Volume Three

Regulatory Review Group

Operating Licenses

U.S. Nuclear Regulatory Commission
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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 Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2 were selected from among all the plants currently licensed to operate.

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

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

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

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

1.3 Assessment Approach

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

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

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

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

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

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

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

An assessment summary was prepared for each item. Each summary contained overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items that were selected

for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

Table 1

SUMMARY OF ASSESSMENT APPROACH

1. Review each operating license item and assign it to a category.

2. Determine which categories contain items that are appropriate to be assessed collectively.

3. Determine which items from the remaining categories will be assessed individually.

4. Assess items in accordance with specified questions; analyze items as necessary.

5. Prepare assessment summaries.

6. Validate results from assessment(s) of previous license(s).

7. Integrate overall results, and develop findings and recommendations.

Table 2

CATEGORIES OF ITEMS

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

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

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

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

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

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

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

Table 3

ASSESSMENT QUESTIONS

1. 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)?

   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 that allows the licensee a tradeoff of items without a reduction in overall safety?

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

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

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

TABLE 3 (Continued)

ASSESSMENT QUESTIONS

4. Enhanced Flexibility Potential

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

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

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

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

   E. Are there NRC programs currently ongoing or under evaluation for implementation that would provide enhanced flexibility to the licensee?

2. ASSESSMENT OF OPERATING LICENSES

2.1 Licenses

As discussed in Chapter 1 of this report, the Review Group assessed the operating licenses of four plants--Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2. The licenses as they were reviewed were as follows:

- The Seabrook Unit 1 operating license was issued on March 15, 1990. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the environmental protection plan, which is Appendix B to the license. The license as reviewed has been amended through Amendment 11, dated May 29, 1992.

- The Surry Unit 1 operating license was issued on May 25, 1972. The operating license consists of the license itself and the Technical Specifications, which are Appendix A to the license. The license as reviewed has been amended through Amendment 170, dated June 1, 1992.

- The Perry Unit 1 operating license was issued on November 13, 1986. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; the environmental protection plan, which is Appendix B to the license; and the antitrust conditions, which are Appendix C to the license. The license as reviewed has been amended through Amendment 43, dated May 28, 1992.

- The Peach Bottom Unit 2 operating license was issued on December 14, 1973. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Technical Specifications, which are Appendix B to the license. The license as reviewed has been amended through Amendment 168, dated July 6, 1992.

The Review Group assessed the Seabrook and Surry licenses separately. The assessment of the Seabrook license, which was performed first, resulted in seven recommendations. The assessment of the Surry license, which was performed next, largely validated the Review Group's assessment of the Seabrook license and resulted in only three additional recommendations. Based on the results of these two assessments, and its knowledge of and experience with other licenses, the Review Group did not expect to find significant information in its reviews of the Perry and Peach Bottom licenses that would result in a substantial number of additional recommendations. Therefore, the Review Group assessed the Perry and Peach Bottom licenses together. The combined assessment was performed using the same methodology as that used previously for the individual plant assessments and resulted in one additional recommendation.

2.2 Assessment of Licenses

The four operating licenses contain a total of 1,127 items. Each item was reviewed and assigned to one of the categories in Table 2. The numbers of items in each of the operating licenses by category are shown in Table 4.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on their regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 297 items or approximately 26 percent of the total number of items.

The number of items in the remaining categories that were assessed individually was determined to be approximately 10 percent for Seabrook and Surry, and, since they were reviewed together, 5 percent for each of Perry and Peach Bottom. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or five of the 50 Category D items in the Perry license were selected for further assessment. With the 297 items that were assessed collectively, this meant that 358 or approximately 32 percent of the 1,127 total items were assessed either collectively or individually.

The items to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. The items that were assessed for each plant are listed in Table 5 of the plant assessment reports (Appendixes A, B, and C for Seabrook, Surry, and Perry and Peach Bottom, respectively).

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table 3. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted. The numbers of items in each of the operating licenses by group are shown in Table 5.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

2.3 Results of Assessment

The assessment summaries for the items assessed for each of the licenses are contained in the attachments of the respective plant assessment reports. The findings and recommendations for each of the licenses are presented in Chapter 3 of the respective plant assessment reports. The separate assessment reports for Seabrook and Surry, and the combined assessment report for Perry and Peach Bottom are provided in Appendixes A, B, and C to this report, respectively.

The overall assessment results, which are based on the findings and recommendations of the assessments of all four of the licenses, are presented in Chapter 3 of this report.

Table 4

OPERATING LICENSE ITEMS BY CATEGORY

	Category	Seabrook	Surry	Perry	Peach Bottom	Totals
A.	Technical Requirements	136	93	136	75	440
B.	Non-Technical License Conditions	4	1	7	2	14
C.	License Conditions That Rely on Other Documents for Requirements	32	16	23	19	90
D.	Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	50	20	48	59	177
E.	Reporting and Recordkeeping Requirements	36	19	39	29	123
F.	Unique Plant Features	10	9	11	10	40
G.	Other	63	34	65	81	243
	Totals	331	192	329	275	1,127

Table 5

FINDINGS (NUMBERS OF ITEMS) BY GROUP

Group	Seabrook	Surry	Perry	Peach Bottom	Totals
Items that appear to exceed applicable regulatory requirements	7	0	0	2	9
Items that should be considered for possible reduction in regulatory burden	4	5	4	4	17
Items that provide inherent flexibility	6	5	3	3	17
Items that should be considered for enhanced flexibility	6	4	7	3	20
Items considered or being considered in other programs	7	4	6	5	22
Items for which no further consideration is warranted	36	19	39	29	123

3. ASSESSMENT RESULTS

3.1 Introduction

As discussed in the previous chapters of this report, the Review Group assessed the operating licenses of four plants--Seabrook, Surry, Perry, and Peach Bottom. The assessment reports for theses licenses are presented in the appendixes to this report. The overall results of the individual license assessments are integrated in this chapter, which discusses the Review Group's overall observations, findings, recommendations, and conclusions.

3.2 Observations

In its assessment of the four licenses, the Review Group made a number of general observations about the licenses themselves. The observations relate to the overall condition of the licenses, the numbers and types of items in the licenses, and the license conditions and Technical Specifications. In addition, some of the observations relate to the plant (license) selection criteria and selection of plants to be assessed, as described in Chapter 1 of this report. The Group's observations are as follows:

3.2.1 Overall Condition of Licenses

The recently issued licenses are generally crisper and cleaner than the ones issued earlier. That is because a concerted effort has been made by the NRC to limit the license conditions to those explicitly required by the Atomic Energy Act or the Commission's regulations. One exception, which is also valid for the older plants, is the incorporation of "contemporary" issues, that is issues that were "hot" at the time the license was issued. An example of this is the TDI diesel-generator license condition in the Perry license. In contrast, the earlier issued licenses appear to contain many more license conditions that address plant-specific issues. Handling of these issues as commitments in licensee-controlled documents would not only increase the flexibility available to the licensee, but also would reduce the regulatory burden on both the licensee and the NRC. In addition, the older licenses contain a number of conditions that overlap or have been superseded, e.g., physical security conditions. Such conditions could lead to confusion and mistakes.

3.2.2 Numbers and Types of Items in Licenses

The average number of items that represent requirements (Categories A through E items in Table 4) in the recently issued licenses (256) is substantially greater than that in the licenses issued earlier (167). However, the percentage of items that represent technical requirements (Category A items in Table 4) in the recently issued licenses (53 percent) is not substantially different from that in the licenses issued earlier (51 percent).

The average number of items that represent requirements (Categories A though E items in Table 4) in the pressurized water reactor plant licenses (204) is not substantially different from that in the boiling water reactor plant licenses (219). Also, the percentage of items that represent technical requirements (Category A items in Table 4) in the pressurized water reactor plant licenses (56 percent) is not substantially different from that in the boiling water reactor plant licenses (63 percent).

The distributions of the items within the categories shown in Table 4 and within the groups shown in Table 5 are not substantially different from one license to another, regardless of the age of the license or the type of reactor. Also as illustrated in Tables 4 and 5, the distributions of the categories of items within each group are not substantially different from one license to another regardless of the age of the license or the type of reactor. These observations imply that no part of an operating license, e.g., the license itself, the Technical Specifications, or a specific Technical Specification section, appears to contain a disproportionate number of items that do not have either a sound regulatory basis or provide inherent flexibility. They also imply that these items may be found in any part of the license.

No significant differences in the Review Group's findings could be attributed to reactor type. Where design and other technical differences based on reactor type were found, no meaningful correlation of these differences with regulatory basis, safety relevance, inherent flexibility, or enhanced flexibility potential was identified.

3.2.3 License Conditions and Technical Specifications

The Technical Specifications of the recently issued licenses were based on the Standard Technical Specifications whereas the Technical Specifications of the early licenses were "custom." This could complicate line-item improvements for the older plants under the Technical Specification Improvement Program as indicated below.

- A number of the Technical Specifications (as well as license conditions) of the early licenses are less restrictive than their later counterparts.

- The Technical Specifications of the early plants are written in a narrative fashion whereas those of the newer plants are written in more coherent format.

- The Technical Specifications of the early plants tend to be more interdependent on others. That is, they frequently cross-reference other Technical Specifications.

- The Technical Specifications exhibit differences in philosophy. A number of the older plant Technical Specifications tend to be system based whereas the newer plant Technical Specifications tend to be function based.

Finally, a number of errors and inconsistencies were identified in the bases of the Technical Specifications, especially those of the older plants. The errors and inconsistencies were often related to Technical Specification revisions for which the bases had not been updated. Given the importance and expanded use of the bases in the Improved Standard Technical Specifications, the existence of errors and inconsistencies, while not an operating license compliance issue, is a problem that nevertheless could have safety impact. Particularly when used as guidance or reference during operation and training, the Technical Specification bases should provide plant operators with accurate and coherent information. In addition, the bases assist both the licensees and the NRC in the interpretation of Technical Specification requirements that might be otherwise ambiguous.

3.2.4 Applicability of Assessment Results

As discussed in Chapter 1 of this report, each of the four licenses selected for review is representative of a significant number of licenses of similar reactor type and containment design configurations. The use of representativeness as a plant selection criterion, along with plant type and license age criteria, is consistent with the Review Group's assessment approach for achieving broad results and generic insights through the review of a limited number of licenses. Further, the findings developed from a specific license review were subjected to additional evaluations during subsequent license reviews to validate the results. Upon the completion of all four plant license assessments, an analysis of the resulting data was conducted. This included efforts to correlate the items supporting specific findings with the operating license categories that had been assigned for those items by the review methodology. The results enumerated in Tables 4 and 5 are typical of the data analyzed in this process.

This overall approach to an integrated assessment process was developed with the intent to evaluate the generic applicability of the findings and conclusions reached as a result of the review of the four specific plant licenses. The consistency of the comparative plant-to-plant data compiled in Tables 4 and 5, along with the proportionate distribution of findings within the operating license categories defined in Table 2, provide evidence of the representative nature of the findings. The generic applicability of the findings was also confirmed by the validation process. Based on these results and the insights gleaned from the assessment of four typical plant operating licenses, it is concluded, therefore, that the recommendations provided in this report are generally relevant and directly applicable to most of the operating licenses.

3.3 Findings

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

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

3.3.1 Items That Appear To Exceed Applicable Regulatory Requirements

Nine, or approximately 3 percent, of the 358 items assessed appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the licenses. Almost half of the instances in which items were found to exceed the applicable regulatory requirements involved situations where the plant design was different from that assumed in the Standard Technical Specifications used to develop the plant-specific Technical Specifications. In most cases, a system that has a design not in conformance with a standard plant design appeared to have additional requirements and/or less flexible provisions imposed in its Technical Specifications compared to the Standard Technical Specifications.

A number of the instances in which items were found to exceed the applicable regulatory requirements involved the elevation of provisions of Commission policy statements, regulatory guides, and other non-requirements to the status of legal requirements. While it is recognized that 10 CFR 50.50 authorizes the Commission to include in licenses such conditions as it deems appropriate, the inclusion of these non-requirements into licenses effectively elevates their status to requirements. In many instances, these non-requirements have not had the benefit of the rigorous regulatory review normally associated with the promulgation of requirements.

The remaining instances in which items were found to exceed the applicable regulatory requirements involved cases where the licensee had not taken advantage of the opportunity to eliminate requirements that are no longer required and at least one case where a licensee voluntarily incorporated a non-requirement into the license (e.g., the

Peach Bottom licensee apparently voluntarily adopted the ISEG function, which was not generically required of pre-TMI accident licensees) and incorporated the requirement into the Technical Specifications.

3.3.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

Seventeen, or approximately 5 percent, of the 358 items assessed appear to have the potential for possible reduction in regulatory burden. Approximately half of these items are Technical Specification requirements that, when compared to the Improved Standard Technical Specifications, are unduly prescriptive or no longer required. Such requirements are, therefore, potential candidates for line-item improvements under the Technical Specification Improvement Program.

A few of the items that should be considered for possible reduction in regulatory burden are license conditions, such as physical security or fire protection, which require license amendments for changes that decrease safeguards effectiveness or the ability to achieve cold shutdown in the event or a fire, respectively. The Review Group believes it is appropriate that such changes be approved by the NRC before they are implemented; however, it sees no benefit in having to amend the license in addition.

A number of the items that have the potential for possible reduction in regulatory burden involve duplicative or otherwise unnecessary reporting requirements. These reporting requirements are (1) surrogates for more appropriate corrective actions, (2) duplicative of those required by the Commission's regulations, especially 10 CFR 50.73, or (3) not essential, at least at the frequency specified, to the accomplishment of the agency's mission. While surrogate and duplicative reporting requirements constitute an unnecessary regulatory burden, those that are redundant may have some value as reminders that reports to the NRC may be required, e.g., if a Technical Specification is violated. For these situations, however, the use of notes that reference the applicable regulations may be more appropriate.

3.3.3 Items That Provide Inherent Flexibility

Seventeen, or approximately 5 percent, of the 358 items assessed appear to have at least some inherent flexibility. These items are generally (1) performance-based requirements, which establish desired objectives without prescriptive details, (2) requirements that are prescriptive only at a high level and allow the implementation details to be specified in licensee-controlled documents, or (3) requirements that allow specified changes to be made without prior NRC approval. A number of items reviewed have demonstrated that performance-based requirements can ensure that adequate safety is provided and at the same time provide the licensee with considerable flexibility in meeting the requirement. Such requirements also reduce the regulatory burden on both the licensee and the NRC. The licenses contain only a relatively small number of exemptions from the Commission's regulations. The most common of these exemptions are from Appendix J to 10 CFR 50. However, some exemptions are not reflected in the licenses; the most common of these exemptions are from Appendix R to 10 CFR 50. The limited areas in which exemptions have been granted and, except for Appendix R to 10 CFR 50, the relatively small number of exemptions that have been granted imply that the regulations afford sufficient flexibility to accommodate wide spectra of plant designs and operations. The Review Group notes that both Appendixes J and R to 10 CFR 50 are presently being considered in the Marginal-to-Safety Program.

3.3.4 Items That Should Be Considered For Enhanced Flexibility

Twenty, or approximately 6 percent, of the 358 items assessed appear to have enhanced flexibility potential. The greatest potential for enhancing flexibility, at least in the short term, appears to lie with the Technical Specification Improvement Program. The program provides the opportunity for licensees not only to totally convert their Technical Specifications to the Improved Standard Technical Specifications but also to pursue generically approved line-item improvements. The Review Group believes that the utility of the program can be greatly enhanced by making available line-item improvements for individual licensees.

The next greatest potential for enhancing flexibility appears to be the expanded use of either performance-based requirements or requirements that are prescriptive only at a high level and allow the implementation details to be specified in licensee-controlled documents. As discussed in Section 3.3.3 of this report, these types of requirements have been shown to ensure that adequate safety is provided and at the same time provide the licensee with considerable flexibility in meeting the requirement. Such requirements also reduce the regulatory burden on both the licensee and the NRC.

The third greatest potential for enhancing flexibility appears to be the increased use of risk assessment methodology in both establishing and implementing requirements. Examples include extending completion times and offering a graded approach to safety such as that already allowed by Appendix B to 10 CFR 50.

The use of allowed outage times, such as those provided for safety-related equipment in the Technical Specifications, could provide additional flexibility in areas such as physical security and fire protection. Also, greater flexibility could be provided by redefining the baselines from which changes to physical security plans, emergency plans, quality assurance plans, and fire protection can be made without prior NRC approval. The underlying regulatory requirements would serve as more appropriate baselines for future changes than the plans themselves.

3.3.5 Items Considered or Being Considered in Other Programs

Twenty-two, or approximately 6 percent, of the 358 items assessed have already been or are being considered in other programs. Items were included here if the other programs offer the potential for reduced regulatory burden or enhanced flexibility. The program in which most of these items have been considered is the Technical Specification Improvement Program. Other programs in which items have already been or are being considered include generic letters, rulemaking efforts, and reporting requirement re-evaluation.

3.3.6 Items for Which No Further Consideration Is Warranted

Three hundred and thirty nine, or approximately 95 percent, of the 358 items assessed were judged to have no bases for further consideration. If an item has already been or is being considered in another program and no further consideration is judged to be warranted, that item is also included here.

3.4 Recommendations

In its assessment of the four licenses, as documented in Appendixes A, B, and C to this report, the Review Group identified 11 recommendations. In the process of integrating the results of the individual plant assessments and preparing this report, the Review Group identified an additional recommendation. The 12 recommendations are presented below in the order in which they were identified. As discussed in Section 3.2.4 of this report, the Review Group believes that the recommendations generally apply to all licenses, therefore, their order of presentation is not important. References to the sections in this report in which the recommendations were identified are provided in parentheses.

To improve the clarity and specificity of the recommendations, many of them have been slightly reworded from the way in which they appear in the individual plant assessments. Also, subsequent to the identification of the recommendations, actions have been taken or are being taken to address a number of them. Where such actions have been or are being taken, it is so noted.

It is important to note that the recommendations have the potential to reduce the regulatory burden on and enhance the flexibility available to the licensees. None of the recommendations needs to be implemented to ensure safety. However, since the implementation of the recommendations would result in a reduction in licensee manpower requirements, this excess manpower could indirectly benefit safety if it were redirected to safety-significant work.

- Ensure the past practice of treating certain Commission policy statements, regulatory guides, and other non-requirements as legal requirements by generally including them in the license without following a disciplined process appropriate to the use. (Sec. A.3.2.1).

- Consider developing NRC staff guidance that considers Technical Specification requirements for design features that provide safety margin in excess of NRC requirements, for example, systems that provide additional redundancy. (Sec. A.3.2.1).

- Provide a consistent approach for making changes to "plans," such as the fire protection, physical security, emergency response, and quality assurance plans, within their proper regulatory and safety contexts. Eliminate the regulatory requirement that compliance with physical security plans be imposed by a license condition. (Sec. A.3.2.2).

- Reevaluate the information/data the NRC needs from nuclear power plant licensees in order to accomplish its mission of protecting the health and safety of the public (taking into consideration the efforts of the CRGR and the Reporting Requirements Task Force). Information/data requirements without a clear nexus to that mission and duplicative reporting requirements should be eliminated. (Sec. A.3.2.2).

Subsequent to its identification of this recommendation, the Review Group reevaluated the reporting requirements contained in a number of the Commission's regulations and several operating licenses. That reevaluation and its results are discussed in volume two of the Review Group's report.

- Invite the industry to provide the NRC with candid insights on licensees' reasons for not taking more advantage of the inherent flexibility afforded them. (Sec. A.3.2.3).

In its report on its assessment of the Seabrook license (Appendix A to this report), the Review Group found that many licensees have not taken advantage of the considerable flexibility that is already available to them. The Group listed possible reasons for this and recommended that the industry provide its views on the subject. The Review Group believes the opportunity for the public to comment on this report provides that occasion. The Review Group will consider any comments received on this subject and, therefore, considers that this recommendation will be adequately addressed.

- Provide additional flexibility in the implementation of the physical security plans, such as providing Technical-Specification-type allowed outage times. (Sec. A.3.2.4).

- Adopt a graded approach to limiting conditions for operation and surveillance requirements wherever practicable, and to the implementation of specific review committee functions, e.g., station onsite review committee procedure and design change reviews. The appropriate application of risk assessment methodology could be valuable in establishing both the bounds and direction of such an approach. (Sec. A.3.2.4).

- Eliminate the practice of including fire protection plans and the provisions for making changes thereto as license conditions. (Sec. B.3.2.2).

- Expand the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan. Eliminate the inconsistencies in the change requirements for these plans. (Sec. B.3.2.2).

- Permit line-item improvements in accordance with the Improved Standard Technical Specifications to be made available to individual licensees (on a plant-specific basis) in addition to lead and subsequent plant licensees. (Sec. B.3.2.4).

The Office of Nuclear Reactor Regulation (NRR) plans to establish an organizational structure to deal with individual plant Technical Specification improvement requests. Therefore, the Review Group considers that appropriate action has been initiated to address this recommendation.

- Expand the use of performance-based requirements to supplant prescriptive criteria in license conditions and technical specifications. In items exhibiting inherent flexibility, the functional requirement is distinguishable from the technical details needed to implement that requirement. As evidenced in the Technical Specification Improvement Program, licensee-controlled programs that govern such implementation details can provide both flexibility and the requisite assurance of system functionality. (Sec. C.3.2.4).

- The licensees should conduct a comprehensive and thorough assessment of their own licenses to identify any items that have the potential for reducing regulatory burden or enhancing flexibility without decreasing the current level of safety. The licensees should inform NRR of any license changes that they would likely pursue and the schedules on which they would pursue them. NRR should consider this information in view of its other regulatorily-mandated work before it decides whether to redirect additional resources to this effort.

3.5 Other Considerations

The Review Group's recommendations, provided in Section 3.4 of this report, have the potential to both reduce the regulatory burden on and enhance the flexibility available to licensees without reducing the current level of safety. Implementation of these recommendations would require that NRR redirect some of its resources not only to deal with the license amendment requests that would be needed to revise the licenses but also to modify some of its own processes. Some of these processes would require rulemaking before license amendments could be issued.

Because NRR's workload has for some time exceeded its resources, it has established a system for setting priorities for its work. License amendments such as those that would result from the Review Group's recommendations and that are not needed to ensure safety are of the lowest priority. Therefore, in order for NRR to implement many of the Review Group's recommendations, it would have to redirect some of its resources away from what is now higher priority work.

The Review Group's recommendations were made without regard to either the magnitude of potential licensee requests for such changes or NRR's ability to accommodate the requests. It would be imprudent for NRR to consider redirecting its resources to this endeavor without first knowing the expected licensee response. Likewise, it would be imprudent for the licensees to proceed with their license amendment requests without the assurance that NRR would be able to process them.

Since the licensees would be the primary beneficiaries of the license changes, the Review Group believes that before NRR considers whether to implement its recommendations, the burden is properly on the licensees to inform NRR of the license changes that they would likely pursue and the schedules on which they would pursue them. In addition, the licensees have an obligation to provide submittals of a quality necessary to support the requests.

These aspects are discussed in more detail in the following sections.

3.5.1 Licensee Burden

The Review Group's recommendations were based on its assessment of a selected number of items in a representative sample of licenses. To determine whether such license changes would be worthwhile, each licensee should conduct a comprehensive and thorough assessment of its own license. Some of the methodology used by the Review Group may prove useful in any such endeavor.

The Review Group found that many licensees have not taken advantage of the considerable flexibility that is already available to them. It is not clear whether they are unaware of this flexibility potential or whether they have chosen to not pursue the changes for other reasons. Nevertheless, it is up to the individual licensee to determine whether changes should be pursued.

In order for NRR to be able to make an informed decision on whether to redirect some of its resources, it must have a reasonable idea of the changes that would be requested by the licensees, including those that would require rulemaking to implement. The information needed includes not only the number of licensees that are expected to pursue changes but also the numbers and types of changes and the schedules on which they would be requested. The Review Group believes the collection of this information could be accomplished most efficiently through a representative licensee organization.

It is imperative that the licensees' amendment requests be of a quality sufficient to avoid the need for NRR to request additional information. Quality submittals can be ensured in several ways. First, the licensees should clearly address the pertinent regulatory requirements and the safety relevance of their request. Secondly, the licensees should try to anticipate the NRC's information needs; they should not just provide a minimal amount of information assuming that if NRR needs more, it will ask for it. Thirdly, the licensees could establish a clearinghouse-like process in which license change requests that have already been approved are made known and readily available to others. A representative licensee organization could effectively provide this function. Finally, pre-application dialogue with the cognizant NRR projects and review personnel can provide valuable insights that can help ensure complete submittals.

3.5.2 NRC Resources

The license changes that would result from the Review Group's recommendations could result in potential resource savings to the licensees. However, substantial upfront investments must be made by both the licensees and the NRC before these savings can be realized. As discussed previously, the resource impact on both the NRC and the licensees must be considered in establishing an efficient process for handling license amendment requests as well as in implementing many of the Review Group's recommendations, particularly those that would require rulemaking. The successful implementation of such changes and process modifications is, therefore, dependent on the effective communication of the expected response of the licensees to the NRC. In addition, if substantially more flexibility is provided to the licensees by allowing implementing details to be relocated to licensee-controlled documents, NRR would have to augment its inspection program, perhaps in the form of additional performance-based inspections as part of its core inspection program, to ensure that the current level of safety is maintained.

Since NRR would have to redirect some of its resources to this effort, it will have to consider not only how many resources would be needed to implement the Review Group's recommendations but also the impact that resource redirection would have on its currently higher priority work. NRR is currently devoting approximately 75 percent of its headquarters resources to operating reactor support and 25 percent to areas such as advanced reactors and plant license renewal. Of those resources that are being devoted to operating reactor support, only about one-fifth are being spent on processing operating license amendment requests. In assessing both the need for and impact of future resource allocations, NRR will have to consider the sometimes competing interests of the licensees and other segments of the industry as well as its own regulatory mandate. The Review Group believes that comments on this aspect from both the licensees and other segments of the industry, and the public in general would be beneficial.

Finally, the Review Group makes no recommendations concerning the details of how its recommendations should be implemented, whether NRR should redirect some of its resources to this effort, and, if so, where those resources should come from and how they should be used. These details are properly the prerogative of NRR.

3.6 Summary and Conclusions

The Review Group assessed the licenses of four plants that it believes are representative of a substantial fraction of the total population of licenses. The Review Group found that the licenses provide not only considerable flexibility to the licensees, but also the potential for further reducing the regulatory burden and enhancing the flexibility for licensees without adversely affecting the current level of safety. With a few exceptions, operating license conditions were noted to generally have a clear relevance to safety and sound regulatory bases. This observation is supported by the Review Group's finding that the number of license items that have either inherent flexibility or enhanced flexibility potential are much larger than the number of items that exceed the applicable regulatory requirements. Also, this suggests that the greatest potential for reducing regulatory burden lies in pursuing additional flexibility instead of making changes to the underlying regulatory requirements.

The Review Group also found that many licensees have not taken advantage of the flexibility that is already available to them. Achieving this additional flexibility through the adoption of the Technical Specification Improvement Program initiatives and other generic guidance appears to represent a significant benefit that is readily attainable at the present time. However, the expenditure of substantial resources devoted to the submittal and processing of license amendments is the upfront cost to both licensees and the NRC. The implementation of such a large effort would also likely adversely impact NRR's other currently higher priority work.

Before making the necessary investment in resources to adequately support such a project, NRR must know whether licensees would take advantage of the enhanced flexibility if the process for achieving it were made more readily available to them. Since the licensees are the primary beneficiaries of amendments that would add flexibility to the licenses, the burden is properly on them to inform NRR of their intentions in this regard. If NRR decides to redirect some of its resources to this effort, it is also incumbent on the licensees to provide license amendment requests of a quality necessary to support the changes.

Finally, while all the Review Group's recommendations have the potential to reduce the regulatory burden and enhance the flexibility for licensee implementation of its license requirements, action to adopt certain of these recommendations would entail NRC process modifications, some of which would require rulemaking. Since none of these recommendations are necessary to ensure safety, the resource impact and other implementing ramifications should be assessed by the affected NRC staff organizations to determine if the perceived benefits to the industry are cost effective. The Review Group notes that action has been initiated or already taken to address a number of the recommendations. While it is believed that the recommendations are generally relevant and directly applicable to most of the existing plant operating licenses, the Review Group recognizes that any decision to initiate actions on the remaining recommendations will be made based on other cogent considerations that could outweigh the findings and conclusions presented in this report.

APPENDIX A

ASSESSMENT OF OPERATING LICENSE

SEABROOK UNIT 1

U. S. Nuclear Regulatory Commission

Regulatory Review Group

February 1993

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A.1 INTRODUCTION

A.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.

A.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 charter and the number needed to be representative of a significant number of plants (licenses).

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

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

Using the above criteria, Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2 were selected from among all the plants currently licensed to operate.

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

Surry was selected because it is one of the earliest licensed PWRs; it is a Westinghouse three-loop plant and is, therefore, representative of a significant number of plants

(licenses); and it has an IPE whose review by the NRC is nearly complete. Surry 1 is also one of the plants evaluated in WASH-1400 and NUREG-1150.

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

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

A.1.3 Assessment Approach

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

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

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

The items were reviewed to determine which categories contained items with similar enough characteristics to be assessed collectively. The items in the remaining categories were then considered to determine the percentage that could be assessed individually.

That percentage was apportioned among the remaining categories and determined the number of items to be assessed in each category.

The items that were to be assessed individually were selected from the remaining categories. The items were selected 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 A.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 that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

The overall results were integrated and the recommendations developed.

Table A.1

SUMMARY OF ASSESSMENT APPROACH

1. Review each operating license item and assign it to a category.

2. Determine which categories contain items that are appropriate to be assessed collectively.

3. Determine which items from the remaining categories will be assessed individually.

4. Assess items in accordance with specified questions; analyze items as necessary.

5. Prepare assessment summaries.

6. Integrate overall results, and develop findings and recommendations.

Table A.2

CATEGORIES OF ITEMS

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

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

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

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

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

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

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

Table A.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)?

   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 that allows the licensee a tradeoff of items without a reduction in overall safety?

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

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

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

TABLE A.3 (Continued)

ASSESSMENT QUESTIONS

4. <u>Enhanced Flexibility Potential</u>

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

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

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

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

   E. Are there NRC programs currently ongoing or under evaluation for implementation that would provide enhanced flexibility to the licensee?

A.2 ASSESSMENT OF SEABROOK OPERATING LICENSE

A.2.1 Seabrook License

The Seabrook Unit 1 operating license was issued on March 15, 1990. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Protection Plan, which is Appendix B to the license. The license as reviewed had been amended through Amendment 11, dated May 29, 1992.

A.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 A.2. The numbers of items in the Seabrook operating license by category are shown in Table A.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 10 percent or 25 of the 254 remaining items. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or five of the 50 items in Category D would be selected for further assessment. With the 77 items that would be assessed collectively, this meant that 102 or approximately 31 percent of the 331 total items would be assessed either collectively or individually.

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

Each item was assessed either collectively or individually as appropriate by considering the answers to specified questions presented in Table A.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.

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

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

The overall results were integrated and the recommendations developed.

A.2.3 Results of Assessment

The summaries of the assessments of each of the items are provided in the attachment to this appendix. The summaries are presented in the order of the categories into which each of the items was assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; next, in the Technical Specifications (TS); and, finally, in the Environmental Protection Plan (EP).

Table A.4

SEABROOK OPERATING LICENSE ITEMS BY CATEGORY

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

Table A.5

INDEX OF SEABROOK OPERATING LICENSE ITEMS ASSESSED

Table A.5 (Continued)

INDEX OF SEABROOK OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category D (5 of 50)**	
TS 6.2.2.a	Minimum shift crew composition	A-51
TS 6.2.3.2	ISEG composition	A-52
TS 6.4.1.7	SORC responsibilities	A-53
TS 6.7.3	Temporary changes of procedures	A-55
EP 3.1	Changes in design and operation	A-57
	Category E (4 of 36)**	
OL 2.G	Violation reporting condition	A-58
TS 3.3.3.4	Meteorological instrumentation	A-59
TS 6.4.1.8	SORC records	A-61
TS 6.8.1.5	Monthly operating reports	A-62
	Category F (10 of 10)***	
OL 2.A	Applicability condition	A-63
TS 5.1.1	Exclusion area	A-63
TS 5.1.2	Low population zone	A-63
TS 5.1.3	Unrestricted areas	A-63
TS 5.2.1	Containment configuration	A-63
TS 5.2.2	Containment design pressure and temperature	A-63
TS 5.3.1	Reactor fuel assemblies	A-63
TS 5.3.2	Reactor control rod assemblies	A-63
TS 5.4.2	Reactor coolant system volume	A-63
TS 5.5.1	Meteorological tower location	A-63
	Category G (63 of 63)***	
OL 1.A	Finding - application	A-64
OL 1.B	Finding - construction completion	A-64

Table A.5 (Continued)

INDEX OF SEABROOK OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
OL 1.C	Finding - conformance with requirements	A-64
OL 1.D	Finding - reasonable assurance	A-64
OL 1.E	Finding - technical qualification	A-64
OL 1.F	Finding - financial protection	A-64
OL 1.G	Finding - issuance of license	A-64
OL 1.H	Finding - satisfaction of requirements	A-64
OL 1.I	Finding - nuclear material	A-64
OL 2.B.1	Authorization - possess, use and operate	A-64
OL 2.B.2	Authorization - possess	A-64
OL 2.D	Exemptions	A-64
TS 1.0	Technical Specification definitions (48 items)	A-64
EP 1.0	Objectives	A-64
EP 4.2.2	Terrestrial monitoring condition	A-64
EP 4.2.3	Noise monitoring condition	A-64

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

* Page number of assessment summary in the attachment to this appendix.

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

*** Items that were assessed collectively; all others were assessed individually.

A.3 ASSESSMENT FINDINGS AND RECOMMENDATIONS

A.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 that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

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

A.3.2 Findings and Recommendations

A.3.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

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

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

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

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

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

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

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

<u>Recommendations</u>: Based on the foregoing, the Review Group recommends the following:

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

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

A.3.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

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

OL 2.E	Physical security condition
TS 3.3.3.3	Seismic instrumentation
TS 3.3.3.4	Meteorological instrumentation
TS 6.8.1.5	Monthly operating reports

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

Recommendations: Based on the foregoing, the Review Group recommends the following:

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

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

A.3.2.3 Items That Provide Inherent Flexibility

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

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

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

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

Of the six items with inherent flexibility, one item--physical security--was also judged to have potential for reduction in regulatory burden and enhanced flexibility. These aspects are addressed in Sections A.3.2.2 and A.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: Based on the foregoing, the Review Group recommends the following:

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

A.3.2.4 Items That Should Be Considered for Enhanced Flexibility

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

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

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

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

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

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

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

Recommendations: Based on the foregoing, the Review Group recommends the following:

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

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

A.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.

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

A.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 also included here.

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

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

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

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ATTACHMENT TO APPENDIX A

ITEM ASSESSMENT SUMMARIES

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 2.1.2**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.0.3**

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

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

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

It is not clear that the Technical Specification could not be made more flexible. Since not all limiting conditions for operation have the same safety significance, the completion times allowed for achieving hot standby, hot shutdown, and cold shutdown could possibly be made more performance oriented, e.g., by considering situation-specific factors. Further, it may not always be safer to change operational modes. For example, if there is reasonable assurance that the situation could be rectified within 1 hour after the completion time for changing modes expires, it might be safer to maintain the reactor in its present mode for that additional period of time than to change modes.

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.1.2.7**

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

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

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

Based upon the above discussion, it is not clear whether either the prescriptive language of this item or the item itself is a needed Technical Specification requirement. The potential for delaying core alterations (e.g., refueling operations) if the LCO is not met exists. However, any change to enhance the flexibility of the item may not 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 that provide the equivalent assurance of acceptable reactivity controls for similar reactors. More review is required to determine whether revision or elimination of this item from the Seabrook 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 SEABROOK ASSESSMENT

Category: A **Item: TS 3.3.3.3**

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.4.6.2**

Seabrook Technical Specification 3.4.6.2, operational leakage, states that the reactor coolant system leakage shall be limited to the following: no pressure boundary leakage, 1 gpm unidentified leakage, 1 gpm total reactor-to-secondary leakage through the steam generators and 500 gpd through any one steam generator, 10 gpm identified leakage, 40 gpm controlled leakage, and reactor coolant system pressure isolation valve leakages as prescribed by formula and the referenced table. This item was chosen because it is representative of a technical requirement that does not provide flexibility.

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

Maintaining the integrity of the reactor coolant system pressure boundary is a primary safety concern. Consistent with that philosophy, it is necessary to maintain the prescriptive requirements related to the leakage limits currently contained in the Technical Specifications. The only requirement where some flexibility may be permissible is related to the 10 gpm identified leakage limit provided that it could be demonstrated that there would be no reduction in the margin of safety if this limit were increased (i.e., the sensitivity of the leakage detection system was not degraded).

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.5.4**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.6.1.7**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.7.1.2**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.7.4**

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

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

There exists no inherent flexibility in this item. The safety-related cooling tower on site is designed with two independent cooling loops and provides an adequate ultimate heat sink option to the normal service water bay cooling path. Additionally, with two 100% capacity pumps in each loop of the service water cooling path, the loss of one pump in each loop would still provide redundant cooling capability to the normal ultimate heat sink, i.e., the Atlantic Ocean. However, given the above scenario (i.e., cooling tower totally available and each service water path functional, but one pump in each loop out of service), the Seabrook plant is placed in a 3-day action requirement to shutdown. By comparison, a plant upon which the Standard Technical Specification requirement was imposed would only have to take similar action if just one service water loop were operable (versus the four available Seabrook loops posed for the above scenario).

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.8.2.1**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.9.4**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 3.12.2**

Seabrook Technical Specification 3.12.2, land use census, requires that a land use census be conducted and identify within a distance of 5 miles in each of the meteorological sectors the location of the nearest milk animal, the nearest residence, and the nearest garden greater than 500 square feet producing broad-leaf vegetation. This item was chosen because it was representative of requirements contained in the radiological environmental monitoring section of the Technical Specifications.

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: A **Item: TS 5.6.3**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: B **Items: All**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: C   Item: OL 2.E

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: C Item: TS 3.4.10

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

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

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

While prescriptive language is used in this Technical Specification and its referenced documents, i.e., the ASME Code and Regulatory Guide 1.14, such details are needed to provide the appropriate technical criteria. Performance-based criteria are already incorporated into the ASME Code Section XI requirements upon which the plant ISI/IST programs are based. Any attempt to use additional performance-based criteria, beyond the ASME Code provisions, would unnecessarily complicate this Technical Specification. Further NRC review of this area for enhanced flexibility does not appear warranted from a regulatory standpoint. However, from a research and technical standpoint, continued NRC liaison with the ASME Code Section XI committees will continue to provide for program revisions and additional flexibility, if appropriate. It is noted that, with Generic Letter 91-18, further flexibility in the form of NRC Inspection Manual Technical Guidance was provided in this area by allowing continued operation with nonconforming piping/support components until the next refueling outage if certain referenced analytical criteria (e.g., Appendix F of Section III of the ASME Code, NRC Bulletins 79-02 and 79-14) are met. Given that such guidance for continued operation can be supported by quantitative analysis, this Technical Specification currently establishes reasonable and acceptable controls. While no further review of this item is warranted, the use of a Generic Letter 91-18 to add flexibility to this area appears to have been beneficial and this approach could be explored further in other areas.

SUMMARY OF SEABROOK ASSESSMENT

Category: C **Item: TS 6.2.2.e**

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

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: TS 6.2.2.a**

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

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: TS 6.2.3.2**

Seabrook Technical Specification 6.2.3.2, Independent Safety Engineering Group (ISEG) composition, states that the ISEG shall be composed of at least five dedicated, full-time engineers located on site with a science or engineering degree and at least 2 years of experience in the degreed field and 1 year of experience in the nuclear field. This item was chosen because of its very prescriptive nature with regard to manpower requirements.

This requirement is based on TMI Action Plan Item I.B.1.2 contained in NUREG- 0737. This particular item was required of applicants for operating licenses only. The purpose of ISEG is to perform independent reviews and audits of plant activities, review other appropriate internal and external information available, and provide recommendations to management where useful improvements can be made. Other than the scope of issues that ISEG reviews, the licensee has nc control over the utilization of the five dedicated plant staff assigned to this function. The Improved Standard Technical Specifications permit the ISEG function to be performed under the review and audit program. This permits more flexible methods of performing the ISEG function (i.e., by a standing committee or by assigning qualified individuals capable of conducting these reviews and audits).

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

Based on the above considerations, it is concluded that the Seabrook Technical Specification requirement related to the composition of ISEG provides little flexibility. However, a Technical Specification change can be submitted adopting the Improved Standard Technical Specification approach; that would provide considerable flexibility in the implementation of this requirement. Based on the viable alternative available, it is concluded that further consideration of this requirement would be unproductive.

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: TS 6.4.1.7**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: TS 6.7.3**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: D **Item: EP 3.1**

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: E **Item: OL 2.G**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: E **Item: TS 3.3.3.4**

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

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

The safety relevance of this item is clearly established by the significance that correct and timely meteorological information has in proper dose assessments and emergency planning decisions. However, the consistency and safety significance of the action requirement of this Technical Specification is not readily evident. Given the inoperability of certain meteorological monitoring instrumentation, the action statement requires the licensee to submit a special report to the NRC outlining the cause of the malfunction and the plans for restoration. Such a reporting requirement within a 10-day deadline after an allowable outage time of 7 days appears to be inconsistent with the fact that, in accordance with the Seabrook Station Emergency Response Manual, an Unusual Event would have to be declared if certain categories of meteorological data (e.g., wind speed) became unavailable.

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: E | Item: TS 6.4.1.8

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: E **Item: TS 6.8.1.5**

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

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: F **Items: All**

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

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

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

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

SUMMARY OF SEABROOK ASSESSMENT

Category: G **Items: All**

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

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

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

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

APPENDIX B

ASSESSMENT OF OPERATING LICENSE

SURRY UNIT 1

U. S. Nuclear Regulatory Commission

Regulatory Review Group

March 1993

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B.1 INTRODUCTION

B.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.

B.1.2 Selection of Plants (Licenses)

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

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

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

Using the above criteria, Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2 were selected from among all the plants currently licensed to operate.

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

Surry was selected because it is one of the earliest licensed PWRs; it is a Westinghouse three-loop plant and is, therefore, representative of a significant number of plants

(licenses); and it has an IPE whose review by the NRC is nearly complete. Surry 1 is also one of the plants evaluated in WASH-1400 and NUREG-1150.

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

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

B.1.3 Assessment Approach

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

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

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

The items were reviewed to determine which categories contained items with similar enough characteristics to be assessed collectively. The items in the remaining categories were 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 that were to be assessed individually were selected from the remaining categories. The items were selected because of their representativeness of a significant number of other items in the category, because of 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 B.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 summary contained overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items that were selected

for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

Table B.1

SUMMARY OF ASSESSMENT APPROACH

1. Review each operating license item and assign it to a category.

2. Determine which categories contain items that are appropriate to be assessed collectively.

3. Determine which items from the remaining categories will be assessed individually.

4. Assess items in accordance with specified questions; analyze items as necessary.

5. Prepare assessment summaries.

6. Validate results from assessment(s) of previous license(s).

7. Integrate overall results, and develop findings and recommendations.

Table B.2

CATEGORIES OF ITEMS

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

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

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

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

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

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

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

Table B.3

ASSESSMENT QUESTIONS

1. <u>Regulatory Bases</u>

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

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

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

2. <u>Safety Relevance</u>

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

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

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

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

3. <u>Inherent Flexibility</u>

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

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

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

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

Table B.3 (Continued)

ASSESSMENT QUESTIONS

4. <u>Enhanced Flexibility Potential</u>

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

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

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

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

E. Are there NRC programs currently ongoing or under evaluation for implementation that would provide enhanced flexibility to the licensee?

B.2 ASSESSMENT OF SURRY OPERATING LICENSE

B.2.1 Surry License

The Surry Unit 1 operating license was issued on May 25, 1972. The operating license consists of the license itself and the Technical Specifications, which are Appendix A to the license. The license as reviewed has been amended through Amendment 170, dated June 1, 1992.

B.2.2 Assessment of License

The Surry operating license contains 192 items. Each of the items was reviewed and assigned to one of the categories in Table B.2. The numbers of items in the Surry operating license by category are shown in Table B.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 44 items or approximately 23 percent of the total number of items.

The number of items in the remaining categories that would be assessed individually was determined to be approximately 10 percent or 15 of the 148 remaining items. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 10 percent, or two of the 20 items in Category D would be selected for further assessment. With the 44 items that would be assessed collectively, this meant that 59 or approximately 31 percent of the 192 total items would be assessed either collectively or individually.

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

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table B.3. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the Surry license, certain items that were assessed previously in the Seabrook license were compared to the corresponding items in the Surry license in order to validate the results of the Seabrook assessment. The items that were selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

B.2.3 Results of Assessment

The assessment summaries for each of the items are provided in the attachment to this appendix. The summaries are presented in the order of the categories into which each of the items was assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; then, in the Technical Specifications (TS).

The overall findings and recommendations are presented in Section B.3.

Table B.4

SURRY OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	93
B. Non-Technical License Conditions	1
C. License Conditions That Rely on Other Documents for Requirements	16
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	20
E. Reporting and Recordkeeping Requirements	19
F. Unique Plant Features	9
G. Other	34
Total	192

Table B.5

INDEX OF SURRY OPERATING LICENSE ITEMS ASSESSED

Table B.5 (Continued)

INDEX OF SURRY OPERATING LICENSE ITEMS ASSESSED

OL = Operating license condition
TS = Technical Specification

* Page number of assessment summary in the attachment to this appendix.

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

*** Items that were assessed collectively; all others were assessed individually.

B.3 ASSESSMENT FINDINGS AND RECOMMENDATIONS

B.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 that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Item TS 3.2.D, chemical and volume control system, appears in three groups. The item appears to have the potential for possible reduction of regulatory burden; it appears to have enhanced flexibility potential; and it has already been or is being considered in another program.

B.3.2 Findings and Recommendations

B.3.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

Findings: None of the items assessed appears to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the Surry operating license. However, the restrictive provision of TS 3.2.D, chemical and volume control system, may exceed the regulatory intent. While the boration capability of the chemical and volume control system design is appropriately based in 10 CFR 50, the specification of rigid operability requirements on all components in the boration flow paths appears to have less foundation. Given that the definition of "operable" in TS 1.D includes support system functionality, the imposition of shutdown provisions for inoperable heat tracing circuits appears to be not only redundant but possibly unwarranted since the intended safety function can be fulfilled by other means.

In its assessment of the Seabrook license, the Review Group found seven items that appear to exceed the applicable regulatory requirements. To validate the Seabrook results, these seven items were also reviewed for the Surry license. Four of the Seabrook items--TS 3.1.2.7, isolation of unborated water sources; TS 6.2.2.a, minimum shift crew composition; TS 6.2.2.e, station staff working hours; and TS 6.8.1.5, monthly operating reports--appear to exceed the applicable regulatory requirements for Surry in the same manner as for Seabrook. Therefore, for these four items, the Seabrook findings also apply to Surry. The three remaining Seabrook items--TS 3.7.1.2, auxiliary feedwater system; TS 3.7.4, service water system; and TS 3.8.2.1, D.C. electrical power system--do not appear to exceed the applicable regulatory requirements for Surry.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license.

B.3.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

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

OL 3.I	Fire protection condition
TS 3.2.D	Chemical and volume control system
TS 3.8.A	Containment integrity and operating pressure
TS 3.12.B.7	Power distribution limits
TS 6.3.A	Action to be taken if a safety limit is exceeded

Although the Surry fire protection license condition, OL 3.I, is atypical, licenses generally contain fire protection conditions that require license amendments for changes to the fire protection plans that adversely affect the ability to achieve and maintain shutdown in the event of a fire. While the need to obtain NRC approval for such changes is evident, it is not clear why a license amendment is necessary. Consideration should be given to eliminating the practice of including fire protection plans and the provisions for making changes thereto as license conditions. In addition, consideration should be given to expanding the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan, and eliminating the inconsistencies in the change requirements for these plans.

Technical Specification 3.2.D represents a surrogate item for core reactivity control. Redundancies were identified in the handling of certain chemical and volume control system components in accordance with the additional requirements of the safety injection system. Such redundancy is not required by the Improved Standard Technical Specifications. Likewise for TS 6.3.A, a redundancy and conflict in the reporting details was identified between this item and the 10 CFR 50.72 and 50.73 reporting requirements. Not only to eliminate any dual reporting but also to provide consistency with 10 CFR 50.36 requirements for safety limit violations, the language of Technical Specification 6.3.A could be modified to reference the above regulations and organized in a more coordinated manner with the safety limit sections of the Technical Specifications.

Technical Specification 3.8.A retains a listing of containment isolation valves. Generic Letter 91-08 provides guidance for preparing a license amendment to remove component lists, such as containment isolation valves, from the Technical Specifications. Therefore, this list could be considered for possible removal.

Technical Specification 3.12.B.7, quadrant power tilt ratio, imposes reporting requirements without time limitations in addition to corrective actions. Further evaluation revealed that this reporting requirement does not appear in the Standard Technical Specifications. Therefore, this requirement could be considered for possible elimination. In its assessment of the Seabrook license, the Review Group found four items that appear to have the potential for possible reduction in regulatory burden. To validate the Seabrook results, these four items were also reviewed for the Surry license. Two of the Seabrook items--OL 2.E, physical security condition, and TS 6.8.1.5, monthly operating reports--should be considered for possible reduction in regulatory burden in the same manner as for Seabrook. Therefore, for these two items, the Seabrook findings also apply to Surry. The two remaining Seabrook items--TS 3.3.3.3, seismic instrumentation, and TS 3.3.3.4, meteorological instrumentation--do not apply to Surry.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license and, in addition, recommends the following:

- Reconsider the practice of including fire protection plans and the provisions for making changes thereto as license conditions.

- Expand the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan. Eliminate the inconsistencies in the change requirements for these plans.

B.3.2.3 Items That Provide Inherent Flexibility

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

OL 3.I	Fire protection condition
OL 3.K	Secondary water chemistry monitoring program
TS 3.5.B	Residual heat removal system
TS 4.0.2	General surveillance requirement
TS 6.4.K	Systems integrity

License Condition OL 3.I provides inherent flexibility in that except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is required for changes to the fire protection plan. However, it is noted that the Surry fire protection license condition is atypical in that licenses generally contain fire protection conditions that require license amendments for changes to the fire protection plans that adversely affect the ability to achieve and maintain shutdown in the event of a fire.

License Condition OL 3.K allows the tailoring of secondary water chemistry programmatic controls to site-specific details and industry guidelines. Major program changes can be handled by the 10 CFR 50.59 process.

Inherent flexibility is also found in TS 3.5.B, given both the allowable outage time and the conditional nature of the need for an operable RHR system depending upon the existing plant situation.

The requirement to perform surveillances at specified time intervals as contained in TS 4.0.2 provides flexibility by permitting some adjustment of these time intervals to accommodate normal test schedules consistent with the guidance contained in Generic Letter 89-14 and the Improved Standard Technical Specifications.

Technical Specification 6.4.K allows the licensee to establish its own program of compliance, coordinated with the ASME Code Section XI provisions, and is not constrained by prescriptive leakage criteria.

In its assessment of the Seabrook license, the Review Group found six items that have at least some inherent flexibility. To validate the Seabrook results, these six items were also reviewed for the Surry license. Five of the Seabrook items--OL 2.E, physical security condition; TS 3.4.10, structural integrity; TS 3.9.4, containment building penetrations; TS 6.2.2.a, minimum shift crew composition; and TS 6.2.2.e, station staff working hours--have at least some inherent flexibility for Surry in the same manner as for Seabrook. Therefore, for these five items, the Seabrook findings also apply to Surry. The remaining Seabrook item--TS 3.12.2, land use census--does not apply to Surry.

Recommendation: Based on the foregoing, the Review Group reaffirms its recommendation in this area from its assessment of the Seabrook license.

B.3.2.4 Items That Should Be Considered for Enhanced Flexibility

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

TS 3.2.D	Chemical and volume control system
TS 3.5.B	Residual heat removal system
TS 3.16.B	Emergency power system
TS 6.1.C.2	Management safety review committee

Technical Specification 3.2.D, as a surrogate for reactivity control, could be made more flexible with respect to required boration activities by addressing the shutdown margin aspect without prescribing conditions on the boration flow paths and components.

For Technical Specification 3.5.B, the insights provided by the application of risk assessment methodology to the evaluation of residual heat removal system operability and shutdown constraints would appear to be valuable in avoiding unnecessary plant transients.

Technical Specification 3.16.B, emergency power, also appears to be an area where the application of risk-based methodology could be used to evaluate the relative risk associated with continued plant operation for limited periods of time compared to shutting down and starting up.

Technical Specification 6.1.C.2, Management Safety Review Committee, appears to be overly prescriptive in that it requires the MSRC to provide the same level of consideration to required procedures and all proposed changes to station systems or equipment that affect nuclear safety. A more performance-based or graded approach that takes into account the relative safety significance of the different areas and items under review would provide additional flexibility. This is also an area in which the application of risk assessment methodology could be considered.

In its assessment of the Seabrook license, the Review Group found six items that have enhanced flexibility potential. To validate the Seabrook results, these six items were also reviewed for the Surry license. Five of the Seabrook items--OL 2.E, physical security condition; TS 3.0.3, general limiting condition for operation; TS 3.6.1.7, containment ventilation system; TS 6.4.1.7, SORC responsibilities; and TS 6.7.3, temporary changes of procedures--have enhanced flexibility potential for Surry in the same manner as for Seabrook. Therefore, for these five items, the Seabrook findings also apply to Surry. The remaining Seabrook item--TS 6.2.3.2, ISEG composition--does not apply to Surry.

Although the review of the Surry license validated the Seabrook results for five out of the six items for which enhanced flexibility potential was identified, significant differences in the age, organization, and functional structure of the Technical Specifications for these two plants were found. These differences could have a direct impact upon the potential for success of any enhanced flexibility initiatives. Similarly, for the four additional Surry items that were assessed in this group, the regulatory philosophy that underlies the Surry

Technical Specifications reflects an approach to component and system controls quite different from that of both the Improved and Standard Technical Specifications. Therefore, even though the Technical Specification Improvement Program provides a mechanism for facilitating increased flexibility and reduced plant operational restrictions, effecting a transition to the Improved Standard Technical Specifications would be more difficult for Surry than for Seabrook.

However, for individual line items, the Surry licensee may be able to justify some of the enhancement options provided by the Technical Specification Improvement Program. Attempts to gain further flexibility by means of significant revisions to the Technical Specifications would have to be weighed by the licensee against the constraints of the plant-specific design and system-based limitations that exist for Surry, as well as for other older plants.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessment of the Seabrook license and, in addition, recommends the following:

- Consider making the Technical Specification Improvement Program available to individual licensees, in addition to lead and subsequent plant licensees, by allowing line-item improvements to be made on a plant-specific basis in addition to the present generic basis.

B.3.2.5 Items Considered or Being Considered in Other Programs

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

TS 3.2.D	Chemical and volume control system
TS 3.4.C	Spray systems
TS 3.8.A	Containment integrity and operating pressure
TS 6.3.A	Action to be taken if a safety limit is exceeded

Technical Specifications 3.2.D, 3.4.C, and 6.3.A have been replaced by better organizationally and technically structured items in the Improved Standard Technical Specifications. The noted revisions eliminate redundancy and establish a more coordinated functional relationship between the safety intent of each Technical Specification and the conditions and surveillances required to meet that intent.

The list of containment isolation valves contained in Technical Specification 3.8.A has already been considered and can be eliminated in accordance with the guidance contained in Generic Letter 91-08.

Recommendation: Based on the foregoing, the Review Group has no recommendations in this area.

B.3.2.6 Items for Which No Further Consideration Is Warranted

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

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

OL 2.C	Nuclear materials condition
OL 3.K	Secondary water chemistry monitoring program
TS 3.4.C	Spray systems
TS 3.8.A	Containment integrity and operating pressure
TS 3.19	Main control room bottled air system
TS 4.0.2	General surveillance requirement
TS 5.4.C	Fuel storage
TS 6.4.K	Systems integrity
Cat. B item	Non-technical license conditions (1 item)
Cat. F items	Unique plant features (9 items)
Cat. G items	Other (34 items)

Recommendation: Based on the foregoing, the Review Group has no recommendations in this area.

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ATTACHMENT TO APPENDIX B

ITEM ASSESSMENT SUMMARIES

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: OL 3.K**

Surry License Condition 3.K, secondary water chemistry monitoring program, requires the licensee to implement a monitoring program of secondary side water to inhibit steam generator tube degradation. General provisions are listed in OL 3.K regarding what this monitoring program shall include. This item was selected for review because it represents a technical requirement incorporated into the operating license, rather than the Technical Specifications, by license amendment.

This item has regulatory basis in the General Design Criteria of Appendix A to 10 CFR 50, which address the integrity of the reactor coolant pressure boundary (e.g., steam generator tubes) and discuss design margins to limit leakage during all operating and accident conditions. The use of secondary water chemistry controls to prevent degradation of the steam generator tubes, representing the barrier between primary and secondary coolant, is a surrogate item. In turn, a secondary water chemistry monitoring program can be viewed as a surrogate for the maintenance of adequate chemistry parameters. In the mid-1970's, such secondary water chemistry parameters were incorporated directly into plant Technical Specifications as limiting conditions for operation with associated surveillance requirements. However, such restrictive requirements were considered to be hindrances to operational flexibility without the realization of commensurate benefits in limiting steam generator tube degradation. Hence, the development of plant-specific secondary water chemistry monitoring and control programs was judged to provide a more effective approach to the same overall goal. The inclusion of such surrogate programs as license conditions appears to have been technically sound as an alternative regulatory approach.

The goal of minimizing steam generator, including reactor coolant pressure boundary, degradation is clearly relevant to safety. Branch Technical Position MTEB 5-3 of the Standard Review Plan (NUREG-0800) provides guidance for secondary water chemistry monitoring and indicates that the NRC will review the individual monitoring program for each plant. It also recommends that steam generator vendor recommendations should be incorporated in the technical requirements of each individual program. This allows a licensee to tailor its programmatic controls to the site-specific design features and needs, while at the same time using industry, e.g., Electric Power Research Institute (EPRI), guidelines where appropriate. Therefore, this Surry license condition has a measure of inherent flexibility, not in the stipulation that a secondary water chemistry monitoring program is required, but rather in the licensee's own development of the implementation details.

Both the Standard and Improved Standard Technical Specifications have incorporated language similar to this Surry item into the appropriate administrative controls section. For Surry, this programmatic requirement could also be more appropriately delineated as an administrative control (similar to item TS 6.4.K) rather than an operating license condition.

It should be noted that individual plant secondary water chemistry control and monitoring programs written to comply with Westinghouse and EPRI guidelines require plant shutdowns if certain action level chemistry limits are exceeded. Such shutdown provisions are not NRC conditions or Technical Specification requirements, but licensee self-imposed controls to not only ensure safe operation, but also maximize overall long-term plant reliability. While major changes to the plant secondary water chemistry program would require processing in accordance with 10 CFR 50.59 provisions, licensee ownership of the implementation details of this operating license condition allows sufficient flexibility for licensees to adequately develop and manage the required programs.

Based on the above considerations, it is concluded that this operating license condition is both appropriate and not unduly restrictive. Although this condition might be more appropriately characterized as an administrative control of the Technical Specifications, it is concluded that further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.2.D**

Surry Technical Specification 3.2.D, chemical and volume control system (CVCS), requires reactor shutdown if specific conditions related to system and component inoperability are not met within stated times. This item applies to one and two-unit operation and takes into consideration the availability of shared systems and commc components. This Technical Specification was selected for review because it represents a system-oriented approach to delineating reactivity controls, somewhat different from the functional (i.e., boration) orientation of newer Technical Specifications.

This item has regulatory basis in the reactivity control and limits criteria of 10 CFR 50, Appendix A. It has clear safety relevance. Given the dual function of certain components (e.g., charging pumps) to perform both emergency core cooling and CVCS roles, a Technical Specification redundancy is recognized and addressed by the provisions of this item. While some inherent flexibility is afforded the licensee by the allowances for inoperable equipment, particularly where components from the opposite unit are available, the item is restrictive not only in its action statements, but also in its stipulated controls over equipment (e.g., heat tracing circuits) with a marginal relation to safety. However, such restrictiveness does not extend to the few direct surveillance requirements that were found to be associated with this Technical Specification.

While General Design Criterion 26 of 10 CFR 50, Appendix A, mandates the design of a reactivity control system with the characteristics of boration capability found in the CVCS design, the need for a Technical Specification to govern such a function is not so clearly based in the regulations. This item represents a surrogate for adequate core reactivity control. Although system-based Technical Specifications can be and are effectively used as surrogate items to properly control design functions, the restrictive provisions, in the case of Technical Specification 3.2.D, may go beyond regulatory intent. As a counterpoint, the Improved Standard Technical Specifications delineate requirements over core reactivity, shutdown margins, and other criticality constraints without the need for specifications governing the relevant CVCS boration flow path components. While boration is a required action for failure to meet the required shutdown margin of the Improved Standard Technical Specifications, controls over the boration flow path and components are not rigidly prescribed. Emergency core cooling system (ECCS) components are not addressed as such in the Surry Technical Specifications. Nevertheless, equipment like the charging pumps are redundantly addressed in both the CVCS and safety injection specifications. Such redundancy is not required by the Improved Standard Technical Specifications.

Given the Surry-specific design features for a two-unit site with common systems, components, and flow paths, the merits of adopting the enhanced flexibility of the Improved Standard Technical Specifications would require additional licensee review. It appears that the benefit of avoiding shutdown requirements if equipment such as heat tracing becomes inoperable may be worth the additional analysis and review effort. However, with the additional flexibility afforded licensees by Generic Letter 91-18 for ensuring the functional capability of a system or component, each licensee may see different advantages, as well as disadvantages, in an item-by-item comparison of the Improved Standard Technical Specifications to plant-specific license conditions.

For Technical Specification 3.2.D, it appears that the Surry licensee would gain not only enhanced flexibility but also a reduction in regulatory burden with the application of Improved Standard Technical Specifications to this item. The impact of such implementation upon the dual-unit system design features and also upon the other technically related areas (e.g., ECCS) warrants further review.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.4.C

Surry Technical Specification 3.4.C, spray systems, requires that the containment pressure and temperature parameters be maintained within certain limits, given a high refueling water storage tank temperature, in order to maintain the functional capability of the containment spray system. Since spray capability for containment depressurization during accident conditions is dependent upon the containment pressure and temperature, this item relies upon additional conditions delineated in TS 3.8, containment. This item was selected for review because it is representative of safety system controls and also because of the referencing relationship between TS 3.4.C and TS 3.8.

This item has regulatory basis in the reactor containment criteria of 10 CFR 50, Appendix A, with related technical bases in 10 CFR 50, Appendix K, as the containment pressure affects emergency core cooling system performance and in 10 CFR 50.49 relative to the environmental qualification of affected electrical equipment. Safety relevance is established by the need to place the reactor in cold shutdown conditions if the maintenance of containment pressure and temperature within design limits cannot be ensured by the containment spray function. The referencing provisions of this item to TS 3.8 requirements, while technically restrictive, provide a nexus between the containment design limits and the containment spray system capabilities. It appears that such restrictiveness is necessary based upon Surry design basis considerations.

While some flexibility in complying with the provisions of this item is provided by the licensee's control of different parameters (e.g., lower service water temperatures to cool the containment atmosphere allow for higher containment air pressure), there is little inherent flexibility, in general, in this Technical Specification. The containment spray system, as an engineered safety feature, is required to cool and depressurize the Surry containment to subatmospheric pressure following a design basis accident. Given the restriction to maintain refueling water storage tank temperature below a specific temperature in order to enable design functionality of the containment spray system during accident conditions, the prescriptive language of this Technical Specification appears warranted.

While the Surry Technical Specifications could be improved in this area from an organizational standpoint by a better order of the technical provisions of TS 3.4, spray systems, and TS 3.8, containment, the existing requirements appear technically sound and appropriate. Similarly, although there is no surveillance requirement directly correlatable with TS 3.4.C, the spray system test provisions (TS 4.5) appear consistent with ASME

Code, Section XI, requirements that are prescribed by 10 CFR 50.55a(g). However, again from an organizational standpoint, separation of the containment spray surveillance specifications from the limiting conditions for operation does not appear to be the most effective means of communicating the requirements.

The Improved Standard Technical Specifications currently represent the best integrated presentation of limiting conditions, actions, and surveillance provisions with a discussion of the background, safety analysis, and bases for each requirement. Based on the above considerations, while TS 3.4.C may appear prescriptive, it is necessarily so for sound design basis and safety reasons. Therefore, it is concluded that consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive unless accomplished as part of a broader organizational restructuring of the Surry Technical Specifications.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 3.5.B

Surry Technical Specification 3.5.B, residual heat removal (RHR) system, requires reactor shutdown if specific component operability provisions are not met within a 14-day time period. This item also requires that immediate attention be directed to making repair of the inoperable equipment for the allowed outage time to be applicable. This Technical Specification was selected for review because it is typical of the Surry system-based specifications.

This item has regulatory basis in General Design Criterion (GDC) 34 with pertinence to GDC 35 of Appendix A to 10 CFR 50, as well as in 10 CFR 50.46. Since Surry is an older plant, the emergency core cooling system (ECCS) functions are addressed separately by the safety injection Technical Specifications. Thus, this item specifying RHR component operability requirements governs the functional control of the capability to bring the reactor coolant system (RCS) to cold shutdown conditions under normal shutdown conditions. While the safety relevance of RHR requirements is clearly established, the coherency in the handling of these provisions in the Surry Technical Specifications is not so evident. As an example, the distinction between the need for operable RHR systems is different when the RCS temperature is above 350°F and when the RCS temperature is below this value. However, while TS 3.1.A.1.d.2 requires single RCS loop or RHR loop operation at or below 350°F, Technical Specification 3.5.B fails to differentiate between the different requirements of the different modes of operation.

In effect, the Surry RHR specification allows the licensee to maintain the reactor in hot or intermediate shutdown conditions with RCS temperature greater than 350°F for an indefinite period of time with no apparent constraints on RHR system operability. While this provides flexibility and may in fact be prudent under certain conditions (i.e., if RHR is unavailable, entering conditions where RHR provides the only cooling connection to a heat sink is not advisable), the intent of Technical Specifications is not to cover all operational situations beyond design basis, as is the better defined role of 10 CFR 50.54(x). Additional inherent flexibility of this item derives from the length of the allowable outage time (i.e., 14 days) for an inoperable component rendering one RHR loop out of service.

Therefore, while detailed operability and shutdown provisions are delineated in this Technical Specification, the licensee has flexibility in both the timing and conditional nature of compliance. For the functional requirements of RHR systems in general, risk-based insights may provide the potential for extended allowable outage times to minimize the alternative risk associated with shutdown operations. However, such conclusions must be based upon a detailed, plant-specific Technical Specification analysis. Thus, while the noted inherent flexibility may have technical merit, such a conclusion can only be substantiated by the appropriate application of a risk-based methodology to evaluate this and any similar items.

The Standard and Improved Standard Technical Specification handle RHR requirements with a similar technical approach and both provide a more coherent focus to the cooling functions of the RHR system than is apparent in this Surry item. However, it is noted that Standard Technical Specifications are written to include RHR components in both ECCS and normal shutdown systems, which does not appear fully applicable to the Surry case. Whether the adoption of Improved Standard Technical Specification provisions is consistent with the Surry RHR system design or would be beneficial in providing additional flexibility to the licensee are questions that may merit further review by the licensee. This area is one where performance-based criteria and risk assessment methodology could provide valuable insights from both a safety perspective and an enhanced flexibility potential. While licensee efforts in these areas and their results may not lead to a reduction in regulatory burden, such initiatives may still be worthwhile in order to provide an organizational structure and coherency to the Surry Technical Specifications. Such an approach would also be more consistent with the current regulatory philosophy.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.8.A**

Surry Technical Specification 3.8.A, containment integrity and operating pressure, specifies the conditions of the reactor and reactor coolant system under which containment integrity shall not be violated. In addition, this Technical Specification contains a list of containment isolation valves. This item was selected because of its potential for reduced regulatory burden.

The legal bases for the containment integrity requirement is contained in 10 CFR 50, Appendix A, General Design Criteria 16, 50, 51, 54, and 55, which specify the design criteria for the containment and piping systems that penetrate containment. Standard Review Plan Sections 6.2.1 through 6.2.7 provide the guidance related to this requirement.

The requirements in this Technical Specification are important to safety since they provide assurance that containment integrity will be maintained in the event of an accident. This Technical Specification is prescriptive and affords little flexibility with regard to the conditions under which containment integrity shall be maintained prior to startup and during reactor operation. The Surry Technical Specifications still retain a listing of the containment isolation valves that are contained in Table 3.8.1 for Unit 1. Generic Letter 91-08, "Removal of Component Lists from the Technical Specifications," provides guidance for preparing a license amendment to remove such component lists from the Technical Specifications. Removal of the list of containment isolation valves from the Technical Specifications would afford the licensee additional flexibility and does not alter the existing Technical Specification requirement for those components to which they apply. It is noted that, in response to Generic Letter 84-13, the Surry licensee has removed the list of shock suppressors (snubbers) from the Technical Specifications.

Considering that a mechanism for a reduction in regulatory burden is available to the licensee through Generic Letter 91-08, it is concluded that further consideration of this item would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.16.B**

Surry Technical Specification 3.16.B, emergency power system, specifies the operational requirements with various offsite and emergency power sources and D.C. battery systems either unavailable or inoperable. This item was selected because it is representative of a Technical Specification limiting condition for operation.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory basis in 10 CFR 50, Appendix A, General Design Criterion 17, Electrical Power Systems, which states that an onsite and offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. Standard Review Plan Sections 8.2, 8.3.1, and 8.3.2 provide the guidance related to satisfying this requirement.

This requirement is important to safety in that it specifies the minimum sources of onsite, offsite, and D.C. power that must be available to continue operation and the allowed outage times for this equipment during which continued plant operation is permissible.

The Technical Specification provides no inherent flexibility to the licensee since it is very prescriptive with regard to the actions to be taken in the event electrical power sources are not restored to operability within the allocated time period. However, it is noted that the allowed outage times for these electrical power sources is longer than those permitted by both the Standard and Improved Standard Technical Specifications. The surveillance requirements for the emergency diesels and batteries are also less prescriptive than for plants that use the Standard Technical Specifications.

This may be an area where risk assessment methodology could be applied to evaluate the relative risks associated with the outage times specified for various electrical power sources to determine if they are appropriate compared to the risk associated with shutting down and starting up.

Based on the above considerations, it is concluded that this is an area where enhanced flexibility may be possible through the application of risk-based methodology to compare the relative risks associated with the outage times specified relative to the risks associated with shutting down and starting up.

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 3.19**

Surry Technical Specification 3.19, main control room bottled air system, specifies that a bottled dry air bank shall be available to pressurize the main control room to a positive differential pressure. The Technical Specification also states that this capability shall be demonstrated by testing in accordance with Technical Specification Section 4.1. This item was selected because it is representative of a technical requirement.

The legal requirement for this Technical Specification is contained in 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, Control Room, which specifies that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions. TMI Action Plan Item III.D.3.4 (NUREG-0737), Control Room Habitability Requirements, provides clarification of guidance contained in GDC 19 and references Standard Review Plan Section 6.4 acceptance criteria and Regulatory Guides 1.78 and 1.95 guidance.

This requirement is relevant to safety in that it is necessary to ensure that control room operators will be adequately protected against the effects of accidental releases of toxic and radioactive gasses so the plant can continue to be safely controlled or shut down under a design bases accident condition. As a result, this requirement has very little flexibility wiun regard to shutdown if a minimum positive differential pressure in the control room cannot be achieved and maintained.

The only unique feature of this item is the reference to Technical Specification Section 4.1, which contains the testing requirements to demonstrate the capability to maintain a positive differential pressure in the control room. This is one of the few places in the Surry Technical Specifications where a surveillance requirement is cross-referenced to a limiting condition for operation. The limiting conditions for operation and their associated surveillance requirements are contiguous in both the Standard and Improved Standard Technical Specifications. The latter approach not only facilitates the use of the Technical Specifications but also clearly establishes the nexus between the limiting conditions for operation and their associated surveillance requirements.

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

SUMMARY OF SURRY ASSESSMENT

Category: A **Item: TS 4.0.2**

Surry Technical Specification 4.0.2, general surveillance requirement, states that the specified time intervals may be adjusted plus or minus 25 percent to accommodate normal test schedules. This item was selected as an example of an item that has inherent flexibility.

This item has regulatory bases in 10 CFR 50.55a and Appendix A to 10 CFR 50. This item is important to safety since it ensures that ASME Code Class 1, 2, and 3 components that are maintained in accordance with the inservice inspection and testing programs for the plant and in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, are tested or inspected within the time intervals specified by the code plus or minus 25 percent to accommodate normal test and outage schedules.

This item has inherent flexibility in that it provides a tolerance for extending the surveillance intervals, which is consistent with guidance contained in Generic Letter 89-14, "Line-Item Improvements in Technical Specifications--Removal of the 3.25 Limit on Extending Surveillance Intervals," and the Improved Technical Specification Program.

Based on the above considerations, it is concluded that, since this item has inherent flexibility consistent with current requirements, further consideration of this item for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: A Item: TS 5.4.C

Surry Technical Specification 5.4.C, fuel storage, is a design feature that requires that, when there is spent fuel in the spent fuel pit, the pit shall be filled with borated water at a concentration of not less that 2,300 ppm to match the boron concentration in the reactor cavity and refueling canal during refueling operations. This item was chosen because it is representative of a number of design feature technical requirements.

The regulatory bases for this item are 10 CFR 50.36 and Appendix A to 10 CFR 50. This requirement is important to safety in that it ensures the prevention of criticality in the fuel storage areas and the reactor.

Although the item is prescriptive, it does not appear to be unduly restrictive in view of its safety significance. The item does not appear to have enhanced flexibility potential.

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

SUMMARY OF SURRY ASSESSMENT

Category: B Items: All

The Surry license contains one item in Category B, "Non-Technical License Conditions." The item, OL 4, effective date and expiration condition, specifies the effective and expiration dates of the license. It is required by 10 CFR 50.51.

The item is not directly related to safety. Although the item is prescriptive, it does not appear to be unduly restrictive. The item does not appear to have enhanced flexibility potential.

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

SUMMARY OF SURRY ASSESSMENT

Category: C Item: OL 2.C

Surry License Condition 2.C, nuclear materials condition, authorizes the licensee to receive, possess, and use byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation, and radiation monitoring equipment calibration and as fission detectors.

The incorporation of this condition into the operating license is primarily a convenience for both the licensee and the NRC. Prior to the issuance of the operating license, this and similar nuclear materials authorizations were issued to the licensee in the form of separate nuclear materials licenses. These separate licenses were incorporated into the operating license upon its issuance. Authority to combine such licenses is provided by Section 161(h) of the Atomic Energy Act and 10 CFR 50.52.

The item is not directly related to safety. Although the item is prescriptive, it does not appear to be unduly restrictive. The item does not appear to have enhanced flexibility potential.

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

SUMMARY OF SURRY ASSESSMENT

Category: C Item: OL 3.I

Surry License Condition 3.I, fire protection condition, requires the licensee to complete certain modifications identified in and in accordance with the schedule prescribed in the NRC's fire protection safety evaluation; to submit certain additional information identified in and in accordance with the schedule prescribed in the NRC's fire protection safety evaluation; and to implement the administrative controls identified in the NRC's fire protection safety evaluation and supplements thereto. This item was selected because it is representative of a number of license conditions that rely on other documents for requirements.

The regulatory bases for the fire protection plan are 10 CFR 50.48, Criterion 3 of Appendix A to 10 CFR 50, and Appendix R to 10 CFR 50. Other than the general authority provided by 10 CFR 50.50 to include in the license any conditions that the Commission deems appropriate and necessary, no explicit regulatory basis exists for the license condition. The item is important to safety in that it ensures that structures, systems, and components important to safety are designed and located to minimize the probability and effects of fires.

Except for completing the specified modifications and submitting the specified information, the license condition only requires the licensee to implement and maintain the administrative controls identified in the NRC's fire protection safety evaluation and supplements thereto. Therefore, except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is needed in order for the licensee to make changes to its fire protection plan. Corresponding license conditions for more recent licenses allow the licensees to make changes to their fire protection plans without prior NRC approval provided the changes would not adversely affect the ability to achieve and maintain shutdown in the event of a fire.

Compared to corresponding license conditions for more recent licenses, the Surry license condition is more flexible in that except for changes to the specified administrative controls, neither a license amendment nor prior NRC approval is required for changes to the fire protection plan. More recent licenses generally contain fire protection conditions that require license amendments for fire protection plan changes that adversely affect the ability to achieve and maintain shutdown in the event of a fire. While the need to obtain NRC approval for such changes is evident, it is not clear why the regulatory burden of a license amendment is necessary. Therefore, consideration should be given to eliminating the practice of including fire protection plans and the provisions for making changes thereto as license conditions. In addition, consideration should be given to expanding the scope of 10 CFR 50.54 to include all the "plans" that are required by the Commission's regulations, including the fire protection plan, and eliminating the inconsistencies in the change requirements for these plans.

SUMMARY OF SURRY ASSESSMENT

Category: D **Item: TS 6.1.C.2**

Surry Technical Specification 6.1.C.2, Management Safety Review Committee (MSRC), requires the MSRC to provide independent review and audit of designated safety-related activities and report to and advise the Senior Vice President-Nuclear on its findings related to those activities. This administrative control implements a continuing monitoring activity that provides independent oversight of safety-related station activities. This item was selected as a representative review activity of a committee required by the Technical Specifications.

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

While a certain degree of inherent flexibility exists for the implementation of this item, there is no inherent flexibility in what functions this Technical Specification requires MSRC to accomplish. This prescriptiveness does not appear to be either consistent or commensurate with the intended safety impact because not all of the referenced functions carry the same safety significance. The recording of meeting minutes and reporting requirements also appear somewhat onerous.

It should be noted that the MSRC has only advisory authority in that it recommends and renders determinations; the Senior Vice President-Nuclear has the responsibility for the resolution of any disagreements on overall station operation. Thus the language in this item, which conveys the administrative control of the MSRC requirements, appears to be overly prescriptive and enhanced flexibility could be provided by the use of performance-based criteria or a graded approach to safety-significant review and audit activities. In addition, the use of risk assessment methodology could possibly provide valuable input into the prioritization of MSRC efforts and into the determination of where limited review and audit time could be most effectively directed.

While the safety intent of the MSRC as an independent review and audit authority is soundly based, achieving enhanced flexibility in the administrative control of the MSRC functions would be a worthwhile initiative. The Improved Standard Technical Specifications do not significantly alter the overall MSRC review and audit responsibilities directed by this item. Therefore, it is recommended that further review of this item be conducted to evaluate not only the need for the current prescriptive language of TS 6.1.C.2 but also the prospects for enhanced flexibility by supporting more of a graded safety approach to the MSRC functions.

SUMMARY OF SURRY ASSESSMENT

Category: D **Item: TS 6.4.K**

Surry Technical Specification 6.4.K, systems integrity, requires the licensee to implement a program to reduce leakage from systems outside containment that may contain highly radioactive fluids during plant transient or accident conditions. Some general inspection and leak test requirements are specified. This item was selected for review because it is representative of a number of programmatic control requirements listed in the administrative controls section of the Surry Technical Specifications.

While this item has regulatory basis in the radiation dose limits of 10 CFR 100, the origin of this Technical Specification is more directly related to the TMI Action Plan (NUREGs 0660 and 0737) requirements addressing primary coolant sources and other highly radioactive systems outside containment (i.e., action item III.D.1.1). Additionally, since the primary coolant in PWR plants contains boric acid, the program requested in NRC Generic Letter 88-05 for the control of boric acid leakage has an indirect relation to the program mandated by this item. Any leakage reduction program established to comply with Technical Specification 6.4.K appears then to be safety relevant and to have a sound technical foundation and a coherent regulatory basis.

This item has inherent flexibility not only in generalizing the leakage criteria to be "as low as practical levels" but also in allowing the licensee to establish its own preventive maintenance, visual inspection, and integrated leakage test provisions. Further, a licensee may take credit for any inservice inspection (ISI) functional testing, accomplished in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, to fulfill the system integrity requirements of this Technical Specification. While the inspection/test provisions of any leakage reduction program may serve as surrogate to the system structural integrity assumed in the plant design, the use of such surrogate items is both appropriate and technically acceptable, since the functional capability of the containment structure would not preclude a potential leakage of radiation from the systems outside containment that are addressed by this Technical Specification.

Both the Improved and Standard Technical Specifications have language similar to this Surry item for the administrative control of leakage of primary coolant sources outside containment. While the establishment of such a leakage reduction program is delineated as a rigid requirement, the implementation details of the program can be reasonably and flexibly set by the licensee without direct NRC involvement in programmatic or procedural revisions. Based on the above considerations, it is concluded that this Technical Specification is appropriate and not unduly restrictive. In addition, it is concluded that consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would be unproductive.

SUMMARY OF SURRY ASSESSMENT

Category: E **Item: TS 3.12.B.7**

Surry Technical Specification 3.12.B.7, power distribution limits, specifies what action must be taken if the power tilt ratio exceeds 2 percent for an additional 24 hours after corrective actions have been taken. This item was chosen because it is an example of corrective actions that include notification and reporting requirements.

Standard Review Plan Section 4.3, "Nuclear Design," states that there are no direct or explicit general design criteria for power densities and power distributions allowed during normal reactor operation. The Standard Review Plan also states that the acceptance criteria in the area of power distribution should satisfactorily demonstrate that reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor. This Technical Specification limits the time the allowable quadrant power tilt ratio exceeds 2 percent without the licensee taking corrective actions in the form of a reduction in nuclear overpower and differential overtemperature trip setpoints. The Technical Specification also requires various combinations of differential temperature trip setpoint reduction and notification and reporting requirements depending on whether the design hot channel factors for rated power have not been exceeded or have been exceeded and reactor power is greater than 10 percent or if the hot channel factors have not been determined.

This requirement is important to safety because it provides protection against exceeding fuel design limits by limiting the time the licensee can continue to operate with possible core power distribution asymmetries (while attempting to correct the problem) without the imposition of trip penalties. This aspect of the requirement is very prescriptive and affords no flexibility if the time restrictions are exceeded. However, this requirement is not as prescriptive for plants that use the Standard Technical Specifications. This Technical Specification also requires a special report evaluating the cause of the power tilt to be submitted to the NRC. However, no time limitation for submittal is specified. Similarly, if hot channel factors for rated power are exceeded at power levels greater than 10 percent or if hot channel factors have not been determined, the Technical Specification requires the licensee to notify the NRC. Again, no time limitation for notification is stated. Since this Technical Specification does not contain any time limitations related to reporting and notifications, the safety importance of this requirement appears to be questionable. A check of the Technical Specifications for Seabrook and several other newer plants that use the Standard Technical Specification format indicates that no NRC reporting or notification requirements related to quadrant power tilt ratio is required.

Based on the above considerations, it is concluded that this item warrants further consideration for possible reduction in regulatory burden related to reporting and notification requirements.

SUMMARY OF SURRY ASSESSMENT

Category: E Item: TS 6.3.A

Surry Technical Specification 6.3.A, action to be taken if a safety limit is exceeded, requires placement of the plant in hot shutdown conditions and the conduct of certain reporting actions in the event that a safety limit is violated. The plant-specific safety limits for the reactor core and the reactor coolant system pressure are delineated in Technical Specifications 2.1 and 2.2. Violation of these safety limits results in operating conditions outside the design constraints of the plant. This item was selected for review because the safety limit action statement is listed as an administrative control and also because the reporting requirements appear to be, in part, redundant to other regulatory requirements.

In addition to 10 CFR 50.36, this item has its regulatory bases in several General Design Criteria of Appendix A to 10 CFR 50, as well as in 10 CFR 100. The safety relevance of the reactor shutdown action is established in 10 CFR 50.36(c)(1) as a general regulatory requirement, as is also the need to notify the NRC. However, the language in the Surry Technical Specification 6.3.A for reports to the NRC is less restrictive than 10 CFR 50.72 for immediate notification and more restrictive than 10 CFR 50.73 for the follow-up written notification. Further, the provision in 10 CFR 50.36(c)(1)--after a safety limit is exceeded, the NRC must authorize the resumption of reactor operation--is not included in this Surry item as it is in Westinghouse Standard Technical Specifications.

As discussed above, the wording of this Technical Specification appears inconsistent with the regulations, thereby resulting in the most restrictive requirement governing the actions to be taken. No inherent flexibility is apparent in this item. For the shutdown provision, such prescriptive language is necessary to comply with the regulations. However, the placement of the action statement in an administrative controls section, rather than with the safety limits section, of the Technical Specifications appears inappropriate. Also, the need to restore compliance with the safety limit that has been exceeded is not specifically addressed in conjunction with the shutdown action, as it is in the Improved Standard Technical Specifications.

For the reporting requirements, the Improved Standard Technical Specifications reference 10 CFR 50.72 and 50.73 requirements for NRC reports and stipulate other provisions for internal licensee reporting and reviews. While these requirements are just as prescriptive as the language in the Surry item, the Improved Standard Technical Specifications provide a more consistent and coherent approach to compliance with the regulations. Further, the Safety Limit Violation Report that is rigidly prescribed in the Surry

Technical Specification could be replaced by a Licensee Event Report, submitted in accordance with 10 CFR 50.73.

Based on the above discussion of the relationship established between the requirements of Technical Specification 6.3.A and 10 CFR 50 and also upon the significance of reactor operation within the bounds of the safety limits, further consideration of this item for enhanced flexibility would be unproductive. However, as a possible reduction in regulatory burden, the Improved Standard Technical Specifications that address safety limits provide a soundly based, clearer, and more consistent approach to the required actions if a safety limit is exceeded. Therefore, the licensee for Surry, as well as other licensees, might consider further evaluation of the benefits of adopting provisions similar to those in the Improved Standard Technical Specifications for handling safety limit violations.

SUMMARY OF SURRY ASSESSMENT

Category: F **Items: All**

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

OL 1	TS 5.1
TS 5.2.A	TS 5.2.B
TS 5.2.C	TS 5.3.A
TS 5.3.B	TS 5.4.A
TS 5.4.D	

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

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

SUMMARY OF SURRY ASSESSMENT

Category: G **Items: All**

The Surry operating license contains 34 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include license conditions and definitions found in the Technical Specifications. Specifically, the Category G items are as follows:

OL a	OL b	OL c	OL d
OL e	OL f	OL 2.A	TS 1.A
TS 1.B	TS 1.C	TS 1.D	TS 1.E.1
TS 1.E.2	TS 1.F	TS 1.G.1	TS 1.G.2
TS 1.G.3	TS 1.G.4	TS 1.H	TS 1.I
TS 1.J	TS 1.K	TS 1.L	TS 1.M
TS 1.N	TS 1.O	TS 1.P	TS 1.Q
TS 1.R	TS 1.S	TS 1.T	TS 1.U
TS 1.V	TS 4.0.1		

Except for the financial qualification part of License Condition OL d, the license conditions are legal findings that appear to be required by the Atomic Energy Act or the Commission's regulations. The financial qualification finding was required at the time the Surry operating license was issued but is no longer required. The remaining items are definitions found in the "Definitions" and other sections of the Technical Specifications. They are judged necessary for the uniform interpretation of the defined terms.

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

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

APPENDIX C

ASSESSMENT OF OPERATING LICENSES

PERRY UNIT 1

PEACH BOTTOM UNIT 2

U. S. Nuclear Regulatory Commission

Regulatory Review Group

April 1993

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C.1 INTRODUCTION

C.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.

C.1.2 Selection of Plants (Licenses)

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

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

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

Using the above criteria, Seabrook Unit 1, Surry Unit 1, Perry Unit 1, and Peach Bottom Unit 2 were selected from among all 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 has been reviewed by the NRC.

Surry was selected because it is one of the earliest licensed PWRs; it is a Westinghouse three-loop plant and is, therefore, representative of a significant number of plants

(licenses); and it has an IPE whose review by the NRC is nearly complete. Surry 1 is also one of the plants evaluated in WASH-1400 and NUREG-1150.

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

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

C.1.3 Assessment Approach

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

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

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

The items were reviewed to determine which categories contained items with similar enough characteristics to be assessed collectively. The items in the remaining categories were 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 that were to be assessed individually were selected from the remaining categories. The items were selected because of their representativeness of a significant number of other items in the category, because of 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 C.3. The questions were designed to determine whether the item has a sound regulatory basis, is related to public health and safety, inherently allows the licensee flexibility in making changes to the plant or operations, or could be modified to provide increased flexibility to the licensee. The questions were written in such a manner that a "no" response would elicit additional review. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

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

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the current license, certain items that were assessed in previous license(s) were compared to the corresponding items in the current license in order to validate the results of the previous assessment(s). The items that were selected

for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility, and (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

Table C.1

SUMMARY OF ASSESSMENT APPROACH

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

Table C.2

CATEGORIES OF ITEMS

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

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

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

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

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

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

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

Table C.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)?

   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 C.3 (Continued)

ASSESSMENT QUESTIONS

4. Enhanced Flexibility Potential

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

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

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

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

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

C.2 ASSESSMENT OF PERRY AND PEACH BOTTOM OPERATING LICENSES

C.2.1 Perry and Peach Bottom Licenses

The Review Group's assessment of the Seabrook license resulted in seven recommendations. The Review Group's assessment of the Surry license largely validated its assessment of the Seabrook license and resulted in only three additional recommendations. Based on the results of the Seabrook and Surry assessments, and its knowledge of and experience with other licenses, the Review Group did not expect to find significant information in its reviews of the Perry and Peach Bottom licenses that would result in a substantial number of additional recommendations. Therefore, the Review Group assessed the Perry and Peach Bottom licenses together. The combined assessment was performed using the same methodology as that used previously for the individual plant assessments.

The Perry Unit 1 operating license was issued on November 13, 1986. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; the environmental protection plan, which is Appendix B to the license; and the antitrust conditions, which are Appendix C to the license. The license as reviewed has been amended through Amendment 43, dated May 28, 1992.

The Peach Bottom Unit 2 operating license was issued on December 14, 1973. The operating license consists of the license itself; the Technical Specifications, which are Appendix A to the license; and the Environmental Technical Specifications, which are Appendix B to the license. The license as reviewed has been amended through Amendment 168, dated July 6, 1992.

C.2.2 Assessment of Licenses

The Perry and Peach Bottom operating licenses contain 329 and 275 items, respectively. Each of the items was reviewed and assigned to one of the categories in Table C.2. The numbers of items in the Perry and Peach Bottom operating licenses by category are shown in Tables C.4A and C.4B, respectively.

The items in each category were reviewed to determine which categories contained items that were similar enough to be assessed collectively. This determination was based on the items' regulatory bases, safety relevance, inherent flexibility, and potential to provide enhanced flexibility. The items in three categories were deemed appropriate to be assessed collectively--Category B, "Non-Technical License Conditions"; Category F, "Unique Plant Features"; and Category G, "Other." These three categories encompassed 83 items or approximately 25 percent of the total number of items for the Perry license, and 93 items or approximately 34 percent of the total number of items for the Peach Bottom license.

The number of items in the remaining categories that would be assessed individually was determined to be approximately 5 percent or 12 of the 246 remaining items for Perry and 9 of the 182 remaining items for Peach Bottom. That percentage was then apportioned among the remaining categories and determined the number of items to be assessed in each category, e.g., 5 percent, or 7 of the 136 items in Category A would be selected for further assessment for the Perry license. With the items that would be assessed collectively, this meant that 95 or approximately 29 percent of the 329 total items would be assessed either collectively or individually for Perry and 102 or approximately 37 percent of the 275 total items would be assessed either collectively or individually for Peach Bottom.

The items that were to be assessed individually were selected because of their representativeness of a significant number of other items in the category, their enhanced flexibility potential, or their special interest. All the items that were assessed are listed in Table C.5A for Perry and Table C.5B for Peach Bottom.

Each item was assessed either collectively or individually as appropriate by considering the answers to the questions presented in Table C.3. The items were analyzed as necessary to ensure an adequate understanding of their regulatory bases, safety relevance, inherent flexibility, and potential for enhanced flexibility.

An assessment summary was prepared for each item. Each summary contains overall conclusions concerning whether the item is appropriate given its safety significance and regulatory basis, whether the item is unduly restrictive, and whether further consideration should be given to the item for possible reduction in regulatory burden or enhanced flexibility. Those items that inherently allow licensees flexibility in making changes to their plants or operations were reviewed in general to determine if the regulatory process may be inhibiting their use of this flexibility.

Following the assessment of the items, they were grouped as follows: (1) items that appear to exceed applicable regulatory requirements, (2) items that should be considered for possible reduction in regulatory burden, (3) items that provide inherent flexibility, (4) items that should be considered for enhanced flexibility, (5) items considered or being considered in other programs, and (6) items for which no further consideration is warranted.

In addition to the items assessed in the Perry and Peach Bottom licenses, certain items that were assessed previously in the Seabrook and Surry licenses were compared to the corresponding items in the Perry and Peach Bottom licenses in order to validate the results of the previous assessments. The items that were selected for validation include (1) those that appear to exceed the applicable regulatory requirements, (2) those that should be considered for possible reduction in regulatory burden, (3) those that provide inherent flexibility and, (4) those that should be considered for enhanced flexibility.

The overall results were integrated and the recommendations developed.

2.3 Results of Assessment

The assessment summaries for each of the items are provided in Attachment A for Perry and Attachment B for Peach Bottom. The summaries are presented in the order of the categories into which each of the items were assigned. Within each category, the items are addressed in the order in which they appear--first, in the operating license (OL) itself; next, in the Technical Specifications (TS); and, finally, for Perry, in the environmental protection plan (EP) and in Appendix C antitrust conditions (ACs), and for Peach Bottom, in the Environmental Technical Specifications (ES).

The overall findings and recommendations are presented in Section C.3.

Table C.4A

PERRY OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	136
B. Non-Technical License Conditions	7
C. License Conditions That Rely on Other Documents for Requirements	23
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	48
E. Reporting and Recordkeeping Requirements	39
F. Unique Plant Features	11
G. Other	65
Total	329

Table C.4B

PEACH BOTTOM OPERATING LICENSE ITEMS BY CATEGORY

Category	No. of Items
A. Technical Requirements	75
B. Non-Technical License Conditions	2
C. License Conditions That Rely on Other Documents for Requirements	19
D. Administrative Controls (Exclusive of Reporting and Recordkeeping Requirements)	59
E. Reporting and Recordkeeping Requirements	29
F. Unique Plant Features	10
G. Other	81
Total	275

Table C.5A

INDEX OF PERRY OPERATING LICENSE ITEMS ASSESSED

Table C.5A (Continued)

INDEX OF PERRY OPERATING LICENSE ITEMS ASSESSED

Item	Subject	Page*
	Category E (2 of 39)**	
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TS 5.1.1	Exclusion area, unrestricted area, site boundary	C-58
TS 5.1.2	Low population zone	C-58
TS 5.2.1	Containment - configuration	C-58
TS 5.2.2	Containment - design temperature and pressure	C-58
TS 5.2.3	Containment - secondary containment	C-58
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TS 5.4.1	RCS - design pressure and temperature	C-58
TS 5.4.2	RCS - volume	C-58
TS 5.5.1	Meteorological tower location	C-58
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OL 1.B	Finding - construction completion	C-59
OL 1.C	Finding - conformance with requirements	C-59
OL 1.D	Finding - reasonable assurance	C-59
OL 1.E	Finding - technical qualification	C-59
OL 1.F	Finding - financial protection	C-59
OL 1.G	Finding - issuance of license	C-59
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OL 2.B.1	Authorization - possess, use and operate	C-59
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Table C.5A (Continued)

INDEX OF PERRY OPERATING LIC!:NSE ITEMS ASSESSED

Item	Subject	Page*
TS 1.0	Technical Specification definitions (52 items)	C-59
EP 1.0	Objectives	C-59

OL = Operating license condition
TS = Technical Specification
EP = Environmental protection plan condition
AC = Appendix C antitrust conditions

* Page number of assessment summary in Attachment A to this appendix.

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

*** Items that were assessed collectively; all others were assessed individually.

Table C.5B

INDEX OF PEACH BOTTOM OPERATING LICENSE ITEMS ASSESSED

Table C.5B (Continued)

INDEX OF PEACH BOTTOM OPERATING LICENSE ITEMS ASSESSED

OL = Operating license condition
TS = Technical Specification
ES = Environmental Technical Specification

* Page number of assessment summary in Attachment B to this appendix.

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

*** Items that were assessed collectively; all others were assessed individually.

C.3 ASSESSMENT FINDINGS AND RECOMMENDATIONS

C.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 that have already been or are being considered in other programs are noted. Finally, those items for which no further consideration is warranted are identified.

The groups mentioned above are not mutually exclusive. That is, a particular item may fall within two or more groups. For example, Peach Bottom Item TS 6.2.3.1, ISEG - function, appears in three groups. The item appears to exceed the applicable regulatory requirements, at least in the manner in which it is implemented in the Peach Bottom operating license; it appears to have the potential for possible reduction of regulatory burden for the licensee; and, because options are already available for the elimination of this item, no further consideration of the requirement by the NRC appears to be warranted.

C.3.2 Findings and Recommendations

C.3.2.1 Items That Appear To Exceed Applicable Regulatory Requirements

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

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

Perry

None

Peach Bottom

TS 6.2.3.1	ISEG - function
ES 7.1.1.B	Organization

Peach Bottom Technical Specification 6.2.3.1 specifies the function of the Independent Safety Engineering Group (ISEG). An ISEG is not required for plants like Peach Bottom for which operating licenses were issued prior to the imposition of the Three Mile Island action plan requirements. However, it is the Review Group's understanding that the Peach Bottom licensee voluntarily chose to have an ISEG and incorporated this function by amendment into its Technical Specifications.

Peach Bottom Environmental Technical Specification 7.1.1.B appears to exceed the regulatory requirements of 10 CFR 50.36b, which delineate the scope of environmental conditions for an operating license. In prescribing reporting responsibilities and referencing a management organization chart, the intent to establish clear lines of authority and communication is distorted by unnecessary specificity, which also makes the item a regulatory burden.

In its assessment of the Seabrook and Surry licenses, the Review Group found seven items that appear to exceed the applicable regulatory requirements, at least in the manner in which they are implemented in the licenses. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Three of the Seabrook and Surry items--minimum shift crew composition, station staff working hours, and monthly operating reports--also appear to exceed the applicable regulatory requirements for both Perry and Peach Bottom. The four remaining Seabrook and Surry items--isolation of unborated water sources, auxiliary feedwater system, service water system, and D.C. electrical power system--do not apply to, or do not appear to exceed the applicable regulatory requirements for, either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.3.2.2 Items That Should Be Considered for Possible Reduction in Regulatory Burden

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

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.3.7.8	Loose parts detection system

TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration

Peach Bottom

TS 3.8.E	Radiological environmental monitoring
TS 3.14.C	Fire detection
TS 6.2.3.1	ISEG - function
ES 7.1.1.B	Organization

Perry License Condition 2.C.9 contains technical requirements related to Transamerica Delaval, Inc. (TDI) diesel generators. The details pertaining to these requirements are provided in an attachment to the license. Since the license for the most recently licensed plant with TDI diesel generators does not contain these conditions, the regulatory burden could presumably be reduced if the licensee submitted an amendment request to remove this license condition in conjunction with an Updated Final Safety Analysis Report change that incorporates the technical requirements contained in the license condition.

Perry Technical Specification 3.3.7.8 imposes a reporting requirement as a surrogate for corrective action. However, further analysis revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.7.4 contains an augmented inservice inspection program that is similar to but more detailed than that described in Generic Letter 84-13. Further analysis, however, revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.11.1.1 prescribes surveillance requirements in accordance with specific criteria for analysis, frequency, and lower limits of detection set forth in a table that is incorporated in this item. Since changes to the tabular data require a Technical Specification revision, such regulatory burden could be reduced by including the specific provisions of this item in the licensee-controlled Offsite Dose Calculation Manual (ODCM). Relocation of such details to the ODCM may be implemented by licensees in accordance with the Technical Specification Improvement Program and associated NRC guidance.

Peach Bottom Technical Specification 3.8.E contains radiological environmental monitoring requirements, including deviations from sampling schedule, land use census, and analysis to be performed on radioactive materials. Generic Letter 89-01 permits a line-item Technical Specification improvement to be made to place the programmatic controls of the Radiological Effluent Technical Specifications (RETS) in the administrative controls section of the Technical Specifications and relocate the procedural details of the RETS, of which Technical Specification 3.14.E is a part, to the Offsite Dose Calculation Manual or the Process Control Program.

Peach Bottom Technical Specification 3.14.C contains the fire detection instrumentation requirements, including operability, surveillance, and reporting requirements. In accordance with Generic Letter 86-10 and the guidance contained in Generic Letter 88-12, all the requirements related to fire protection systems and fire brigade staffing, including those contained in Section 3.14.C, can be removed from the Technical Specifications and placed in a licensee-controlled technical requirements document.

Peach Bottom Technical Specification 6.2.3.1 specifies the function of the Independent Safety Engineering Group (ISEG). Under the Improved Standard Technical Specifications, the ISEG function may be performed as a staff function under the independent reviews and audits program in the Technical Specifications. Therefore, this item can be pursued by the licensee for possible line-item elimination.

Peach Bottom Environmental Technical Specification 7.1.1.B intends to establish the proper management line of authority for environmental matters. However, in specifying titles and referencing a management organization chart, this item unintentionally creates a burden and a need for amending the Environmental Technical Specifications as a result of any organizational changes affecting this area.

In its assessment of the Seabrook and Surry licenses, the Review Group found nine items that appear to have the potential for possible reduction in regulatory burden. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Seven of the Seabrook and Surry items--physical security condition, seismic instrumentation, meteorological instrumentation, monthly operating reports, fire protection condition, containment integrity and operating pressure, and action to be taken if a safety limit is exceeded--appear to have the potential for possible reduction in regulatory burden for Perry or Peach Bottom or both. The two remaining Seabrook and Surry items--chemical and volume control system and power distribution limits--do not apply to either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.3.2.3 Items That Provide Inherent Flexibility

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

Perry

TS 3.6.1.5	Containment structural integrity
TS 6.5.1.2	PORC composition
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.6.F	Recirculation pumps
TS 3.7.E	Large primary containment purge/vent valves
TS 6.5.3.1	Procedure review and approval

Perry Technical Specification 3.6.1.5 allows the licensee to establish the performance criteria against which the technical requirements are measured. It also provides some time for the conduct of repair activities before directing a plant shutdown that would otherwise be required when primary containment integrity is in doubt.

Perry Technical Specification 6.5.1.2 provides inherent flexibility with regard to the composition of the Plant Operations Review Committee (PORC) in that the chairman of this committee can appoint up to two alternates to participate as voting members at any one time. The number of persons required to perform the PORC function is determined by the licensee; however, any changes to this number requires a Technical Specification amendment.

Perry Technical Specification 6.5.3.1 provides for a technical review and control function that relieves the Plant Operations Review Committee of some of its review responsibilities. The shifting of these responsibilities to Technical Review and Control has provided inherent flexibility in the performance of this function.

Peach Bottom Technical Specification 3.6.F provides inherent flexibility by permitting continued plant operation with only one recirculation loop in service. Even with the adjustments required of certain safety limits, limiting safety system settings, and various scram setpoints, the additional flexibility inherent in avoiding a shutdown by allowing plant operation with a reduced power level and tighter setpoint controls is advantageous to the licensee.

Peach Bottom Technical Specification 3.7.E provides inherent flexibility by only limiting the cumulative time that a purge or vent flow path can exist during a calendar year; the licensee has unlimited flexibility within that constraint. However, the Technical Specification is also prescriptive in this and other aspects.

Peach Bottom Technical Specification 6.5.3.1 contains a technical review and approval function that relieves the Plant Operations Review Committee of some of its responsibilities. The shifting of these responsibilities to the technical review and approval function provides inherent flexibility in the performance of this function.

In its assessment of the Seabrook and Surry licenses, the Review Group found 11 items that appear to have at least some inherent flexibility. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Nine of the Seabrook and Surry items--physical security condition, structural integrity, land use census, minimum shift crew composition, station staff working hours, fire protection condition, residual heat removal system, general surveillance requirement, and systems integrity--appear to have at least some inherent flexibility for Perry, Peach Bottom or both. The two remaining Seabrook and Surry items--containment building penetrations and secondary water chemistry monitoring program--do not apply to, or do not appear to have at least some inherent flexibility for, either Perry or Peach Bottom.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses.

C.3.2.4 Items That Should Be Considered for Enhanced Flexibility

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

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.1.3.1	Control rod operability
TS 3.4.1.1	Recirculation loops
TS 3.6.2.1	Drywell integrity
TS 3.7.2	Control room emergency recirculation system
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.7.E	Large primary containment purge/vent valves
TS 3.8.D	40 CFR 190
TS 6.5.3.1	Procedure review and approval

Perry License Condition 2.C.9 contains technical requirements related to Transamerica Delaval, Inc. (TDI) diesel generators. In addition to a reduction in regulatory burden if the license condition was eliminated, as discussed in Section C.3.2.2 of this report, enhanced flexibility would also result since the 10 CFR 50.59 review process could be used to determine changes that can be made without prior NRC approval.

Perry Technical Specification 3.1.3.1 exhibits a potential for enhanced flexibility with certain implementation options available to the licensee. The lead-plant efforts of the LaSalle County Station could be followed in initiating a line-item revision to the provisions governing scram discharge volume component controls. For even greater flexibility, the Improved Standard Technical Specifications, which extend the enhanced flexibility to the control rod requirements as well, could be adopted.

Perry Technical Specification 3.4.1.1 could be made more flexible if the licensee opted to submit an amendment request including a plant-specific analysis justifying single recirculation loop operation. In addition, if the licensee adopted the Improved Standard Technical Specifications, a more consistent approach to the applicable safety limits, setpoints, and direct flow measurements would be provided.

Perry Technical Specification 3.6.2.1 may be unduly prescriptive in that it requires that the plant be shut down within a specified time when actions taken to restore drywell integrity within specified completion times have failed. However, it does not consider the risk of extending the completion times relative to that of shutting down the plant. This is an area where more performance-based methods could be used and where the application of risk assessment methodology could be considered.

Perry Technical Specification 3.7.2 appears to have the potential for enhanced flexibility based upon a plant-specific application of risk assessment methodology. The provisions for control room emergency recirculation system operability at any plant could be tailored to the unique system design and reliability of that plant.

Perry Technical Specification 3.11.1.1 and Peach Bottom Technical Specification 3.8.D, which both address areas covered by a Radiological Environmental Monitoring Program (REMP), can be enhanced by adopting the more flexible provisions of the Improved Standard Technical Specifications. Generic Letter 89-01 provides guidance on the relocation of the REMP to a licensee-controlled document (the Offsite Dose Calculation Manual), the procedural details of which are delineated as program requirements in the administrative controls section of the Technical Specifications. Thus, the handling of these Technical Specifications as administrative controls, which is also endorsed by the Technical Specification Improvement Program, provides a more flexible method of ensuring compliance with 10 CFR 50.36a.

Perry Technical Specification 6.5.3.1 relieves the Plant Operations Review Committee (PORC) of some of its review responsibilities. However, it appears that some of these responsibilities are duplicated between these two functions and that the interfaces between these organizations could be burdensome. The adoption of the staff functional approach, as permitted by the Improved Standard Technical Specifications, would result in enhanced flexibility.

Peach Bottom Technical Specification 3.7.E is prescriptive in that if specified conditions are not met, the penetration must be isolated within 4 hours or the reactor must be shut down. Further, the item requires that the inflatable seals for the large containment isolation valves be replaced at least once every third refueling outage. These are areas in which the use of risk assessment methodology could be considered, e.g., to compare the relative risks of extending the completion times and shutting down the reactor and, under the provisions of the maintenance rule, 10 CFR 50.65, determining a more flexible replacement frequency for the inflatable seals.

Peach Bottom Technical Specification 6.5.3.1, relieves the PORC of some of its review responsibilities. Although there appears to be an effective interface between both organizations, the adoption of a staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in a more flexible requirement.

In its assessment of the Seabrook and Surry licenses, the Review Group found 10 items that appear to have enhanced flexibility potential. To validate the Seabrook and Surry results, these items were also reviewed for the Perry and Peach Bottom licenses. Nine of the Seabrook and Surry items--physical security condition, general limiting condition for operation, containment ventilation system, ISEG composition, SORC responsibilities, temporary changes of procedures, residual heat removal system, emergency power system, and management safety review committee--appear to have enhanced flexibility potential for Perry or Peach Bottom or both. The one remaining Seabrook and Surry item--chemical and volume control system--does not apply to either Perry or Peach Bottom.

An assessment of the findings documented in Section C.3.2.3 of this report reveals that, for many items that provide inherent flexibility, the specifications are less prescriptive and more performance based. Such an approach to the delineation of license requirements allows for the restrictive details to be governed by licensee-controlled documents and also provides an effective means for achieving additional flexibility. For example, Perry Technical Specification 3.6.1.5 allows the licensee to establish specific performance criteria against which the technical requirements for containment structural integrity are measured. Further, several licensee programs were found to have an ample degree of operational latitude, while at the same time providing adequate control over a wide range of technical and administrative areas. This is exemplified in both the Perry and Peach Bottom Technical Specifications TS 6.5.3.1 where certain Plant Operations Review Committee responsibilities have been shifted to a Technical Review and Approval function.

In reviewing the items to be considered for enhanced flexibility, both technical requirements and programmatic areas were identified to have the potential for greater flexibility under licensee control without sacrificing system or program functionality. Additionally, the specification of prescriptive provisions in licensee-controlled documents reduces the regulatory burden on both the licensee and the NRC by allowing changes to be made without having to amend the license. As an example, both the Perry and Peach Bottom Radiological Effluent Technical Specifications could be relocated to the administrative controls section of the Technical Specifications with the implementing details of the applicable programs moved to the Process Control Program or the Offsite Dose Calculation Manual, both licensee-controlled documents.

This approach, allowing licensee-controlled documents to provide the programmatic controls and implementing details for needed technical requirements, could be used to a much greater extent than is currently in practice. For many items, including those where the Perry and Peach Bottom license reviews validate previous assessment findings (e.g., physical security), performance-based guidance could replace prescriptive criteria. The use of such performance-based direction would help clarify the difference between requirements and their implementation details and thus enhance the flexibility of the latter without adversely affecting compliance with the former.

Recommendations: Based on the foregoing, the Review Group reaffirms its recommendations in this area from its assessments of the Seabrook and Surry licenses and, in addition, recommends the following:

- Expand the use of performance-based direction to supplant prescriptive criteria in license conditions and Technical Specifications. In items exhibiting inherent flexibility, the functional requirement is distinguishable from the technical details needed to implement that requirement. As evidenced in the Technical Specification Improvement Program, licensee-controlled programs that govern such implementation details provide both flexibility and the requisite assurance of system functionality.

C.3.2.5 Items Considered or Being Considered in Other Programs

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

Perry

TS 3.1.3.1	Control rod operability
TS 3.3.7.8	Loose parts detection system
TS 3.4.1.1	Recirculation loops
TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.3.1	Technical review and control

Peach Bottom

TS 3.6.F	Recirculation pumps
TS 3.8.D	40 CFR 190
TS 3.8.E	Radiological environmental monitoring
TS 3.14.C	Fire detection
TS 6.5.3.1	Procedure review and approval

Perry Technical Specifications 3.1.3.1 and 3.4.1.1 have more coherent, better organized, and more flexible counterparts in the Improved Standard Technical Specifications, while Perry Technical Specification 3.11.1.1 has been replaced as part of the Technical Specification Improvement Program with an administrative control provision that is more appropriate to the functional requirement.

Perry Technical Specification 3.3.7.8 imposes a reporting requirement as a surrogate for corrective action. Further analysis, however, revealed that this Technical Specification does not appear in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry Technical Specification 3.7.4 and Peach Bottom Technical Specification 3.8.E, in accordance with the provisions of Generic Letters 84-13 and 89-01, respectively, are not included in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Perry and Peach Bottom Technical Specifications 6.5.3.1, which are similar, could be integrated into a staff functional approach in accordance with the Improved Standard Technical Specifications. This approach simplifies and provides additional flexibility to the entire review and audit function.

Peach Bottom Technical Specification 3.6.F, which is similar to Perry Technical Specification 3.4.1.1, and Peach Bottom Technical Specification 3.8.D, which is similar to Perry Technical Specification 3.11.1.1, have both been replaced with better organized and more appropriate functional items in the Improved Standard Technical Specifications.

Peach Bottom Technical Specification 3.14.C, in accordance with the provisions of Generic Letters 86-10 and 88-12, is not included in the Improved Standard Technical Specifications and, therefore, can be pursued by the licensee for possible line-item elimination.

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

3.2.6 Items for Which No Further Consideration Is Warranted

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

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

Perry

OL 2.C.9	TDI diesel generator reliability
TS 3.1.3.1	Control rod operability
TS 3.3.7.8	Loose parts detection system
TS 3.4.1.1	Recirculation loops
TS 3.6.1.5	Containment structural integrity
TS 3.7.4	Snubbers
TS 3.11.1.1	Liquid effluents - concentration
TS 6.5.1.2	PORC composition
TS 6.5.3.1	Technical review and control
TS 6.9.4	Special reports - fire protection program
Cat B items	Non-technical license conditions (7 items)
Cat F items	Unique plant features (11 items)
Cat G items	Other (65 items)

Peach Bottom

TS 3.6.D	Safety and relief valves
TS 3.6.F	Recirculation pumps
TS 3.8.D	40 CFR 190
TS 3.8.E	Radiological environmental monitoring
TS 3.14.C	Fire detection
TS 6.2.3.1	ISEG - function
TS 6.5.3.1	Procedure review and approval

ES 7.1.1.B	Organization
Cat B items	Non-technical license conditions (2 items)
Cat F items	Unique plant features (10 items)
Cat G items	Other (81 items)

Recommendations: Based on the foregoing, the Review Group has no recommendations in this area.

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- ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

- Atomic Energy Act of 1954, as amended.

- Code of Federal Regulations, Title 10 - Energy.

- Code of Federal Regulations, 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

- Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit No. 1, issued November 13, 1986, through Amendment 43, May 28, 1992.

- Facility Operating License No. DPR-44, Peach Bottom Atomic Power Station, Unit No. 2, issued December 14, 1973, through Amendment 168, July 6, 1992.

- Facility Operating License No. NPF-86, Seabrook Station, Unit No. 1, issued March 15, 1990, through Amendment 11, May 29, 1992.

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- NUREG-1150, "Reactor Risk Reference Document," Volume 1, (Draft) February 1987; and "Severe Accident Risks: Assessment for Five U.S. Nuclear Power Plants - Final Summary Report," December 1990.

- NUREG-1216, "Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc." August 1986.

- NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," Generic Letter 89-01, Supplement No. 1, April 1991.

- NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," September 1992, (referred to in the Assessment Report as "Improved Standard Technical Specifications" for Peach Bottom).

- NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6," September 1992, (referred to in the Assessment Report as "Improved Standard Technical Specifications" for Perry).

- USNRC Generic Letter 84-13, "Technical Specifications for Snubbers," May 3, 1984.

- USNRC Generic Letter 86-10, "Implementation of Fire Protection Requirement," April 24, 1986.

- USNRC Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," August 2, 1988.

- USNRC Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program," January 31, 1989.

- USNRC Letter, dated January 15, 1993, issuing Amendment No. 89 to Facility Operating License No. NPF-11 and Amendment No. 74 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively, with safety evaluation enclosed.

- USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for the Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, March 1978.

- USNRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

- USNRC Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Nuclear Reactors," May 1981.

- USNRC Regulatory Review Group, "Draft Assessment of Operating License, Seabrook Station, Unit No. 1," February 1993.

- USNRC Regulatory Review Group, "Draft Assessment of Operating License, Surry Power Station, Unit No. 1," March 1993.

- WASH-1400 (NUREG-75/014), "Reactor Safety Study--An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," October 1975.

ATTACHMENT A TO APPENDIX C

PERRY ITEM ASSESSMENT SUMMARIES

SUMMARY OF PERRY ASSESSMENT

Category: A Item: OL 2.C.9

Perry License Condition 2.C.9, Transamerica Delaval, Inc. (TDI) diesel generator reliability, references Attachment 2 to the license, which requires the licensee to perform crankshaft and cylinder block inspections and air roll tests on the diesels. In addition, it is stated that changes to the maintenance and surveillance program approved by the staff are subject to the provisions of 10 CFR 50.59 and, if cracks in the crankshaft are found, this condition should be reported to the NRC. This item was selected because it not only represents a technical requirement incorporated into the operating license, rather than the Technical Specifications, but also contains the details of the requirement in a referenced attachment to the license. Another attachment to the license, related to the detailed control room design review imposes similar types of requirements.

This item has a regulatory basis in General Design Criterion 17 (GDC-17) of Appendix A to 10 CFR 50, which addresses electrical power systems. Concerns regarding the reliability of TDI diesels were first prompted by a crankshaft failure at Shoreham Nuclear Power Station in August 1983. In response to the problem, nuclear utility owners formed the TDI Diesel Generator Owners Group, which developed recommendations related to replacements, modifications, inspections, testing, and maintenance and surveillances. The NRC staff's evaluation, which is contained in NUREG-1216,"Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc.," concluded that implementation of the Owners Group's recommendations, plus additional actions identified in NUREG-1216, established the adequacy of the TDI diesels for nuclear service as required by GDC-17. The recommendations in NUREG-1216 are contained in Attachment 2 to the operating license.

This license condition is important to safety since it ensures the operability of the diesels manufactured by a specific company. However, the need for the incorporation of requirements related to this issue as a license condition and the inclusion of prescriptive technical requirements as an attachment to the license condition appear unnecessary. The Comanche Peak license, which is for the plant most recently licensed with TDI diesels, neither includes this as a license condition nor incorporates it into the Technical Specifications. Presumably, this license condition could be removed if the licensee submitted an amendment request to remove it in conjunction with an Updated Final Safety Analysis Report change that incorporates the requirements contained in the license condition. The regulatory burden would be reduced since a license amendment would no longer be required to change the testing and inspection requirements related to the TDI diesels that were imposed by the NRC. In addition, this would also provide enhanced flexibility since the licensee could use the 10 CFR 50.59 review process to determine changes that could be made without prior NRC approval.

Based on the above considerations, it is concluded that further consideration of this issue would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A **Item: TS 3.1.3.1**

Perry Technical Specification 3.1.3.1, control rod operability, requires that all control rods be operable and specifies actions for various conditions of untrippable or otherwise inoperable control rods and scram discharge volume (SDV) valves. Surveillance requirements for scram discharge volume components as well as control rods are delineated. This item was selected for review because it is representative of reactivity control provisions and it references the surveillance requirements of other Technical Specifications for operability determinations.

In addition to the general discussion of Technical Specifications in 10 CFR 50.36, this item has regulatory bases in Appendix A to 10 CFR 50, as well as in 10 CFR 100. It has clear safety relevance to the reactivity controls required to maintain the reactor with an acceptable shutdown margin and in accordance with minimum critical power ratio limits. Adequate controls over the SDV components also ensure that offsite radiation doses are limited to the levels allowed by the regulations. This item is consistent with other General Electric BWR/6 plant provisions in specifying both control rod and SDV component requirements in the same Technical Specification. The Improved Standard Technical Specifications, however, address control rod and SDV component provisions separately. A coherent application of control rod operability is established in the Perry Technical Specifications by the recognition that this item applies when control rods are declared inoperable as a result of other action items. However, the surveillance requirement of Technical Specification 3.1.3.1, which references the surveillances of four other action items, appears to be redundant and somewhat confusing in that the acceptance criteria of other surveillance tests would be more appropriately applied through their own action item requirements alone.

This item has limited inherent flexibility in the varying level of actions required, dependent upon the number of control rods inoperable and the various causes of inoperability. Additionally, the restriction that not more than eight control rods may be inoperable at power provides some operational latitude while at the same time considers the possibility that a generic problem may exist that requires reactor shutdown for resolution. This provision was retained in the Improved Standard Technical Specifications. However, the Improved Standard Technical Specifications provide other forms of enhanced flexibility in this area, not only in increasing the allowable outage times for inoperable control rods and SDV valves, but also in allowing separate action item entry conditions to apply for each control rod or SDV vent and drain line. This increase in flexibility has already been incorporated, in part, as a Technical Specification amendment to the SDV pro :sions of the control rod operability requirements (i.e., also TS 3.1.3.1) for the LaSalle 'Units 1 and 2. Thus, the enhanced flexibility potential for this item at Perry and other similar plants is good, not only because of the lead-plant efforts of LaSalle in the area of a line-item improvement to the SDV limiting conditions for operation, but also because the Improved Standard Technical Specifications offer additional flexibility in the control rod limiting conditions for operation as well.

While the NRC safety evaluation for the LaSalle operating license amendments, NUREG-0803, was issued based on a site-specific analysis and request, the staff review considered elements and criteria of a generic nature related to the SDV system piping. This safety evaluation also documented the consistency between the LaSalle request and the Improved Standard Technical Specifications in the handling of SDV controls. Thus, while extension of such flexibility to control rod operability, in general, requires additional plant-specific analysis for Perry, LaSalle, and other similar plants, the concept of implementing line-item improvements based on the presentation of valid analysis has been demonstrated to be both practical and workable.

As discussed above, while Technical Specification 3.1.3.1 has potential for enhanced flexibility, such improvement has already been considered by the NRC and can be pursued on a plant-specific basis through adoption of the appropriate Improved Standard Technical Specifications. Further review of this item for a reduction of regulatory burden is not deemed to be worth the effort. Individual licensees must determine if Technical Specification revisions at their plants are warranted to take advantage of the enhanced flexibility options available in this area. Therefore, additional consideration of this item by the NRC would be unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A **Item: TS 3.4.1.1**

Perry Technical Specification 3.4.1.1, recirculation loops, requires that both reactor coolant system recirculation loops be in operation with added conditions on total core flow and its relationship to thermal power. The action statements require a reactor shutdown if both recirculation loops are not in operation. This item was selected for review because single recirculation loop operation is not permitted for Perry as it is for certain other boiling water reactors.

This Technical Specification has regulatory basis in Appendix A to 10 CFR 50, relative to the maintenance of core operating conditions within fuel design limits and is also related to the fuel cladding integrity criteria delineated in 10 CFR 46 and Appendix K to 10 CFR 50. Such regulatory bases clearly establish the safety relevance of this item. However, even though a special test exception (i.e., TS 3.10.4) allows suspension of the two-loop, matched flow requirements during low-power physics testing, this item is restrictive in both its stipulation of two-loop operation and its constraints on thermal power relative to core flow conditions.

Increased flexibility is available to the licensee by performing a plant-specific analysis that would justify the adequacy of emergency core cooling system (ECCS) performance during single recirculation loop operation. As documented in the Technical Specification bases for this item, operation with one reactor coolant recirculation loop inoperable is prohibited until such an ECCS analysis is performed, evaluated, and determined to be acceptable. Operation with only one recirculation loop in service is authorized at several BWR plants (e.g., Peach Bottom, reference item TS 3.6.F in Attachment B to this report). A licensee can pursue such an option if technically justifiable based upon plant ECCS design and analysis. Thus, although Perry Technical Specification 3.4.1.1 has little inherent flexibility, the licensee has the ability to add flexibility with a technically justified submittal to the NRC for a line-item amendment.

Additionally, the Improved Standard Technical Specifications not only allow single recirculation loop operation if supported by accident analysis, but also increase the allowable outage time for loop flow mismatch beyond that specified in the Perry Technical Specifications. The coherency of the Improved Standard Technical Specifications is evidenced for this item by the focus of the surveillance requirements on flow mismatch rather than the neutron flux noise levels of the Perry Technical Specification. While this allows for a more consistent application of the safety limits and setpoints of the Core Operating Limits Report as it relates to flow requirements, further evaluation of an individual plant's stability region on the core power/flow map would be required prior to the licensee's pursuing adoption of the Improved Standard Technical Specifications. Furthermore, while flow mismatch (i.e., Perry TS 3.4.1.3) is incorporated into this item in the Improved Standard Technical Specifications, the provisions governing the recirculation loop flow control valves have been relocated from this item to a separate Technical Specification.

While enhanced flexibility potential exists for this item, a licensee's pursuit of the above-noted flexibility would probably not result in a reduction in regulatory burden. On a plant-specific basis, each licensee must determine if exercising the available options is worth the effort. It is, therefore, concluded that further consideration of this item would be unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A   Item: TS 3.6.1.5

Perry Technical Specification 3.6.1.5, containment structural integrity, requires that the structural integrity of the containment be maintained during the operational conditions in which primary containment operability is prescribed as applicable, i.e., power operation, startup, and hot shutdown. The action statement dictates placing the reactor in cold shutdown conditions if the containment is found in nonconformance with structural integrity criteria. This item was selected for review because both the limiting condition for operation (LCO) and the surveillance requirements, which include written reports, appear to be redundant to other regulatory requirements.

In addition to the discussion of LCOs and reporting requirements in 10 CFR 50.36, this item has regulatory bases in the reactor containment criteria of Appendix A to 10 CFR 50 and in the containment inspection provisions of Appendix J to 10 CFR 50. The structural integrity of the containment has an important safety significance--not only from a design basis perspective but also with respect to 10 CFR 100 requirements. While this item provides no flexibility in its stipulated compliance provision, some inherent flexibility is evident in allowing the licensee to establish the criteria that determine compliance. Since the required visual inspections are intended to verify no apparent changes in appearance or other abnormal degradation of the containment surfaces and annulus fill concrete, the licensee is permitted to develop its own inspection program and acceptance criteria to fulfill this surveillance requirement.

The stipulation that the inspection be performed prior to the Type A containment leakage rate test not only reiterates the regulatory requirement of Appendix J to 10 CFR 50, but also points out somewhat of an inconsistency in this Perry item. With the surveillance inspections accomplished with the plant already in cold shutdown conditions, directing shutdown actions for failure to conform to the acceptance criteria appears to be a moot requirement. However, it is possible that some structural integrity problems could be identified in the containment during power operations or hot shutdown conditions. In this case then, the licensee is allowed 24 hours for repair before initiating a shutdown that would be otherwise required when primary containment integrity is in doubt.

By comparison, the Improved Standard Technical Specifications allow only a 1 hour completion time for structural integrity repairs but extend the time allotted to achieve cold shutdown conditions. This difference in the allowable outage times might reflect the consideration of repair-time impact upon the various containment designs (e.g., free-standing steel versus prestressed concrete containments). In any case, Perry Technical

Specification 3.6.1.5 appears more flexible than its Improved Standard Technical Specification counterpart. While one of the surveillance provisions of this item specifies a written reporting requirement that is redundant to 10 CFR 50.73, elimination of this surveillance/reporting requirement, although appropriate, will not result in a reduction in regulatory burden. Furthermore, from a practical standpoint, the discovery of problems during surveillance inspections would only prevent a plant startup, not dictate a shutdown, given the plant is already shut down. Such redundancy and inconsistency might justify the elimination of this item as a Technical Specification, but the allowance for a 24-hour repair period for degraded containment structural conditions identified at power provides the licensee additional flexibility that would not be otherwise available.

Therefore, this item is not a candidate for elimination or enhanced flexibility by way of the Technical Specification Improvement Program. The existing requirements already provide some inherent flexibility for regulatory compliance. While the inconsistent action times between the Improved Standard Technical Specifications and Perry TS 3.6.1.5 may warrant additional technical review, it is concluded that further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.6.2.1

Perry Technical Specification 3.6.2.1, drywell integrity, requires that the licensee restore drywell integrity within 1 hour or be in hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours (37 hours total) if drywell integrity is lost. This item was chosen because of its potential for enhanced flexibility.

This item has its regulatory basis in the General Design Criteria of 10 CFR 50, Appendix A, Section V, Reactor Containment. This section addresses the integrity of the reactor containment, including the drywell, and the methods of ensuring containment integrity. Drywell integrity is demonstrated by verification of closure of drywell penetrations that do not have automatic isolation and verification that drywell airlock, suppression pool and drywell bypass leakage are all in compliance with corresponding Technical Specification requirements.

This requirement is important to safety to ensure that in the event of an accident, 10 CFR 100 limits are not exceeded. No inherent flexibility is provided by this Technical Specification. However, it is not clear that the Technical Specification could not be made more flexible. Since not all limiting conditions for operation have the same safety significance, the completion times allowed for achieving hot standby, hot shutdown, and cold shutdown could possibly be made more performance oriented, e.g., by considering situation-specific factors. Further, it may not always be safer to change operational modes. For example, if there is reasonable assurance that the situation could be rectified within 1 hour after the completion time for changing modes expires, it might be safer to maintain the reactor in its present mode for that additional period of time than to change modes.

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

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.7.2

Perry Technical Specification 3.7.2, control room emergency recirculation system, requires that two independent control room emergency recirculation subsystems be operable. These provisions ensure that the control room will remain habitable for operations personnel during and following all design basis accident conditions. This item was selected for review because it is representative of a two-train plant safety system for which the application of a risk assessment methodology may be beneficial.

This Technical Specification has a clear regulatory basis in General Design Criterion 19 of Appendix A to 10 CFR 50, which specifies a maximum whole body dose of 5 rem to control room personnel for the duration of an accident. The safety relevance of this item is further established in Regulatory Guide 1.52, which sets forth criteria for the reliable performance of such engineered-safety-feature atmosphere cleanup systems. While less restrictive action requirements are stipulated in the cold shutdown and refueling modes, the action items and surveillance requirements direct generally prescriptive provisions. Particularly in its detail and reference to the position statements of Regulatory Guide 1.52, little inherent flexibility can be found in the scope of the surveillance tests and their quantitative acceptance criteria.

The Improved Standard Technical Specifications provide some increased flexibility, primarily in cold shutdown and refueling conditions where system operability is only required during specific plant activities. Additionally, the surveillance requirements of the Improved Standard Technical Specifications, while also endorsing Regulatory Guide 1.52 positions, appear to be generally less restrictive. However, the shutdown requirements mandated in both Technical Specifications are similarly restrictive under hot operational conditions.

This item appears to have the potential for enhanced flexibility based on a plant-specific application of risk assessment methodology. Different plant control room design features, including proximity of the control room and air intakes to the reactor, have a varying impact upon both the need to enter the recirculation mode of operation and the length of time until shutdown, as dictated by component inoperability. System design features and maintenance practices can contribute significantly to the control room emergency recirculation system reliability. Thus, it is reasonable to expect that the provisions governing the control room emergency recirculation system operability requirements at a plant, if based upon performance and analyzed with a risk perspective, could be tailored to the individual plant, just as the environmental conditions of a design basis accident are calculated and analyzed uniquely for each plant. The current language of this item, as well as that of the Improved Standard Technical Specifications, appears to adopt regulatory guidance and conservative safety margins appropriately. However, the specification of such standard requirements may be unnecessarily restrictive to plants whose design and maintenance practices reduce the exposure risk to control room personnel in other ways.

Based on the above considerations, it is concluded that this Technical Specification is appropriate; however, it may be unduly restrictive based on plant-specific considerations. It is recommended that further consideration be given this item for enhanced flexibility, particularly where risk assessment methodology could be used by a licensee to demonstrate that the accident dose in the control room would remain acceptably low with less prescriptive requirements.

SUMMARY OF PERRY ASSESSMENT

Category: A Item: TS 3.7.4

Perry Technical Specification 3.7.4, snubbers, requires that all snubbers be operable; that any inoperable snubbers be restored to operable status within 72 hours and an engineering evaluation be performed to determine the cause of the failure; or that the attached system be declared inoperable and the applicable action statement be followed. This item was selected for review because of its potential for reduction in regulatory burden.

The regulatory bases for this item are 10 CFR 50.55a and General Design Criteria 1, 2, 4, 14, and 15 of Appendix A to 10 CFR 50. Regulatory guidance for this item is provided by Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures," and Generic Letter 84-13, "Technical Specifications for Snubbers."

The item is relevant to safety in that the operability of snubbers is necessary to ensure that the structural integrity of the reactor coolant system and other safety systems is maintained during and following a seismic event or other events that initiate dynamic loads.

The Perry snubber Technical Specification was based on similar requirements for previously licensed BWR/6 plants. In accordance with the provisions of Generic Letter 84-13, the list of snubbers is not included in the Technical Specification. However, the associated surveillance requirement contains an augmented inservice inspection program that is similar to but more detailed than that described in Generic Letter 84-13.

The Improved Standard Technical Specifications do not contain detailed snubber Technical Specifications. Only the requirement to include the snubbers in the inservice testing program for ASME Code Class 1, 2, and 3 components remains. The list of snubbers and the details of the associated inservice inspection program have been relegated to licensee-controlled documentation.

Based on the above considerations, it is concluded that possible reduction in regulatory burden can be achieved by the licensee by pursuing the line-item elimination of this item under the Technical Specification Improvement Program. In addition, it is concluded that no further review of this item is warranted.

SUMMARY OF PERRY ASSESSMENT

Category: B **Items: All**

The Perry operating license contains seven items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. They deal with sale and leaseback, antitrust, emergency planning, financial protection, and the effective and expiration dates of the license. Specifically, the Category B items are as follows:

OL 2.B.7.a	OL 2.C.3.a
OL 2.C.3.b	OL 2.C.8
OL 2.G	OL 2.H
AC (all)	

The sale and leaseback, antitrust, and emergency planning conditions are not required by either the Atomic Energy Act or the Commission's regulations but are authorized by 10 CFR 50.50, which provides that the license may contain such conditions as the Commission deems appropriate. The financial protection license condition is based on Section 170 of the Atomic Energy Act and 10 CFR 140. The effective and expiration dates license condition is required by Section 103 of the Atomic Energy Act and 10 CFR 50.51.

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

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

SUMMARY OF PERRY ASSESSMENT

Category: C **Item: TS 3.11.1.1**

Perry Technical Specification 3.11.1.1, liquid effluents - concentration, requires that the concentration of radioactive material released in liquid effluents to unrestricted areas be limited to a specified concentration for dissolved noble gases and to the concentration delineated in the regulations for all other radionuclides. The limiting condition for operation (LCO) refers to Appendix B to 10 CFR 20 for limits, while the surveillance requirements refer to the Offsite Dose Calculation Manual (ODCM) for calculational methods and sampling analysis. This item was selected for review to determine if redundant requirements are being imposed.

This item has regulatory basis in 10 CFR 50. It limits the "instantaneous" concentration of radioactive materials in liquid effluents to help ensure that the dose objectives of Appendix I to 10 CFR 50 are not exceeded. Additionally, 10 CFR 50.36a requires that operating licenses contain Technical Specifications that require compliance with 10 CFR 20.106, which governs the radioactivity in effluents to unrestricted areas.

Additional flexibility is provided by Generic Letter 89-01, which allows licenses to restructure their Radiological Effluent Technical Specifications (RETS). A license amendment implementing this restructuring moves the requirements of TS 3.11.1.1 to a radioactive effluents controls program in the administrative controls section of the Technical Specifications. The implementing details of the program are moved to the ODCM. The Improved Standard Technical Specifications adopted this restructuring of the RETS. Licensee adoption of the guidance provided in Generic Letter 89-01 and its supplement, NUREG-1302, not only enhances licensee flexibility in implementing the program that governs RETS compliance, but also reduces the regulatory burden by eliminating unnecessary and redundant limiting conditions for operation and action items.

It is noted that licensees may need to amend TS 3.11.1.1 or the radioactive effluent controls program to avoid unnecessarily restrictive limits created by the interface of the Technical Specifications with the newly revised 10 CFR 20. Thus, consideration of this item for enhanced flexibility at this time may be unrealistic given the priority attention needed to establish consistency with the new 10 CFR 20 provisions.

The prescriptiveness of Technical Specification 3.11.1.1 must be viewed in balance with the limited flexibility it exhibits in referencing the ODCM and the need for compliance with 10 CFR 50 and the new 10 CFR 20. As discussed above, Generic Letter 89-01 and the Technical Specification Improvement Program offer both enhanced flexibility and the possibility for a reduction in the regulatory burden associated with this item. Based on these considerations, it is concluded that no further review of this item is warranted since options have been provided under Generic Letter 89-01 and the Technical Specification Improvement Program for revisions to this item.

SUMMARY OF PERRY ASSESSMENT

Category: D **Item: TS 6.5.1.2**

Perry Technical Specification 6.5.1.2, PORC composition, identifies the composition of the Plant Operating Review Committee (PORC). This item was selected as a representative administrative controls Technical Specification.

The regulatory basis for the plant onsite review function is 10 CFR 50.40(b), Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for this requirement is contained in Standard Review Plan (NUREG-0800), Section 13.4, which specifies that the qualification levels for the PORC members should be at least equivalent to those described in ANSI N18.1, Section 4.4, and the PORC composition should provide for interdisciplinary reviews of the subject matter.

The PORC at Perry, which satisfies the above requirements, has nine permanent members including the chairman, the Director of the Nuclear Engineering Department. The remaining members are managers or staff members that have expertise in various technical disciplines. There appears to be some flexibility in the composition of the PORC with regard to the technical disciplines of the permanent members and the fact that two alternate members can be appointed and participate as voting members of PORC at any one time. In reviewing the composition of PORC or its equivalent for the other three plants assessed by the Review Group, it was found that Seabrook had 10 members, Surry had six members, and Peach Bottom had nine members. There are no specific regulatory requirements regarding the number of members that constitute the onsite review committee, as is evident by the differences among the four plants assessed. In addition, the composition also varies among plants. Surry and Seabrook use only plant management and supervisors while Perry and Peach Bottom include staff members. The blend of technical expertise, however, is fairly consistent among the plants. It appears that each licensee proposed the composition of its onsite review committee and the staff reviewed the acceptability on a plant-specific basis, which has resulted in the variation of the number of members among plants. Changes to the number of members on the onsite review committee may be and have been made by licensees through the Technical Specification amendment process on a plant-specific basis. It is, therefore, up to each licensee to determine the number of members needed for the onsite review committee to perform its function in accordance with the regulations.

Based on the above considerations, it is concluded that since there is no specific regulatory requirement related to the number of persons that constitute the onsite review committee, the licensee has the flexibility to determine its plant-specific needs to satisfy this function. Therefore, further consideration of this requirement for possible reduction in regulatory burden or enhanced flexibility would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: D **Item: TS 6.5.3.1**

Perry Technical Specification 6.5.3.1, technical review and control, requires the review of the procedures/instructions, proposed modifications, and proposed tests and experiments that affect nuclear safety by individuals other than those who prepared the documents. This item was selected because it is an example of a Technical Specification requirement that has inherent flexibility and also the potential for enhanced flexibility.

The regulatory basis for this function is contained in 10 CFR 50.40(b), Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for meeting this requirement is contained in Standard Review Plan (NUREG-0800) Section 13.4, "Operational Review." The licensee has satisfied this regulation through the onsite and offsite committees and the Independent Safety Engineering Group (for plants licensed after the TMI accident). To reduce the workload of the Plant Operations Review Committee (PORC) the licensee submitted a Technical Specification amendment that shifted the review of procedures/instructions, modifications, and tests and experiments that affect nuclear safety from PORC to a newly created technical review and control function. Since the PORC in conjunction with the technical review and control function provide the equivalent of the responsibilities performed previously by PORC, the amendment was approved by the staff in March 1992.

The creation of the technical review and control function, although not required by the NRC, provides the licensee inherent flexibility since it reduces the review responsibilities of the PORC by permitting qualified independent individuals to perform these reviews subject to the approval of the General Manager of the Perry plant. However, it appears that with the creation of this new function, some of the responsibilities (e.g., the review of proposed modifications to plant structures, systems, and components) are redundant to those of PORC.

As a result of this amendment, the Perry Technical Specification requires four organizations (PORC, Nuclear Safety Review Committee, Independent Safety Engineering Group, and technical review and control) to satisfy the plant review and audit functions. Pursuit of the staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in enhanced flexibility for the licensee. For example, the prescriptive Independent Safety Engineering Group could be replaced by a staff function, and the review and audit process could be simplified if a line-item improvement in accordance with the Improved Technical Specifications was pursued by the licensee.

Based on the fact that a viable alternative is available to the licensee that would permit enhanced flexibility, it is concluded that further consideration of this issue would prove unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: E Item: TS 3.3.7.8

Perry Technical Specification 3.3.7.8, loose parts detection system, requires that the specified loose parts detection system instrumentation be operable during startup and power operation. If one or more channels of the loose parts detection system is inoperable for over 30 days, the only action required is the submittal of a special report to the NRC addressing the cause of the malfunction and the plans for restoring the channel(s) to operable status. This item was selected as a representative Technical Specification in which the only action is a reporting requirement.

Although not identified explicitly, the regulatory bases for a loose parts detection system are 10 CFR 50.36 and General Design Criteria 1 and 13 of Appendix A to 10 CFR 50. Regulatory guidance on loose parts monitoring systems is provided in Standard Review Plan Section 4.4, "Thermal and Hydraulic Design," and Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." The Standard Technical Specifications contain a loose parts detection system Technical Specification similar to Perry's; the Improved Standard Technical Specifications contain no loose parts detection system Technical Specification.

The Technical Specification is safety-relevant in that it provides assurance that loose metallic parts in the reactor coolant system will be detected prior to causing damage to the reactor coolant system internals, including the reactor fuel. While the required action for system inoperability affords no inherent flexibility in submitting the special report, continued operation is permissible with the instrumentation inoperable. It appears that submittal of the special report is a surrogate to the real goal of restoring the instrumentation to operable status in a timely manner.

Reduction in regulatory burden could be achieved by the elimination of the requirement to submit the special report. Since no loose parts detection system Technical Specification is contained in the Improved Standard Technical Specifications, even further reduction in regulatory burden could be accomplished if the licensee pursued the line-item elimination of the entire requirement.

Based on the above considerations, it is concluded that no further review of this item is warranted.

SUMMARY OF PERRY ASSESSMENT

Category: E **Item: TS 6.9.4**

Perry Technical Specification 6.9.4, special reports - fire protection program, states that violations to the requirements of the fire protection program that could have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported via the License Event Report system within 30 days. This item was selected because it is an example of a redundant reporting requirement.

The regulatory basis for the fire protection program is 10 CFR 50.48, Criterion 3 of Appendix A to 10 CFR 50, and Appendix R to 10 CFR 50. The reporting requirements associated with this issue are referenced to 10 CFR 50.73. This reporting requirement is necessary to ensure that the staff is informed of events that affect the ability of the plant to achieve and maintain safe shutdown in the event of a fire and that adequate compensatory measures have been taken by the licensee.

Section 2.F of the operating license requires that for violations of the requirements in Section 2.C, which includes the fire protection license condition, initial notification be made within 24 hours with written followup within 30 days. This 30-day followup report is redundant to the reporting requirement in Technical Specification 6.9.4. Since any violations of the fire protection program that would adversely affect the ability to achieve and maintain safe shutdown in the event of a fire would have to be reported in accordance with the initial notification and reporting requirements contained in 10 CFR 50.72 and 10 CFR 50.73, respectively, both the reporting requirements in the license and the Technical Specifications are redundant and unnecessary. In recognition of this fact, these reporting requirements have not been included in the Improved Standard Technical Specifications.

Based on the above considerations, it is concluded that, although the reporting requirements are redundant, there would be no reduction in regulatory burden or enhanced flexibility if the redundancy were eliminated, and therefore further consideration of this issue would probably be unproductive.

SUMMARY OF PERRY ASSESSMENT

Category: F **Items: All**

The Perry operating license contains 11 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 5.2.1
TS 5.2.2	TS 5.2.3
TS 5.3.1	TS 5.3.2
TS 5.4.1	TS 5.4.2
TS 5.5.1	

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

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

SUMMARY OF PERRY ASSESSMENT

Category: G **Items: All**

The Perry operating license contains 65 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include license conditions, Technical Specification definitions, and the environmental protection plan objective. 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.O (52 items)	EP 1.0

The license conditions are legal findings and authorizations that appear to be required by the Atomic Energy Act or the Commission's regulations. The Technical Specification definitions are judged necessary for the uniform interpretation of the defined terms. The environmental protection plan objective is deemed to be appropriate.

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

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

ATTACHMENT B TO APPENDIX C

PEACH BOTTOM ITEM ASSESSMENT SUMMARIES

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: A **Item: TS 3.6.D**

Peach Bottom Technical Specification 3.6.D, safety and relief valves, requires that both safety valves and the safety mode of all the safety-relief valves be operable prior to reactor startup and during pressurized reactor coolant system conditions. The two safety valves and 11 safety-relief valves provide sufficient pressure relief margin for the reactor coolant system overpressure system safety limit. A shutdown is required if either safety valve is inoperable. However, continued operation is permissible for 30 days if one safety-relief valve exhibits an inoperable safety valve function, or for 7 days if two safety-relief valves have inoperable safety valve functions. This item was selected for review because it is representative of the limiting conditions for operation (LCO) for primary system boundary components and because of its relationship to the safety limit for the reactor coolant system integrity.

This item has regulatory bases in General Design Criteria 14 and 15 of Appendix A to 10 CFR 50, as well as in 10 CFR 50.55a, which references the ASME Boiler and Pressure Vessel Code requirements. Given the overpressure protection criteria of the ASME Code, which are necessary to ensure the integrity of the reactor coolant pressure boundary, and considering the importance of ensuring the integrity of the reactor coolant system as a fission product barrier, the safety relevance of this item is clearly established and reinforced by the inclusion of safety and relief valve setpoints as limiting safety system settings. While the capacity of the safety and safety-relief valves is sufficient to provide overpressure protection, the relief system capacity of the safety-relief valves also satisfies design intent by precluding a challenge of the safety valves during normal plant operational transients.

Although the surveillance criteria are somewhat performance based in allowing the bench testing of 50 percent of the valves during refueling shutdowns, the overall LCO and surveillance requirements are prescriptive. Each relief valve is required to be manually tested once per operating cycle, normally during startup, when the primary system pressure reaches the desired testing plateau. The relief valves additionally function as part of the automatic depressurization system and, correspondingly, must also comply with the LCO and surveillance requirements of Technical Specification 3.5.E. However, while the shutdown provisions are prescriptive, the allowable outage times appear to provide sufficient operational latitude, particularly when only one safety-relief valve is inoperable. Thus, some flexibility is afforded the licensee both in the time to respond to an inoperable valve and in the routine implementation of the bench testing-inspection program.

The Improved Standard Technical Specifications have similar LCO and surveillance requirements, differing primarily in the 14-day allowable outage time for one or two inoperable safety-relief valves. There appears to be little additional flexibility that can be provided for this item. Furthermore, the differences in allowable outage times between this item and its Improved Standard Technical Specification counterpart may not have much practical significance except for delaying a shutdown if the valve itself requires repair or replacement.

Based on the above considerations, it is concluded that this Technical Specification is appropriate. While restrictively worded, the details have proper technical bases in the ASME Code and in regulatory requirements. Given plant design constraints and safety limits for ensuring system integrity, the overpressure controls provided by this item appear reasonable. While risk assessment methodology could probably be used to determine optimum allowable outage times, such effort does not appear to be warranted. Therefore, it is concluded that further consideration of this item for enhanced flexibility or reduction in regulatory burden would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: A **Item: TS 3.6.F**

Peach Bottom Technical Specification 3.6.F, recirculation pumps, establishes conditions governing core flow and thermal power criteria for one- and two-recirculation loop operation. Reactor shutdown is directed if the prescribed conditions and criteria are not met within specified times. This item was selected for review because it permits single-loop operation but conditions such operation upon setpoint changes controlled by other Technical Specifications.

Similar to Perry Technical Specification 3.4.1.1 (reference Attachment A to this report), this item has regulatory basis in Appendix A to 10 CFR 50 and is related to the fuel cladding integrity criteria of 10 CFR 46 and Appendix K to 10 CFR 50. A safety limit and limiting safety system settings are referenced and are affected by single-loop operation in that certain trip setpoints and core operating limits must be adjusted accordingly. This item has a sound and consistent relevance to safety in ensuring that appropriate operational margins are maintained to prevent exceeding fuel design limits.

Unlike the Perry plant, Peach Bottom plant operation with one recirculation loop in service is permitted. The applicable General Electric thermal power and core flow limitations are endorsed and prescribed. This provides inherent operational flexibility. The requirements, therefore, are technically based in the unique plant design and the controls necessary to maintain the reactor core in a stable region of operation relative to core flow and thermal power.

The Improved Standard Technical Specifications provide some enhanced flexibility in a longer allowable outage time and a simpler, more coherent approach to surveillance requirements that uses flow mismatch as the measured parameter. However, given that Peach Bottom already is taking advantage of the allowance for single-loop operation, the other flexibility enhancements of the Technical Specification Improvement Program may not be practically applicable to this item because of the Peach Bottom fuel design, power stability, and other neutron flux limitations. Since this Technical Specification references several other specifications and, in turn, the Core Operating Limits Report (COLR), pursuing a line-item adoption of the applicable Improved Standard Technical Specification may not be technically justifiable or reasonable from the standpoint of reducing regulatory burden. Therefore, since only limited enhanced flexibility potential exists for this item, pursuit of the improved flexibility may not be worth the effort.

Based on the above considerations, although revision of this item has been effected by the Technical Specification Improvement Program, individual licensees may determine that adoption of such changes may not be beneficial as a line-item revision. It is, therefore, concluded that additional review of this area requires a plant-specific evaluation, and further generic consideration of this requirement for enhancement would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: A **Item: TS 3.7.E**

Peach Bottom Technical Specification 3.7.E, large primary containment purge/vent isolation valves, specifies operability conditions for the six- and 18-inch valves; limits the purposes, times, and conditions for which purge/vent paths may exist; and requires that if these conditions are not met, the penetration must be isolated or the reactor shut down. This item was selected for review because of its apparent potential for enhanced flexibility.

The regulatory bases for this item are General Design Criteria 16, 54, and 56 of Appendix A to 10 CFR 50, Appendix J to 10 CFR 50, and 10 CFR 100. Regulatory guidance for this item is provided by Standard Review Plan Section 6.2.4, "Containment Isolation System," including Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." This item is relevant to safety in that it must balance the need to purge and vent the containment for the purposes of inerting, de-inerting, and pressure control with that of limiting the release of containment atmosphere to the environs following a loss-of-coolant accident.

The item provides inherent flexibility by only limiting the cumulative time that a purge or vent flow path can exist during a calendar year; the licensee has unlimited flexibility within that constraint. However, the item is prescriptive in that in the event the specific conditions are not met, the penetration must be isolated within 4 hours or the reactor must be shut down. Further, the item requires that the inflatable seals for the large containment isolation valves be replaced at least once every third refueling outage.

It appears that this item has enhanced flexibility potential, especially in its requirement to shut the reactor down within 4 hours if the specified conditions are not met and in its requirement to replace the inflatable seals at least every third refueling outage. Perhaps these are areas in which risk assessment methodology could be applied, e.g., to compare the relative risks of extending the completion times and shutting down the reactor and, under the provisions of the maintenance rule, 10 CFR 50.65, determining a more flexible replacement frequency for the inflatable seals.

Based on the above considerations, it is concluded that the Technical Specification is appropriate. However, it is recommended that further consideration be given to this item for possible enhanced flexibility, perhaps through the use of risk assessment methodology.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: A **Item: TS 3.14.C**

Peach Bottom Technical Specification 3.14.C., fire detection, identifies the fire detection instrumentation and provides the operability, surveillance, and reporting requirements associated with the fire detection instrumentation. This item was selected because it is representative of a Technical Specification requirement that could be relocated to a licensee-controlled document and, therefore, result in a reduction in regulatory burden.

The regulatory bases for the fire protection program are 10 CFR 50.48, Criterion 3 of Appendix A to 10 CFR 50 and Appendix R to 10 CFR 50. This item is important to safety in that it ensures that structures, systems, and components important to safety are protected by detection systems that are properly located for early detection and, therefore, minimumization of the effects of fires.

Peach Bottom Technical Specification 3.14 (including Section 3.14.C) contains the requirements for the licensee's fire protection program. Generic Letter 88-12 provides guidance for the preparation of a license amendment request to implement Generic Letter 86-10. Generic Letter 86-10 recommends that the fire protection program requirements be removed from the Technical Specifications. Implementation of Generic Letter 88-12 would result in the removal of the fire protection systems and fire brigade staffing requirements from the Technical Specifications and would add administrative controls. Relocation of the fire protection systems and fire brigade staffing requirements into a licensee-controlled document would not result in a reduction in the level of fire protection control, but would provide a reduction in regulatory burden, since a Technical Specification amendment would not be required when changes to the portions of the fire protection program related to the items being removed are made.

Based on the above considerations, it is concluded that, since Generic Letter 88-12 provides guidance for removal of fire protection requirements from the Technical Specifications that would result in a reduction in regulatory burden, further consideration of this item would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: B **Items: All**

The Peach Bottom operating license contains two items in Category B, "Non-Technical License Conditions." These items were deemed appropriate to be assessed collectively. Specifically, the Category B items are OL 3.d, NPDES permit change condition, and OL 4.0, effective date and expiration condition.

The NPDES permit change condition is not required by either the Atomic Energy Act or the Commission's regulations but is authorized by 10 CFR 50.50, which provides that the license may contain such conditions as the Commission deems appropriate. The effective date and expiration condition is required by 10 CFR 50.51.

Neither of the items is directly related to safety. Although the license conditions are prescriptive, they do not appear to be unduly restrictive. Neither of the items appears to have enhanced flexibility potential.

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

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: C Item: TS 3.8.E

Peach Bottom Technical Specification 3.8.E, radiological environmental monitoring, addresses deviations from the sampling schedule, land use census, and analysis to be performed on radioactive materials. This item was selected because it is representative of a Technical Specification requirement that could be relocated to a licensee-controlled document and, therefore, result in a reduction in regulatory burden.

The regulatory bases for this item are 10 CFR 20.106, which is related to the release of radioactivity in effluents to unrestricted areas; Appendix I to 10 CFR 50, which provides numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion for nuclear power plant effluents; 10 CFR 50.36a, which is related to Technical Specifications on effluents from nuclear power plants; and 10 CFR 50, Appendix A, Section VI, which addresses radioactivity control. The guidance for meeting these requirements is contained in Standard Review Plan (NUREG-0800) Section 11, which addresses radioactive materials, and Regulatory Guide 1.109, which addresses the calculation of annual doses to the public from routine releases of reactor effluents for the purpose of evaluating compliance with Appendix I to 10 CFR 50.

This specification ensures that the doses to the public will be within 10 CFR 20 limits and as low as reasonably achievable in accordance with Appendix I. Peach Bottom Technical Specification 3.8 (which includes Section 3.8.E) contains the procedural details associated with Radiological Effluent Technical Specifications (RETS). Generic Letter 89-01 permits a line-item Technical Specification improvement to be made by permitting the licensee to place the programmatic controls of the RETS in the administrative controls section of the Technical Specifications and relocate the procedural details of the RETS to the Offsite Dose Calculation Manual (ODCM) or the Process Control Program (PCP). Relocation of the details of the RETS into the ODCM or PCP would not result in a reduction in the level of radiological control but would provide a reduction in regulatory burden since a Technical Specification amendment would not be required to change the procedural details of the RETS.

Based on the above considerations, it is concluded that, since Generic Letter 89-01 permits a line-item improvement that would reduce the regulatory burden, further consideration of this item would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D **Item: TS 6.2.3.1**

Peach Bottom Technical Specification 6.2.3.1, function of the Independent Safety Engineering Group (ISEG), prescribes the functions that are to be performed by this group. This item was selected because it is an example of a Technical Specification that appears to exceed regulatory requirements.

This requirement is based an TMI Action Plan Item I.B.1.2 contained in NUREG-0737. This TMI item was only required of applicants for an operating license. Since Peach Bottom Unit 2 received its operating license in 1973, which was prior to the TMI requirements contained in NUREG-0737, there is no regulatory requirement that the licensee's Technical Specifications include an ISEG function.

Based on discussions with the staff, it is our understanding that the licensee endorsed this concept and voluntarily incorporated the ISEG function into its Technical Specifications. As noted in the Review Group's assessment of the Seabrook license, the Improved Standard Technical Specifications permit the ISEG function to be performed as a staff function under the reviews and audits program in the Technical Specifications.

Since the Peach Bottom licensee voluntarily incorporated the ISEG function into the Technical Specifications, removal of this function through the Technical Specification amendment process could be pursued because an ISEG is not a regulatory requirement for Peach Bottom. Alternatively, the ISEG function could be performed as a staff function in accordance with the Improved Standard Technical Specifications.

Based on the above considerations, it is concluded that further consideration of this item would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D Item: TS 6.5.3.1

Peach Bottom Technical Specification 6.5.3.1, procedure review and approval, specifies the procedure review responsibilities of the Station Qualified Reviewer, the Plant Operation Review Committee (PORC), the Plant Manager, and the designated superintendent responsible for the procedure. This item was selected because it is an example of a requirement that has inherent flexibility and the potential for enhanced flexibility.

The regulatory basis for this function is 10 CFR 50.40b, Standards for Licenses and Construction Permits, which states that a licensee must be technically qualified to engage in the operation of a nuclear power plant. Guidance for meeting this requirement is contained in Standard Review Plan (NUREG-0800) Section 13.4, "Operational Review." The licensee has satisfied this regulation through the onsite and offsite review committees. In addition, the licensee has voluntarily adopted the Independent Safety Engineering Group (only required for plants licensed after the TMI accident). To reduce the workload of the PORC, the licensee's Technical Specifications contain a technical review and approval function, which also is not required by the NRC. The PORC in conjunction with the technical review and approval function provides the equivalent of the responsibilities traditionally performed by PORC alone.

Unlike that of Perry, the Peach Bottom technical review and approval function is limited to the review of procedures and procedure changes. The responsibilities of the PORC and technical review and approval functions are well defined so there does not appear to be any review responsibilities that are duplicated by both functions. Since PORC has the option of reviewing these procedures and procedure changes instead of the Station Qualified Reviewer, inherent flexibility within the organization has been provided.

Similar to Perry, the Peach Bottom Technical Specifications require four organizations (PORC, Nuclear Safety Review Committee, Independent Safety Engineering Group, and technical review and approval) to satisfy the review and audit functions. Pursuing the adoption of the staff functional approach, as permitted by the Improved Standard Technical Specifications, could result in enhanced flexibility for the licensee in the performance of the functions currently handled by the four existing organizations. Therefore, it is concluded that further consideration of this issue would be unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: D **Item: ES 7.1.1.B**

Peach Bottom Environmental Technical Specification 7.1.1.B, organization, requires the Plant Superintendent to report to and consult with other designated licensee superintendents in all matters pertaining to the operation of the facility or to the Environmental Technical Specifications. The Peach Bottom management organization chart is included in the Environmental Technical Specifications and is referred to by this item. This item was selected for review because of its general, administrative nature and to determine whether it has appropriate basis as license condition.

The issuance of Environmental Technical Specifications as license conditions (i.e., Appendix B to the Operating License) has regulatory basis in 10 CFR 50.36b and is related to 10 CFR 51 in that the required conditions are derived from the environmental assessment conducted as part of the plant licensing process. However, this specific item appears to have no sound regulatory basis as an obligation of the licensee in the environmental area as would be required as a valid license condition in accordance with 10 CFR 50.36b. In fact, the designated chain of command specified by this item conflicts not only with the management organization chart that is referenced and included in the Environmental Technical Specifications, but also with the managerial titles and responsibilities specified in the administrative controls section of the Technical Specifications (i.e., Appendix A to the operating license).

Furthermore, while the organizational requirements of the Technical Specifications generally discuss lines of authority, responsibility, and communication, this item is prescriptive in its detail, making it prone to error unless revised with every licensee management organization change. The above-noted conflicts more than likely reflect the fact that this item has not been updated when changes affecting the license conditions were implemented.

The requirement to update such a license condition represents an unnecessary regulatory burden upon the licensee. It is concluded that this item should be considered for possible elimination or, at a minimum, revision to reflect general consistency with the existing organizational requirements of the administrative controls section of the Technical Specifications. Other similar line items in the Environmental Technical Specifications appear to warrant the same attention. However, this problem appears to be unique to the Peach Bottom Environmental Technical Specifications, which have undergone major item revisions or deletions over time. Thus, additional consideration of this requirement for enhancement by the NRC would prove unproductive and no further review of this area is warranted.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: E **Item: TS 3.8.D**

Peach Bottom Technical Specification 3.8.D, 40 CFR 190, requires compliance with the provisions of 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." Using the doses calculated from radioactive effluent releases that are governed by other Technical Specifications in combination with doses associated with direct radiation (turbine shine, storage tanks, etc.), a total dose is calculated. If the limits of Technical Specification 3.8.D are exceeded, a special report is required to be submitted to the NRC. This item was selected for review because its provisions appear to be redundant to existing regulations and the only stipulated action is a reporting requirement.

This item has regulatory bases in 10 CFR 20 and Appendix I to 10 CFR 50. Since 10 CFR 20 specifically requires compliance with 40 CFR 190 and also specifies reporting requirements for releases of radioactive material in excess of the limits of 40 CFR 190, this Technical Specification appears to be redundant to the regulations. However, 10 CFR 20 and 40 CFR 190 merely provide an overall, broadly defined limit. Technical Specification 3.8.D provides the details of how to meet the regulations as well as actions to be taken when the limits are exceeded. Since the 40 CFR 190 dose levels are generally the more limiting requirements for permissible levels of radiation in unrestricted areas governed by 10 CFR 20, this item is safety relevant; it also includes the practice of ALARA principles.

While 10 CFR 50.36a requires Radiological Effluent Technical Specifications (RETS), Generic Letter 89-01 allows licensees to relocate the procedural details of the RETS to the Offsite Dose Calculation Manual (ODCM). Further, the Technical Specification Improvement Program governs the ODCM, Radiological Environmental Monitoring Program, and Radiological Effluent Control Program requirements as administrative controls in the Improved Standard Technical Specifications. One of the Radiological Effluent Control Program administrative provisions is the requirement that it include limitations to annual doses in accordance with 40 CFR 190. Thus, while the technical requirements that represent the origin of this item are retained in the Improved Standard Technical Specifications, they are handled there more coherently and flexibly as program requirements.

The recent revision to 10 CFR 20 impacts this area. The manner in which Technical Specification revisions relative to 10 CFR 20 changes will be handled is still under review by the NRC to effect a coherent and consistent, yet technically correct, process for compliance with the regulations.

Based on the above considerations, this item has the potential for enhanced flexibility if a change is pursued in accordance with the Technical Specification Improvement Program. However, given the recent revision to 10 CFR 20, further evaluation of the most effective way to implement regulatory compliance in this regard is ongoing by the NRC staff. Therefore, in light of the Technical Specification Improvement Program and the ongoing reviews, it is concluded that any additional consideration of this item for a further reduction in regulatory burden would prove unproductive.

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: F **Items: All**

The Peach Bottom operating license contains 10 items in Category F, "Unique Plant Features." These items were deemed appropriate to be assessed collectively. They identify the plant and its location and delineate the plant's major design features. Specifically, the Category F items are as follows:

OL 2.A	TS 5.1
TS 5.2.A	TS 5.2.B
TS 5.3	TS 5.4.A
TS 5.4.B	TS 5.4.C
TS 5.5	TS 5.6

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

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

SUMMARY OF PEACH BOTTOM ASSESSMENT

Category: G **Items: All**

The Peach Bottom operating license contains 81 items in Category G, "Other." These items were deemed appropriate to be assessed collectively. They include license conditions, Technical Specification definitions, and Environmental Technical Specification definitions. Specifically, the Category G items are as follows:

OL 1.A	OL 1.B
OL 1.C	OL 1.D
OL 1.E	OL 1.F
OL 1.G	OL 1.H
OL 1.I	OL 2.B.1
TS 1.O (56 items)	ES 1.0 (15 items)

Except for the financial qualification part of License Condition OL 1.E, the license conditions are legal findings that appear to be required by the Atomic Energy Act or the Commission's regulations. The financial qualification finding was required at the time the Peach Bottom operating license was issued but is no longer required. The Technical Specification definitions and Environmental Technical Specification definitions are judged necessary for the uniform interpretation of the defined terms.

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

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