combinations of failures that can lead to the occurrence of the accident sequence) are also identified. Importance measures can be calculated which (1) indicate the importance of a given component to the overall severe core damage frequency, (2) indicate how the core damage frequency would increase if it were assumed a given component was failed (commonly called the Risk Achievement Worth), and (3) indicate how the core damage frequency would decrease if the component in question was perfectly reliable (commonly called the Risk Reduction Worth). This quantitative determination of the importance to safety of the various components modeled is common practice and is a requested result of the Individual Plant Examinations called for by Generic Letter 88-20.

When considering the importance to safety as it applies to the possibility of a graded approach to quality assurance, the "Risk Achievement Worth" importance measure may be of greatest utility. It identifies how badly the severe core damage frequency would be impacted if the reliability of the component was degraded to the point where failure were certain. (A similar measure examining the incremental increase in core damage frequency resulting from an incremental decrease in reliability - the Fussell-Vesely importance measure - is also routinely calculated my most computer codes used in the quantification of the plant PRA models.) A brief examination of risk analyses performed in the NUREG-1150 study indicates that there are only 100 to 200 specific components whose certain failure would increase the severe core damage frequency by more than 1×10^{-6} per year. This number decreases to 40 to 190 when considering increases in core damage frequency greater than 1×10^{-5} per year.

This list of important components could then form a portion of list of those items requiring the highest level of quality assurance. It would need to be supplemented in two ways. First, there are certain components which are not usually directly modeled in the PRA because their failure probability is believed to be very low. (An example is the reactor pressure vessel.) The boundary conditions and assumptions used in the probabilistic study need to be examined to determine which components, if any, should be added to the list because they would have considerable impact if their reliability seriously degraded, and they were not modeled in the plant-specific risk analysis. Second, PRAs generally models items at the component level or higher. Some of these components are complex (e.g., diesel generators) and have many piece parts or sub-components. Since the Q-List may extend to these piece parts, a deterministic engineering evaluation may be necessary to determine if failure of the piece part will cause failure of the component to perform its function. If so, the piece part may need to be added to the listing of those items receiving the highest level of quality assurance.

Note that this discussion focuses on those systems and components important to safety, but does not yet consider structures. In principle, the same approach as outlined above could be used, treating the external events portion of the PRA in a manner analogous to that outlined above. In practice, however, it appears that the lack of flexibility in quality assurance requirements has its greatest impact in the procurement of replacement parts for operating plants. If this is correct, the treatment of structures in a graded quality assurance program for an existing plant may be of second-order importance.

Having assembled a list of those components and their piece parts most important to safety, and thus deserving the most extensive quality assurance attention, it is necessary to consider the effects of modeling and data uncertainties and the general completeness of the results before employing this list solely for the highest level of quality assurance. We will have available shortly the results of Individual Plant Examinations for all power reactors in the country. It will be possible to prepare plant-specific lists of those components with greatest safety significance for each plant. These then can be aggregated to compose a generic listing of important components for a given class of reactors where any item found important at any one plant in the list would be included in the aggregate listing for that class of plant. This aggregate list could then be used (in similar fashion to that described above) to identify those items requiring the second highest level of quality assurance, i.e., any component found important to safety at any plant of a given type would receive a heightened level of quality assurance, while those components found important at a specific plant would be treated in a manner similar to that afforded to safety grade components today.

Items not found important to risk in systems which are currently regarded as safety grade would still receive quality assurance consideration. However, this might rely primarily on pre-operational testing, and analysis of reliability of components built to the same industrial standard, rather on the maintenance of a pedigree on the component.

The general process identified here is similar in many respects to the process that would be followed under the guidelines prepared by the industry in response to the Maintenance Rule. Efforts should be taken to coordinate efforts and avoid duplication.