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Date: August 31, 2000
NOC-AE-00000918
File: G03.15
STI: 31166047

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Revised Request for Exemption to Exclude Certain Components
From The Scope of Special Treatment Requirements Required by Regulations

Reference 1: Letter from J.J. Sheppard to the Document Control Desk dated July 13, 1999 (NOC-AE-000518)

In Reference 1, the STP Nuclear Operating Company (STPNOC) requested an exemption from various special treatment requirements in the regulations of the Nuclear Regulatory Commission (NRC). On January 18, 2000, the NRC issued a request for additional information (RAI) on the exemption request. As requested by NRC, STPNOC has prepared responses to the RAIs and has revised its exemption request to account for those responses and additional issues raised by NRC during meetings with STPNOC. Specifically:

- Attachment 1 to this letter provides STPNOC's revised exemption request.
- Attachment 2 to this letter provides a revised list of the specific regulations from which STPNOC is seeking an exemption.
- Attachment 3 to this letter provides STPNOC's revised responses to NRC questions identified as a result of the March 15, 1999, meeting between NRC and STPNOC on its draft exemption request.
- Attachment 4 to this letter provides STPNOC's responses to NRC's RAIs dated January 18, 2000.
- Attachment 5 demonstrates that the exemption request provides reasonable assurance that low safety significance (LSS) and non-risk significant (NRS) components will be able to perform their safety-related functions under design basis conditions.
- Attachment 6 demonstrates that the STP Updated Final Safety Analysis Report (UFSAR) is the proper location for STPNOC's commitment in this exemption request.
- Attachment 7 provides a proposed section for the UFSAR containing STPNOC's commitments with respect to this exemption request.
- Attachment 8 provides a description of STP's commercial practices for safety-related LSS and NRS components.

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This letter and its attachments supercede Reference 1 in its entirety.

The exemption request is based on risk-informed insights from the South Texas Project (STP) Probabilistic Risk Assessment (PRA) and is an extension of risk-informed STP programs (e.g. Graded Quality Assurance (GQA)) that have already been reviewed and approved by the NRC. The proposed exemption is consistent with the NRC's proposed Option 2 for risk-informed revisions to 10 CFR Part 50 as described in SECY-98-300, Options for Risk-Informed Revisions to 10 CFR Part 50, and the Advanced Notice of Proposed Rulemaking (ANPR) on Risk-Informing Special Treatment (65 Fed. Reg. 11488 (March 3, 2000)). STPNOC believes that this prototype application will assist the NRC in the development and implementation of risk-informed regulations.

As discussed in SECY-98-300 and the ANPR, NRC's regulations in Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of components that are safety-related, important to safety, or otherwise within the scope of the regulations. These special treatment requirements go beyond normal commercial and industrial practices, and include quality assurance (QA) requirements, environmental and seismic qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. The scope of these regulations applies to some components that have little or no safety or risk significance.

The NRC has approved a risk-informed process for determining the safety/risk significance of structures, systems, and components (SSCs) as part of the GQA program for STP. In accordance with this approved risk-informed process, STPNOC has determined that certain safety-related components in the plant are LSS or NRS. STPNOC now requests an exemption to exclude LSS and NRS components from the scope of NRC's regulations that impose special treatment requirements. STPNOC also requests an exemption from the associated process-related regulations that require prior NRC approval of certain changes in the special treatment provisions in the licensing basis (or alternatively requests that the NRC take certain actions that would obviate the need for prior NRC approval of such changes).

This exemption request is being developed primarily to enhance nuclear safety at the site by allowing more resources to be focused on safety/risk. In addition, STPNOC anticipates savings that will exceed $2 million per year for component and part replacements and for streamlining and enhancing processes for LSS and NRS components.

STPNOC requests that the exemption be approved not later than April 15, 2001 to facilitate its implementation and to facilitate industry planning for developing risk-informed regulations.

If you have any questions, please call Mr. Glen E. Schinzel at 361-972-7854 or me at 361-972-8757.

J. J. Sheppard
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STI: 31166047

LIST OF ATTACHMENTS

Attachment 1:	Revised Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations
Attachment 2:	Revised List of Regulations for Which an Exemption is Requested
Attachment 3:	Revised Responses to Preliminary NRC Questions
Attachment 4:	Responses to the Request for Additional Information
Attachment 5:	Assurance of Functionality of LSS and NRS Components
Attachment 6:	Location of STP's Commitments Related to the Exemption Request
Attachment 7:	Proposed UFSAR Section 13.7, "Risk-Informed Special Treatment Requirements"
Attachment 8:	Elements of the South Texas Project Commercial Treatment for Safety-Related LSS and NRS SSCs

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ATTACHMENT 1

REVISED REQUEST FOR EXEMPTION TO EXCLUDE CERTAIN COMPONENTS FROM THE SCOPE OF SPECIAL TREATMENT REQUIREMENTS REQUIRED BY REGULATIONS

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Revised Request for Exemption to Exclude Certain Components
From The Scope of Special Treatment Requirements Required by Regulations

1.0 Introduction

As discussed in SECY-98-300, *Options for Risk-Informed Revisions to 10 CFR Part 50*, and the Advanced Notice of Proposed Rulemaking (ANPR) *on Risk-Informing Special Treatment Requirements* (65 Fed. Reg. 11488 (March 3, 2000)), the regulations of the Nuclear Regulatory Commission (NRC) in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of components that are safety-related, important to safety, or otherwise within the scope of the regulations. These special treatment requirements go beyond normal commercial and industrial practices, and include quality assurance (QA) requirements, environmental and seismic qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. The scope of these regulations applies to some components that have little or no safety or risk significance.

The NRC has approved a risk-informed process for determining the safety/risk significance of structures, systems, and components (SSCs) as part of the Graded Quality Assurance (GQA) program for the South Texas Project (STP). In accordance with this approved risk-informed process, STP has determined that certain components in the plant are low safety/risk significant (LSS) or non-risk significant (NRS), and that other components are high safety/risk significant (HSS) or medium safety/risk significant (MSS).

STP requests an exemption to exclude LSS and NRS components from the scope of NRC's regulations that impose special treatment requirements. STP also requests an exemption from the associated process-related regulations that require prior NRC approval of certain changes in the special treatment provisions in the licensing basis (or alternatively requests that the NRC take certain actions that would obviate the need for prior NRC approval of such changes, as discussed in Section 4.2 of this request). The specific regulations in question are identified in Attachment 2.

Although STP is requesting an exemption to exclude LSS and NRS components from the scope of the special treatment requirements, STP will apply normal industrial and commercial practices to these components. These practices provide reasonable assurance that these components will be able to perform their safety functions, commensurate with their significance to safety. Additionally, STP will be evaluating non-safety-related HSS and MSS components to determine whether enhanced treatment is warranted for their safety-significant functions. Thus, STP expects that this exemption will result in an overall risk benefit, or at least will be risk neutral.

This exemption request provides the information required by 10 CFR 50.12 and 21.7 and is organized as follows:

- Section 2.0 identifies the objectives of the exemption request.

- Section 3.0 provides background regarding the NRC-approved process for determining the safety/risk significance of SSCs at STP, and STP's proposed treatment for various categories of components.

- Section 4.0 identifies the regulations to which the exemption would apply.

- Section 5.0 provides the technical justification for the exemption, including a discussion of the relevant provisions in Regulatory Guide 1.174 applicable to risk-informed requests for NRC approval of changes to the licensing basis.

- Section 6.0 demonstrates that special circumstances exist under 10 CFR 50.12 for the granting of this exemption request.

- Section 7.0 describes STP's planned approach for a phased implementation of the exemption and identifies STP's provisions to facilitate the NRC's review and inspection of implementation of the exemption.

- Section 8.0 concludes that the exemption request is justified and should be granted.

2.0 Objectives of the Exemption Request

NRC's regulations governing special treatment requirements have different scopes. Some of the regulations apply to safety-related components (e.g., Part 21 and Appendix B to Part 50); some apply to components that are important to safety (e.g., the General Design Criteria in Appendix A to Part 50); and some apply to safety-related and a specific subset of non-safety-related components (e.g., 10 CFR 50.49 and 50.65). Some (but not all) LSS and NRS components come within the scope of each of these regulations.

The GQA program only allows STP to apply a reduced level of quality assurance to MSS, LSS, and NRS components, commensurate with their safety/risk significance. In practice, a fully risk-informed set of controls defined by a risk informed program cannot be implemented due to the special treatment requirements imposed by 10 CFR Parts 21, 50, and 100 for SSCs that are safety-related, important to safety, or otherwise within the scope of the regulations. Regardless of their risk significance, SSCs that have been classified as safety-related are subject to requirements for quality assurance (QA) in Appendix B to Part 50, environmental qualification (EQ) under 10 CFR 50.49, inservice testing and inspection under 10 CFR 50.55a, and monitoring the effectiveness of maintenance under 10 CFR 50.65. Given the lack of safety/risk importance of LSS and NRS components in protecting the health and safety of the public as determined by the approved risk-informed categorization process, an exemption is appropriate to exclude LSS and NRS components from the scope of the special treatment requirements in these regulations.

The exemption will enable STP to remove LSS and NRS components from the scope of special treatment requirements, to the extent that such components come within the scope of NRC's regulations. As a result, this exemption will reduce occupational radiation exposures and costs that would otherwise be incurred in providing unnecessary qualifications, quality assurance controls, maintenance monitoring requirements, testing, and inspection for the LSS and NRS components. LSS and NRS components have no significant effect on public health and safety and do not warrant such comprehensive levels of control.

STP is not requesting an exemption from the provisions in 10 CFR Parts 50 and 100 that specify design or functional requirements for SSCs; i.e., the requirements that specify the safety functions to be performed by a system or component (including features to prevent adverse impacts upon the safety function of one SSC due to the failure of another SSC). Furthermore, STP is not proposing to use this exemption to change any design or functional requirements in the Updated Final Safety Analysis Report (UFSAR) or the Technical Specifications. However, because STP will not be applying the special treatment requirements in Parts 21, 50, and 100 to such components, replacement LSS and NRS components could be procured from non-Appendix B suppliers or as commercial grade without meeting the rigorous quality assurance and qualification requirements normally imposed on components that are safety-related, important to safety or otherwise within the scope of the regulations.

In summary, STP is requesting a limited exemption from the regulations that impose special treatment requirements. The exemption would enable STP to exclude only LSS and NRS components from the scope of the special treatment requirements. To the extent applicable, HSS and MSS components would still be subject to those regulations (and will still be governed by the GQA program as approved by the NRC in 1997). LSS and NRS components would still be subject to those regulations to the extent that they also impose design or functional requirements.

Furthermore, the regulations in 10 CFR 50.59 require a written evaluation for changes in the special treatment provisions in the UFSAR, and some of these UFSAR changes may require prior NRC approval. Preparing an evaluation and seeking prior NRC approval for each of these changes would be extremely inefficient and overly burdensome for both STP and the NRC. Therefore, an exemption from Section 50.59 is essential to permit STP to implement these changes without the need for a separate evaluation and approval of each change. For similar reasons, exemptions are needed from 10 CFR 50.54(a), which restricts changes in the QA program description. In the alternative, STP requests that the NRC take certain actions, as described in more detail in Section 4.2, that would obviate the need for prior NRC approval of such changes.

3.0 Background

3.1 GQA Risk-Informed Program

On March 28, 1996, STP submitted a request under 10 CFR 50.54(a) for approval of the GQA program. The purpose of that request was to enable STP to implement its QA program in a graded manner, commensurate with the SSC's significance to safety. The NRC conducted an extensive review of the proposed GQA program, including holding many meetings with STP, evaluations at the site, telephone discussions, and requests for additional information. In

response, STP submitted supplementary information, followed by a final submittal on August 4, 1997. By letter dated November 6, 1997, the NRC determined that the GQA program was acceptable and transmitted a Safety Evaluation that documented its detailed review. The final approved version of the GQA program was made effective for implementation on February 1, 1998, as part of Revision 13 of the Operations Quality Assurance Program.

The GQA program, as approved by the NRC, applies to adjustments in STP quality assurance controls based on a safety/risk significance evaluation process. The GQA program utilizes a risk-informed process for determining the safety/risk significance of SSCs, including:

- The use of an Expert Panel and associated Working Groups to implement a risk-informed comprehensive risk management program.
- A methodology, which combines probabilistic risk assessment (PRA) data with deterministic insights and performance-based information, to determine the safety/risk significance of SSCs.
- A configuration control program which documents the safety/risk significance evaluation process in the form of system-specific risk significance basis documents.
- A feedback loop to reassess SSC safety/risk significance when new information becomes available, such as PRA changes and SSC performance changes.

The process for determining the safety/risk significance of SSCs places a component into one of four categories: 1) high safety/risk significant (HSS), 2) medium safety/risk significant (MSS), 3) low safety/risk significant (LSS), and 4) non-risk significant (NRS). STP experience to date demonstrates that the majority of the safety-related components are categorized as LSS and NRS because they support ancillary functions (e.g., vents, drains, and local indications) for safety-related systems and do not provide, support, or affect the primary safety functions of the systems.

In the first 29 systems evaluated under the GQA program, approximately 90% of all components (out of more than 43,600 components in total) are either LSS or NRS. This component population includes approximately 16,700 safety-related components. Of these safety-related components, approximately 23% are safety/risk significant (i.e., HSS or MSS), and approximately 77% are not safety/risk significant (i.e., LSS or NRS). Currently, there are 372 non-safety-related components that have been risk ranked as MSS or HSS. Of these, 220 are fire dampers in the Mechanical Auxiliary Building HVAC System. The response to RAI 35 in Attachment 4 provides a representative sample of non-safety-related MSS and HSS components.

To date, STP has applied its categorization process to tagged components; i.e., to components identified on the Master Equipment Database (MED), and to date has not applied this process to items that are not listed in the MED, such as pipes, cables, and structures. However, STP desires the option of applying its categorization process to items that are not in the MED. Therefore, this exemption request applies to both MED components and non-MED components. A table of representative MED components is included in the response to RAI 1 in

Attachment 4. STP currently plans to implement the exemption for non-MED components on a case-by-case basis.

Using PRA and deterministic insights, STP's categorization process establishes the safety/risk importance of a component based upon its failure probability/impact on the safety functions supported by the component. As a result, the results of this categorization process are independent of, and are not affected by, the application of the results of the process. Thus, the categorization process is appropriate for application in any risk-informed context - not only for use in graded quality assurance applications, but also for applications in maintenance, engineering, and operations.

3.2 STP's Current Risk-Informed Categorization Process

STP determines the safety/risk significance of components using an integrated decision-making process that uses both risk insights and deterministic insights. In categorizing a component, STP considers the relationship of the component to plant risk by using the STP PRA. Additionally, STP considers the contribution of the component to defense-in-depth and safety margins. These factors are discussed in more detail below.

3.2.1 Capability of the PRA to Support the Categorization Process

STP's PRA models internal and external initiating events at full power, and calculates both core damage frequency (CDF) and large early release frequency (LERF). Some of the external events that are addressed in the STP PRA include: external floods from main cooling reservoir breach; tornado that fails offsite power and the essential cooling pond; seismic events from 0.1 to 0.6g (Note: the safe shutdown earthquake for STP is 0.1g); and internal fires. The evaluation of seismic events and other external events are well beyond the design basis external events. All of these external events are included in the STP PRA results and are explicitly included in all risk categorizations that are based on the PRA. STP's PRA also includes equipment failure contributions due to environmental effects. The environmental effects are found in the spatial interactions analysis for the fire PRA and previous studies submitted to the NRC. Finally, STP considers shutdown and mode changes in its deterministic categorization process, as discussed in more detail in Section 3.2.2.2 below.

The PRA also accounts for common cause failures. However, the process has evolved since the GQA program was implemented. The PRA for the GQA program used an overly conservative estimation of risk by double and triple counting certain event risks. Specifically, as each component's PRA was computed, certain events were determined to be common to several components. Rather than distributing the event's risk among the various components, the GQA analysis applied the full risk to each affected component – thus unnecessarily multiplying the event's risk. In order to eliminate some of the conservatism associated with the above process, STP now splits the importance of multiple term common cause failure events evenly among their constituent components.

The quality of STP's PRA is discussed in more detail in the response to RAI 32 in Attachment 4. As this response shows, the STP PRA has undergone extensive NRC reviews in support of license amendments. Most recently, the NRC found the quality of STP's PRA to be acceptable as part of its review of STP's GQA program.

The PRA configuration control program incorporates a feedback process to update the PRA Model. The updates are segregated into two categories:

- The first category, called a plant operating update, incorporates plant design changes and procedure changes that affect PRA-modeled components, initiating event frequency updates, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months.

- The second category, called a comprehensive data update, incorporates changes to plant-specific failure rate distributions and human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months. The "comprehensive data update" will allow for the accumulation of enough data to properly update distributions without creating an undue burden to perform these time-consuming data updates.

The 36-month frequency specified above is based upon the proposed Nuclear Energy Institute's (NEI) frequency for PRA model updates. In addition, the 60-month frequency is based upon data self-assessments performed by STP personnel earlier this year. Any update will entail an engineering evaluation to determine if applicable 'state-of-the-art' changes relative to PRA technology should be incorporated.

The PRA model may be updated on a more frequent basis if an update would result in a significant increase in the CDF. All potential model changes are evaluated on a monthly basis with respect to the impact on CDF. If a potential change would result in an increase in CDF of greater than 10%, the model can be updated at the discretion of the PRA Administrator. Also, during the "plant operating update", each of the data variables updated normally during the "comprehensive data update" will be evaluated for changes in the data. If plant specific history demonstrates an approximate order of magnitude change, then that PRA variable can be updated during the "plant operating update" at the discretion of the PRA Administrator.

The results of the updated PRA and the associated PRA risk categorizations are furnished to the Working Group, which, as part of the feedback and corrective action process, recategorizes the safety/risk significance of components as needed.

3.2.2 Component Categorization Using the Integrated Decision-Making Process

STP's current risk-informed categorization process evaluates the safety/risk significance of individual components. Under STP's risk-informed categorization process, a component's importance is separately rated based upon PRA risk insights and deterministic insights. The higher of the two ratings is then used as the component's final categorization. Each of these categorization processes is discussed below.

3.2.2.1 PRA Risk Categorization Process

The relative importance of components modeled in the PRA is determined using PRA importance measures. The results of this process are used as inputs for the risk categorization of components. Additionally, STP has performed sensitivity studies to verify that its risk categorization of components will not be adversely affected by the removal of LSS and NRS components from the scope of the special treatment requirements. Each of these is discussed below.

3.2.2.1.1 Use of PRA Importance Measures

Only components that are modeled in the PRA are given a PRA risk categorization. The risk significance of a component that is modeled in the PRA is based upon its Fussell-Vessely (FV) importance, which is the fraction of the CDF and LERF to which failure of the component contributes, and its risk achievement worth (RAW), which is the factor by which the CDF and LERF would increase if it were assumed that the component is guaranteed to fail. Specifically, risk significance is based upon the following:

PRA Ranking	Criteria
High	**RAW ≥ 100.0 or** **FV ≥ 0.1 or** **FV ≥ 0.005 and RAW ≥ 2.0**
Medium (Further Evaluation is Required)	**FV < 0.005 and 100.0 > RAW ≥ 10.0**
Medium	**FV ≥ 0.005 and RAW < 2.0 or** **FV < 0.005 and 10.0 > RAW ≥ 2.0**
Low	**FV < 0.005 and RAW < 2.0**

These importance measures are calculated using the STP PRA, which is a fully integrated model of plant risk from all categories of initiators, both internal and external. This means that all initiating events are included in all model quantifications. Thus, the resulting risk importance measures are determined from sequences that are representative of all the initiating events.

Because the STP PRA is an integrated model, STP does not routinely calculate risk importance measures for specific classes of initiating events. However, in response to RAIs 32 and 36 in Attachment 4, STP performed a special evaluation that separately looked at the risk importance measures by class of external event (fires, seismic, external floods). Based upon this evaluation, STP concluded that a component's risk ranking would not be increased from a separate calculation of the risk importance measures for different classes of external events. This result is attributable to the overall small importance that external events have on the PRA for STP.

As mentioned above, a component's categorization based upon this risk categorization process may be increased, but may not be decreased, as a result of the deterministic categorization process discussed in Section 3.2.2.2 below.

3.2.2.1.2 Evaluation of Overall Risk Impacts

For the purposes of this exemption request, STP performed an evaluation to determine the overall risk impacts of changing the treatment to be afforded to LSS and NRS components.

As explained in detail in Section 5.2.4.1 below, none of the NRS components and only a small fraction of the LSS components are modeled in the PRA. Therefore, even if changes were to occur in the availability and reliability rates for these components, such changes would not affect the results of the PRA or the results of the risk categorization process based on PRA insights.

As shown by available industry and STP data discussed in Section 5.2.4.1, the failure rates for non-safety-related components are generally similar to the failures for the same type of safety-related components. Therefore, the change from special treatment to commercial treatment is not expected to have any appreciable effect on the failure rates of LSS and NRS components. Consequently, to the extent that some LSS components are modeled in PRA, the results of the PRA should not be affected as a result of this exemption.

Finally, even if it were assumed that the failures rates of LSS and NRS components will significantly increase as a result of the exemption, there would be little impact upon the results of the PRA. For the purposes of this exemption request, STP has performed special sensitivity analyses to determine the impact of increasing the failure rates of LSS and NRS components by up to a factor of ten. The results of those sensitivity analyses demonstrate that the impact of such increased failure rates would be insignificant (e.g., an increase in CDF and LERF by about 2%) and would satisfy the applicable quantitative criteria in Regulatory Guide 1.174. Furthermore, this postulated increase in failure rates had no impact on the risk categorization of the components.

3.2.2.2 Deterministic Categorization Process

Components are subject to a deterministic categorization process, regardless of whether they are also subject to the risk categorization process using PRA insights. This deterministic categorization process can result in an increase, but not a decrease, in a component's categorization.

The first step in the deterministic categorization process consists of identifying the system functions performed by the selected plant system. Next, each component is evaluated to identify the system function(s) supported by that component. This is documented in a Risk Significance Basis Document (RSBD). Then, the component is initially assigned the same risk as the most limiting system function that it supports. If the Working Group reaches consensus that the initial risk is satisfactory, no additional documentation is required since the justification has already been provided within the function categorization. Only in the case where component redundancy or other insight is used to rank the component category lower than the category of its most limiting supported system function would additional documentation be required. These steps are described below in more detail.

A component's deterministic importance is directly attributable to the importance of the function supported by the component. In cases where a component supports more than one system function, the component is categorized based on the highest categorization based on the functions supported. In categorizing the functions of a system, the Working Group considers five critical questions regarding the function, each of which is provided a weighting factor. These questions and their weight are as follows:

QUESTION	WEIGHT
Is the function used to mitigate accidents or transients?	5
Is the function specifically called out in the emergency operating procedures (EOPs) or Emergency Response Procedures (ERPs) (which provides beyond design basis configuration)?	5
Does the loss of the function directly fail another risk-significant system?	4
Is the loss of the function safety significant for shutdown or mode changes?	3
Does the loss of the function, in and of itself, directly cause an initiating event?	3

Based on the impact on safety if the function is unavailable and the frequency of loss of the function, each of the five questions is given a numerical answer ranging from 0 to 5. This grading scale is as follows:

"0" - Negative response

"1" - Positive response having an insignificant impact and/or occurring very rarely

"2" - Positive response having a minor impact and/or occurring infrequently

"3" - Positive response having a low impact and/or occurring occasionally

"4" - Positive response having a medium impact and/or occurring regularly

"5" - Positive response having a high impact and/or occurring frequently

STP has developed definitions for the terms used in this grading scale. The definitions are as follows:

Frequency Definitions –

- Occurring Frequently – continuously or routinely demanded
- Occurring Regularly – demanded > 5 times per year
- Occurring Occasionally – demanded 1-2 times per cycle
- Occurring Infrequently – demanded < once per cycle
- Occurring Very Rarely – demanded once per lifetime

Impact Definitions –

- High Impact – a system function is lost which likely could result in core damage and/or may have a negative impact on the health and safety of the public
- Medium Impact – a system function is lost which may, but is not likely to, result in core damage and/or is unlikely to have a negative impact on the health and safety of the public
- Low Impact – a system function is significantly degraded, with very low likelihood of core damage, and no negative impact on the health and safety of the public is expected
- Minor Impact – a system function has been moderately degraded, but there is no core damage or negative impact on the health and safety of the public
- Insignificant Impact – a system function has been challenged, but there is no core damage or negative impact on the health and safety of the public

Although some of these definitions are quantitative, both of these sets of definitions are applied based on the collective judgment and experience of the Working Group.

The numerical values, after weighting, are summed; the maximum possible value is 100. Based on the sum, functions are categorized as follows:

SCORE RANGE	CATEGORY
0 – 20	NRS
21 – 40	LSS
41 – 70	MSS
71 – 100	HSS

A function with a low categorization due to a low sum could still receive a higher risk classification if one of its five questions received a high numerical answer. Specifically, a weighted score of 25 on any one question would result in an HSS categorization; a weighted score of 15-20 on any one question would result in a minimum categorization of MSS; and a weighted score of 9-12 on any one.question would result in a minimum categorization of LSS. This is done to ensure that a component with a significant risk in one area does not have that risk masked because of its low risk in other areas.

In general, a component is given the same categorization as the most limiting (highest category) system function that the component supports. However, a component may be ranked lower than the associated system function(s), based upon several criteria. For example:

- A component's categorization may be considered for one level lower than the most limiting system function when there are diverse means of satisfying the system function.
- If there are multiple, independent means of satisfying the system function, a reduction in categorization may be considered. However, merely having multiple trains of a component available in a system does not automatically result in a lower categorization for a component.
- The final category of a component cannot be NRS if the system function is LSS (or higher), and generally cannot be more than one category level lower than the category of the most limiting system function (an exception to this is that Local and Auxiliary Shutdown Panel handswitches may be ranked two levels lower for HSS functions).

When considering component redundancy or diversity, the Working Group evaluates redundancy based on system operating configuration, reliability history, recovery time available, and other factors. The Working Group examines the effect of the component failure on each system function supported by that component. The primary consideration is whether failure of the component will fail or severely degrade the function. If the answer is no, then component redundancy may be factored in, as long as the component's reliability and that of its redundant counterpart have been satisfactory. A component can be considered reliable when the component demonstrates strong operating performance with few deficiencies, the component has no open concerns that are applicable to STP based on industry operating experience, and site operating experience reflects no negative reliability trends or concerns. In all cases, the component's final categorization cannot be lower than the PRA risk categorization if the component is modeled in the PRA.

The categorization process is an iterative process based upon the Working Group's professional judgment. In Working Group meetings, the members' insights and varied experience are used to ensure that the final result reflects a comprehensive and justifiable deterministic judgment. It should also be emphasized that the Working Group and Expert Panel can deviate, in a risk-conservative direction, from the category assigned to a component using the above process.

3.2.2.3 Maintaining Defense-in-Depth

STP's categorization process includes consideration of a component's contribution to defense-in-depth. In particular, consideration is given to the following factors:

- Balance among prevention of core damage, prevention of containment failure or bypass, and mitigation of offsite releases - - The deterministic categorization process requires a consideration of the contribution of a component to prevention of initiating events, to mitigation of accidents, and to emergency actions under the EOPs and ERPs.

- Preservation of system redundancy, independence, and diversity - - The deterministic categorization of a component is based upon the significance of the system function supported by that component. A component's categorization may be considered for one level lower than the categorization of the most limiting system function if there are diverse means or multiple, independent means of satisfying the system function. However, merely having multiple trains of a component available in a system does not automatically result in a lower categorization for a component.

- No over-reliance on programmatic or operator actions - - The deterministic categorization process is based upon the system function supported by a component, and does not allow for any reduction in a component's safety significance based upon an over-reliance on programmatic or operator action.

- Consideration of common cause failures - - The PRA and associated sensitivity analyses combine to account for the potential for common cause failures.

In addition, as discussed in detail in Section 5.2.2 below, the exemption itself will preserve defense-in-depth, because its does not involve any change in the design of the plant.

3.2.2.4 Maintaining Safety Margins

STP's categorization process includes consideration of a component's contribution to maintenance of safety margins, including uncertainty in the analysis and supporting data. In particular:

- The categorization process does not allow for changes in the design or function of SSCs (e.g., setpoints, procedures). Changes to the design of functions are addressed in the design control process. STP has not requested an exemption from the requirements of Appendix B, Criterion III, Design Control.

- As discussed in more detail in Section 5.2.4.1, STP performed sensitivity analyses to determine the impact on CDF and LERF from a decrease in the reliability of LSS and NRS components. The sensitivity analyses demonstrate that there is insignificant impact on CDF and LERF with a ten-fold increase in component failure rates. Increases of this magnitude would be sufficiently obvious and undesirable as to result in an appropriate corrective action in accordance with the Corrective Action Program. A ten-fold increase in failure rates bounds any reasonable estimate of the increase that may be expected as a result of the exemption.

- As discussed above, the deterministic categorization of a component is based upon the significance of the system function supported by that component. A component's categorization may be considered for one level lower than the most limiting system function if there are diverse means or multiple, independent means of satisfying the system function. However, merely having multiple trains of a component available in a system does not automatically result in a lower categorization for a component.

- The categorization process using PRA insights and the deterministic categorization process each account for the consequences if a component fails to perform its function. In particular, the risk categorization process includes criteria based upon the Risk Achievement Worth (RAW), which explicitly evaluates the impact on CDF and LERF if the component is postulated to fail. Additionally, the deterministic categorization process explicitly considers the impact on system safety functions in the component fails. A component is only classified as LSS or NRS if the impact is no more than minor.

3.2.3 Documentation of the Integrated Decision-Making Process and the Decision Criteria Used

3.2.3.1 Working Group and Expert Panel

The integrated decision-making process used by STP is documented by procedure. The procedure requires the use of an Expert Panel and a Working Group. The roles and responsibilities of these groups are described below.

The Expert Panel:

- is composed of a group of senior-level personnel with diverse backgrounds
- approves the criteria for assessing the safety/risk significance of SSCs.
- approves and issues documents communicating risk-informed decisions.
- appoints the Working Group.
- assesses the overall station risk impact due to SSC performance and implemented risk-informed programs after each plant-specific data update of the PRA.
- retains appropriate documented decisions and supporting documents as quality records.

The Working Group:

- is composed of experienced personnel who possess diverse knowledge and insights into the plant and uses deterministic knowledge and risk insights in making component classifications.
- analyzes performance data and considers available risk information when developing recommendations.
- documents recommendations and includes rationale that forms the bases for the recommendations.
- provides recommendations to the Expert Panel for approval.
- following Expert Panel approval, takes appropriate action to facilitate implementation of the decisions.

Procedures control the composition of and processes used by the Expert Panel and Working Group. In particular, procedures designate the managers who are to act as the chairman, members, and alternates for the Expert Panel, and the procedures also identify the groups to be represented on the Working Group. The Expert Panel and Working Group have expertise in the areas of risk assessment, quality assurance, licensing, engineering, and operations and maintenance. Additionally, the cognizant system engineer participates in Working Group discussions for that system. Procedures also identify training requirements for members of the Expert Panel and Working Group, including training on probabilistic risk assessment, risk ranking, and the graded quality assurance process. Finally, the procedures specify the requirements for a quorum of the Expert Panel and Working Group, meeting frequencies, the decision-making process for determining the categorization of components, the process for resolving differing opinions among the Expert Panel and Working Group, and periodic reviews of the appropriateness of the programmatic control and oversight provided to categorized components.

The deterministic categorization process is an iterative process based upon the Working Group's professional judgment. In Working Group meetings, the members' insights and varied experience are used to ensure that the final result reflects a comprehensive and justifiable deterministic judgment. If during this iterative process a consensus agreement cannot be reached by the Working Group members, a 'Dissenting Opinion' is documented and forwarded to the Expert Panel for resolution prior to documenting a final categorization. Similarly, if there is a dissenting opinion on the Expert Panel, it is forwarded to senior management for resolution.

As discussed in more detail in Section 3.2.1 above, the PRA is updated on a periodic basis, and may be updated more frequently in response to other factors. The results of the updated PRA are furnished to the Working Group which, as part of the feedback and corrective action process, recategorizes the risk significance of components as needed based upon the process described in Section 3.2.2.1 above.

3.2.3.2 Documentation of the Integrated Decision-Making Process

The following documentation of the integrated decision-making process exists and is available for NRC review:

- The results of the relative risk importance of components modeled in the PRA
- The results of the sensitivity analyses discussed in Section 5.2.4.1
- The Risk Significance Basis Documents (RSBD), which include system functions, the identification of the risk significance of components, the categorization of components, and the bases for determinations (including PRA inputs, performance analysis results, and deterministic inputs)
- The values for CDF and LERF
- The procedures described in the preceding section
- The PRA

As discussed above, a component is initially assigned the same category as the most limiting system function that it supports. If the Working Group reaches consensus that the initial category is satisfactory, the RSBD simply records that determination and no additional documentation is required since the justification has already been provided within the function categorization. Only in the case where component redundancy or other insight is used to rank the component category lower than the category of its most limiting supported system function is additional documentation required.

As part of its review of the GQA program, the NRC staff reviewed the Risk Significance Basis Documents used to categorize components in the Radiation Monitoring, Essential Cooling Water, and Emergency Diesel Generator systems. The staff concluded in Section 3.3.4 of its Safety Evaluation for the GQA program that the documents are "comprehensive, well organized, and capable of providing a scrutable record of the functional relationships linking system functions to individual component attributes for proper categorization" of components. The documentation of the safety/risk significance evaluation process and the evaluation results for the other completed systems will also be available for NRC review at the STP site.

3.3 Treatment for Component Categories

3.3.1 Safety-Related HSS and MSS Components

Safety-related HSS and MSS components continue to receive the treatment required by NRC regulations and STP's associated implementing programs. Additionally, STP evaluates the risk-significant functions performed by these components to determine whether there are any functions that are not being treated under STP's current programs. If STP does identify any risk-significant function that is not currently being treated as safety-related: 1) it will be evaluated for enhanced treatment as discussed in Section 3.3.6 below, 2) an engineering evaluation will be performed to ensure that the component can perform the function, and 3) the function will be subject to STP's design control program and Corrective Action Program.

The risk-significant functions of the safety-related HSS and MSS components will be subject to monitoring under the Maintenance Rule at the system/ train level. Furthermore, these functions will be subject to the monitoring described in Section 5.2.5 below.

3.3.2 Non-Safety-Related HSS and MSS Components

Non-safety-related HSS and MSS_components will continue to receive any existing special treatment required by NRC regulations and STP's implementing programs. Additionally, the risk-significant functions of these components will receive consideration for enhanced treatment. This process is described in Section 3.3.6 below.

The risk-significant functions of the non-safety-related HSS and MSS components will be subject to monitoring under the Maintenance Rule at the system/train level. Furthermore, these functions will be subject to the monitoring described in Section 5.2.5 below.

3.3.3 Safety-Related LSS and NRS Components

Safety-related LSS and NRS_components receive normal commercial and industrial practices. These practices are described in Section 3.3.7 below.

The safety-related LSS and NRS components will be subject to monitoring under the Maintenance Rule at the plant, system, or train level as appropriate. Monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. Although LSS and NRS component performance will not be explicitly monitored, degrading performance in these components, if significant, would be observed at the plant, system, or train level. Failures of LSS or NRS components that result in the performance criteria being exceeded will be identified and resolved under a corrective plan of action, which includes actions to prevent recurrence. Furthermore, the safety-related functions of these components will be subject to the monitoring described in Section 5.2.5 below.

As discussed in Section 4.2.1 below, STP is requesting an exemption from 10 CFR 50.59 with respect to changes in special treatment requirements in the UFSAR as applicable to those components categorized as either LSS or NRS. However, the 50.59 evaluation process will still be used for LSS or NRS components for issues unrelated to changes in special treatment (e.g., design changes in LSS and NRS components).

3.3.4 Non-Safety-Related LSS and NRS Components

The treatment of non-safety-related LSS and NRS components is not subject to regulatory control.

3.3.5 Uncategorized Components

Until a component is categorized, it is not considered in the scope of this exemption request and continues to receive the treatment required by NRC regulations and STP's associated implementing programs, as applicable.

3.3.6 Enhanced Treatment for Non-Safety Related HSS and MSS Components

Non-safety-related HSS and MSS components may perform risk-significant functions that are not addressed by STP's current treatment programs. As discussed below, STP evaluates these components to identify whether enhanced treatment for such components is warranted.

When a non-safety-related component is categorized as HSS or MSS, STP documents the condition under the corrective action program and determines whether enhanced treatment is warranted to enhance the reliability and availability of the function. In particular, STP evaluates the treatment applied to the component to ensure that the existing controls are sufficient to maintain the reliability and availability of the component in a manner that is consistent with its categorization. This process evaluates the reliability of the component, the adequacy of the existing controls, and the need for any changes. If changes are needed, additional controls are applied to the component. In addition, the component is placed under the Maintenance Rule monitoring program, if not already scoped in the program (i.e., failures of the component are evaluated and Maintenance Rule Functional Failures (MRFF) involving the component are counted against the performance criteria at the plant/system/train level, as applicable). Additionally, as provided in the approved GQA program, non-safety-related HSS and MSS components are subject to the TARGETED QA program. TARGET controls apply to non-safety-related, safety significant components that have been risk-categorized as HSS or MSS. TARGET controls are subject to specific regulatory special treatment requirements and to additional administrative controls. These controls will be specifically 'targeted' to the critical attributes that resulted in the component being categorized as HSS or MSS. Components under these controls will remain non-safety-related and will be procured commercial, but the special treatments will be appropriately applied to give additional assurance that the component will be able to perform its function when demanded. Examples of process enhancements that could be applied to non-safety-related HSS and MSS components include:

- Performing routine preventive maintenance (PM) tasks more frequently to ensure component reliability
- Ensuring that the component's critical attributes are functionally validated following maintenance activities
- Ensuring that controls for replacement parts are enhanced (e.g., TARGETED components receive engineering evaluation for item equivalency replacements).
- Increasing the quality oversight of work activities and work documentation (e.g., audits, surveillances, and monitoring activities are performed for TARGETED components).

These identified processes provide reasonable assurance that non-safety related HSS and MSS components will be able to perform their safety significant functions.

For safety related HSS and MSS components, STP considers that these components and their critical attributes are within the bounds of the licensing and design basis (in other words, for safety-related components, STP does not take credit for any functions in the PRA that are not already part of the design and licensing basis). Of all of the functions that a component is designed to perform, those that are associated with its risk significance are identified as critical attributes. Therefore, critical attributes are a subset of the design functions of a component and are within the bounds of the licensing and design basis. While there may be risk significant events beyond the scope of the design basis where safety related HSS and MSS components would be required to function, they would not be called upon to perform functions that have not already been identified as critical attributes. Therefore, existing special treatment requirements and controls regulating facility changes are considered to be adequate to preserve these risk significant critical attributes.

3.3.7 Normal Commercial and Industrial Practices for Safety-Related LSS and NRS Components

STP does not have a specific procedure or program labeled as a "Commercial Treatment Program". Rather, the commercial treatment elements are contained in numerous plant programs and procedures (most of which are applicable to both safety-related and non-safety-related equipment and processes). These programs and procedures have been successfully implemented on the non-nuclear Balance of Plant portion of the Station and have been effective in maintaining its availability and reliability. In addition, commercial treatment has been proven to provide reasonable assurance that the Balance of Plant SSCs can satisfy their design functional requirements. These same commercial treatment elements have historically been used to provide an appropriate level of assurance that HSS and MSS non-safety-related SSCs can satisfy their design functional requirements. Since these commercial treatment elements have successfully ensured the design functional requirements of HSS and MSS non-safety-related SSCs, these commercial treatment practices will be adequate to provide an appropriate level of assurance that LSS and NRS safety-related SSCs can satisfy their design functional requirements.

The paper entitled *Assurance of Functionality of LSS and NRS Components* provides further support for the conclusion that commercial treatment will provide reasonable assurance that safety-related LSS and NRS SSCs will be able to perform their design functional requirements. This paper is enclosed in Attachment 5.

The following summarizes the Commercial Treatment elements currently in place at STP, as they will be implemented on LSS and NRS safety-related SSCs when the exemption request is granted. This summary is also provided in Attachment 8.

Note: The description of STP's commercial treatment practices lists, by number and title, examples of relevant procedures. The listing of such procedures should not be construed as incorporating those procedures by reference.

3.3.7.1 Accountability, Responsibility, and Organization

The general authorities, responsibilities, and accountabilities for STP personnel engaged in power plant activities (both safety-related and non-safety-related) are defined and described in approved Station procedures and policies. Examples of the subjects addressed by these procedures and policies include:

- Conduct of Operations (OQAP 3.0)
- Professionalism and Standards of Performance (NGP0110)
- Goals and Objectives Program (NGP0114)
- Qualification of Plant Staff Personnel (0PGP03-ZA-0065)
- Radiation Protection (0PGP03-ZR-0050)
- General Employee Training (0PGP03-ZT-0133)
- Indoctrination/Training Licensing Personnel (0PLP01-ZA-0003)

Organizational responsibilities are established to assure the accomplishment of corporate, site, and performance goals. To support Station goals and objectives, technical and administrative procedures are established and implemented. Procedural guidance is provided to evaluate whether the performance of a particular activity requires a procedure. Factors that are considered in determining whether a procedure is warranted include the complexity of the work and its impact on the SSC safety functions. Procedural controls include the provision to allow personnel to stop the process, and through the corrective action program, to change the controls or processes to ensure safety functions are satisfied. The process for the development, review, and approval of new procedures, procedure revisions, procedure changes and procedure deletions as well as the performance of periodic procedure reviews is also established and controlled.

Plant activities are conducted in a manner that is consistent with a conservative, technically-sound approach to plant operations which properly values nuclear safety and public protection. Goals and performance criteria are established commensurate with the importance to safety and plant reliability of the service, SSC, process or function. Examples of the subjects addressed by policies and procedures which govern goals and performance criteria include:

- Goals and Objectives Program (NGP0114)
- System Performance Monitoring (0PGP03-ZE-0038)
- Performance Indication Tracking Guide (0PGP05-ZV-0013)
- Quality Surveillance/Performance Monitoring (0PQP02-ZA-0003)
- Tests/Evaluations Requiring Additional Controls (0PGP03-ZA-0506)

Personnel are responsible for assuring that goals and performance criteria are met, and are responsible for recommending changes that support the accomplishment of company and site goals and performance criteria.

3.3.7.2 Performance Expectations for Personnel

Performance expectations and standards for personnel are specified in approved procedures, policies, and guidelines. These expectations and standards are satisfied through the use of performance criteria. Examples of the subjects covered by these procedures, policies, and guidelines include:

- Performing/Verifying Station Activities (0PGP03-ZA-0010)
- Minor Maintenance Guideline (MG-0002)
- Design Change Implementation (0PGP04-ZE-0312)
- Work Direction (0PGP03-ZA-0113)

Personnel are responsible for performing their activities consistent with these expectations, as well with established company standards. It is the responsibility of personnel to propose recommendations (when improvements are recognized) and to refine work processes to achieve expectations and resolve deficiencies.

3.3.7.3 Design Control Process

The Station's Design Control Program is used for both safety-related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications which are in the Configuration Management Program. The Design Control Program complies with 10 CFR Part 50, Appendix B and is used for maintaining design inputs and functional requirements, for repair, replacement, or modification of pressure-retaining capability of ASME systems, and for the accomplishment of design activities. For example, LSS/NRS ASME substitutions will be performed using design control procedures and comply with Quality Group D codes and standards per Regulatory Guide 1.26. The Design Control Program conforms to Regulatory Guide 1.64 (Quality Assurance Requirements for the Design of Nuclear Power Plants) and ANSI N45.2.11-1974 (Quality Assurance Requirements for the Design of Nuclear Power Plants). The STP Design Control Program is described in the Operations Quality Assurance Plan (OQAP), which has been reviewed and accepted by the NRC. Examples of subjects addressed in the procedures that implement the Design Control Program include:

- Design Change Implementation (0PGP04-ZE-0312)
- Plant Modifications (0PGP04-ZE-0310)
- Design Change Package (0PGP04-ZE-0309)
- Control of Configuration Changes (0PGP03-ZM-0021)
- Design & Modification Control (OQAP 6.0)
- 10CFR50.59 Evaluations (0PGP05-ZA-0002)

Upon grant of the exemption, STP may replace LSS and NRS ASME components with non-ASME components by ensuring that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements. The replacement component remains safety-related and 10CFR50 Appendix B design controls still apply.

For LSS and NRS ASME items that are replaced with non-ASME items, STP will manufacture or procure, install, and perform examinations of the replacement items in accordance with the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks). However, consistency with the original material specification and design will be maintained by using the original ASME Section II material or equivalent standard material (e.g., ASTM), applying the original ASME Section III Design Specification loadings and load combinations, and using the ASME Section III calculation methodology and allowables. (Note: The use of ASME Section III allowables is justified because it is consistent with the ASME Section III design methodology. Also, the use of identical or equivalent material and original design methodology will, in general, eliminate the need for reanalysis provided the wall thickness exceeds original design minimum wall requirement). It is not the intent of this request to require manufacturing and quality requirements of ASME Section III such as impact testing and NDE. These are considered permissible reductions in special treatment requirements consistent with the low risk classification. In addition, STP will conduct post-installation pressure tests in accordance with ASME Section XI. Other additional requirements of ASME Section III, including Subsection NCA, General Requirements, would not apply.

As an alternative to the above preferred approach, STP would have the option to design, manufacture and test a replacement component in accordance with the technical requirements of the applicable nationally recognized non-nuclear Code, Standard or specification, including design methodology and allowables, except that the design loadings of the original design specification will apply. In this case a recalculation may be required.

For example, the replacement of an LSS or NRS ASME valve would consist of the following steps:

1. Initiate a design change package to implement the replacement of a code valve with a non-code valve.
2. Identify the proposed ANSI B16.34 valve
3. Ensure and document that the material specifications of the replacement valve are consistent with the requirements of ASME Section II or equivalent standard material (e.g., ASTM).
4. Ensure that the wall thickness exceeds the original design minimum wall thickness. (Note: this step, along with step 3 above, will ensure that the stresses are within allowables).
5. Prepare a Work Package that identifies installation requirements required by ANSI B31.1 such as welding, NDE, etc.
6. Install the valve and conduct post-installation testing in accordance with ASME Section XI and the requirements of the work package.
7. Maintain associated records for the life of the plant.

These measures will be sufficient to provide reasonable assurance, commensurate with the low risk significance, that the technical requirements of the Code will be satisfied for LSS and NRS replacement items, and that the replacement items will remain functional under design basis conditions.

3.3.7.4 Procurement Process

Station procurement activities, including the identification, procurement, and receipt of replacement components or parts, are performed in accordance with approved procedures and processes. These processes include the following:

- Identification – The need for replacement components, parts, or material is identified and documented through either the Corrective Action Program (CAP) or through the Procurement and Inventory Control System (PICS). CAP is used to identify needed replacement parts when deficiencies are noted or when improvements are planned, and PICS will identify the need to procure replacement parts when inventory levels fall below specified criteria.

- Procurement Evaluation – A procurement request form is prepared and evaluated to ensure that technical requirements and quality requirements have been adequately described and specified. This evaluation ensures that the original design inputs and assumptions for the SSC have been properly factored in. Detailed procurement information, catalog identifications, and specifications, as applicable, are documented in a formal Purchase Order to the supplier. This documented information supports the procurement of the desired replacement component or part.

- Engineering Evaluation - Whenever conditions exist such that the proposed replacement components or parts deviate from the design specifications (technical requirements) of the original item, an engineering evaluation is performed to assess the design adequacy of the proposed replacement. Such an evaluation compares the form, fit, and function of the replacement component to the original component. The comparison of form, fit, and function includes the capability of the replacement component to perform its function under design basis conditions, including applicable seismic and environmental conditions. This evaluation is documented and performed in accordance with either the Item Equivalency Evaluation procedure, the Commercial Dedication Process, or the Condition Report Engineering Evaluation (CREE). These processes are consistent with the guidance provided by EPRI report NP-6406, 'Technical Evaluation of Replacement Items Guideline'. Alternatively, a design change may be developed (in accordance with the Design Control process described above) to validate the acceptability of the replacement item. Procurement specifications are developed to delineate specific technical or quality requirements when a detailed, formal Purchase Order is developed.

- Purchase Order – Procurement requirements are specified on a Purchase Order which includes design specifications, quality requirements, conformance with nationally recognized consensus standards (if required by the design specifications), and shipping requirements. Substitutions are not allowed unless written authorization is granted by Engineering. In the alternative, catalog identifications may be placed on a Purchase Order in support of a replacement part procurement.

- Receipt Inspection – At the time of receipt, the received item is inspected by warehouse personnel to ensure that the item was not damaged in the process of shipping, and that the item received is the item ordered. In addition, when the item is delivered to the field for use, a similar inspection is performed by Maintenance personnel.

- Special Handling & Storage – The procurement program provides for the identification and implementation of special handling and storage requirements (if required) to ensure that the item is not damaged or degraded during shipment to the site or during storage on site. These handling and storage requirements consider available recommendations from the vendor. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

Examples of the subjects addressed in approved procedures which govern the procurement process includes the following:

- Procurement of Materials (0PGP03-ZP-0011)
- Replacement Item Equivalency Evaluation (0PGP03-ZE-0072)
- Purchase Order/Control Management (0PGP03-ZP-0013)
- Receiving Material (0PNP01-ZP-0026)
- Marking, Handling, Storage, Maintenance of Materials (0PNP01-ZP-0033)

3.3.7.5 Installation Process

The Station's component and part installation activities are accomplished through the Design Change Programs and/or through the Maintenance processes. Components procured under the commercial treatment program may be installed in plant systems that are governed by special treatment requirements, including but not limited to ASME Code, Class 1E, and seismic requirements. In such instances, proceduralized measures are taken to ensure that the most limiting requirements are applied at the interface between the commercial component and the rest of the system. Examples of such requirements include design, installation, welding, non-destructive examination, inspection, and testing requirements.

Upon completion of installation activities, a post-maintenance test (PMT) (or other tests as specified by the Work Planner, Work Supervisor, System Engineer, or Plant Operations) is performed, if the installation could affect an SSC's design function. Post-maintenance testing is performed to provide an appropriate level of assurance that the SSC is performing within expected parameters and component functionality is verified prior to being returned to service. The post-maintenance testing may necessitate that the SSC be placed in service to validate the acceptance of its performance. Post-maintenance testing is not necessarily performed under design basis conditions. Examples of post-maintenance attributes which may be checked include:

- no visible leaks of process fluid
- smooth and expected operation of stroked valves
- expected fluid pressure and flow from pumps
- rotating equipment operating with no abnormal noises
- expected circuit continuity and indication
- expected vibration responses on rotating equipment

The component is only returned to service after Operations accepts the results of the maintenance activities and/or the post-maintenance or operability tests. If expected performance is not achieved, the work order document remains open to continue replacement/repair/rework until the activity is satisfactorily completed.

Examples of subjects addressed by approved procedures and policies used to support the installation process include:

- Work Process (0PGP03-ZA-0090)
- Post Maintenance Testing Program (0PGP03-ZM-0025)
- Minor Maintenance Guidelines (MG002)
- Tool Pouch Maintenance (MG001)
- Condition Reporting Process (0PGP03-ZX-0002)
- Design Change Implementation (0PGP04-ZE-0312)
- Control of Configuration Changes (0PGP03-ZA-0021)

3.3.7.6 Maintenance Process

The Station's maintenance process activities include corrective, preventive, and predictive maintenance. These maintenance activities are performed in accordance with approved procedures, processes, and guidelines. Each of these types of maintenance activities is discussed below:

Corrective Maintenance

- Identification – An SSC deficiency is identified, documented, and tracked through the Corrective Action Program (i.e., a condition report is written). All personnel on site are encouraged to generate condition reports as soon as a deficiency is identified.

- Evaluation - The deficiency is evaluated by a senior reactor operator (SRO)-qualified individual to determine the appropriate corrective maintenance to be performed. Troubleshooting actions may be required if the cause of the deficiency cannot be readily ascertained. Engineering support may also be required to identify the cause of the deficiency. The evaluation includes a determination of the level of planning, work instructions, and documentation needed to perform the work activity. Examples of factors that are considered include the complexity of the work, impact on SSC critical attributes, potential impact on operating equipment, and testing requirements.

- Scheduling – Based on the results of the above evaluation, the deficiency is determined to be acceptable as-is, or corrective maintenance activities are scheduled. This schedule is based upon a consideration of restraints such as material, engineering, plant conditions, craft resources, and the relative impact of the deficiency upon plant safety.

- Planning – Based on the results of the above evaluations, the approach to work package development is determined.

  - If the work activity is straightforward and within the scope of the skill of the craft, no planned package is necessary and the work activity does not require scheduling. Maintenance personnel will coordinate work performance with Operations personnel. Only electronic documentation of the work completed is required. This approach is known as Tool Pouch Maintenance.

  - If the work activity is straightforward, however, procedure support is needed to aid in work performance and documentation, then a streamlined planned package is developed by either the work planner or work supervisor. The work activity is normally scheduled, however, scheduling is not required. If scheduling is not required, then Maintenance personnel will coordinate work performance with Operations personnel. This approach is known as Minor Maintenance.

  - If the work activity is complex, a detailed planned work package is developed. This package describes the necessary work instructions, measuring and test equipment requirements (M&TE), initiation of necessary permits, and identification of any hold points, post-maintenance testing requirements, and documentation requirements. The work activity is scheduled to coordinate work performance with Operations personnel. This approach is known as a detailed, planned package.

- Implementation – Maintenance work is performed as planned to correct the deficient condition.

- Closure – Maintenance actions taken to correct the deficient condition, as well as the results of any required post-maintenance tests, are documented. Hard-copy documentation is archived as required per procedure and is available for retrieval. Input into the Corrective Action Program, Maintenance Rule Program, and the component history is provided as applicable.

Preventive Maintenance (PM)

- Identification – Preventive maintenance is an action taken to maintain a piece of equipment within design operating condition and to extend its life, and is performed for the purpose of preventing equipment failure. Preventive maintenance tasks and work packages are developed for active structures, systems, or components factoring in vendor recommendations and sound technical insights. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

- Evaluation – Engineering and Maintenance personnel perform an evaluation to determine an appropriate periodic and/or condition-based PM for the specific component. This evaluation includes a consideration of vendor recommendations, industry experience, site experience, service conditions, safety significance, INPO and EPRI guidelines, commitments, NRC and industry notices, Qualification Maintenance Database, codes, standards, and technical insight.

- Scheduling – Preventive Maintenance tasks are scheduled based upon a consideration of factors such as the ability to perform the PM activity on-line, as low as reasonably achievable (ALARA) dose considerations, and minimization of individual equipment outages. In addition, PMs may be utilized to accomplish certain corrective maintenance activities based upon plant requirements and efficiency of performance.

- Planning – Preventive Maintenance work instructions are developed in accordance with the PM Planners Guide. Work instructions are written to provide an appropriate level of assurance that the PM can be performed by qualified craftsmen. Work instructions include necessary permits, PMT requirements, M&TE requirements, radiation work permit (RWP) requirements, and documentation requirements.

- Implementation – The Preventive Maintenance task is performed to maintain equipment reliability. If a deficiency is noted during PM task performance which is not within the scope or intent of the PM, the deficiency may be corrected by either making a "one-time change" to the PM instructions, or if conditions do not permit immediate correction, issuing a condition report. Both methods are administratively controlled by approved procedures. Post maintenance testing, if required by the work instructions, is completed and the equipment is released to Operations for return to service.

- Closure – Preventive Maintenance actions as well as post maintenance test results are documented. The completed package is turned-over to Operations for review and acceptance. The equipment is returned to service after Operations accepts the results of the task activity and the post maintenance testing. If required by the work package, an operability test and/or other desired testing is performed. The documentation is archived as required by procedure and is available for retrieval. Inputs into Corrective Action Program, the Maintenance Rule Program, and the component history are provided, as applicable.

Predictive Maintenance

Predictive maintenance activities are intended to anticipate, monitor, and/or include degradation mechanisms which, if left unattended, would result in component failure. The frequency and scope of these maintenance actions are established and documented based on various considerations such as vendor recommendations, environmental operating conditions, safety significance, and operating performance history. STP may deviate from vendor recommendations based on specific circumstances and sound business practices.. Such deviations are not required to be documented.

Examples of predictive maintenance activities include:

- periodic lube oil analyses on large motors and pumps
- vibration analyses of rotating equipment
- thermographic analysis of both mechanical and electrical SSCs to identify improper temperature conditions or electrical hot-spots
- motor potential diagnostic testing

Deficiencies that are discovered during the performance of predictive maintenance activities are either immediately corrected or identified for correction under the Corrective Action Program.

Post maintenance testing, if required by the work instructions, is performed to provide an appropriate level of assurance that the SSC is performing within expected parameters prior to being returned to service.

Examples of the subjects addressed by approved procedures and guidelines used to support the corrective, preventive, and predictive maintenance process includes:

- Work Process (0PGP03-ZA-0090)
- PM Program (0PGP03-ZM-0002)
- Work Direction (0PGP03-ZA-0113)
- Minor Maintenance Guidelines (MG001)
- Tool Pouch Maintenance Guidelines (MG002)
- Post Maintenance Testing (0PGP03-ZM-0025)
- Troubleshoot/Rework Process (0PMP01-ZA-0041)

3.3.7.7 Inspection, Test, and Surveillance Process

The Station's inspection and test process is primarily addressed and implemented through the Maintenance process. As stated above, the Maintenance process addresses inspections and tests through corrective, preventive, and predictive maintenance activities. These activities factor in vendor recommendations into the selected approach. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

The Station's surveillance process is governed by approved test procedures to ensure that Technical Specification requirements are met and satisfied. These Technical Specification requirements apply to some safety-related and non-safety-related SSCs, including some components that are categorized as LSS. These surveillances will continue to be performed in the same manner as they are currently. However, if future Technical Specification relief is granted, appropriate surveillances will be modified as needed.

In addition to the surveillances required by Technical Specifications, additional monitoring is performed to validate the acceptable performance of SSCs. Examples of programs used to monitor SSC performance include the following.

- Maintenance Rule - Monitoring will be performed at the system, train, or plant level to ensure SSC reliability levels are maintained. LSS and NRS component performance is not explicitly monitored, but degrading performance in these components, if significant, would be observed at the system or train level. Failures of LSS or NRS components that result in the system's performance criteria being exceeded are identified and resolved under a corrective plan of action, which includes actions to prevent recurrence.

- Operator Rounds – Routine inspections and observations of plant systems and equipment are performed on a per shift basis by both licensed and non-licensed plant operators.

- System Engineer Reviews – The operation of plant systems is monitored by the cognizant system engineer through physical walkdowns and reviews of condition reports and other factors that may impact the health of the system. System Engineer reviews (per the System Engineering Responsibilities and Expectations Guideline) include:

  - visual inspections
  - performance reviews
  - material deficiency reviews
  - feedback from Operations and Maintenance personnel to receive additional insights into the performance of SSCs

- Management Observations – Station management spends time in the field to monitor equipment and personnel performance. These field observations by experienced nuclear managers provide additional insight and assurance that SSCs are performing within expectations. In addition, a report of all condition reports (CRs) generated within the previous 24-hours is available to all managers for review.

- Graded Quality Assurance Working Group evaluations – Periodic reviews of SSC performance are conducted by the Graded Quality Assurance (GQA) Working Group to ensure that the expected level of SSC performance is achieved. These in-depth reviews are conducted at least once per cycle. The periodic system reviews performed by the GQA Working Group include:

  - comprehensive corrective action reviews

- reviews of System Engineer system health status and insight on changes
- PRA model updates
- system modification reviews
- licensing, quality, and operational document reviews
- performance trend reviews

Examples of subjects addressed by approved procedures and guidelines used to support the inspection, test, and surveillance process include:

- Work Process (0PGP03-ZA-0090)
- PM Program (0PGP03-ZM-0002)
- Plant Surveillances (0PGP03-ZE-0004)
- Post Maintenance Testing (0PGP03-ZM-0025)
- Maintenance Rule Program (0PGP04-ZE-0313)
- Graded Quality Assurance Working Group Process (0PEP02-ZA-0001)

3.3.7.8 Corrective Action Program

The Station's Corrective Action Program is used for both safety-related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications. The Corrective Action Program complies with 10 CFR Part 50 Appendix B, and is used to identify and correct equipment deficiencies and nonconformances, to determine the cause of significant conditions adverse to quality, and to initiate actions to prevent recurrence. The STP Corrective Action Program is described in the Operations Quality Assurance Program (OQAP), which has been reviewed and accepted by the NRC.

Examples of subjects addressed in approved procedures and policies that implement the Corrective Action Program include:

- Condition Reporting Program (0PGP03-ZX-0002)
- Corrective Action Program (NGP0131)
- Control of Conditions Adverse to Quality (OQAP 13)
- CR Engineering Evaluation Program (0PGP04-ZA-0002)

3.3.7.9 Management and Oversight Process

The Station's management and oversight process is accomplished through approved procedures and guidelines. This process includes the following:

- Assessments - Assessments are performed to provide a systematic examination of activities, processes, and systems, structures, or components to evaluate the effectiveness of work practices and/or management controls. The type, frequency and degree of specificity of assessments are determined by the importance to the SSC's safety functions, by the performance history of structures, systems, and components, or by the work activity being evaluated. There are two categories of assessments:

- Independent Oversight – These assessments consist of monitoring, surveillance, audit, and inspection activities. These assessments are routinely performed, and the level of detail of the assessment is based on the safety significance of the SSC and its performance history.

- Line Self-Assessments – These assessments may be continuous or focused. Examples of continuous self-assessments include peer-checking activities, independent/dual verification activities, and field observation activities. Continuous self-assessments are performed daily based upon the task scope. Focused assessments evaluate specific programs, processes, or activities. Focused self-assessments evaluate compliance with Station commitments and requirements, and are performed at least once every two years. The frequency of performing focused self-assessments may be adjusted based on actual performance feedback.

Assessments are performed by line organizations or personnel, by management, or by independent internal or external organizations or groups. The importance to the safety function and performance history determines the degree of management and technical oversight. Personnel performing assessments are qualified through training, work experience, or certification.

Actions and findings identified during assessment activities are documented on a condition report in accordance with the Corrective Action Program. Action ownership and management oversight are provided to these open items to ensure proper follow-up and closure of identified issues.

- Maintenance Rule - Monitoring is performed at the system, train, or plant level to ensure SSC reliability levels are maintained. LSS and NRS component performance is not explicitly monitored, but degrading performance in these components, if significant, would be observed at the system or train level. Failures of LSS or NRS components that result in the system's performance criteria being exceeded are identified and resolved under a corrective plan of action, which includes actions to prevent recurrence.

- Graded Quality Assurance (GQA) Working Group - SSC performance is periodically assessed (at least once per cycle) by the GQA Working Group to ensure that the expected level of SSC performance is achieved. If an SSC fails to satisfy performance expectations, the commercial controls are evaluated to determine if a weakness in the controls is a cause of or contributing factor to the degraded performance. If the controls are the cause of or a contributor to the degraded performance, the controls are enhanced as necessary to address the weakness. Following the enhancement of the controls, SSC performance will continue to be assessed to ensure that the adjustment to controls has accomplished the desired level of reliability and availability.

- PRA Model Updates - The PRA configuration control program incorporates a feedback process to update the PRA Model. The updates are segregated into two categories:

- The first category, called a plant operating update, will incorporate plant design changes and procedure changes that affect PRA-modeled components, initiating event frequency updates, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months.

- The second category, called a comprehensive data update, will incorporate changes to plant-specific failure rate distributions, human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months. The "comprehensive data update" will allow for the accumulation of enough data to properly update distributions without creating an undue burden to perform these time-consuming data updates.

The 36-month frequency specified above is based upon the proposed Nuclear Energy Institute's (NEI) frequency for PRA model updates. In addition, the 60-month frequency is based upon data self-assessments performed by STP personnel earlier this year. Any update will entail an engineering evaluation to determine if applicable 'state-of-the-art' changes relative to PRA technology should be incorporated.

The PRA model may be updated on a more frequent basis if an update would result in a significant increase in the CDF. All potential model changes are evaluated on a monthly basis with respect to the impact on CDF. If a potential change would result in an increase in CDF of greater than 10%, the model can be updated at the discretion of the PRA Administrator. Also, during the "plant operating update", each of the data variables updated normally during the "comprehensive data update" will be evaluated for changes in the data. If plant specific history demonstrates an approximate order of magnitude change, then that PRA variable can be updated during the "plant operating update" at the discretion of the PRA Administrator.

In addition, technical and administrative procedures are established and implemented as described throughout this section. Administrative procedures identify the factors to be considered in determining whether a procedure is warranted; such factors include the complexity of the work and its impact on the SSC safety functions. Procedural controls include the provision to allow personnel to stop the process, and through the corrective action program, to change the controls or processes to ensure safety functions are satisfied. The process for the development, review, and approval of new procedures, procedure revisions, procedure changes, and procedure deletions, as well as the performance of periodic procedure reviews, is also established and controlled.

Procedures provide for the qualification, training, and certification of personnel. The procedures provide for General Employee Training, indoctrination/nuclear training, certification when required by applicable commercial code or standard incorporated in a design specification, and continuing training and retraining. Personnel are qualified, trained, and certified commensurate with the functions they perform. Experienced personnel may be exempted from prerequisite training.

Documentation, review, and retention requirements of the completed work activities are governed by administrative procedures and work instructions. Examples of documentation include:

- Design drawings and specifications
- Procurement documents, such as procurement request forms, engineering evaluations of item equivalency, and purchase orders or catalog orders
- Documentation of completed installations and the results of applicable post-modification tests
- Documentation of completed maintenance activities and the results of applicable post-maintenance tests
- Documentation of Technical Specification surveillances, documentation required by the Maintenance Rule program, reports of periodic evaluations by the GQA Working Group, and documentation of deficiencies identified through operator rounds, system engineer reviews, and management observations.
- Condition reports of conditions adverse to quality
- Documentation of actions and findings resulting from assessments.

Procedures identify the types of inspection, test, and surveillance equipment requiring control and calibration, and the interval of calibration. When calibration is required, each organization is responsible for assuring that the M&TE that it uses has been calibrated to the accuracy required for its intended use. Equipment that is in error or defective is removed from service or properly tagged to indicate the error or defect.

Examples of subjects addressed in approved procedures and guidelines used to support the management and oversight process includes:

- Plant Audits (0PQP01-ZA-0001)
- Oversight Planning and Scheduling Process (0PQP01-ZA-0015)
- Quality Surveillance/Monitoring (0PQP02-ZA-0003)
- Maintenance Rule Program (0PGP04-ZE-0313)
- Graded Quality Assurance Working Group Process (0PEP02-ZA-0001)
- Probabilistic Risk Assessment Program (0PGP04-ZA-0604)
- Independent Plant Assessments (0PQP01-ZA-0006)
- Self-Assessment Process (STP-0423)

3.3.7.10 Configuration Control Process

The Station's configuration control process is controlled through approved procedures and policies. The design control process ensures that the configuration of the Station is properly reflected in design documents and drawings. Changes to the Station are controlled through design change packages (modifications) which require that control drawings and documents be updated prior to closeout of the modification package.

In addition, configuration control addresses the status of components day-to-day in the field. SSCs are tagged and are manipulated by qualified Operations personnel per procedure. The configuration control process manages and controls the physical changes (procedural and equipment) to the facility to assure that the plant configuration and practices correctly reflect the licensing bases. Non-ASME components installed in ASME Code systems are identified and tracked.

Examples of subjects addressed in approved procedures and policies that implement the configuration control process include:

- Design Change Implementation (0PGP03-ZE-0031)
- Operations Configuration Management (0PGP03-ZO-0039)
- Performing/Verifying Station Activities (0PGP03-ZA-0010)
- Configuration Management Program (0PGP03-ZA-0109)
- Control of Configuration Changes (0PGP03-ZA-0021)
- Plant Modifications (0PGP04-ZE-0310)

3.4 Licensing Basis Documentation and Change Control

STP is proposing to control its commitments regarding the categorization process and the treatment to be provided to components by adding a new Section 13.7 to the STP UFSAR. STP's proposed Section 13.7 for the UFSAR is contained in Attachment 7. This new section in the UFSAR will be controlled in accordance with the requirements in 10 CFR 50.59. STP is also proposing to use the following additional criteria for controlling changes to Section 13.7 of the UFSAR:

- STP may make changes in the categorization process described in Section 13.7 of the UFSAR, without prior NRC approval, provided that the revised process:

  - has both PRA and deterministic categorization processes;
  - designates the category of a component based upon the higher of the PRA risk categorization and deterministic categorization;
  - provides a risk categorization of a component as LSS or NRS only if the component has little or no impact on CDF or LERF (using commonly accepted assessment techniques such as Fussell-Vessely (FV) and RAW); and
  - provides a deterministic categorization of a component as LSS or NRS only if the component has no more than a minor impact on the ability of the plant to prevent or mitigate accidents.

- STP may make changes in the treatment described in Section 13.7 of the UFSAR, without prior NRC approval, provided that the revised treatment provides reasonable assurance that components:

  - can perform their HSS and MSS functions, and
  - can perform their safety-related functions under design basis conditions.

- STP may make changes in the evaluations and assessments described in Section 13.7 of the UFSAR, without prior NRC approval, provided that the revised evaluations and assessments include the following elements:

  - Processes to monitor the effect of changes in special treatment on the reliability/availability of safety-significant and safety-related functions;
  - Processes to ensure that significant changes in reliability/availability of safety-significant and safety-related functions are evaluated for impact on component risk categorization, application of special treatment, and corrective action; and
  - Periodic assessments to update the PRA and to reassess the risk categorization of components.

STP's proposed process represents the first-of-a-kind program, and it may be expected that changes will be needed or desirable as experience with implementation of the process is accumulated. 10 CFR 50.59 and STP's additional criteria will provide flexibility for STP to make changes in the process based upon this experience, while ensuring that significant changes will be subject to prior NRC approval. As described in more detail in Attachment 6, STP believes that the USFAR is the appropriate location for this information (rather than the Operations Quality Assurance Program), because the risk-informed process applies to more subjects than just quality assurance and because the change control process for the OQAP would be unduly restrictive if applied to Section 13.7. Additionally, it would be inappropriate (and inconsistent with NRC's policy toward the technical specifications and other license conditions) to incorporate Section 13.7 as a license condition. Such a condition would create an undue burden on both STP and the NRC, because it would require STP to apply for and the NRC to review and approve relatively minor changes in the process.

4.0 Regulations to Which the Exemption Would Apply

NOTE: As used in this section, reference to LSS and/or NRS components is understood to apply only to LSS or NRS components that are safety-related, important to safety, or otherwise currently under the scope of NRC's regulations. Other non-safety-related LSS and NRS components are not subject to regulatory control.

STP is seeking to exclude LSS and NRS components from the scope of NRC's special treatment requirements, without the need to seek prior NRC approval for each specific change. To accomplish this objective, STP is requesting an exemption 1) to exclude LSS and NRS components from the scope of the special treatment requirements in Parts 21, 50, and 100; and 2) from several process-related regulations that govern changes in the special treatment provisions in the licensing basis (in the alternative, STP requests that the NRC take certain actions, as described in more detail in Section 4.2, that would obviate the need for prior NRC approval of such changes). A list of the specific regulations in question is provided in Attachment 2. Each of the exemptions is discussed in Sections 4.1 and 4.2 below.

In preparing this request, STP has employed a conservative process to identify the regulations from which an exemption may be needed to enable STP to implement its risk-informed approach toward special treatment. For example, STP has identified a regulation as a candidate for an exemption if STP could not implement its risk-informed approach consistently with NRC's previous guidance or interpretations for that regulation. If the NRC determines that an exemption from a particular regulation identified in this exemption request is not needed to enable STP to implement its risk-informed approach as described in this request, STP requests that the NRC document the basis for that conclusion in the safety evaluation for the exemption.

4.1 Exemption to Exclude LSS and NRS Components from the Scope of Special Treatment Requirements

There are a number of NRC regulations that impose special treatment requirements on SSCs that are safety-related, important to safety, or otherwise within the scope of the regulations. These regulations apply to the designated components, even if those components are LSS or NRS. As discussed in Section 5.0 below, it is not necessary for the protection of the public health and safety to apply comprehensive special treatment requirements to LSS and NRS components, and an exemption to exclude these components from the scope of these requirements would reduce unnecessary burdens on both STP and the NRC, and allow more attention to be focused on safety/risk significant SSCs. Therefore, STP seeks an exemption to exclude LSS and NRS components from the scope of the special treatment requirements in Parts 21, 50, and 100.

LSS components have "minimal" risk significance. As their name implies, NRS components have no risk significance. However, as discussed previously in Section 2.0, STP is not requesting that LSS and NRS components be exempted from every regulation in Parts 21, 50, and 100. For example, there are numerous regulations that govern the designs and functions of SSCs (e.g., 10 CFR 50.46, which applies to the Emergency Core Cooling System, and most of the General Design Criteria (GDC) in Appendix A to Part 50). STP is not requesting an exemption from the regulations governing the design and functional requirements for SSCs. This request seeks to exclude LSS and NRS components only from the scope of the special treatment requirements in Parts 21, 50, and 100. Furthermore, although LSS and NRS components would be excluded from the scope of certain special treatment requirements, STP will ensure that LSS critical attributes are documented and incorporated into corrective action and design control processes.

4.1.1 Special Treatment Regulations Subject to this Request

Attachment 2 identifies the specific special treatment regulations in Parts 21, 50, and 100 to which this exemption request applies. In summary, these regulations are as follows:

- Qualification Requirements in 10 CFR 50.49, GDC 2, GDC 4, and Part 100, Appendix A.VI (which is incorporated in Part 50 through 10 CFR 50.34(b)(11)) - These regulations impose environmental and seismic qualification requirements on certain classes of SSCs. In particular, these regulations require testing and inspection to demonstrate that components can perform their safety functions under the conditions in question, and require that the

components be maintained in their qualified condition. These requirements apply to components that are safety-related or important to safety, without regard to their actual safety/risk significance. Given the lack of safety/risk significance of LSS and NRS components, the requirements for their testing, inspection, and documentation are unduly burdensome and divert resources from items with safety/risk significance. Therefore, in order to be able to procure and maintain components in a manner commensurate with their safety/risk significance, STP is requesting an exemption to exclude LSS and NRS components from the scope of these qualification requirements. However, STP's exemption request with respect to GDC 4 pertains only to environmental qualification of LSS and NRS components. All applicable SSCs (whether HSS, MSS, LSS, or NRS), including piping, cable trays, and conduit, will continue to be protected or otherwise designed to withstand the dynamic effects as described in GDC 4.

- Maintenance Rule Requirements in 10 CFR 50.65 - This regulation imposes requirements for monitoring the effectiveness of maintenance for a broad category of SSCs, including safety-related SSCs and certain non-safety-related SSCs. These requirements apply to components that come within the scope of the Maintenance Rule, without regard to their actual safety/risk significance. Given the lack of safety/risk significance of LSS and NRS components, requirements for monitoring of maintenance effectiveness and for performing unnecessary preventive maintenance are unduly burdensome. Instead, normal commercial and industrial maintenance practices and application of the STP corrective action program are an appropriate and acceptable means for assuring the function of LSS and NRS components. Therefore, in order to be able to implement the maintenance program for these components in a manner commensurate with their safety/risk significance, STP is requesting an exemption to exclude LSS and NRS components from the scope of the Maintenance Rule. STP would still conduct monitoring at the plant/system/train level; however, failure of an LSS or NRS component would not count as a Maintenance Rule Functional Failure (MRFF) at the plant/system/train level unless that failure caused a failure of an HSS or MSS function.

- Quality Assurance Requirements in GDC 1, 10 CFR Part 21, and 10 CFR Part 50, Appendix B (which is incorporated in 10 CFR 50.34(b)(6)(ii)) - These regulations impose QA, procurement/dedication, and reporting requirements on SSCs that are important to safety or safety-related. These requirements apply to components within the scope of the regulations, without regard to their actual safety/risk significance. As discussed in Section 3.5.1 of the NRC's Safety Evaluation for the GQA Program, the GQA Program was designed to comply with the requirements in Appendix B. Therefore, under the GQA Program, STP is still required to apply a certain level of quality assurance to LSS and NRS components that are safety-related or important to safety. However, LSS and NRS components have no significant effects on public health and safety and do not warrant such comprehensive levels of control. Given the lack of safety/risk significance of LSS and NRS components, these formal QA requirements are unduly burdensome. Instead, normal commercial and industrial practices and application of the STP corrective action program are an appropriate and acceptable means for assuring the adequacy of LSS and NRS components. For example, STP would be able to purchase replacement components that meet the design and functional requirements (including seismic and environmental considerations) without necessarily adhering to the strict QA requirements currently in place. By purchasing replacement

components that meet the design and functional requirements, and by performing a receipt inspection to validate that what was received is what was ordered, reasonable assurance is provided that the component can, and will, perform its risk significant design functional requirements. Therefore, in order to be able to implement a QA program in a risk-informed manner, STP is requesting an exemption to exclude LSS and NRS components from the scope of these regulations. However, as discussed more fully later, STP will continue to apply its design control and corrective action programs to LSS and NRS components.

- Inspection and Testing Requirements in GDC 18, and 10 CFR Part 50, Appendix J - These regulations impose inspection and testing requirements for certain types of components. These requirements apply to components within the scope of the regulations, without regard to their actual safety/risk significance. LSS and NRS components have no significant effects on public health and safety and do not warrant such comprehensive levels of control. Furthermore, as discussed in response to RAIs 16 and 17 in Attachment 4, the exemption from Appendix J should have no effect on containment leakage rates, since the components exempted cannot credibly be a source of leakage. Given the lack of safety/risk significance of LSS and NRS components, requirements for formal inspection and testing programs for these components are unduly burdensome. Instead, normal commercial and industrial practices and application of the STP corrective action program are an appropriate and acceptable means for assuring the function of LSS and NRS components. Therefore, in order to be able to implement inspection and testing programs in a risk-informed manner, STP is requesting an exemption to exclude LSS and NRS components from the scope of these regulations.

- Code Requirements in 10 CFR 50.55a. - STP seeks an exemption for LSS and NRS components from requirements for repair/replacement, inservice inspection, and inservice testing in ASME Code Section XI and/or the Operation and Maintenance (O&M) Code which is incorporated by reference in 10 CFR 50.55a. For repair and replacement activities, LSS and NRS components would be exempt from ASME Code, Section XI, except that material specifications and the design (design loadings, design methodology, and stress allowables) would be consistent with the original requirements. STP would otherwise be able to purchase replacement components that meet the technical requirements of a nationally recognized Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks). Installation and examination of LSS and NRS replacement components would be performed in accordance with ANSI B31.1. Post-installation testing would be performed in accordance with ASME Section XI and the requirements of the work package. For inservice inspection and inservice testing activities, LSS and NRS components would be exempt from the requirements of ASME Section XI. STP also requests exemption from 10 CFR 50.55a(h) with respect to Sections 4.3 and 4.4 of IEEE 279 in order to exclude LSS and NRS components from the scope of the quality and environmental qualification requirements described in those sections of IEEE 279.

In summary, STP is requesting an exemption to exclude LSS and NRS components from the scope of special treatment requirements that are intended to provide a high level of assurance of the quality of safety significant components. In lieu of these requirements, STP would apply normal commercial and industrial practices for these components, similar to the type of practices applied to the power production side of the plant. These proven practices will result in an appropriate and acceptable level of assurance that these components will be capable of performing their functions, commensurate with their safety/risk significance. These practices have been demonstrated to be effective in maintaining the reliability and availability of power production components, and are similar to or more rigorous than practices used effectively throughout the world in commercial and industrial applications.

4.1.2 Preservation of Safety Functions

This exemption will not alter the design and functional requirements applicable to LSS and NRS components. In particular, these components will still be available to function under applicable design conditions even though they are not important from a risk perspective to mitigate or prevent accidents.

The exemption will, however, eliminate the need to qualify LSS and NRS components for such conditions. Thus, for example, STP will utilize purchase requirements or other evaluations to provide an appropriate level of assurance that replacement components will be available to function under design conditions, but will not necessarily perform or require qualification tests.

Although LSS and NRS components would not need to be qualified, STP will take actions to provide an appropriate level of assurance that the components will be available to function under design basis conditions. These actions are described in detail in Section 3.3.7.

In addition, as explained below, STP will be implementing additional provisions to ensure the function of electrical components, ASME components, and seismically designed components.

4.1.2.1 Electrical Components

STP is requesting an exemption from the requirements in 10 CFR 50.49 regarding the qualification of electrical components. STP is also requesting an exemption from 10 CFR 50.55a(h)(2), to the extent that it incorporates the quality assurance and environmental qualification requirements in IEEE 279 for Class 1E electrical equipment. If this exemption is granted, LSS and NRS Class 1E electrical equipment could be replaced with equipment that meets the functional and design requirements in IEEE 279, but that was not procured, installed, and maintained in accordance with the quality assurance and environmental qualification requirements in Sections 4.3 and 4.4 of IEEE 279 and 10 CFR 50.49. Additionally, under the requested exemption, existing LSS and NRS Class 1E components would not need to be maintained in accordance with those requirements.

Thus, when the need arises, STP would be able to replace an existing LSS or NRS Class 1E component with a commercial grade component that meets the existing functional and design requirements, including environmental parameters. An engineering evaluation would be performed to ensure that the replacement component will not degrade other Class 1E components and will satisfy the required form, fit, and function (including the ability to function under specified environmental conditions). Thus, the replacement component will be capable of meeting the design and functional requirements (including environmental considerations); however, the component would not be specifically qualified or subject to all of the quality assurance provisions associated with Class 1E components. By purchasing replacement components that meet the design and functional requirements, and by procuring from reputable vendors, reasonable assurance will be provided that a quality product (which will meet the various challenges of service operation) will be received. Upon receipt, an inspection will be performed to validate that the received component is the component that was ordered and was not damaged during shipment. If discrepancies are noted during this receipt inspection, the discrepancies will be resolved or the component will be returned to the vendor. As a result, even if the subject component were to be exposed to conditions for which the original Class 1E component was specifically qualified for, there will be reasonable assurance, commensurate with the risk significance, that the component will be able to perform satisfactorily and will not electrically degrade other components. Therefore, STP intends to continue to identify the replacement component as Class 1E, similar to maintaining the safety-related classification for applicable LSS and NRS mechanical components.

If STP determines that a proposed replacement commercial grade component does not satisfy all of the design functional requirements applicable to a Class 1E component, STP will either purchase and install a Class 1E component that is fully qualified and subject to the applicable special treatment requirements, or the replacement component will be classified as non-1E. In the latter case, the replacement device will be properly isolated from the Class 1E circuitry, and the circuit downstream of the required isolation device(s) would also be classified as non-1E and maintain proper separation.

STP will take other actions as necessary to ensure that HSS and MSS Class 1E components are appropriately protected from Class 1E LSS and NRS components. In particular, for cabling, STP will maintain current separation requirements. Existing Class 1E cables will not be replaced with commercial grade cables if this would result in violating separation requirements. Thus, for example, qualified Class 1E cables from the isolation device to the safety-related LSS or NRS SSCs will not be replaced with non-qualified commercial grade cables if such replacement would result in the non-qualified cables being located in raceways or electrical containment penetrations that contain fully qualified Class 1E cables.

The design provisions listed above will provide adequate protection of the public health and safety. In order for a failure of a LSS or NRS Class 1E component to have a risk significant impact on the plant, the following would have to occur at the same time: 1) the component must fail in such a way as to affect the upstream electrical circuitry; 2) the redundant train must fail; and 3) the above failures must occur during a design basis event. STP does not consider this scenario to be credible.

Furthermore, components that are categorized as LSS or NRS, by definition, serve little, if any, function in mitigating the consequences of an accident or protecting the core. Therefore, the loss of the LSS or NRS component itself (e.g., failure of a motor operator to provide motive force to its associated valve when demanded) has been determined through the categorization process to have little or no impact on protecting the public health and safety during a design basis event or any other credible event.

4.1.2.2 ASME Components

STP is requesting an exemption from the ASME Code requirements incorporated in 10 CFR 50.55a, with respect to the repair and replacement of ASME components. For example, the exemption would enable STP to replace LSS and NRS ASME components with non-ASME components, subject to ensuring that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements.

STP's compliance with the ASME Code was originally established on a system level basis. Therefore, most LSS and NRS components in ASME systems are currently subjected to ASME Code requirements. For LSS and NRS components, STP seeks to re-establish ASME Code class boundaries on a subsystem level basis rather than on a system level basis without prior NRC approval. If this exemption is granted, LSS and NRS ASME components may be repaired in accordance with or replaced with non-Code components that meet the technical requirements of a nationally recognized Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks) without prior NRC approval, as the need arises. Thus, individual systems could contain both Code and non-Code components.

Since the plant has already been designed and constructed in accordance with the ASME Code, STP's exemption request addresses only the portion of the Code that involves the repair and replacement of items, when the need arises. This activity is covered under Section XI of the Code. Section XI of the Code generally requires that replacement items meet the requirements of the original Construction Code, i.e., ASME Section III. However, Section XI does provide relief for piping, valves, and fittings 1-inch nominal pipe size and less. These items are exempt from the requirements of Section XI and, by reference, from the requirements of Section III, as long as the materials and stress levels are consistent with the requirements of the applicable Construction Code. The Code provides this relief in consideration of the burden of the extensive controls required by Section III, especially subsection NCA, when compared to the likelihood and consequences of a failure. Under STP's exemption request, this relief would be expanded to include repair and replacement of safety-related LSS and NRS ASME items, regardless of size or product form.

Thus, for LSS and NRS ASME items, STP would be able to repair the items or fabricate or procure replacement items in accordance with the technical requirements of a nationally recognized Code, Standard or Specification suitable for that item as long as the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements. Installation and examination of LSS and NRS replacement components would be performed in accordance with ANSI B31.1. In addition, STP will conduct post-installation pressure tests in accordance with ASME Section XI. Other additional requirements of ASME Section III, including Subsection NCA, *General Requirements*, would not apply.

For example, the replacement of an LSS or NRS ASME valve with a non-ASME valve would consist of the following steps:

1. Initiate a design change package to implement the replacement of a code valve with a non-code valve.
2. Identify the proposed ANSI B16.34 valve
3. Ensure and document that the material specifications of the replacement valve are consistent with the requirements of ASME Section II or equivalent standard material (e.g., ASTM).
4. Ensure that the wall thickness exceeds the original design minimum wall thickness. (Note: this step, along with step 3 above, will ensure that the stresses are within allowables).
5. Prepare a Work Package that identifies installation requirements required by ANSI B31.1 such as welding, NDE, etc.
6. Install the valve and conduct post-installation testing in accordance with ASME Section XI and the requirements of the work package.
7. Maintain associated records for the life of the plant.

These measures will be sufficient to provide reasonable assurance, commensurate with the LSS or NRS risk significance, that the technical requirements of the Code will be satisfied for LSS and NRS replacement items, and that the replacement items will remain functional under design basis conditions.

4.1.2.3 Seismic Components

STP is requesting an exemption from a subset of the seismic requirements in 10 CFR Part 100. Under STP's exemption request, safety-related LSS and NRS components will still be required to function in the event of a safe shutdown earthquake (SSE), but will not be specifically qualified. As described in Section 3.3.7, STP will perform engineering evaluations to provide an appropriate level of assurance that procured LSS and NRS components will be able to perform their safety-related functions under design basis conditions. Thus, LSS and NRS components would still be available to function in an SSE, even though they are not important from a risk perspective to mitigate the accident or transient.

Additionally, LSS and NRS components will continue to be designed and installed so as not to impact HSS and MSS safety-related SSCs during a SSE. Seismic interaction is a standard which is currently applied between non-safety-related and safety-related equipment at STP. Under STP's exemption request, seismic interaction would be considered a functional requirement and would be maintained in accordance with existing special treatment requirements. Thus, STP would maintain provisions that would preclude adverse interactions between HSS/MSS components and LSS/NRS components where required.

4.2 Exemptions from Process-Related Requirements in Part 50

The changes in special treatment discussed above would normally be governed by several of NRC's process-related regulations. These regulations are 10 CFR 50.59 (which governs changes in the facility and procedures as described in the UFSAR) and 10 CFR 50.54(a) (which governs changes in the QA program description referenced in the UFSAR). As described more fully below, STP requests an exemption from each of these process-related requirements for LSS and NRS components to facilitate its implementation of the exemption from the special treatment requirements.

As described in more detail below, the exemptions from Sections 50.59 and 50.54(a) may not be necessary, when certain pending rule changes become effective and if the NRC interprets the revised rules as obviating the need for NRC review and approval of the individual changes to the special treatment provisions in STP's licensing basis as applicable to LSS and NRS components.

4.2.1 Exemption from 10 CFR 50.59

As currently effective, Section 50.59 requires a separate evaluation for each change in the facility or procedures as described in the UFSAR and requires NRC approval for any changes that involve an unreviewed safety question. In order to remove LSS and NRS components from the scope of special treatment requirements, STP will need to change the UFSAR.

Performing a 50.59 evaluation (and, as necessary, obtaining NRC approval) for each change in a special treatment requirement would be extremely burdensome and prohibitively costly for both STP and the NRC. Therefore, STP desires a permanent and global exemption from the Section 50.59 evaluation process, but only for changes in special treatment requirements as applicable for components that are categorized as either LSS or NRS. The 50.59 evaluation process will still be used for LSS or NRS components for issues unrelated to changes in special treatment (e.g., design changes in LSS and NRS components). The exemption will eliminate the need for individual 50.59 evaluations (and prior NRC approval) for each change in special treatment described in the UFSAR for LSS and NRS components. Following approval of this exemption, STP will be able to screen changes in UFSAR special treatment provisions for LSS and NRS components from further evaluation under Section 50.59.

The requested exemption from Section 50.59 applies not only to UFSAR special treatment provisions required by the regulations, but also to other UFSAR special treatment provisions that are not directly required by a regulation. STP would not need to perform a detailed 50.59 evaluation or seek prior NRC approval for UFSAR changes involving other inspection, maintenance, testing, qualification, and QA provisions applicable to LSS and NRS components that are not specifically covered by regulations (for example, such changes could involve the testing provisions for motor operated valves or air operated valves, and inspection of snubbers).

In 1999, the NRC revised Section 50.59 (64 Fed. Reg. 53613). When the revised rule becomes effective, it will raise the threshold of changes requiring prior NRC approval under Section 50.59. It might be possible to interpret the revised rule as not requiring prior NRC approval for revision of the special treatment described in the UFSAR for LSS and NRS SSCs.

For example:

- The revised rule defines "change" in terms of impacts on "functions." Since STP's proposal on special treatment requirements is not intended to affect the functions of LSS and NRS components, it could be considered that STP would not be making a "change" as defined in the revised rule. Similarly, it might be possible to consider the definition of "procedures" and "facility" in the revised rule as excluding changes in UFSAR special treatment requirements.

- The revised rule provides eight criteria for determining when a change requires prior NRC approval. The most relevant criterion requires prior NRC approval for changes that "result in more than a minimal increase in the likelihood of occurrence of a malfunction" of an SSC previously evaluated in the UFSAR. It might be possible to interpret this criterion (and the other criterion in the revised rule) as not requiring prior NRC approval for the change in special treatment requirements given the lack of safety/risk significance of LSS and NRS components.

However, the NRC has not issued sufficient guidance on the revised rule to permit a conclusive determination on these issues. Therefore, there is substantial uncertainty whether STP could make these changes in the UFSAR without seeking prior NRC approval.

Once the revision to Section 50.59 becomes effective, STP's request for an exemption from Section 50.59 may be unnecessary if the NRC were to interpret the revised rule as described above. Consequently, STP will withdraw its request for an exemption from Section 50.59 if the NRC staff issues a written position or guidance clearly stating that the types of changes in special treatment requirements being proposed by STP do not require prior NRC approval under Section 50.59.

4.2.2 Exemption from 10 CFR 50.54(a)

Section 50.54(a) requires a separate evaluation for each change in the QA program description referenced in the UFSAR, and requires NRC approval for any changes that involve a reduction in commitments. Removal of LSS and NRS components from the scope of the QA program will entail a change (and a reduction in commitment) in the QA program description. To accomplish these changes, it may require multiple changes to the QA Program. Performing a 50.54(a) evaluation and obtaining NRC approval for each change would be extremely burdensome and prohibitively costly for both STP and the NRC. Therefore, STP desires an exemption from Section 50.54(a) to eliminate the need for a separate 50.54(a) evaluation and prior NRC approval for removal of LSS and NRS components from the scope of the QA program.

STP notes that the NRC has issued a direct final rule (64 Fed. Reg. 9030) that raises the threshold of changes requiring prior NRC approval. This rule became effective on April 26, 1999. It would be reasonable to interpret the revised rule as not requiring prior NRC approval for the exclusion of LSS and NRS components from the scope of the QA program. In particular, Section 50.54(a)(3)(ii) of the Direct Final Rule states that prior NRC approval is not required for changes in the QA program that make use of a quality assurance alternative or exception approved by an NRC safety evaluation. This section could be interpreted as not requiring NRC approval for changes in the STP QA program description made in accordance with NRC's safety evaluation for this exemption request.

However, the NRC has not issued sufficient guidance on the rule to permit a conclusive determination on these issues. Therefore, there is uncertainty whether STP could make these changes in the QA program description without seeking prior NRC approval.

STP's request for an exemption from Section 50.54(a) may be unnecessary if the NRC were to develop their interpretation of the rule as described above. Consequently, STP will withdraw its request for an exemption from Section 50.54(a) if the NRC staff issues a written position or guidance clearly stating that the NRC's safety evaluation for STP's exemption request obviates the need for prior NRC approval under Section 50.54(a) of the types of changes in the QA program description being proposed by STP.

5.0 Technical Justification for the Exemption

5.1 Section 50.12 and Section 21.7

10 CFR 50.12(a)(1) allows exemptions that are "Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security." Similar provisions are in 10 CFR 21.7. Although 10 CFR Part 100 does not include a provision for exemptions, Part 100 is incorporated by reference in Part 50 through Section 50.34(b)(11); therefore, the exemption provisions in Section 50.12 are applicable to requests for exemptions from Part 100.

The requested exemption satisfies the requirements in Sections 50.12 and 21.7 described above:

- The requested exemption is not inconsistent with any law - There is nothing in the Atomic Energy Act or other statute applicable to the NRC that requires that the special treatment provisions in question be applied to LSS or NRS components, or that the NRC establish or implement the process-related regulations in question.

- The requested exemption is consistent with the common defense and security - The requested exemption relates solely to special treatment requirements and the process-related regulations, and does not pertain to the security or safeguards plans. Therefore, it does not have any effect on the common defense and security.

- The requested exemption would not present an undue risk to the public health and safety - The exemption would apply only to LSS and NRS components. From a risk-informed perspective, components that are categorized as LSS and NRS do not significantly affect the performance of any significant safety function. Furthermore, the exemption would exclude such components only from special treatment requirements and the process-related regulations, but not from the design and functional requirements for these components. As explained in the next section, the exemption from the special treatment requirements and the process-related regulations will not impact defense-in-depth or safety margins, and will result in no net increase in risk. In addition, the approval and implementation of this exemption request will focus more attention on safety/risk significant SSCs. This will result in a risk positive outcome. Therefore, the exemption will not adversely impact the public health and safety.

NRC has raised a question on whether the exemption to exclude LSS and NRS components from the scope of the term "basic component" would be inconsistent with the definition of "basic component" as provided in Section 223 of the Atomic Energy Act. For several reasons, STP believes that such an exemption would not be inconsistent with Section 223:

- As provided in 10 CFR 21.1, Section 206 of the Energy Reorganization Act, not Section 223 of the Atomic Energy Act, is the governing statute for Part 21. Section 206 itself only requires the reporting of defects and noncompliances that could create a substantial safety hazard. Section 206 does not define the term "basic component".

- Section 223.b of the Atomic Energy Act does use the term "basic component" and defines it similarly to "basic component" as defined in Part 21. However, Section 223.b is a criminal statute - - it is not the governing statute for Part 21. In particular, Section 223.b expressly states that its definition of "basic component" is "for the purposes of this subsection," and does not require that this definition be used for other sections in the Act or the Commission's regulations.

In summary, the definition of "basic component" in Section 223.b is restricted to that section, does not apply to Section 206, and does not require that the NRC use the same definition of "basic component" in Part 21. In this regard, the Commission has previously taken the position that Section 206 does not require Part 21 to apply to all safety-related SSCs and that the NRC has discretion to determine what kinds of SSCs should be considered "basic component," and this position has been accepted by the courts. *See Natural Resources Defense Council v. NRC*, 666 F.2d 595, 603 (D.C. Cir. 1981). Therefore, NRC is free to grant exemptions from the definition of "basic component" in Part 21.

5.2 Regulatory Guide 1.174

Regulatory Guide 1.174 provides technical guidance for licensees who request NRC approval for changes in the licensing basis using a risk-informed approach. This guidance establishes five principles governing risk-informed changes to the licensing basis. As demonstrated in the following sections, the requested exemption satisfies these principles.

5.2.1 Compliance with Current Regulations

Regulatory Guide 1.174 states that the proposed change in the licensing basis should meet current regulations, unless it is explicitly related to a requested exemption or rule change. This exemption request implements this principle.

5.2.2 Defense-in-Depth

Regulatory Guide 1.174 states the proposed change in the licensing basis should be consistent with the defense-in-depth philosophy.

The impact of this exemption on defense-in-depth is similar to the GQA program impact, which was previously approved by the NRC in Section 3.5.2 of the Safety Evaluation for the GQA program. Consistency with defense-in-depth is assured by the following factors:

- Functional requirements and the design configuration of systems are retained.
- No existing plant barriers will be removed or altered.
- Design provisions for redundancy, diversity, and independence will be maintained.
- The plant's response to transients or other initiators will not be affected.
- Preventive or mitigative capability of components will be preserved.
- Normal commercial and industrial practices will provide an appropriate and acceptable level of assurance that LSS and NRS components will be able to perform their intended functions.
- As discussed in the following sections, even if it is unrealistically assumed that the use of commercial and industrial practices would result in substantial increases in failure frequencies, STP has performed bounding sensitivity analyses which demonstrate that there would be little, if any, impact. It is important to note that STP's corrective action program will continue to be used to identify and correct any adverse trends for plant systems and/or equipment (including safety-related LSS and NRS components), and as such, is a significant barrier against allowing unacceptable levels of performance degradation to occur.

Therefore, based on the above and on additional details discussed in the NRC's Safety Evaluation for the approved GQA program, the exemption will be consistent with the defense-in-depth philosophy.

5.2.3 Safety Margins

Regulatory Guide 1.174 states that the proposed change in the licensing basis should maintain sufficient safety margins.

Similar to the GQA program approved by the NRC, the requested exemption does not involve a change in any functional requirements or the configuration of SSCs. Because of the low risk significance of the components in the scope of the exemption and the lack of any change in the functional requirements for SSCs, STP does not expect the need to change any of the safety analyses in the UFSAR. Therefore, sufficient safety margins associated with the design will be maintained by the exemption.

The requested exemption will enable STP to remove LSS and NRS components from the scope of special treatment requirements in Parts 21, 50, and 100 related to qualifications, quality assurance controls, maintenance requirements, and inspection and testing. This change will result in an appropriate and acceptable level of assurance of the quality of these components for the following reasons:

- STP does not intend to replace the existing safety-related LSS and NRS components, absent good cause (e.g., obsolescence or failure). Since the existing safety-related LSS and NRS components were designed, procured, manufactured, and installed in accordance with the existing special treatment requirements, these components have inherent design margins to perform their intended functions that will not be adversely affected by this exemption.

- If STP replaces an LSS or NRS ASME component with a non-ASME component, STP will ensure that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements. This will provide an appropriate level of assurance that replacement components for ASME items have sufficient margin.

- The exemption will allow LSS and NRS components to be inspected, tested, and maintained under normal commercial and industrial practices. In addition, replacement components may be designed, manufactured, procured, and installed under STP's normal commercial and industrial practices. These practices provide an appropriate and acceptable level of assurance that the components will satisfy their functional requirements. Given the minimal safety/risk significance of the components in question, these practices are sufficient to provide an appropriate level of assurance that the components will satisfy their functional requirements and have sufficient safety margin. Furthermore, performance of the systems/trains containing safety-related LSS and NRS components will be monitored under the Maintenance Rule program by tracking MRFFs if the failure causes loss of an HSS/MSS function and under the Corrective Action Program to ensure that these systems/trains maintain satisfactory levels of performance.

- STP will continue to apply its corrective action program to safety-related LSS and NRS components. This program provides reasonable assurance that deficiencies involving safety-related LSS and NRS components will be identified and corrected, and that the necessary action will be taken to ensure acceptable performance levels are maintained.

Although there is reasonable assurance that the LSS and NRS components will continue to function, in the unlikely event that they fail, there would be no significant consequences as discussed below:

- Components categorized as NRS are not sufficiently significant to warrant modeling in the PRA. The safety-related NRS components generally include local indicators, vents, drains, and root valves sized 1" and under. Even if it is assumed that an NRS component were to fail, the results of the PRA would be unaffected.

- In the systems reviewed to date using the approved risk categorization process, approximately 30% of the safety-related components have been categorized as LSS. These LSS components generally include manual process valves, handswitches, and instrumentation. As described in more detail in Section 5.2.4.1, STP has performed various sensitivity studies to bound the impact on overall risk of changes in the reliability of LSS components. In performing these analyses, STP very conservatively assumed that the failure rate for these components could be a factor of ten greater than indicated by operating experience. Despite this overly conservative assumption, the sensitivity studies showed that the risk-informed categorization of the components was not adversely affected, and that the impact on CDF and LERF was insignificant. Therefore, even if it is assumed that the relaxation of special treatment controls would result in increase failure rates for LSS components, there would be little, if any, impact on risk.

- In determining the categorization of a component, STP considers whether failure of the component could directly cause an initiating event or result in failure of another risk-significant SSC; whether the system is needed to mitigate an accident or transient, or is used in the EOPs; and whether the SSC is significant to the safety of mode changes and shutdown conditions. If any of these conditions could exist, a consistent, proceduralized methodology is applied to determine the significance and probability of the component failure. If failure of the component is determined to have a significant impact and/or if the event in question is considered probable, the component is classified as HSS or MSS. As a result, LSS and NRS components do not have significant impacts on safety significant functions.

In summary, STP's normal commercial and industrial practices provide an appropriate and acceptable level of assurance that LSS and NRS components will continue to have sufficient margin to satisfy their functional requirements. Nevertheless, even if it were assumed that these components would not perform as expected, there would be no significant impact on safety.

5.2.4 Changes in Risk

5.2.4.1 Increases in Risk

Regulatory Guide 1.174 states that, when proposed changes in the licensing basis result in an increase in core damage frequency or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

STP uses a site-specific PRA in the risk-informed process for categorizing SSCs. The NRC has previously approved the quality of the PRA, quality assurance for the PRA, the scope of the PRA, and the application of the PRA results and insights to the risk-informed process for categorizing components, as described in Section 3.2 of the NRC's Safety Evaluation for the GQA program. In this safety evaluation, the NRC also approved the overall risk significance categorization process as an acceptable means of determining the risk significance of SSCs. This process combines the PRA results with deterministic insights developed by a group of experienced individuals. These same tools will be used for this exemption.

As explained below, most LSS and NRS components are not modeled in the PRA; therefore, even if there are changes in the availability and reliability rates for these components, such changes will not affect the results of the PRA. Furthermore, to the extent that such components are modeled, the exemption is not expected to have any appreciable effect on the failure rates of LSS and NRS components; therefore, the results of the PRA should not be affected as a result of this exemption. Finally, even if it were assumed that the failures rates of LSS and NRS components will significantly increase as a result of the exemption, there would be little impact upon the results of the PRA, and the impact would satisfy the applicable quantitative criteria in Regulatory Guide 1.174.

5.2.4.1.1 NRS Components and Most LSS Components Do Not Affect the Results of the PRA

As discussed previously, NRS components are not sufficiently significant to warrant modeling in the PRA. Therefore, the exemption to exclude NRS components from the scope of special treatment requirements will not affect the results of the PRA for STP.

Furthermore, the vast majority (approximately 90%) of the safety-related LSS components were so categorized solely for deterministic reasons; i.e., they were not sufficiently significant to warrant modeling in the PRA. Therefore, the exemption to exclude these components from the scope of the special treatment requirements will not affect risk levels as determined by STP's PRA.

In summary, all of the NRS components, and the great majority of the LSS components, are not sufficiently important to warrant modeling in the PRA. Therefore, any changes in the special treatment requirements for these components will not affect the results of the PRA.

5.2.4.1.2 The Exemption Should Not Affect Failure Frequencies of LSS and NRS Components and Therefore Should Not Affect the Results of the PRA

The changes in the special treatment requirements do not lend themselves to a quantitative risk assessment, because the relationship between the special treatment requirements and equipment performance has not been firmly established. However, the change in the special treatment requirements for LSS and NRS components is not expected to impact system performance levels, because STP will continue to monitor SSC performance under the Corrective Action Program and the Maintenance Rule program and take appropriate corrective actions as necessary to maintain system performance. Additionally, as discussed in Section 3.3.7 of this exemption request, STP will apply commercial treatment to safety-related LSS and NRS components. Although this treatment will not reach the levels mandated by NRC's current regulations, it will be sufficient to ensure that the components have a high reliability level.

Furthermore, the change in special treatment requirements should have no adverse impact upon the availability of LSS and NRS components. Failures of such components will still be subject to the STP corrective action program. Furthermore, since STP may be taking these components out-of-service less frequently for unnecessary maintenance, inspection, and testing, the availability of such components may actually increase as a result of the exemption.

Finally, to the extent that data is available, the data demonstrate that the failure frequencies for similar types of safety-related and non-safety-related components are not significantly different. STP has performed an analysis of data from the Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX). Nuclear industry data reporting to the Nuclear Plant Reliability Data System (NPRDS) spans the time period from 1977 through 1996. The EPIX Maintenance Rule and Reliability Information (MRRI) database includes component failure data since 1996. NPRDS component engineering data includes indication of safety class, thus enabling a distinction between safety-related component and non-safety-related component failure rates. While the MRRI database does not include a safety-class distinction, INPO was able to provide STP an MRRI database file for 1997-1999 data that is "back-linked" to NPRDS, thus providing indication of safety class. The NPRDS data and MRRI data were first analyzed separately then merged to provide a large-scope analysis for the purposes of this exemption. STP has developed a report, entitled "Safety-Related Versus Non-Safety-Related Equipment Failure Frequency Data Analysis for Nuclear Power Plants in the United States" dated April 6, 2000, describing this NPRDS-MRRI data analysis. This report is available upon request.

The scope of this merged NPRDS-MRRI analysis includes over 670,000 component records and over 166,000 component failure records. The historical data analyzed consisted of over 74 billion component-hours of experience. The Response to Request for Additional Information (RAI) 42 in Attachment 4 includes tables that provide information for all 33 component type data categories contained in the merged NPRDS-MRRI database. These tables show that the calculated safety-related failure frequencies are generally greater than or roughly

equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS-MRRI data. This analysis shows that, of 33 component type categories investigated, 21 had higher safety-related failure frequencies than corresponding non-safety-related failure frequencies. Non-safety-related failure frequency values were significantly higher than corresponding safety-related failure frequencies in only one of the 33 categories (the "containment penetration" component type category). The analysis shows that, for most component types, the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data.

In addition to the analysis of the data contained in the EPIX database, STP has performed limited data collection in support of an on-going Balance-of-Plant (BOP) model. The data collected covers active equipment necessary to support power production (e.g., feedwater and condensate pumps). The collected data indicate no apparent difference in the failure rates for normally operating motors between safety and non-safety-related equipment. These results support the conclusions of the data analysis of the EPIX data.

Based upon this data, STP concludes that the changes in the special treatment requirements for LSS and NRS components will not significantly impact the failure frequencies for those components. Consequently, the results of the PRA should not be significantly affected by the exemption.

5.2.4.1.3 Postulated Increases in Failure Frequencies of NRS and LSS Components Would Not Significantly Impact the Results of the PRA

Even if it were postulated that the change in special treatment requirements for LSS and NRS components would result in a significant increase in the failure frequencies for such components, such an increase would not have a significant impact upon the results of the PRA. This conclusion is supported by the lack of risk importance of LSS and NRS components, and by sensitivity studies performed by STP.

As stated above, none of the NRS components and only a small subset of the LSS components are sufficiently significant to warrant modeling in the PRA. With regard to the LSS components that are modeled in the PRA, a component is categorized as LSS under the PRA only if it has a FV value of less than 0.005 and a RAW of less than 2. Given these values and STP's estimated CDF of approximately 10^{-5}/yr and estimated LERF of approximately 10^{-7}/yr (excluding shutdown risk), it is apparent that LSS components contribute very little to the overall CDF and LERF or to changes in CDF and LERF. Therefore, there will be little or no increase in CDF and LERF, even assuming unrealistically high failure rates for these components.

Furthermore, STP has performed sensitivity studies in which unreliability was simultaneously increased for MSS and LSS components of a similar type within the scope of the PRA. STP selected check valves for analysis in this sensitivity study because most of the valves would have an LSS ranking in the PRA. Check valves experience both a passive (transfer close/open) and active failure (fail to open/close on demand) mode, and check valves in general have low failure rates which is ideal for evaluating the effect of changes in the failure frequency. STP increased the failure rates of the check valves by factors of 2, 5 and 10. Despite these increases, there was only one component that changed categories (i.e., it increased from LSS to MSS). This component was just inside the LSS boundaries and changed to MSS when the failure rate was increased by a factor of 10. However, the composite rank for the check valve in question had already been ranked MSS due to the importance of the valve during several planned maintenance evolutions. Therefore, this sensitivity study indicates that the overall categorization for the check valves was not changed even assuming a significant increase in failure rates.

STP expects that results similar to those for the check valves would be obtained if other component types were subjected to similar sensitivity analyses.

Additionally, STP has completed a sensitivity study for solid state protection system relays. STP investigated the effects of increasing failure rates by factors of ten and one hundred. No significant change in core damage frequency was seen with an increase in relay failure rates of one hundred. This is primarily due to redundancy (two out of four relay logic).

Finally, STP performed a sensitivity study to determine the impact on CDF and LERF from postulating a factor of 10 increase in the failure rates of <u>all</u> LSS components (there are currently 431 components that are currently categorized as LSS and modeled in the PRA). In all cases, increasing the failure rates of LSS components by a factor of 10 resulted in a failure rate that was greater than the 95th percentile for each of the LSS component failure rate distributions. The cumulative impact to the annual average CDF and LERF of the increased failure rates for all LSS components categorized to date is shown below:

	Current Average (events/reactor year)	**Sensitivity Study λ_{LSS}*10 (events/reactor year)**	**Increase**	**% Increase**
CDF	**9.0781E-6**	**9.3232E-6**	**2.4510E-7**	**2.7%**
LERF	**1.3742E-7**	**1.3911E-7**	**1.6900E-9**	**1.2%**

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF). These results show that the aggregate effects of increased failure rates are well within acceptance guidelines and are consistent with the Commission's Safety Goal Policy.

Based upon the above, STP concludes that, even if it is unrealistically assumed that the change in the special treatment requirements were to result in significant degradation of NRS and LSS components, there would be little or no increase in risk.

5.2.4.2 Decreases in Risk

Regulatory Guide 1.174 states that licensees asking for relief using risk-informed approaches should also identify areas in which requirements should be increased based upon the results of risk analyses. For STP, this provision would apply to those HSS and MSS components that are currently classified as non-safety-related (to date, STP has identified almost 400 such components).

As described in Sections 3.3.2 and 3.3.6, STP will evaluate components categorized as HSS or MSS to ensure that the existing controls are sufficient to maintain the reliability and availability of the safety-significant functions of these components in a manner that is consistent with their categorization. Once these components are identified as HSS or MSS through the categorization process, STP's process evaluates the existing controls that are placed on these components to identify what, if any, additional controls are needed to provide reasonable assurance that the component can satisfy its safety-significant functional requirements. In addition, these components are subject to the Maintenance Rule monitoring program, and the reliability of these components is assessed on a periodic basis through the feedback process. This process will evaluate the reliability of the component, the adequacy of the existing controls and categorization, and the need for any changes.

At this time, STP has not quantified the effect of these changes on the risk profile of the plant. However, STP expects that these changes will result in some improvements in the reliability of the HSS and MSS components, which in turn will result in a decrease in risk.

5.2.4.3 Conclusions Regarding Changes in Risk

As discussed above, the exclusion of LSS and NRS components from the scope of the special treatment requirements is expected to result in little or no decrease in reliability of these components, but the improvements in the controls for HSS and MSS components are expected to result in some increase in reliability of these components. Based upon its experience with the risk-informed categorization process, STP estimates that the increased controls for MSS and HSS components and removal of LSS and NRS components from the scope of special treatment requirements will result in a net decrease in risk (or will be risk neutral). The reason for this determination is that some MSS and HSS components will receive augmented controls, and LSS and NRS components will still be maintained sufficient to meet their associated system performance levels, such that resulting risk levels are maintained at current levels (i.e., risk neutral) or improved (i.e., risk positive). The NRC reached similar conclusions in Section 3.5.4 of its Safety Evaluation for the GQA program.

5.2.5 Monitoring of the Impact of the Proposed Change

Regulatory Guide 1.174 states that the impact of the proposed change in the licensing basis should be monitored using performance measurement strategies.

In order to monitor the impact of the changes in the special treatment requirements, STP will use the monitoring and feedback programs established as part of its GQA program. The NRC approved these programs in Sections 3.3 and 3.5.5 of the Safety Evaluation for the GQA program. Since these programs cover the SSCs that are the subject of this exemption request, the existing programs can be used for monitoring of both the GQA program and the changes in the special treatment requirements.

Some of the requested exemptions from the special treatment requirements relate to elements of the monitoring programs. STP will take the following steps to ensure that these exemptions do not adversely impact the effectiveness of the monitoring programs:

- Maintenance Rule Monitoring - As indicated in Regulatory Guide 1.160, although there may be a need to perform monitoring under the Maintenance Rule at the component level, most monitoring is performed at the train/system/plant level. STP will still be required to monitor the effectiveness of maintenance for the trains/systems/plant containing these components. Therefore, the Maintenance Rule monitoring program will continue to ensure the reliability and availability of the HSS and MSS functions of the systems containing the LSS and NRS components affected by this exemption. Furthermore, although LSS and NRS component performance is not explicitly monitored, degrading performance in these components, if significant, would be observed at the plant, system, or train level.

- Corrective Action Program - STP's request includes an exemption to exclude safety-related LSS and NRS components from the scope of the QA requirements in Parts 21 and 50. However, STP will continue to apply its corrective action program to components that are categorized as safety-related LSS and NRS, as approved by the NRC in Section 3.4.1.9 of the Safety Evaluation for the GQA program.

The approved monitoring and feedback programs include provisions for evaluating performance at a component level. These provisions are described in the following subsections.

5.2.5.1 Corrective Action Program

The corrective action program is controlled by procedure and permits anyone at the plant site who identifies a deficiency to document that condition for correction. Condition reports (CRs) document degraded equipment performance or conditions, including conditions identified as a result of operator rounds, system engineer walk-downs, and corrective/preventive maintenance activities. These documented deficiencies are available for review each day by Station personnel, and are acted upon to implement appropriate remedial and/or corrective actions.

Safety-related LSS and NRS components will continue to be subject to the corrective action program. This will ensure that input from the corrective action program regarding degradation or failures of safety-related LSS and NRS components (including information from trending and causal analyses) will continue to be provided to the monitoring and feedback program to assess the impact of the changes in the special treatment requirements and the need for future adjustments to ensure the continued reliability and availability of the components.

For example, if degradation is identified in LSS or NRS components, restoration activities could include testing to provide an appropriate level of assurance that the component is fully restored and functionally capable. This could include the performance of surveillance testing, stroke time testing, leakage testing, or other refurbishments as required to provide reasonable assurance that the component is capable of performing its intended functions.

5.2.5.2 Performance Reporting & Identification Database

The performance reporting and identification database collects both positive and negative indicators from the performance of plant activities, such as corrective maintenance, installation of modifications, inspection activities, and conduct of testing. The Quality organization provides oversight of this database. Negative indicators are documented through the initiation of a condition report. The scope of this program includes LSS and NRS components.

5.2.5.3 Maintenance Rule Monitoring

STP will continue to monitor performance on a plant, system, or train level, as appropriate. Data used for monitoring is obtained from various sources, such as work orders, condition reports, and test results. When performance goals are not satisfied, STP will perform evaluations to determine the cause and corrective action, including, as necessary, corrective action for LSS and NRS components.

5.2.5.4 Operating Experience Group (OEG)

The OEG collects and evaluates problems reported in the STP corrective action program database, which includes problems with individual safety-related LSS and NRS components. Condition reports generated through the Corrective Action Program against system components are reviewed. The review also includes related industry experience, review of performance indicators from the Performance Reporting & Identification Database, and identification of adverse trends and causes. The OEG identifies adverse equipment trends, possible common cause or common mode failures, and similarities in station and industry operating experience. When such problems are identified, a condition report is generated to obtain corrective action.

Additionally, management periodically reviews System Health Reports, which identify performance indicators, corrective action plans and status, as applicable.

5.2.5.5 Component Categorization Reviews

At least once per cycle, the Operating Experience Group performs a comprehensive evaluation of conditions generated within the previous reporting period against each specific risk-categorized system designator, and reports the results to the Working Group. This report includes information for the current reporting period, as well as the two previous reporting periods. The Working Group is tasked with determining if any categorization revisions are warranted based on:

- a degradation of equipment performance,
- System Engineer input,
- Maintenance Rule input, or
- Licensing, Quality, or Operations organization input.

Whenever degraded performance is attributed to the reduction or relaxation of special treatment controls, the Working Group will recommend appropriate remedial action up to and including the reinstatement of the subject special treatment control(s) and the potential re-categorization of the component to a higher level. Any proposed categorization changes are submitted to the Expert Panel for approval. Once approved, the categorization change is reflected electronically in the controlled Master Equipment Database and through a revision to the Risk Significance Basis Document for that system. In addition, if the categorization was changed from LSS/NRS to MSS or HSS, or if a special treatment control was reinstated, a new condition report would be generated to assess the impact of returning the subject component to the scope of the appropriate special treatments. This assessment would include an evaluation of activities performed on, with, or for the component during the time that the component was excluded from the scope of special treatment requirements.

While no specific timeframe is identified for reinstatement of the special treatment controls, it is expected that these controls will be reinstated in a timely manner (i.e., generally within the next available 12 week Functional Equipment Group (FEG) full cycle, if possible; if operational conditions necessitate that these additional controls be applied sooner, appropriate action will be taken to incorporate the controls.). The generated condition report remains open until the corrective actions, if any, are implemented as appropriate. These corrective actions may include, but are not limited to, an evaluation of the component's impact on current operating conditions and the Technical Specifications. The component's performance would continue to be monitored as part of future periodic reviews to provide an appropriate level of assurance that the applied controls are effective. It should be noted that the component's impact on current operating conditions is evaluated in accordance with the standard operability review that is performed following initiation of the condition report.

5.2.5.6 PRA Updates

The Risk and Reliability Analysis Department performs periodic updates of the PRA as described in Section 3.2.1 above. The updates include equipment failure rate data to reflect observed equipment performance. The results of the revised PRA are furnished to the Working Group, which, as part of the feedback and corrective action process, can recategorize the safety/risk significance of components as needed. These results are provided in a report that generally contains the following type of information: overall updated CDF and LERF, the change in CDF and LERF from the previous reporting period, a characterization of the dominant sequences, the initiating event contributions to CDF and LERF, a list of modeled component PRA risk rankings, and a list of components whose PRA categorization has changed. This information is presented to the responsible Working Group, and then to the Expert Panel who determines if additional corrective actions are required based on approved criteria (such as the FV and RAW criteria, and the criteria in Regulatory Guide 1.174 on CDF and LERF). If corrective actions are necessary, the actions are addressed by the Working Group. If no corrective actions are identified, then the Expert Panel approves the report, and in so doing, approves the updated risk levels until the next reporting period.

In summary, the existing monitoring program described above and approved by the NRC for the GQA program will be sufficient to monitor the effects of the changes in the special treatment requirements for LSS and NRS components and to provide an appropriate level of assurance that SSCs will continue to satisfy their functional requirements.

6.0 Special Circumstances

10 CFR 50, Section 50.12(a)(2) provides that "special circumstances" must be present before the NRC will grant an exemption. Similarly, Section 21.7 requires that exemptions be in the public interest. As discussed below, this exemption request satisfies several of the criteria for special circumstances identified in Section 50.12(a)(2) and also demonstrates that the exemption is in the public interest under Section 21.7.

6.1 Satisfaction of the Second Criterion for Special Circumstances

The second criterion states: "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The purpose of Part 50 is to provide reasonable assurance that the facility will be operated safely. The exemption is consistent with this purpose. The requested exemption will not change the functional requirements of plant systems or components. Furthermore, components that are HSS or MSS will continue to be subject to the special treatment regulations, thereby providing assurance that components with safety significance will perform their functions.

As discussed in Section 5.0 above, application of the special treatment requirements in Parts 21, 50, and 100 to LSS and NRS components is not necessary to provide adequate protection of public health and safety. The scope of the current regulations is based upon a deterministic approach. The scope of the existing regulations was developed without benefit of the insights currently available from probabilistic risk assessments and is more extensive than necessary for providing adequate protection of safety. Specifically, these regulations apply to LSS and NRS components that are not risk significant as determined by the PRA, and are not significant to safety based upon other deterministic considerations (e.g., their ability to mitigate accidents or transients and the consequences of their failure). Thus, it is not necessary to apply the special treatment regulations to LSS and NRS components to achieve the underlying purpose of the regulations. Therefore, the exemption satisfies the second criterion of Section 50.12.

Similarly, it is not necessary to apply the process-related regulations that are the subject of this exemption request in order to achieve their underlying purpose. The process-related regulations are intended to ensure that significant changes in the licensing basis for a plant are reviewed and approved by the NRC. This purpose will be served by NRC's approval of this exemption request. It is not necessary for the NRC to review and approve the numerous individual changes in special treatment requirements, because those changes will be implemented in accordance with NRC-approved criteria for this exemption.

6.2 Satisfaction of the Fourth Criterion for Special Circumstances

The fourth criterion is: "The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption."

As discussed in Section 5.2.4.2, there are some components at STP that are not safety-related but have been categorized as MSS or HSS. As part of STP's risk-informed approach to safety, STP will evaluate the controls currently applied to these components and will apply additional controls to these components, if appropriate. These controls will provide additional assurance that these components will satisfy their safety significant functional requirements and as such represents a benefit to public heath and safety.

STP has not performed a quantitative analysis to determine the decrease in risk from application of special controls to such components. However, based upon its experience with the risk-informed categorization process, STP estimates that the increased controls for some MSS and HSS components and removal of LSS and NRS components from the scope of special treatment requirements will result in a net decrease in risk, or will be risk neutral. The reason for this is that some MSS and HSS components will receive augmented controls, and LSS and NRS components will still be maintained sufficient to meet their associated system performance levels, such that resulting risk levels are maintained at current levels or improved.

Additionally, the exemption will lead to reduced occupational radiation exposures that would otherwise be incurred in providing unnecessary maintenance, testing, and inspection for the LSS and NRS components.

STP concludes that the exemption either will result in a net benefit to the public health and safety or will be risk neutral. Therefore, the exemption satisfies the fourth criterion of Section 50.12.

6.3 Satisfaction of the Sixth Criterion for Special Circumstances

This exemption request also satisfies the sixth criterion for special circumstances; namely, that "There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption."

At the time the special treatment regulations were issued, STP and other plants did not have an approved PRA or risk-informed process for categorizing the safety/risk significance of components. After the special treatment regulations were issued, STP established a risk-informed process for assessing the relative safety/risk significance of SSCs, and the NRC has reviewed and accepted this process for implementation. Application of this process provides a sound basis for identifying components that do and do not warrant special treatment, commensurate with each component's safety/risk significance. Such information on the relative safety/risk significance of SSCs was not available when the special treatment regulations were issued.

Furthermore, when the regulations were issued, no consideration was given to the costs and benefits of applying special treatment requirements to LSS and NRS components. As discussed above, the special treatment regulations, as applied to LSS and NRS components, do not provide any significant benefit to safety. Furthermore, these regulations impose unnecessary burdens with respect to LSS and NRS components. STP estimates that, upon full implementation of the GQA program and implementation of this exemption, component and part replacement savings will exceed $1 million per year. Additionally, STP conservatively estimates that the streamlining and enhancements of processes for LSS and NRS components could result in a total savings of more than $1 million per year from the Station's present operating and maintenance budget. In addition, requiring special treatment for LSS and NRS components increases occupational radiation exposure and diverts management and station personnel attention from activities that are more significant to safety and plant reliability. Therefore, an exemption would be beneficial from both a safety and cost perspective.

This new information demonstrates that LSS and NRS components provide little or no safety benefits, and that the costs of the special treatment regulations for LSS and NRS components outweigh the minimal safety benefits of such treatment. This information constitutes a changed circumstance, which justifies exempting LSS and NRS components from these special treatment regulations.

Similarly, when the process-related regulations addressed by this exemption request were issued, it was not considered that a licensee would want to make changes in the special treatment requirements for thousands of components. Consequently, the unnecessary burden posed by the need to evaluate each of these changes under these process-related regulations was also not foreseen. This constitutes a changed circumstance, which justifies the exemption from these process-related requirements.

6.4 Conclusions Related to Special Circumstances

The requested exemptions for both the special treatment requirements and the process-related regulations satisfy several of the criteria for special circumstances and are in the public interest. Therefore, this exemption request meets the requirements of Sections 50.12 and 21.7, and should be granted.

7.0 Phased Approach to Implementation of the Exemption

Upon approval of the exemption, STP will begin the implementation process. Due to the unique nature of this exemption, a deliberate, phased approach will be pursued. This phased approach will enable the NRC to gain additional confidence that STP is implementing the exemption in an appropriate manner. In addition, with more than 43,000 components already classified through the approved risk-informed process for categorizing components, STP cannot reasonably implement all aspects of the exemption at one time.

7.1 Phased Implementation of the Risk-Informed Process for Categorizing Components

As part of its approval of the GQA program, the NRC approved a risk-informed process for categorizing the safety/risk significance of components. The component evaluations (and the associated bases for the resulting safety/risk significance), system function categorization, component critical attributes, and assessments performed by Licensing, Operating Experience, Quality, Operations, PRA, and Engineering personnel are documented in Risk Significance Basis Documents which are generated for each system evaluated. The development of the Risk Significance Basis Documents is governed by approved Station procedures, and the Risk Significance Basis Documents, once approved, are controlled through the STP Document Control process. The approved safety/risk significance evaluation information is electronically loaded into the Station's Master Equipment Database for reference and use by Station personnel. Use of the available safety/risk significance information cannot occur until the associated controlling processes or programmatic procedures have been revised.

Using this process, STP has already completed the categorization of components in the following systems:

AF – Auxiliary Feedwater
CC – Component Cooling Water
CH – Essential Chilled Water
CV – Chemical and Volume Control
DG – Emergency Diesel Generator (and eight auxiliary systems)
EW – Essential Cooling Water
FC – Spent Fuel Pool Cooling
FW – Main Feedwater
HC – Reactor Containment Building HVAC
HE – Electrical Auxiliary Building Heating Ventilation and Air Conditioning (HVAC)
HF – Fuel Handling Building HVAC

HM – Mechanical Auxiliary Building HVAC
IA – Instrument Air
MS – Main Steam
PS – Primary Sampling
RA – Radiation Monitoring
RC – Reactor Coolant
RH – Residual Heat Removal
SB – Steam Generator Blowdown
SI – Safety Injection
WL – Liquid Waste Processing

STP will implement the exemption on the above systems first. STP may complete component categorization for subsequent systems and anticipates that it will then implement the exemption for these systems on a routine basis.

7.2 Planned Implementation Approach

Initially, implementation of the exemption will focus on the systems for which the risk-informed categorization process has been completed at the time the exemption is granted. Upon approval of the exemption, STP will begin a phased implementation of the exemption for those systems as described in the following sections.

7.2.1 Procedure Changes

STP will evaluate and revise its procedures applicable to each of the areas addressed by the exemption. STP will revise its procedures, as necessary, to reflect the description of the treatment described in Section 3.3.7 of this exemption request.

Until these procedures are revised, STP will continue to ensure that safety-related LSS and NRS components are subject to applicable safety-related and quality-related procedures. In addition, as appropriate, STP will modify various programs (e.g., provisions for motor operated valve (MOV) program, air operated valve (AOV) program, snubber testing program, molded case circuit breaker program) to remove LSS and NRS components from the scope of these programs. Although an exemption is not required to modify these programs, the modifications will be processed as required, in accordance with STP's commitment change process, as described in the following section.

7.2.2 Changes in Licensing Basis Documents

In conjunction with the evaluations and changes in procedures discussed above, STP will evaluate the need for corresponding changes in the UFSAR and the Operations Quality Assurance Program (OQAP). STP will provide the NRC with reports of these changes as required by 10 CFR 50.71(e) and 50.54(a). Since STP is requesting an exemption from Sections 50.59 and 50.54(a) for changes to special treatment requirements for LSS and NRS components, STP will not need to perform the evaluations normally required by these regulations, and will not need to seek prior NRC approval of such changes. Following the grant of the exemption, STP will simply notify the NRC of any changes in the UFSAR or OQAP special treatment requirements for LSS and NRS components in accordance with 10 CFR 50.71(e).

Currently, STP does not anticipate that such changes will impact the Technical Specifications. However, to the extent that an individual change in the UFSAR will also entail a change in the Technical Specifications, STP will not implement the modification without first applying for and receiving NRC approval of a separate application for an amendment of the Technical Specifications.

The Technical Specifications apply to some systems that include LSS and NRS components. These Technical Specifications provisions will continue to apply after the exemption is granted. Therefore, STP will continue to evaluate nonconforming and degraded conditions involving LSS and NRS components to determine whether such conditions affect the operability of systems controlled by the Technical Specifications. Additionally, STP will continue to perform the surveillances required by the Technical Specifications for those components.

Changes to other STP commitments related to special treatment requirements will be controlled using STP's commitment control process. STP's process is consistent with the guidance of the Nuclear Energy Institute (NEI) NEI-99-04, entitled *Guidelines for Managing NRC Commitment Changes*, which the NRC found acceptable in SECY-00-0045.

7.2.3 Changes in Activities and Components in the Field

Following revisions of the procedures applicable to the LSS and NRS components, STP will begin to implement the revised procedures.

For example, following the revisions of the governing procurement procedures, STP will begin procuring commercially available items on an as-needed basis for currently installed LSS and NRS components. The procurement of these commercial items will occur as needs develop and as required to support regularly scheduled maintenance activities.

The NRC will have the opportunity to inspect the implementation of the exemption for these initial systems. STP will factor the results of these inspections into its feedback and monitoring process described below.

7.2.4 Feedback and Monitoring of Implementation Activities

As discussed in Section 5.2.5, STP will maintain a feedback mechanism in place to periodically evaluate system and equipment performance results, and to assess the impact of the changes that have been made. This assessment will include HSS and MSS non-safety-related components to determine what additional controls are necessary to ensure equipment performance, availability, and reliability. Plant/system/train performance levels will be monitored in accordance with Maintenance Rule requirements by tracking MRFFs (if the failure causes loss of an HSS/MSS function) to provide an appropriate level of assurance that changes to the controls for LSS and NRS components will not have an adverse effect on performance. Also, the assessment will evaluate if reductions in quality controls or other requirements have resulted in a decrease in component reliability, and if it would be prudent to enhance the oversight controls or simply procure another component. This process provides an appropriate level of assurances that any negative performance changes that are attributed to the relaxation of special treatment controls are addressed in a timely manner.

Whenever degraded performance is attributed to the reduction or relaxation of special treatment controls, the Working Group will recommend the appropriate remedial action up to and including the reinstatement of the subject special treatment control(s) and the potential re-categorization of the component's significance to a higher level. Any proposed categorization changes are submitted to the Expert Panel for approval. Once approved, the categorization change is reflected electronically in the controlled Master Equipment Database and through a revision to the Risk Significance Basis Document for that system. In addition, if the categorization was changed from LSS/NRS to MSS or HSS, or if a special treatment control was reinstated, a new condition report would be generated to assess the impact of returning the subject component to the scope of the appropriate special treatments. This assessment would include an evaluation of activities performed on, with, or for the component during the time that the component was excluded from the scope of special treatment requirements.

STP's monitoring and feedback process ensures that any changes in equipment performance are evaluated for impact on risk significance. Condition reports are initiated to document component failures or performance degradations and the resulting corrective actions. Condition reports may also be used to initiate and document the results of Preventive Maintenance activities. For each system whose components have been risk ranked, the associated condition reports are reviewed and evaluated periodically for evidence of negative performance trends. Any such evidence is brought to the attention of the Working Group where it is evaluated for impact on the risk ranking of the associated components. The Working Group, with Expert Panel approval, then adjusts the risk ranking, as appropriate. This feedback loop provides an appropriate level of assurances that any negative performance changes that are attributed to the relaxation of special treatment controls are addressed by the reinstatement of applicable controls up to and including the re-categorization of the component's risk significance, as appropriate.

7.2.5 Summary

In summary, STP will implement this exemption in a phased approach. This will enable STP to account for lessons learned from initial implementation, and will enable the NRC to perform inspections and reviews to gain additional confidence in STP's proposed implementation process before the process is applied across additional systems at the plant.

8.0 Conclusions

The requested exemption satisfies the requirements in Sections 50.12 and 21.7 and the intent of Regulatory Guide 1.174, will reduce unnecessary burdens, and will enable STP to focus attention and resources on matters that are truly significant to safety. Therefore, STP requests that the NRC grant STP an exemption from the regulations identified in Attachment 2. These exemptions will enable STP to exclude LSS and NRS components from the scope of the special treatment requirements, and to make associated changes in its licensing basis without seeking prior NRC approval for each change. In addition, STP believes that this prototype application will assist the NRC in developing the risk-informed approach to 10 CFR Part 50.

ATTACHMENT 2

REVISED LIST OF REGULATIONS FOR WHICH AN EXEMPTION IS REQURESTED

REVISED LIST OF REGULATIONS FOR WHICH AN EXEMPTION IS REQUESTED

Regulation	Scope of Exemption	Justification for Exemption
10 CFR 21.3 – Request an exemption to exclude safety-related LSS and NRS components from the scope of the definition of "basic component." [See Note 1]	The procurement, dedication, and reporting requirements in Part 21 would not be applied to safety-related LSS and NRS components. Section 3.3.7 of Attachment 1 identifies the procurement controls that would be applied to safety-related LSS and NRS components.	Part 21 imposes procurement and dedication requirements and requires the reporting of defects and noncompliances involving components whose failure could cause a "substantial safety hazard." Reporting of defects and noncompliance involving safety-related LSS and NRS components is not necessary to meet the intent of Part 21, because failure of such components would not result in a substantial safety hazard.
10 CFR 50.34(b)(6)(ii) - Request an exemption to the extent that it incorporates provisions from 10 CFR Part 50, Appendix B.	Refer to request for exemption from Appendix B.	Refer to request for exemption from Appendix B.
10 CFR 50.34(b)(11) – Request an exemption to the extent that it incorporates seismic qualification requirements in Part 100.	Refer to request for exemption from Part 100.	Refer to request for exemption from Part 100.
10 CFR 50.49(b) – Request an exemption to exclude LSS and NRS components from the scope of electric equipment important to safety for the purposes of environmental qualification of electrical components.	• The qualification documentation and files specified in Section 50.49 would not be applicable to LSS and NRS components. • LSS and NRS components would not be required to be maintained in a qualified condition under Section 50.49.	Section 50.49 ensures that electrical components important to safety can perform their safety function in a harsh environment during and following a design basis event. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. Therefore, it is not necessary to maintain such equipment in a qualified condition or to replace such components with qualified components in order to meet the intent of Section 50.49.

Regulation	Scope of Exemption	Justification for Exemption
	• LSS and NRS components could be replaced with components that are not qualified under Section 50.49. • LSS and NRS components, as applicable under Section 50.49, would be designed to function in the installed environment. Section 3.3.7 of Attachment 1 identifies the design and procurement controls that would be applied to LSS and NRS components to achieve this requirement.	
10 CFR 50.54(a)(3) – Request an exemption from the requirement to seek prior NRC approval for reductions in the commitments in the QA program description involving LSS and NRS components.	STP would not need prior NRC approval for reductions in commitments in the QA program description related to LSS and NRS components.	It would be extremely burdensome and prohibitively costly to seek prior NRC approval for each such change. NRC's approval of this exemption request serves the same purpose as the approval required by this section of the regulations.
10 CFR 50.55a(f) and (g) – Request an exemption from the requirements of ASME Section XI, for repair and replacement of safety-related LSS and NRS components (except that the material specifications and the design (design loadings, design methodology, and stress	Safety-related LSS and NRS components could be repaired and replaced with non-Code components, subject to meeting the original requirements for materials and design methodology and conducting post-installation pressure tests in accordance with Section XI of the ASME Code. Sections 3.3.7 and 4.1.2.2 of Attachment 1 identifies the quality, design and procurement controls that	These provisions ensure that mechanical systems and important components within these systems can perform their safety function. By definition, components that are categorized as LSS and NRS do not significantly contribute to mitigating or preventing accidents. Therefore, when the need arises to repair and replace these components, it is not necessary to repair, or procure and install, the component pursuant to the ASME Code to satisfy the purpose of these provisions, as long as materials and stress levels meet the referenced

Regulation	Scope of Exemption	Justification for Exemption
allowables) are consistent with the original requirements.	would be applied to safety-related LSS and NRS components that are repaired or replaced.	requirements.
10 CFR 50.55a(f) – Request an exemption from meeting the requirements of ASME Section XI for testing of safety-related LSS and NRS components.	Safety-related LSS and NRS components would not be in the scope of component-specific inservice testing requirements. System-level testing requirements would continue to be applied. Additionally, Section 3.3.7 of Attachment 1 identifies other controls that would be applied to ensure the functionality of safety-related LSS and NRS components.	These provisions ensure that mechanical systems and important components within these systems can perform their safety function. By definition, components that are categorized as LSS and NRS do not significantly contribute to mitigating or preventing accidents. Therefore, it is not necessary to perform ASME inservice testing of these components to satisfy the purpose of these provisions.
10 CFR 50.55a(g) – Request an exemption from meeting the requirements of ASME Section XI for inspection of safety-related LSS and NRS components.	Safety-related LSS and NRS components would not be in the scope of component-specific inservice inspection requirements. Section 3.3.7 of Attachment 1 identifies controls that would be applied to ensure the functionality of safety-related LSS and NRS components.	These provisions ensure that mechanical systems and important components within these systems can perform their safety function. By definition, components that are categorized as LSS and NRS do not significantly contribute to mitigating or preventing accidents. Therefore, it is not necessary to perform ASME inservice inspection of these components to satisfy the purpose of these provisions.
10 CFR 50.55a(h) – Request an exemption to exclude safety-related LSS and NRS components from the scope of components required to meet sections 4.3 and 4.4 of IEEE 279.	Sections 4.3 and 4.4 of IEEE 279 would not apply to safety-related LSS and NRS components. The other requirements listed in IEEE 279, including functional and design requirements, would be applicable. Additionally, Sections 3.3.7 and 4.1.2.1 of Attachment 1 identifies other controls that would be applied to ensure the functionality of safety-related LSS and NRS components.	The quality assurance requirements and environmental qualification requirements in Sections 4.3 and 4.4 of IEEE 279 are not necessary for these components. By definition, components that are categorized as LSS and NRS do not significantly contribute to mitigating or preventing accidents. Therefore, it is not necessary to apply these requirements to satisfy the purpose of these provisions.

Regulation	Scope of Exemption	Justification for Exemption
10 CFR 50.59(a)(1), (a)(2) and (b)(1) (pre-1999 version); 10 CFR 50.59(c)(1), (c)(2), and (d)(1) (2000 version) – Request an exemption from the requirement to perform a written evaluation of changes in special treatment requirements for LSS and NRS components. Also request an exemption from the requirement to seek prior NRC approval for such changes to the extent that they fall within the listed criteria in 50.59.	STP would not be required to perform 50.59 evaluations for changes in the special treatment requirements for LSS and NRS components, and would not be required to seek prior NRC approval for those changes.	It would be extremely burdensome and prohibitively costly to perform a 50.59 evaluation and seek prior NRC approval for each such change. NRC's approval of this exemption request serves the same purpose as the approval required by this section of the regulations.
10 CFR 50.65(b) – Request an exemption to exclude LSS and NRS components from the scope of SSCs covered by the Maintenance Rule.	• STP would be required to monitor performance on a plant/system/train level, and to evaluate failures of LSS and NRS components that cause a Maintenance Rule Functional Failure (MRFF) at the applicable plant/system/train level if the failure results in loss of an HSS/MSS function.	Section 50.65 requires monitoring of the effectiveness of maintenance activities for "safety significant plant equipment" to minimize the likelihood of failures and events caused by lack of effective maintenance. LSS and NRS components do not fall within the intent of Section 50.65. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function.
10 CFR Part 50 Appendix A, GDC 1 – Request an exemption to exclude LSS and NRS components from	• LSS and NRS components would not be required to satisfy the QA requirements in GDC 1 (except to the extent that they incorporate Criteria	Quality assurance provides adequate confidence that SSCs, which prevent or mitigate the consequences of accidents that could cause undue risk to the public health and safety, will perform satisfactorily in service. By

Regulation	Scope of Exemption	Justification for Exemption
the scope of SSCs important to safety under GDC 1.	III, XV, and XVI of Appendix B to Part 50). • Section 3.3.7 of Attachment 1 identifies other controls that would be applied to ensure the functionality of LSS and NRS components.	definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. Therefore, exclusion of such components from the scope of the QA program is consistent with the intent of these regulations. Furthermore, this exemption will not affect any of the functional requirements for the components.
10 CFR Part 50, Appendix A, GDC 2 – Request an exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 2, to the extent that GDC 2 requires tests and inspections to demonstrate that SSCs are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.	• LSS and NRS components would not be required to be maintained in a qualified condition under GDC 2. • LSS and NRS components could be replaced with components that are not qualified under GDC 2. • LSS and NRS components, as applicable under GDC 2, would be designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Section 3.3.7 of Attachment 1 identifies the design and procurement controls that would be applied to LSS and NRS components to achieve this requirement.	These qualification requirements ensure that components important to safety can perform their safety function during and following a design basis event. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. It is unnecessary to maintain the qualification of such components or to replace them with qualified components to meet the intent of these regulations.
10 CFR Part 50, Appendix A, GDC 4 – Request an exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 4, to the extent that GDC 4 requires	• LSS and NRS components would not be required to be maintained in a qualified condition under GDC 4. • LSS and NRS components could be replaced with components that are not qualified under GDC 4.	GDC 4 ensures that components important to safety can perform their safety function during and following a design basis event. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. Therefore, it is not necessary to maintain such equipment in a qualified condition or to replace such components with

Regulation	Scope of Exemption	Justification for Exemption
inspection and testing to demonstrate that SSCs are able to withstand environmental effects. GDC 4 requirements related to dynamic effects would not be exempted.	• LSS and NRS components, as applicable under GDC 4, would be designed to withstand applicable environmental effects. Section 3.3.7 of Attachment 1 identifies the design and procurement controls that would be applied to LSS and NRS components to achieve this requirement. • GDC 4 requirements with respect to dynamic effects would continue to be applicable.	qualified components in order to meet the intent of GDC 4.
10 CFR Part 50, Appendix A, GDC 18 – Request an exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 18, to the extent that GDC 18 requires that inspection and testing be performed for individual features, such as wiring, insulation, connections, switchboards, relays, switches, and buses.	• Individual LSS and NRS components within the electrical power systems would not be required to be inspected or tested. • System functional tests would still be required in accordance with GDC 18. • Additionally, Section 3.3.7 of Attachment 1 identifies other controls that would be applied to ensure the functionality of LSS and NRS components.	These provisions ensure that Electric Power Systems and important components within these systems can perform their safety function. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. Therefore, it is not necessary to inspect or test these components to satisfy the purpose of these provisions.
10 CFR Part 50 Appendix B, Introduction – Request an exemption to exclude safety-related LSS and NRS components from the scope	• Safety-related LSS and NRS components would not be required to satisfy the QA requirements in Appendix B, except for design control, control of nonconformances,	Quality assurance provides adequate confidence that SSCs, which prevent or mitigate the consequences of accidents that could cause undue risk to the public health and safety, will perform satisfactorily in service. By definition, components that are categorized as LSS and

Regulation	Scope of Exemption	Justification for Exemption
of safety-related SSCs covered by Appendix B (except for Criterion III pertaining to Design Control and Criteria XV and XVI governing nonconformances and corrective actions).	and corrective action. • Section 3.3.7 of Attachment 1 identifies other controls that would be applied to ensure the functionality of safety-related LSS and NRS components.	NRS do not involve the performance of any safety significant function. Therefore, exclusion of such components from the scope of the QA program is consistent with the intent of these regulations. Furthermore, this exemption will not affect any of the functional requirements for the components, control of nonconformances, and corrective actions.
10CFR Part 50, Appendix J, B.III – Request an exemption to exclude safety-related LSS and NRS components, subject to the additional limitations listed under Justification for Exemption, from the scope of components requiring local leak rate tests and containment isolation valve leak rate tests.	• Local leak rate tests of LSS containment isolation valves and other safety-related LSS or NRS components would not be required. With respect to LSS containment isolation valves, this exemption only applies to valves that satisfy one or more of the criteria in the Justification for Exemption. • Cumulative limits for containment leakage would be based upon the tested components, with the assumption that the exempted components contribute zero leakage. • Section 3.3.7 of Attachment 1 identifies controls that would be applied to ensure the functionality of safety-related LSS and NRS components.	By definition, components that are categorized as LSS and NRS do not significantly contribute to mitigating or preventing accidents. Furthermore, to be exempt, a containment isolation valve would be required to meet one or more of the following criteria: • The valve is not required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers). • The valve is normally closed and in a physically closed, water-filled system (e.g., containment isolation valves in the Demineralized Water system) • The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Main Feedwater system). • The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment

Regulation	Scope of Exemption	Justification for Exemption
		isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve. • The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).
10 CFR Part 100, Appendix A.VI(a)(1) and (2) – Request an exemption to exclude safety-related LSS and NRS components from the scope of SSCs covered by these sections, to the extent that these sections require testing and inspection to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.	• LSS and NRS components would not be required to be maintained in a qualified condition under Part 100. • LSS and NRS components could be replaced with components that are not qualified under Part 100. • LSS and NRS components, as applicable under Part 100, would be designed to withstand the effects of seismic events without loss of capability to perform their safety function. Section 3.3.7 of Attachment 1 identifies the design and procurement controls that would be applied to LSS and NRS components to achieve this requirement.	These qualification requirements ensure that components important to safety can perform their safety function during and following a design basis event. By definition, components that are categorized as LSS and NRS do not involve the performance of any safety significant function. It is unnecessary to maintain the qualification of such components or to replace them with qualified components to meet the intent of these regulations.

1. In the alternative to granting an exemption from 10 CFR 21.3, the NRC could determine that such an exemption is not necessary. First, as provided in Section 21.3, the provisions in Part 21 and the quality assurance provisions in Appendix B to 10 CFR Part 50 are not applicable to a commercial grade item until it is dedicated. In fact, Section 21.3 states the purpose of dedication is to provide reasonable assurance "equivalent to an item designed and manufactured under a 10 CFR part 50, appendix B, quality assurance program." Since STP is requesting an exemption from the QA requirements in Appendix B with respect to LSS and NRS components, the NRC could decide that the dedication (and other provisions) in Part 21 are inapplicable to LSS and NRS components (i.e., there is no reason to dedicate a component to make its quality equivalent to that provided by Appendix B, if Appendix B is not applicable to the component). Second, as discussed in 10 CFR 21.1, the purpose of Part 21 is to require notification of the NRC of defects and noncompliances that could create a "substantial safety hazard," which is defined in Section 21.3 as a "major reduction in the degree of protection" provided by a facility. Since LSS and NRS components, by definition, do not provide a significant degree of protection to the public health and safety, a defect or noncompliance involving such components could not lead to a "major reduction" in protection. Therefore, the NRC could decide that Part 21 does not apply to LSS and NRS components because their failure could not create a substantial safety hazard.

ATTACHMENT 3

REVISED RESPONSES TO PRELIMINARY NRC QUESTIONS

REVISED RESPONSES TO PRELIMINARY LIST OF NRC QUESTIONS

Probabilistic Risk Assessment

1. Explain why STPNOC is proposing that both low safety significant (LSS) and non-risk significant (NRS) structures, systems and components (SSCs) be removed from scope and not, for example, only NRS. If not all LSS and NRS SSCs are to be removed from scope, explain how some will be selected and others not.

Response:

As explained in Section 5.2.4.1 of Attachment 1, NRS components by definition are not modeled in the probabilistic risk assessment (PRA), and therefore no risk increases would result from changes in special treatment for NRS components. LSS components may be modeled in the PRA, however, little or no risk increases would result from changes in special treatment for these components. In particular, STP performed a sensitivity study to determine the cumulative impact on core damage frequency (CDF) and large early release frequency (LERF) from postulating a factor of 10 increase in the failure rates of all LSS components modeled in the PRA. As a result of this analysis, STP determined that the annual average CDF and LERF would increase by 2.451 E-7 and 1.69 E-9, respectively, as a result of the increased failure rates for all LSS components categorized to date. The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF).

It is our opinion that in a risk informed regulatory environment these impacts are acceptable, and regulatory oversight can be reduced and/or eliminated for LSS and NRS components without impact to public health and safety. Although LSS and NRS components would be removed from the scope of the special treatment requirements, LSS and NRS components would still be programmatically controlled by station programs and procedures similar to the manner in which balance-of-plant equipment is currently controlled. As described in Attachment 5 and Section 3.3.7 of Attachment 1, these commercial controls will be sufficient to provide reasonable assurance that the LSS and NRS components will perform their functions.

2. **Explain why the SSC categorization process approved by the staff solely to determine which SSCs could not be subjected to a reduction in some Quality Assurance (QA) controls, is also applicable to support the much more encompassing change arising from placing SSCs outside the scope of regulations.**

Response:

The approved risk significance evaluation process is sufficient to support removal of SSCs from the scope of regulations. As described in Section 3 of Attachment 1, the categorization process is independent of any particular program (such as the QA program) and instead is based upon an evaluation of the safety significance of system functions and components.

The categorization process is appropriate and acceptable for application in any risk-informed context for the following reasons.

1) Using PRA and deterministic insights, STP's categorization process establishes the safety/risk importance of a component based upon its failure probability/impact on the safety functions supported by the component.
2) A review of the methodology used in this process, and described in Section 3 of Attachment 1, shows that the process is not affected by the type of application that utilizes the results.
3) This process is analogous to the safety related classification process where a component is classified independent of which regulations it may or may not fall into the scope of after it is classified.

Under the deterministic process in NRC's regulations, station programs addressing special treatment requirements are applicable to components defined as "safety-related" (or important to safety). The risk significance evaluation is a more appropriate tool than the current deterministic process for defining safety-significant components. Based on the items above and by assessing likelihood of occurrence and associated performance information, the risk significance evaluation process is a more robust and rigorous method for establishing the scope to which special treatment requirements are appropriate and for which they will provide measurable benefits relative to nuclear safety. SSCs that are not risk significant do not warrant any special treatment (QA or otherwise).

3. **During the Graded QA (GQA) application review, the staff determined that the LSS SSCs would be subject to QA controls appropriate to their low safety significance. Given that for GQA, LSS and NRS are treated identically, the staff did not review the differentiation between LSS and NRS beyond confirming that NRS was at least LSS. Explain how SSCs placed in the NRS are so much less significant than SSCs placed in LSS.**

Response:

The first distinction to be noted is that LSS components may be included in the scope of the PRA whereas NRS components are not. Some LSS components could have quantifiable impacts on the PRA based on assumptions of their reliability or availability even though those impacts could be exceedingly small. Depending upon the magnitude of the assumed change in reliability or availability, the aggregate effects for quantifiable LSS items would range from no impact to linear impact effects proportional to the magnitude of the assumed change. NRS components do not readily lend themselves to quantification since they are not within the scope of the PRA. Therefore, increased failure rates in NRS components do not translate to risk significance or public health effects in and of themselves.

Another distinction between LSS and NRS components is the response to the key deterministic questions contained in the risk significance evaluation process. The NRS components have responses to the key deterministic questions ranging from zero (i.e., negative or no impact) to minor impacts (i.e., very little impact, if any, or little impact under restrictive or specific assumed initial conditions). Components are classified as NRS based on having composite scores from the key deterministic questions less than 20 (i.e., out of a possible maximum score of 100). It should be noted that the deterministic questions associated with risk determination are weighted such that a clear distinction exists between components classified as LSS and NRS. Thus, the risk significance evaluation process provides not only objective distinctions between LSS and NRS components but is biased to classify components to higher risk significance for key areas of concern. Components not reaching a composite score of 20 are not risk significant and merit no regulatory oversight.

Even though NRS components have less risk significance than LSS components, it is appropriate to treat LSS and NRS components similarly with respect to special treatment requirements. As discussed in Section 5.2.4.1 of Attachment 1, changes in special treatment requirements for LSS components are expected to have little or no impact on risk. Therefore, the effects of changes in special treatment requirements for LSS components are similar to the effects of changes in special treatment requirements for NRS components, and from a risk-informed perspective there is no reason to have a different special treatment program for each of these types of components.

4. **A claim that there will be no decrease in equipment availability following a reduction in QA controls, environmental qualification, seismic qualification and various other special treatment requirements, will almost certainly not be demonstrable. Assuming that some impact on margins or availability is possible, explain how the potential aggregate risk impact of removing NRS (and possibly LSS) SSCs from the scope of regulations will be determined.**

Response:

NRS components are not sufficiently important to warrant modeling in the PRA. Therefore, changes in special treatment for NRS components would have no impact on risk. Similarly, most LSS components (90%) are not sufficiently important to warrant modeling in the PRA. Therefore, changes in special treatment for those LSS components would have no impact on risk. With respect to the remaining LSS components, STP has performed sensitivity analyses to bound the potential aggregate risk impact of changes in special treatment.

As stated in Section 5.2.4.1 of Attachment 1, STP first performed sensitivity studies in which failure rates were simultaneously increased for MSS and LSS components of a similar type (i.e., check valves) within the scope of the PRA. These studies evaluated the impact of increasing the failure rates of the check valves by as much as an order of magnitude. Based upon these studies, STP determined that increases in the failure rate by as much as an order of magnitude had little, or no, impact on the final component risk categorization.

Additionally, as described in Section 5.2.4.1, STP performed a sensitivity study to determine the impact on CDF and LERF from postulating a factor of 10 increase in the failure rates of <u>all</u> LSS components. The cumulative impact to the annual average CDF and LERF of the increased failure rates for all LSS components categorized to date is shown below:

	Current Average (events/reactor year)	**Sensitivity Study λ_{LSS}*10 (events/reactor year)**	**Increase**	**% Increase**
CDF	**9.0781E-6**	**9.3232E-6**	**2.4510E-7**	**2.7%**
LERF	**1.3742E-7**	**1.3911E-7**	**1.6900E-9**	**1.2%**

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF).

Based upon the above, STP concludes that, even if it is unrealistically assumed that the change in the special treatment requirements were to result in significant degradation of LSS components, there would be little or no increase in risk.

Finally, if there is an effect on equipment availability from reduced or eliminated special treatment requirements, the effect will be seen based on future PRA performance data updates and the periodic performance evaluation and feedback process. The aggregate risk impact for LSS and NRS components will be assessed quantitatively using the following:

1) Station performance indicators, and
2) Periodic updates of the PRA with respect to performance data,

Additionally, qualitative assessments will be performed using the Corrective Action Program. Equipment deficiencies and nonconformances will be identified on all components regardless of their risk significance. LSS and NRS components will be reviewed, and if necessary, adjustments made, in special treatment as part of the periodic feedback and reassessment.

5. Explain why, if current Inservice Testing (IST) policy is only to allow increased inspection intervals for LSS, your proposal to remove them from regulatory scope (e.g., allowing no testing) is acceptable.

Response:

STP requests that LSS and NRS components be exempted from the ASME IST program. The basis for this request is that compliance with ASME Code requirements for LSS and NRS components is not necessary to provide reasonable assurance that these components are capable of performing their intended function(s). Reasonable assurance for LSS and NRS components can be achieved through other programs currently in effect. For example, the Maintenance Rule program will monitor on a plant/system/train level for Maintenance Rule Functional Failures (MRFFs) whose failures result in loss of an HSS/MSS function, which provides a mechanism for regulatory oversight for equipment performance. Also, the Corrective Action Program is effective in identifying equipment nonconformances and deficiencies, regardless of their risk significance. Finally, commercial testing requirements will be applied as appropriate for currently installed or replacement components in order to provide reasonable assurance of component functional capability.

Further support for STP's exemption request is provided by STP's PRA sensitivity analysis, in which all LSS component failure rates were increased by a factor of 10. This factor bounds any reasonable estimate of the increase in failure rates due to STP's exemption request. The results of this study identify only a small increase (i.e., about 2%) in CDF and LERF. This increase is well within the limits identified in Regulatory Guide 1.174 and demonstrates that compliance with ASME Code IST requirements for LSS and NRS components is not necessary to provide adequate protection of public health and safety.

It is not STP's understanding that NRC has a policy of only allowing increased intervals for LSS components and not to allow removal of such components from the scope of the IST program. For example, NRC Regulatory Guide 1.175, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing*, Section 2.1.1, explicitly states that licensees make seek exemptions under 10 CFR 50.12 to permit the use of risk-informed IST. Furthermore, Section 2.1.1 also states that "alternatives to the acceptable RI-IST approaches presented in this guide may be proposed by licensees so long as supporting information is provided that demonstrates that the key principles discussed in Chapter 2 of this guide are maintained." As shown below, STP's exemption request does address the key principles in Regulatory Guide 1.175:

Key Principle in RG 1.175	**Relevant Section in STP Exemption Request (Attach. 1)**
1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.	5.2.1
2. The proposed change is consistent with the defense-in-depth philosophy.	5.2.2
3. The proposed change maintains sufficient safety margins.	5.2.3
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.	5.2.4
5. The impact of the proposed change should be monitored using performance measurement strategies.	5.2.5

Therefore, STP believes that this exemption is consistent with the Commission's current policy on risk-informed relief requests. Additionally, this exemption request is also consistent with Option 2 in SECY 98-300, and the Advanced Notice of Proposed Rulemaking (ANPR), *Risk-Informing Special Treatment Requirements* (65 Fed. Reg. 11488 (March 3, 2000)), both of which propose to remove low safety significant components from the scope of the ASME Code and 10 CFR 50.55a (which include the requirements for IST).

STP has concluded that the IST program should be within the scope of this exemption request. The IST program is a special treatment process, and should be treated similarly to other special treatment requirements. In that regard, the risk significance evaluation process can be used to establish the importance of components within the IST program, and components with low safety significance do not need to be subject to the IST program to provide adequate protection of public health and safety. Thus, STP has included IST within the scope of this exemption request.

6. As a pilot, does STPNOC plan to identify regulatory changes that would allow the changes you propose without requiring exemption requests?

Response:

The regulations that STP believes to be affected are tabulated in Attachment 2 of the submittal. Recommending specific changes to these regulations is beyond the scope of this exemption request. STP would expect to provide that input through industry efforts such as the NEI efforts to risk inform 10 CFR Part 50. In addition, STP would be pleased to discuss specific regulatory changes and provide input identifying changes as a parallel effort.

Quality Assurance

1. Explain why STPNOC does not propose using the existing Regulatory Guides for developing a risk-informed QA program.

Response:

STP has already applied the provisions of Regulatory Guide 1.176 in development of its original GQA submittal. However, the approved GQA program only allows STP to apply a reduced level of quality assurance to MSS, LSS and NRS components, commensurate with their safety/risk significance. In practice, a fully risk-informed set of controls defined by a risk informed program cannot be implemented due to the special treatment requirements imposed by 10 CFR Parts 21, 50, and 100 for SSCs classified as safety-related, important to safety, or otherwise within the scope of the regulations. Given the lack of safety/risk importance of LSS and NRS components in protecting the health and safety of the public as determined by the approved risk-informed categorization process, an exemption is appropriate to exclude LSS and NRS components from the scope of the special treatment requirements in these regulations. Such a change is beyond the scope of the existing Regulatory Guides.

2. **STPNOC should maintain the GQA program commitment for the feedback loop to make adjustments, when necessary, in the application of the QA controls for LSS/NRS equipment based on plant/industry performance, and plant modifications.**

Response:

As described in Sections 5.2.5 and 7.2.4 of Attachment 1, STP will maintain a feedback mechanism in place to periodically evaluate the system and equipment performance results, and to assess the impact of the changes that have been made. This assessment will include HSS and MSS non-safety-related components to determine what additional controls are necessary to ensure equipment performance, availability, and reliability. Plant/system/train performance levels will be monitored, as applicable, in accordance with Maintenance Rule requirements to ensure that changes to the controls for LSS and NRS components will not have an adverse effect on system performance. Also, the assessment will evaluate if reductions in quality controls or other requirements have resulted in a decrease in component reliability, and if it would be prudent to enhance the controls applicable to LSS and NRS components.

3. **Page 6 of the March 1, 1999, letter describes other programs (such as motor operated valves (MOVs), air operated valves (AOVs), molded case circuit breakers that are not listed in Attachment 1, where LSS and NRS equipment will be treated differently. STPNOC should explicitly identify all technical areas where LSS and NRS equipment will no longer be within scope of the current program controls.**

Response:

The technical areas where LSS and NRS equipment will no longer be within the scope of the current program controls addressed in NRC regulations are covered in Section 4.1 of Attachment 1. STP does not believe approval of the proposed exemption is required for scope changes to other programs (such as MOV and AOV programs) that are not subject to separate NRC regulations. Therefore, the exemption request does not identify all of the technical areas where LSS and NRS equipment will no longer be within scope of programs not governed by existing regulations.

4. **STPNOC should provide amplifying information on how commercial vendors, not necessarily the original equipment manufacturer (OEM), will provide a suitable engineering analysis to demonstrate the LSS/NRS equipment conforms to the STPNOC procurement specifications which include typical nuclear requirements (i.e., seismic loading, harsh environment profiles).**

Response:

LSS and NRS components will be available to function in the environments that have been identified for them since the procurement process will contain controls to provide reasonable assurance that the components will be able to withstand such environments. If the component has critical attributes, STP procurement processes will provide an appropriate level of assurance that these attributes are satisfied.

As discussed in more detail in Section 3.3.7 of Attachment 1 and the response to RAI 13(b) in Attachment 4, LSS and NRS components will be procured using normal commercial practices. Before procuring such components, STP will perform an engineering evaluation to compare the design functional requirements of the component to be procured with the specifications provided by the vendor for commercial grade items (or will provide the vendor with procurement specifications that address the critical characteristics). This evaluation will provide reasonable assurance that the specifications for the commercial grade item satisfy the requisite form, fit, and function, any critical attributes that the component may have, and any seismic or environmental conditions that the component may be exposed to. Following this evaluation, a purchase order will be issued to the vendor, which will identify the commercial grade item to be procured. Upon receipt of the component, an inspection will be performed to ensure that the component received is the component identified in the purchase order.

These normal commercial practices will provide an appropriate and acceptable level of assurance that LSS and NRS components will be able to perform their intended functions. As discussed in Section 5.2.4.1 of Attachment 1, data collected by STP demonstrate that, for most component types, the calculated failure rates for safety-related components are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data. This data indicates that components procured without special treatment are, in general, able to meet their design and procurement specification requirements. Therefore, LSS and NRS components will be capable and available to function under the range of environmental conditions (similar to other components that are purchased using commercial and industrial practices).

Furthermore, as discussed in Section 5.2.4.1 of Attachment 1, even if it is unrealistically assumed that the use of commercial and industrial practices would result in substantial increases in failure rates, STP has performed bounding sensitivity analyses which demonstrate that there would be little, if any, impact on the safety/risk categorization of the components. Therefore, STP believes that engineering analyses, qualification testing, or other specialized efforts to provide empirical evidence or other justifications of an SSC's ability to function in adverse environments is overly burdensome and not necessary for LSS and NRS components (i.e., not commensurate with their safety significance).

5. **STPNOC should re-consider waiving the entire scope of the QA program for LSS/NRS equipment. For example, why are the nonconformance/corrective action, monitoring programs, and design controls eliminated in their entirety? Why is the basic QA program no longer acceptable for LSS/NRS equipment?**

Response:

STP is not waiving the entire scope of the QA program for LSS and NRS equipment, and has retained the nonconformance/corrective action, monitoring programs, and design controls, as discussed in Sections 4.1.1 and 5.2.5 of Attachment 1.

As discussed in the response to Question 1 above, NRC's regulations do not allow STP to reduce QA controls below the level provided in the basic QA program. Some of the basic QA controls are unduly restrictive, given the lack of safety significance of the LSS and NRS components (e.g., the requirement that suppliers have a QA program that satisfies Appendix B to Part 50). STP desires to reduce quality assurance for LSS and NRS components in a manner that is commensurate with the lack of safety significance of the LSS and NRS components. Therefore, STP is requesting an exemption from these regulations.

6. **STPNOC should provide amplifying justification on the usefulness of normal plant operational feedback mechanisms to demonstrate the functionality of LSS/NRS equipment with respect to special treatment provisions like equipment qualification and seismic qualification provisions.**

Response:

In order to monitor the impact of the changes in the special treatment requirements, STP will use the monitoring and feedback programs established as part of its GQA program. The NRC approved these programs in Sections 3.3 and 3.5.5 of the Safety Evaluation for the GQA program. Since these programs cover the SSCs that are the subject of this exemption request, the existing programs can be used for monitoring of both the GQA program and the changes in the special treatment requirements. This is covered in Section 5.2.5 of Attachment 1.

As recognized in Regulatory Guide 1.174, plant performance under actual design conditions (e.g., seismic events or harsh environments) may not be readily measurable, and in such cases the licensee should use whatever information most closely approximates actual performance data. STP believes that its monitoring and feedback programs achieve this goal. Specifically, STP's programs will account for the results of tests on the component or its associated system; STP will trend nonconformances involving LSS and NRS components; STP will use procurement evaluations to provide reasonable assurance that the components can perform their functions under applicable seismic and environmental conditions; and STP will perform post-installation tests of replacement LSS and NRS components.

7. **Page 20 and 21 of the March 1, 1999, letter discuss a commitment for enhanced treatment of non-safety related high safety significant (HSS) and medium safety significant (MSS) equipment with respect to technical areas. How will this commitment be integrated into a licensing basis document, and what specific technical enhancements will be considered? STPNOC should confirm that the GQA provisions for treatment of non-safety related HSS and MSS equipment remains intact.**

Response:

This exemption request does not affect the GQA provisions for non-safety-related HSS and MSS components, and such provisions will remain in effect. Sections 3.3.6 and 5.2.4.2 of Attachment 1 discuss STP's provision for enhanced treatment of non-safety MSS and HSS components. The commitment for enhanced treatment of non-safety-related high safety significant (HSS) and medium safety significant (MSS) equipment will be integrated into appropriate station documents. Additionally, as discussed in Attachment 7 and the response to the RAI 9 in Attachment 4, STP will include within the UFSAR a description of the process for evaluating the need for enhanced treatment of non-safety-related HSS and MSS components.

8. **Page 24 and 25 of the March 1, 1999, letter provide a high level discussion as to why the composite set of changes will ensure adequate protection of public health and safety. The basis for this conclusion should be augmented to justify how the whole set of LSS and NRS could be found so unimportant.**

Response:

By definition LSS and NRS components generally support ancillary functions (e.g., vents, drains, and local indications) for safety-related systems and do not provide, support, or affect the primary safety functions of the systems. Given the lack of safety/risk importance of LSS and NRS components to protect the health and safety of the public as determined by the approved risk-informed categorization process, an

exemption is appropriate to exclude LSS and NRS components from the scope of the special treatment requirements in these regulations. This is discussed in Sections 3.0 and 5.2.2 to 5.2.4 of Attachment 1.

Since the existing safety-related LSS and NRS components were designed, procured, manufactured, and installed in accordance with the existing special treatment requirements, these components have inherent design margins to perform their intended functions that will not be adversely affected by this exemption. STP's normal commercial and industrial practices provide an appropriate and acceptable level of assurance that replacements for safety-related LSS and NRS components will continue to have sufficient margin to satisfy their functional requirements. Although there is reasonable assurance that the LSS and NRS components will continue to function, in the unlikely event that they fail, there would be no significant consequences as discussed in Sections 5.2.3 and 5.2.4 of Attachment 1.

Additionally, as described in Section 5.2.4.1, the NRS components and most of the LSS components are not sufficiently important to warrant inclusion in the PRA. For the remaining LSS components identified to date, STP performed a sensitivity study to determine the impact on CDF and LERF from postulating a factor of 10 increase in the failure rates of all PRA-modeled LSS components. The cumulative impact to the annual average CDF and LERF of the increased failure rates for all LSS components categorized to date is shown below:

	Current Average (events/reactor year)	**Sensitivity Study λ_{LSS}*10 (events/reactor year)**	**Increase**	**% Increase**
CDF	**9.0781E-6**	**9.3232E-6**	**2.4510E-7**	**2.7%**
LERF	**1.3742E-7**	**1.3911E-7**	**1.6900E-9**	**1.2%**

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF).

Finally, the requested exemption does not involve a change in any functional requirements or the configuration of SSCs. Because of the low risk significance of the components in the scope of the exemption, STP does not expect the need to change any of the safety analyses in the UFSAR.

9. **STPNOC should first take advantage of current mechanisms to resolve the problems on burdensome special treatment controls, such as using the design control program to re-classify equipment as non-safety related.**

Response:

STP plans to take advantage of current mechanisms to resolve the problems on burdensome special treatment controls as stated in the letter to the NRC from M. A. McBurnett dated April 8, 1999, "Request for Concurrence with Process for Reclassification of Non-Risk Significant Components."

However, even after these mechanisms are utilized, there will still be a large number of safety-related components, important to safety components, and other LSS and NRS components that fall within the scope of the regulations. An exemption from NRC's regulations is needed to risk-inform the special treatment requirements applicable to such components.

Maintenance Rule

1. **With respect to the Maintenance Rule, monitoring LSS and NRS SSCs at the system/train level is acceptable (STPNOC is requesting an exemption from monitoring at the component level). There are no requirements to monitor at the component level even for HSS SSCs. Also, the Regulatory Guide/NUMARC guidance calls for plant level monitoring (scrams, transients) for most LSS/NRS SSCs.**

Response:

Although monitoring at the component level is not a requirement for all components, Regulatory Guide 1.160 states that some component-level monitoring may be necessary. A similar statement is contained in the statement of considerations for the Maintenance Rule. (56 Fed. Reg. 31306, 31309 (July 10, 1991)) Therefore, STP believes an exemption to 10 CFR 50.65 is necessary to clarify that the Maintenance Rule is not applicable to individual NRS and LSS components.

To clarify the STP request, LSS/NRS SSCs will be monitored at the plant/system/train level, as appropriate. Monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. When performance criteria are not satisfied, STP will perform evaluations to determine the cause and corrective action, including, as necessary, corrective action for LSS and NRS components.

General Design Criterion 17

1. **With the utilization of commercial grade electrical components (i.e., components determined to be LSS or NRS that previously met all Class 1E requirements), explain how independence is preserved between redundant safety system (i.e., how will the availability of the minimum required equipment be assured during any design basis event?).**

Response:

STP does not plan to revise the design basis separation criteria. Separation will be maintained between 1E and non-1E equipment as described in the STP UFSAR.

Upon grant of the exemption, STP will not apply the special treatment requirements to currently installed LSS and NRS components. For replacement activities, safety-related electrical equipment will be evaluated to determine the extent that special treatment requirements can be eliminated. Additionally, even though we will apply our commercial practices to procurement of replacement components, we will treat the replacement commercial device as Class IE by performing an engineering evaluation and procuring it with the required design and functional attributes commensurate with an IE classification. Where this cannot be achieved, the component will be procured as a fully qualified, safety related item.

The following justifications are provided to support our position:

Loss of the function of an LSS or NRS electrical component (e.g., failure of a motor operator to provide motive force to its associated valve when demanded) has been determined through the risk categorization process to have little or no impact on protecting the core, mitigating the consequences of an accident, or protecting the public health and safety.

For handswitches, where the associated end device has been risk categorized as MSS and the handswitch has been ranked at a lower risk, STP will confirm and document that credible failure(s) of the handswitch could not prevent the associated end device from performing its MSS function(s) or that cause the loss of other critical required attributes. In the case where this cannot be confirmed, STP will revise the risk categorization of the handswitch to match that of the end device. For additional discussions on handswitches, please refer to the response to Question 10.

When the need arises, STP may replace an existing LSS or NRS Class IE component with a commercial grade component that meets the existing functional design requirements, including environmental parameters. Procuring a commercial component is accomplished by performing an engineering evaluation to provide reasonable assurance that the replacement component satisfies the required form, fit, and function, will not degrade other IE equipment, and is capable of meeting the design functional requirements (including seismic and environmental considerations); however, the component will not

be specifically qualified. If a replacement commercial grade component does not satisfy all of the design functional requirements applicable to a Class IE component, it will be classified as Non IE, and the replacement device will be properly isolated from the Class IE circuitry. The circuit downstream of the required isolation device(s) would be also classified as Non IE and would be routed in a non-IE raceway. Where the above is not feasible, the component will be procured as a fully qualified, safety related item.

Upon grant of the exemption, by purchasing functionally equivalent replacement components from commercial grade vendors, reasonable assurance is provided that the new component will perform is design function and will not challenge the IE distribution system.

The above measures will provide reasonable assurance, commensurate with the risk significance, that LSS and NRS electrical components will be able to satisfactorily perform their design functions and will not electrically degrade other components or challenge the IE distribution system. Therefore, STP intends to continue to identify safety related LSS and NRS electrical components as Class IE, similar to maintaining the safety related classification for LSS and NRS safety related mechanical components.

For cabling, STP intends to maintain current separation requirements. Therefore, non-IE circuit cabling will be properly separated from IE circuits, and will be run in non-IE raceways.

In order for a failure of the LSS and NRS component to have a risk significant impact on the plant, the following must occur at the same time:

- The replacement component must fail in such a way as to affect the upstream electrical circuitry.
- Any fully qualified Class IE isolation device(s) must fail to protect the upstream circuits.
- The above two failures occur during a design basis event.

  STP does not consider this scenario to be credible.

Finally, STP will perform monitoring to ensure that LSS and NRS electrical components will perform in accordance with the assumptions in the PRA. All equipment necessary to mitigate the consequences of initiating events is included in the plant PRA. Changes to the risk significance of components included in the PRA will not result in removal of the equipment from the model. As component replacements are made, changes in equipment failure rates, if they occur, will be identified by the Corrective Action Program or the Maintenance Rule Program (if failure of the LSS or NRS SSC could affect an HSS/MSS function) and the new failure rates incorporated into the PRA model during the cycle updates. Requantification of the model with the changed failure rates may result in a change to the components' risk ranking. In addition, any change in equipment performance will be deterministically evaluated during the periodic reviews by the GQA Working Group. These evaluations may also result in a change to the components' risk ranking.

a) **Present the criteria and their basis (i.e., current UFSAR commitments and/or new commitments) that establish the minimum requirements for preserving the independence between redundant safety systems).**

Response: (SEE RESPONSE ABOVE)

b) **If an isolation device is utilized to isolate (Class 1E HSS or MSS from LSS or NRS), will this isolation device be required to be HSS or MSS? If not, provide justification.**

Response:

STP would expect the device to be classified consistent with the equipment it protects. STP will take actions as necessary to ensure that HSS and MSS Class 1E components are appropriately protected as discussed above (reference section 4.1.2.1). See the response to RAI question 12 for amplifying details.

c) **Provide a description of the extent to which the recommendations of Regulatory Guide 1.75 are followed.**

Response:

As a result of the exemption, STP expects no change in its commitment to the design and functional requirements in Regulatory Guide 1.75. However, STP would not apply the qualification and quality assurance requirements in Regulatory Guide 1.75 to LSS and NRS components.

Environmental Qualification

1. **Since 50.49 only requires components to be qualified to 50.49 requirements if they are exposed to the design basis accident (DBA) environment and must function in order to mitigate the consequences of the DBA, it is not clear which components need exemption. Provide clarification.**

   Response:

   STP has not identified specific components that need exemption from 10 CFR 50.49. Instead, STP requests an exemption from the qualification requirements in 10 CFR 50.49 to the extent that any LSS and NRS may come within the scope of this section. In this regard, the scope of Section 50.49 is deterministic (i.e., it applies to any components that meet the criteria of Section 50.49(b), without regard to the probability that such components will actually be needed to function during design basis conditions). In contrast, under STP's risk-informed categorization process, a component may be classified as LSS or NRS even though it falls within the scope of Section 50.49(b), provided that both PRA-based and deterministic risk insights show that the component has insignificant or minor contribution to or impact on the ability to mitigate design basis events.

   LSS and NRS components will still be procured and maintained in accordance with design and environmental requirements; however, this exemption would enable STP to procure and install these components without formal equipment qualification testing as required by 10CFR50.49. Since those components will be suitable and capable of performing their intended function(s), there would be no risk increase due to this change.

Inservice Testing

1. **Explain why STPNOC did not use the existing Regulatory Guide 1.175 for risk-informed IST?**

   Response:

   Regulatory Guide 1.175 provides for a reduction in the frequency of testing of components that are not risk significant. For the reasons explained in the response to PRA Question 5 above, STP believes that LSS and NRS components do not warrant any IST under the ASME Code. Since Regulatory Guide 1.175 does not address reductions in the scope of the IST program, STP has not used the specific provisions in Regulatory Guide 1.175. However, as discussed in the response to PRA Question 5, STP's exemption request does address the key principles in Regulatory Guide 1.175 and shows that those principles are satisfied by the request.

2. **Discuss why, without monitoring at the component level, common cause failures of LSS components are not important to safety.**

Response:

As discussed in Section 5.2.5 of Attachment 1, the Operating Experience Group (OEG) collects and evaluates problems reported in the STP corrective action program database, which will include problems with individual safety-related LSS and NRS components. The OEG also supports identification of adverse equipment trends, possible common cause or common mode failures, and similarities to station and industry operating experience.

Under the Maintenance Rule, STP will continue to monitor performance on a plant, system, or train level, as appropriate. Monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. When performance criteria are not satisfied, STP will perform evaluations to determine the cause and corrective action, including, as necessary, corrective action for LSS and NRS components.

Therefore, the existing monitoring program approved by the NRC for the GQA program will be sufficient to monitor the effects of the changes in the special treatment requirements for LSS and NRS components and to provide an appropriate level of assurance that SSCs will continue to satisfy their functional requirements.

3. **Discuss how you can assure that LSS and NRS components in safety-related systems purchased without QA documentation (e.g., material certifications) will meet design and procurement specification requirements (e.g., strength, ductility, and corrosion-resistance).**

Response:

As discussed in more detail in Section 3.3.7 of Attachment 1 and the response to RAI 13(b) in Attachment 4, LSS and NRS components will be procured using normal commercial practices. Before procuring such components, STP will perform an engineering evaluation to compare the design functional requirements of the component to be procured with the specifications provided by the vendor for commercial grade items (or will provide the vendor with procurement specifications that address the critical characteristics). This evaluation will provide reasonable assurance that the specifications for the commercial grade item satisfy the requisite form, fit, and function, any critical attributes that the component may have, and any seismic or environmental conditions that the component may be exposed to. Following this evaluation, a purchase order will be issued to the vendor, which will identify the commercial grade item to be procured. Upon receipt of the component, an inspection will be performed to ensure that the component received is the component identified in the purchase order.

These normal commercial practices will provide an appropriate and acceptable level of assurance that LSS and NRS components will be able to perform their intended functions. As discussed in Section 5.2.4.1 of Attachment 1, data collected by STP demonstrate that, for most component types, the calculated failure rates for safety-related components are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data. This data indicates that components procured without special treatment are, in general, able to meet their design and procurement specification requirements. Therefore, LSS and NRS components will be capable and available to function under the range of environmental conditions (similar to other components that are purchased using commercial and industrial practices).

Furthermore, as discussed in Section 5.2.4.1 of Attachment 1, even if it is unrealistically assumed that the use of commercial and industrial practices would result in substantial increases in failure rates, STP has performed bounding sensitivity analyses which demonstrate that there would be little, if any, impact on the safety/risk categorization of the components. Therefore, STP believes that engineering analyses, qualification testing, or other specialized efforts to provide empirical evidence or other justifications of an SSC's ability to function in adverse environments is overly burdensome and not necessary for LSS and NRS components (i.e., not commensurate with their safety significance).

4. Explain why it is acceptable for failures of LSS and NRS components in a safety-related system to occur when those components are not modeled in your probabilistic risk assessment (PRA). Could failures of LSS and NRS components impact the ability of a safety-related system to perform its intended functions?

Response:

As discussed in Section 5.2.4.1 of Attachment 1, all NRS components and most LSS components are not sufficiently significant to warrant modeling in the PRA. Therefore, the exemption to exclude such components from the scope of special treatment requirements will not affect the results of the PRA for STP.

The fact that these components are not modeled in the PRA does not affect the adequacy of the PRA. STP uses a site-specific PRA in the risk-informed process for categorizing SSCs. The NRC has previously approved the quality of the PRA, quality assurance for the PRA, the scope of the PRA, and the application of the PRA results and insights to the risk-informed process for categorizing components, as described in Section 3.2 of the NRC's Safety Evaluation for the GQA program. These same tools will be used for this exemption.

Furthermore, as discussed in more detail in Section 3.2.2.2 of Attachment 1, STP considers deterministic criteria in categorizing the safety significance of a component (whether or not it is modeled in the PRA). Such criteria include whether the component

could directly cause an initiating event, whether it could fail a risk significant system, and whether it mitigates an accident or transient. Such consideration ensures that a safety-related component is not categorized as LSS or NRS, unless it has no or low impact on a safety significant function.

5. **Describe any feedback that will occur when LSS and NRS components are found in a degraded or nonconforming condition. Will the feedback explicitly address each special treatment area being affected by the exemption request (e.g., to re-establish QA or IST requirements as may be appropriate)?**

Response:

As discussed in Section 5.2.5 of Attachment 1, STP's corrective action program will continue to be used to identify and correct nonconforming conditions and adverse trends for LSS and NRS components. Information from the corrective action program will be provided to the monitoring and feedback program to assess the impact of the changes in the special treatment requirements and the need for future adjustments to ensure the continued reliability and availability of the components. Whenever degraded performance is attributed to the reduction or relaxation of special treatment controls, the Working Group will recommend the appropriate remedial action, up to and including the reinstatement of the subject special treatment control(s) and the potential re-categorization of the component's risk significance to a higher level. As such, the corrective action and monitoring and feedback programs are a significant barrier against allowing unacceptable levels of performance degradation to occur.

6. **Quantitatively assess, to the extent practical, the aggregate effect on core damage frequency (CDF) and large early release frequency (LERF) of deleting the various special treatment requirements. What would be the effect on CDF and LERF if all LSS and NRS components were assumed to be unavailable?**

Response:

As discussed in Section 5.2.4.1 of Attachment 1, data collected by STP demonstrates that, for most component types, the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data. Therefore, STP expects that there will be no change in CDF and LERF as a result of removal of LSS and NRS components from the scope of the special treatment requirements.

Furthermore, as also discussed in Section 5.2.4.1 of Attachment 1, all NRS components and most LSS components are not sufficiently significant to warrant modeling in the PRA. Therefore, the exemption to exclude those components from the scope of special treatment requirements will not affect the results of the PRA for STP.

For the remaining LSS components, STP performed a sensitivity study to determine the impact on CDF and LERF from postulating a factor of 10 increase in the failure rates of all PRA-modeled LSS components. The cumulative impact to the annual average CDF and LERF of the increased failure rates for all LSS components categorized to date is shown below:

	Current Average (events/reactor year)	**Sensitivity Study λ_{LSS}*10 (events/reactor year)**	**Increase**	**% Increase**
CDF	**9.0781E-6**	**9.3232E-6**	**2.4510E-7**	**2.7%**
LERF	**1.3742E-7**	**1.3911E-7**	**1.6900E-9**	**1.2%**

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF).

It is not realistic (and contrary to accepted procedures for performing a PRA) to assume that all modeled LSS components fail simultaneously during an accident, merely because such components are subject to normal commercial and industrial practices rather than special treatment requirements. In fact, as discussed above, data collected by STP demonstrates that, for most component types, the calculated failure frequencies for safety-related components are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components. This data indicates that components procured without special treatment are, in general, able to meet their design and functional requirements. Therefore, it would be inappropriate to calculate CDF and LERF assuming that all LSS components are unavailable.

7. **Discuss the basis for deleting IST requirements for LSS components (components with a Fussell-Vessely (FV) less than 0.005 and a risk achievement worth (RAW) less than 2.0) in light of the fact that the ASME component-specific and risk-informed Code cases, currently under development, define IST requirements for these components.**

Response:

STP is requesting an exemption from 10 CFR 50.55a(f) to the extent that it incorporates the IST requirements in the ASME Code (including ASME Code cases). As discussed in the response to PRA Question 5 above, this exemption request is consistent with the NRC's five key principles for risk-informed relief requests. Additionally, this exemption request is also consistent with Option 2 in SECY 98-300, and the Advanced Notice of Proposed Rulemaking (ANPR), *Risk-Informing Special Treatment Requirements* (65 Fed. Reg. 11488 (March 3, 2000)), both of which propose to remove low safety significant components from the scope of the ASME Code and 10 CFR 50.55a (which include the requirements for IST).

The IST program is a special treatment process, and should be treated similarly to other special treatment requirements. In that regard, STP's risk significance evaluation process can be used to establish the importance of components within the IST program, and components with low safety significance do not need to be subject to the IST program to provide adequate protection of public health and safety.

Compliance with ASME Code requirements for LSS components is not necessary to provide reasonable assurance that these components are capable of performing their intended function(s). Reasonable assurance for LSS components can be achieved through other programs currently in effect. For example, the Maintenance Rule program will continue to require monitoring on a plant/system/train level, which provides a mechanism for regulatory oversight for equipment performance. Monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. Also, the Corrective Action Program is effective in identifying equipment nonconformances and deficiencies, regardless of their risk significance. Finally, industry accepted testing requirements may be applied as appropriate for currently installed or replacement components in order to provide reasonable assurance of component functional capability.

Finally, STP performed a PRA sensitivity analysis, in which all PRA-modeled LSS component failure rates were increased by a factor of 10. This factor bounds any reasonable estimate of the increase in failure rates due to STP's exemption request. The results of this study identified only a small increase (i.e., about 2%) in CDF and LERF. This increase is well within the limits identified in Regulatory Guide 1.174 and demonstrates that compliance with ASME Code IST requirements for LSS components is not necessary to provide adequate protection of public health and safety.

8. Describe the deterministic factors considered by the expert panel in categorizing components. How does the licensee's overall categorization process ensure that appropriate level of QA, testing, maintenance, etc., is conducted on components categorized as LSS and NRS?

Response:

The deterministic factors considered by the expert panel in categorizing components were discussed in STP's application for the GQA program and were approved by NRC in its Safety Evaluation for the GQA program. These factors are also discussed in Section 3.2.2.2 of Attachment 1.

This categorization process is based upon the importance of a component to the performance of safety functions. The categorization process is not dependent upon, and does not specify, the level of special treatment requirements for a component.
As discussed in Sections 3.3.7 and 5.2.4 of Attachment 1, the exemption will allow LSS and NRS components to be inspected, tested, and maintained under normal commercial and industrial practices. In addition, replacement components may be designed, manufactured, procured, and installed under STP's normal commercial and industrial

practices. These practices provide an appropriate and acceptable level of assurance that the components will satisfy their functional requirements. Data collected by STP demonstrate that, for most component types, the calculated failure frequencies for safety-related components are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data. This data indicates that components procured without special treatment are, in general, able to meet their design and functional requirements.

Given the minimal safety/risk significance of the components in question, normal commercial and industrial practices are sufficient to provide an appropriate level of assurance that the components will satisfy their functional requirements and have sufficient safety margin. Furthermore, performance of the plant/systems/trains containing LSS and NRS components will be monitored under both the Maintenance Rule program (monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function) and the Corrective Action Program to ensure that these plant/systems/trains maintain satisfactory levels of performance.

Inservice Inspection

1. Explain why STPNOC does not propose using the existing Regulatory Guides for developing a risk-informed Inservice Inspection (ISI) Program (for piping).

Response:

The applicable document is Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection*. Regulatory Guide 1.178 provides for a reduction in the frequency of inspection of components that are not risk significant, but does not provide for taking LSS and NRS components out of the scope of the ISI program. STP believes that LSS and NRS components do not warrant any ISI under the ASME Code. Since Regulatory Guide 1.178 does not address reductions in the scope of the ISI program, STP has not relied on the specific provisions in Regulatory Guide 1.178 to support this Exemption Request.

However, Regulatory Guide 1.178, Section 2.1.1, explicitly states that "alternatives to the acceptable RI-ISI approaches presented in this guide may be proposed by licensees so long as supporting information is provided that demonstrates that the key principles discussed in this guide are maintained." As shown below, STP's exemption request does address the key principles in Regulatory Guide 1.178:

Key Principle in RG 1.178	Relevant Section in STP Exemption Request (Attach. 1)
1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.	5.2.1
2. The proposed change is consistent with the defense-in-depth philosophy.	5.2.2
3. The proposed change maintains sufficient safety margins.	5.2.3
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.	5.2.4
5. The impact of the proposed change should be monitored using performance measurement strategies.	5.2.5

Additionally, this exemption request is also consistent with Option 2 in SECY 98-300, and the Advanced Notice of Proposed Rulemaking (ANPR), *Risk-Informing Special Treatment Requirements* (65 Fed. Reg. 11488 (March 3, 2000)), both of which propose to remove low safety significant components from the scope of the ASME Code and 10 CFR 50.55a (which include the requirements for ISI).

The ISI program is a special treatment process, and should be treated similarly to other special treatment requirements. In that regard, the risk significance evaluation process can be used to establish the importance of components within the ISI program, and components with low safety significance do not need to be subject to the ISI program to provide adequate protection of public health and safety.

Compliance with ASME Code ISI requirements for LSS and NRS components is not necessary to provide reasonable assurance that these components are capable of performing their intended function(s). Reasonable assurance for LSS and NRS components can be achieved through other programs currently in effect. For example, the Maintenance Rule requires monitoring on a plant/system/train level, which provides a mechanism for regulatory oversight for equipment performance. Also, the Corrective Action Program is effective in identifying equipment nonconformances and deficiencies, regardless of their risk significance. Finally, industry accepted inspection requirements may be applied as appropriate for currently installed or replacement components in order to provide reasonable assurance of component functional capability.

Finally, STP performed a PRA sensitivity analysis, in which all PRA-modeled LSS component failure rates were increased by a factor of 10. This factor bounds any reasonable estimate of the increase in failure rates due to STP's exemption request. The results of this study identify only a small increase (i.e., about 2%) in CDF and LERF. This increase is well within the limits identified in Regulatory Guide 1.174 and demonstrates that compliance with ASME Code ISI requirements for LSS and NRS components is not necessary to provide adequate protection of public health and safety.

2. How does the change in risk category impact ASME classification for the purpose of ISI?

Response:

STP's categorization process identifies a safety/risk classification for components (i.e., HSS, MSS, LSS, or NRS). However, this categorization process does not affect existing classifications (e.g., safety-related, ASME, and seismic). Therefore, components will retain their existing ASME classification after application of the risk-informed categorization process. However, as discussed in Section 4.1.2.2 of Attachment 1, ASME components categorized as LSS and NRS could be replaced with non-Code components subject to ensuring that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements.

Regardless of whether the LSS and NRS component in question is the original Code item or a replacement non-Code item, STP will not include such component within the scope of the ISI program for the reasons stated in the response to Question 1.

Seismic Qualification

1. **A seismic event is a common cause failure initiator. Real earthquake experience data had indicated that some equipment is vulnerable to earthquake motions. Reliance on design and purchase specifications may not be sufficient to demonstrate the functions of some mechanical and electrical equipment. Discuss what criteria and procedures you plan to use to assure that LSS and NRS components will function during and after an earthquake without performing equipment seismic qualification in accordance with the current provisions in the UFSAR.**

   Response:

   As explained in Sections 3.3.7 and 4.1.2.3 of Attachment 1, the procurement process will include appropriate controls that provide reasonable assurance that LSS and NRS components are able to function during and/or following a seismic event, to the extent required by their critical attributes. Normal commercial and industrial practices will be utilized to provide reasonable assurance that such components satisfy these requirements. Experience with commercial and industrial components indicates that there is reasonable assurance that the components will survive those seismic events for which they are designed. Nevertheless, even if some LSS and NRS components were to fail during a design basis earthquake, there would be no significant impact on a safety function, as discussed below.

   As stated in Sections 5.2.3 and 5.2.4 of Attachment 1, there is reasonable assurance that the LSS and NRS components will continue to function in the event of an earthquake. In the unlikely event that they fail, there would be no significant consequences.

   - Components categorized as NRS are not sufficiently significant to warrant modeling in the PRA. The safety-related NRS components generally include local indicators, vents, drains, and root valves sized 1" and under. Even if it is assumed that an NRS component were to fail, the results of the PRA would be unaffected, and there would no significant impact on safety.

   - Most of the components categorized as LSS are not sufficiently significant to warrant modeling in the PRA. Even if it is assumed that such an LSS component were to fail, the results of the PRA would be unaffected, and there would no significant impact on safety.

   - For the remaining LSS components, STP performed a sensitivity study to determine the cumulative impact on CDF and LERF from postulating a factor of 10 increase in the failure rates of <u>all</u> LSS components. As a result of this analysis, STP determined that the annual average CDF and LERF would increase 2.451 E-7 and 1.69 E-9, respectively, as a result of the increased failure rates for all LSS components categorized to date. The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF).

ATTACHMENT 4

RESPONSES TO THE

REQUEST FOR ADDITIONAL INFORMATION

ON

SOUTH TEXAS PROJECT

REQUEST FOR EXEMPTIONS TO EXCLUDE CERTAIN COMPONENTS

FROM THE SCOPE OF SPECIAL TREATMENT REQUIREMENTS

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1. *In discussions with the licensee during the August 31, and September 1, 1999, public meetings, it was unclear what components were included (or excluded) from the exemption request. For example, the licensee stated that piping was not included in the exemption request - only "tagged" components were included in the scope of the proposed exemption. The staff requests that the licensee provide a list of the groups or types of components included in the exemption request (not individual components).*

RESPONSE:

The risk significance of any component in the plant can be determined using the established risk categorization process. This process, which is detailed in STP's procedures and elsewhere in this response, consists of the following major elements:

1. **For the system in which the component resides, the identification of system functions and a determination of the risk significance of each function.**
2. **Identification of all system functions that the subject component supports.**
3. **Identification of the component's risk in the PRA, where applicable.**
4. **A determination of the risk significance of the component based on steps 1, 2 and 3 above along with additional insight regarding the impact and probability of the component's failure on the system functions supported.**
5. **Identification of component critical attributes for safety related LSS/MSS and non-safety related MSS/HSS components.**
6. **Review and approval by an Expert Panel.**
7. **Periodic performance feedback to ensure the appropriateness of the risk categorization.**

Until a component is categorized in this manner, it is not in the scope of this exemption request and, if it is safety related, remains conservatively under the Full QA program (i.e., special treatment requirements are still applied).

Most components in the plant are included in the Master Equipment Database (MED) and are identified by a unique tag number assigned in accordance with the Total Plant Numbering System (TPNS). The following table provides a representative list of MED components. Components that are not in the MED include, but are not limited to:

- **structures,**
- **piping,**
- **cables,**
- **relays,**
- **fuses,**
- **terminal blocks**
- **125 VAC lighting, and**
- **skid-mounted components.**

To date, STP has chosen to apply the risk categorization process to MED components. This was done in order to maintain the number of components in any given system at a manageable level and also because most maintenance and procurement activities are performed on MED components.

STP considers that additional benefits can be achieved from risk categorizing non-MED components. As long as the above process is followed, the risk significance of a non-MED component can be determined with the same robustness and controls as has been done for MED components. Due to the low level of plant activity on these types of components, STP may perform the risk categorization on a case-by-case basis as the need arises.

For example, the Working Group may be asked to determine the risk significance of a portion of system piping in order to support a maintenance activity. Assuming the system's MED components have already been risk ranked, the Working Group would convene and reach consensus on the system functions that are supported by the piping (typically, pressure boundary). The subject piping would then be ranked, factoring in the risk that was previously assigned to the system's pressure boundary function. Critical attributes would then be established. This risk significance determination and the supporting justification would be provided to the Expert Panel for approval, after which it would be disseminated to the plant staff.

Therefore, this exemption request also applies to non-MED components that have been categorized as LSS or NRS. It should be emphasized that regulatory requirements not in the scope of this exemption request would continue to be applied for LSS and NRS non-MED components. In addition, as with MED components, design functional requirements would still apply and could not be changed without being first evaluated under the design change process.

MASTER EQUIPMENT DATABASE COMPONENTS

TYPE	DESCRIPTION
ACCUMU	ACCUMULATOR
ACCUMULATOR	RECEIVER
ACCUMULATOR	TANK
AIRDRYER	DRYER, AIR
BATTRY	BATTERY
BATTRY	CHARGER, BATTERY
BLOWER	BLOWER
BLOWER	COMPRESSOR
BLOWER	FAN
BLOWER	FAN, EXHAUST
BLOWER	FAN, SUPPLY
BLOWER	TURBOCHARGER
BLOWER	UNIT, AIR HANDLING
BLOWER	VALVE
CHILER	CHILLER
CHILER	COOLER, SAMPLE
CHILER	SWITCH
CKTBRK	BOX, JUNCTION
CKTBRK	CUBICLE
CKTBRK	PANEL, NOISE SUPPRESSOR
CKTBRK	STARTER, AC
CKTBRK	SWITCH, DISCONNECT
CKTBRK	SWITCH, HAND
COMPER	CABINET, COMPUTER
COMPER	CRT
COMPER	KEYBOARD
CRANE	HOIST
CRDRVE	MECHANISM, CONTROL ROD DRIVE
DAMPER	ACTUATOR
DAMPER	DAMPER
DAMPER	DAMPER, AOV, DIAPHRAGM
DAMPER	DAMPER, AOV, PISTON
DAMPER	DAMPER, BACKDRAFT
DAMPER	DAMPER, BUBBLE
DAMPER	DAMPER, FIRE
DAMPER	DAMPER, ISOLATION
DAMPER	DAMPER, MANUAL
DAMPER	DAMPER, OPPOSED
DAMPER	DAMPER, PARALLEL
DAMPER	DAMPER, SINGLE
DAMPER	DAMPER, TORNADO
DAMPER	VALVE
ELCFUN	BOX, DISCONNECT
ELCFUN	BOX, JUNCTION

TYPE	DESCRIPTION
ELCFUN	CABINET, INSTRUMENT, LOCAL
ELCFUN	CABINET, LOCAL INST
ELCFUN	CABINET, TERMINATION
ELCFUN	FILTER, VENT
ELCFUN	PANEL
ELCFUN	PANEL, CONTROL
ELCFUN	POWER, SCR
ELCFUN	SNUBBER, BALL
ELCFUN	SNUBBER, RACK
ELCFUN	TERM BOARD
ELCFUN	TERM BOX
ELCFUN	TRAY, CABLE
ENGINE	GENERATOR, DIESEL
ENGINE	NOZZLE, INJECTOR
FILTER	DEMISTER
FILTER	FILTER
FILTER	FILTER, CARBON
FILTER	FILTER, LUBRICATION
FILTER	FILTER, OIL
FILTER	FILTER, PARTICLE
FILTER	FILTER, REPLACEABLE
FILTER	FILTER, SILENCER
FILTER	FILTER, SPECIAL
FILTER	FILTER, VENT
FILTER	PLENUM
FILTER	SKIMMER
FILTER	STRAINER
FILTER	STRAINER, DUPLEX
FILTER	STRAINER, MAGNETIC
FILTER	STRAINER, SELF CLEANING
FILTER	VALVE, SOV, NON-PROCESS
GENERA	GENERATOR, EMERGENCY
HEATER	CABINET, INSTRUMENT, LOCAL
HEATER	HEATER, IMMERSION
HEATER	HEATER, PRESSURIZER
HEATER	HEATER, SPACE
HEATER	HEATER, UNIT
HEATER	HEATER, VENT
HTEXCH	COIL
HTEXCH	CONDENSER
HTEXCH	COOLER
HTEXCH	EXCHANGER, HEAT
HTEXCH	HEAT EXCHANGER
HTEXCH	REHEATER
IBISSW	ELEMENT, MOISTURE

TYPE	DESCRIPTION
IBISSW	ELEMENT, POSITION
IBISSW	ELEMENT, TEMPERATURE
IBISSW	SWITCH
IBISSW	SWITCH, ANALYSIS
IBISSW	SWITCH, CONDUCTIVITY
IBISSW	SWITCH, CURRENT
IBISSW	SWITCH, DENSITY
IBISSW	SWITCH, FLOW
IBISSW	SWITCH, LEVEL
IBISSW	SWITCH, LEVEL DIFF
IBISSW	SWITCH, MOISTURE
IBISSW	SWITCH, POSITION
IBISSW	SWITCH, PRESSURE
IBISSW	SWITCH, PRESSURE DIFF
IBISSW	SWITCH, TEMPERATURE
ICLOOP	ALARM, PRESSURE
ICLOOP	LOOP, LEVEL
ICNTRL	CABINET, LOCAL INSTRUMENT
ICNTRL	CONTROLLER
ICNTRL	CONTROLLER, FLOW
ICNTRL	CONTROLLER, HAND
ICNTRL	CONTROLLER, LEVEL
ICNTRL	CONTROLLER, PRESSURE
ICNTRL	CONTROLLER, PRESSURE DIFF
ICNTRL	CONTROLLER, TEMPERATURE
ICNTRL	CONVERTER, CURRENT
ICNTRL	CONVERTER, TIME
ICNTRL	LOOP
ICNTRL	LOOP, FLOW
ICNTRL	LOOP, LEVEL
ICNTRL	LOOP, PRESSURE
ICNTRL	LOOP, SPEED
ICNTRL	LOOP, TEMPERATURE
ICNTRL	ORIFICE, FLOW
ICNTRL	RACK, RELAY
ICNTRL	RECORDER, CONTROL STATION
ICNTRL	RECORDER, TEMPERATURE
ICNTRL	VALVE, FLOW
ICNTRL	VALVE, SOV, NON-PROCESS
INDREC	CONTROLLER, FLOW
INDREC	GAUGE, FLOW
INDREC	GAUGE, LEVEL
INDREC	INDICATOR
INDREC	INDICATOR, CURRENT
INDREC	INDICATOR, FLOW
INDREC	INDICATOR, LEVEL
INDREC	INDICATOR, MISC
INDREC	INDICATOR, MOISTURE

TYPE	DESCRIPTION
INDREC	INDICATOR, NUCLEAR
INDREC	INDICATOR, PRESSURE
INDREC	INDICATOR, PRESSURE DIFF
INDREC	INDICATOR, RADIATION
INDREC	INDICATOR, SPEED
INDREC	INDICATOR, TEMPERATURE
INDREC	INDICATOR, TIME
INDREC	RECORDER
INDREC	RECORDER, ANALYSIS
INDREC	RECORDER, CONTROL STATION
INDREC	RECORDER, FLOW
INDREC	RECORDER, LEVEL
INDREC	RECORDER, LEVEL DIFF
INDREC	RECORDER, PRESSURE
INDREC	RECORDER, PRESSURE DIFF
INDREC	RECORDER, RADIATION
INDREC	RECORDER, TEMPERATURE
INDREC	RECORDER, TURBIDITY
INDREC	STATION, MEASURING
INDREC	SWITCH, FLOW
INDREC	SWITCH, LEVEL
INDREC	SWITCH, POSITION
INDREC	SWITCH, PRESSURE DIFF
INDREC	SWITCH, TEMPERATURE
INTCPM	AHU MICROPROCESSOR
INTCPM	AMPLIFIER
INTCPM	CONVERTER
INTCPM	COUNTER, FLOW
IPWSUP	POWER SUPPLY
IPWSUP	POWER SUPPLY, CONVERTER
IPWSUP	POWER, SUPPLY
IXMITR	BOX, JUNCTION
IXMITR	CABINET, LOCAL INSTRUMENT
IXMITR	ELEMENT
IXMITR	ELEMENT, ANALYSIS
IXMITR	ELEMENT, BORON
IXMITR	ELEMENT, CONDUCTIVITY
IXMITR	ELEMENT, CURRENT
IXMITR	ELEMENT, DENSITY
IXMITR	ELEMENT, FLOW
IXMITR	ELEMENT, LEVEL
IXMITR	ELEMENT, MOISTURE
IXMITR	ELEMENT, POSITION
IXMITR	ELEMENT, RADIATION
IXMITR	ELEMENT, SPEED
IXMITR	ELEMENT, TEMPERATURE
IXMITR	SWITCH, FLOW
IXMITR	SWITCH, HAND

TYPE	DESCRIPTION
IXMITR	TRANSMITTER
IXMITR	TRANSMITTER, ANALYSIS
IXMITR	TRANSMITTER, CURRENT
IXMITR	TRANSMITTER, DENSITY
IXMITR	TRANSMITTER, FLOW
IXMITR	TRANSMITTER, LEVEL
IXMITR	TRANSMITTER, LEVEL DIFF
IXMITR	TRANSMITTER, MOISTURE
IXMITR	TRANSMITTER, POSITION
IXMITR	TRANSMITTER, PRESSURE
IXMITR	TRANSMITTER, PRESSURE DIFF
IXMITR	TRANSMITTER, RADIATION
IXMITR	TRANSMITTER, TEMPERATURE
IXMITR	UNIT, AIR HANDLING, MICROPROCR
MECFUN	EVAPORATOR
MECFUN	EXTRACTOR, VAPOR
MECFUN	GEAR, TURNING
MECFUN	GOVERNOR, OVERSPEED
MECFUN	GOVERNOR, SPEED
MECFUN	MIXER
MECFUN	MIXER, VENT
MECFUN	PENETRATION, MECHANICAL
MECFUN	RING, SEAL
MECFUN	ROD, MAGNETIC
MECFUN	SAMPLER
MECFUN	SCREEN, TRAVELING
MECFUN	SHIELD, MISSILE
MECFUN	SKID, ATTEMPERATOR
MECFUN	SKID, BRS
MECFUN	SKID, COND COLLECTING
MECFUN	SKID, COND RETURN
MECFUN	SKID, LWPS
MECFUN	SKID, SEAL WATER
MECFUN	SNUBBER, BALL
MECFUN	SNUBBER, RACK
MECFUN	TURNING GEAR
MOTOR	MOTOR
MOTOR	TURNING GEAR
PANEL/CAB	ALARM
PANEL/CAB	BOX, JUNCTION
PANEL/CAB	CABINET, INSTRUMENT, LOCAL
PANEL/CAB	CABINET, LOCAL INSTRUMENT
PANEL/CAB	CABINET, TERMINATION
PANEL/CAB	PANEL, CONTROL
PANEL/CAB	RESISTOR, GROUND
PANEL/CAB	TERM BOARD
PIPE	DISC, PRESSURE

TYPE	DESCRIPTION
PIPE	EDUCTOR
PIPE	EJECTOR
PIPE	EXPANSION JOINT
PIPE	HOSE, FLEX
PIPE	JOINT, EXPANSION
PIPE	ORIFICE, FLOW
PIPE	PLUG, STUD
PIPE	SILENCER
PIPE	THERMOWELL
PIPE	WELL, TEST
PUMP	INTERNALS, PUMP
PUMP	PUMP
PUMP	PUMP, CENTRIFUGAL
PUMP	SKID, CHEM FEED
PUMP	SKID, FILTER
RELAY	AMPLIFIER
RELAY	CONTROLLER, PRESSURE
RELAY	INDICATOR, NUCLEAR
RELAY	RACK, AUX RELAY
RELAY	RACK, LOCAL INSTRUMENT
RELAY	RACK, RELAY
RELAY	RACK, STUD
RELAY	STARTER, DC
RELAY	SWITCH, HAND
RELAY	VALVE, PCV
RELAY	VALVE, SOV, NON-PROCESS
SECURI	DOOR
SECURI	TRANSMITTER, LEVEL
SW,HAND	SWITCH, HAND
SW,PROCESS	COUNTER, FLOW
SW,PROCESS	SWITCH
SW,PROCESS	SWITCH, DIFF PRESS IND
SW,PROCESS	SWITCH, DIFF PRESSURE
SW,PROCESS	SWITCH, LEVEL
SW,PROCESS	SWITCH, POSITION
SW,PROCESS	SWITCH, PRESSURE
SW,PROCESS	SWITCH, PRESSURE DIFF
SW,PROCESS	SWITCH, SPEED
SW,PROCESS	SWITCH, TEMPERATURE
SW,PROCESS	SWITCH, VIBRATION
SWGEAR	PANEL, DISTRIBUTION
SWGEAR	SWITCHGEAR
TOOL	CART, RECHARGING
TOOL	RIG, LIFTING
TOOL	STAND, STORAGE
TOOL	TENSIONER, STUD
TRANSF	TRANSFORMER
TRANSF	TRANSFORMER, BYPASS

TYPE	DESCRIPTION
TURBIN	TURBINE, AUX
VALVE	ACTUATOR, MOTOR OPERATED
VALVE	DAMPER
VALVE	DAMPER, AOV, PISTON
VALVE	DAMPER, BACKDRAFT
VALVE	DAMPER, FIRE
VALVE	DAMPER, OPPOSED
VALVE	DAMPER, PARALLEL
VALVE	TRAP, MOISTURE
VALVE	TRAP, STEAM
VALVE	VALVE
VALVE	VALVE, AOV, DIAPHRAGM
VALVE	VALVE, AOV, PISTON
VALVE	VALVE, BALL
VALVE	VALVE, BUTTERFLY
VALVE	VALVE, CHECK
VALVE	VALVE, CONTROL
VALVE	VALVE, DIAPHRAGM
VALVE	VALVE, DIAPHRAGM (METAL)
VALVE	VALVE, DRAIN
VALVE	VALVE, FLOW
VALVE	VALVE, GATE
VALVE	VALVE, GLOBE
VALVE	VALVE, HOV
VALVE	VALVE, HYDRAULIC
VALVE	VALVE, MANUAL
VALVE	VALVE, MOTOR OPERATED
VALVE	VALVE, NEEDLE
VALVE	VALVE, PCV
VALVE	VALVE, PRESSURE
VALVE	VALVE, SAFETY
VALVE	VALVE, SOLENOID
VALVE	VALVE, SOV, NON-PROCESS
VALVE	VALVE, SOV, PROCESS
VALVE	VALVE, SPEED
VALVE	VALVE, TEMPERATURE
VALVE	VALVE, TV
VALVE	VALVE, VENT
VALVE	VALVE, VIBRATION SHUTDOWN
VALVE	VALVE, VIBRATION TRIP
VALVE	VIBRATION, RELAY
VALVE	WELL, TEST
VALVOP	ACTUATOR, MOTOR OPERATED
VALVOP	MOV
VALVOP	VALVE, MOTOR OPERATED
VESSEL	COLUMN, INSTRUMENT
VESSEL	DEMINERALIZER
VESSEL	DISCONNECT, RV INTERNALS

TYPE	DESCRIPTION
VESSEL	DRYER, AIR
VESSEL	INTERNALS, REACTOR
VESSEL	PRESSURIZER
VESSEL	RECEIVER
VESSEL	RESERVOIR
VESSEL	SEPARATOR, MOISTURE
VESSEL	SHROUD
VESSEL	STUD
VESSEL	TABLE, THIMBLE SEAL
VESSEL	TANK
VESSEL	THIMBLE, DETECTOR
VESSEL	TIE ROD, SEISMIC
VESSEL	TRAP, MOISTURE
VESSEL	TUBE, THIMBLE GUIDE
VESSEL	VESSEL, REACTOR

2. *The licensee's proposed exemption request is not clear in terms of which ASME Code requirements will continue to be applied to safety-related components that are categorized as LSS or NRS. In the August 31, 1999, meeting to discuss the exemption request, the licensee stated that safety-related piping 1-inch nominal pipe size (NPS 1) and less were not subject to ASME Code requirements. Although the ASME Code, Section XI excludes ISI requirements for piping and components of NPS 1 and smaller (except for steam generator tubing), it is not clear whether the licensee was intending to exclude safety-related piping NPS 1 and smaller from ASME Code, Section III design requirements or from the ASME Code, Section XI repair and replacement requirements. Please clarify which requirements of the ASME Code (Section III and Section XI) will not be applied to safety-related piping NPS 1 and smaller. Please either confirm that the ASME Code requirements will continue to be satisfied at STP, or provide a technical basis for why this piping will remain functional under all design conditions (i.e., with the design, repair, and replacement requirements eliminated).*

RESPONSE:

STP requests an exemption from 10CFR50.55a as it relates to ASME Code requirements. Since the plant has already been designed and constructed in accordance with the ASME Code, STP's exemption request addresses only the portion of the Code that involves the repair or replacement of items, when the need arises; and in-service testing and inspection. This activity is covered under Section XI of the Code. For replacement activities, Section XI of the Code generally requires that replacement items meet the requirements of the original Construction Code, i.e., ASME Section III. However, Section XI does provide relief for piping, valves, and fittings 1-inch nominal pipe size (NPS) and less. These items are exempt from the requirements of Section XI and, by reference, from the requirements of Section III, as long as the materials and applicable stress levels are consistent with the requirements of the applicable Construction Code. The Code provides this relief in consideration of the burden of the extensive controls required by Section III, especially subsection NCA, when compared to the likelihood and consequences of a failure.

STP considers that the above relief should be expanded to include safety related LSS and NRS replacement items, regardless of size or product form. In procuring such items, STP would ensure that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements and are documented. This process would be similar to the existing relief allowances provided in Section XI for NPS 1 and smaller items.

In order to effect the above relief, STP is seeking an exemption from the requirements of the ASME Code that would allow replacement items to be fabricated or procured, installed, and examined in accordance with the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks). Consistency with the original material specification and design will be maintained by using the original ASME Section II material or equivalent standard material (e.g., ASTM), applying the original ASME Section III Design Specification

loadings and load combinations, using the ASME Section III calculation methodology and allowables. (Note: The use of ASME Section III allowables is justified because it is consistent with the ASME Section III design methodology. Also, the use of identical or equivalent material and original design methodology will, in general, eliminate the need for reanalysis provided the wall thickness exceeds original design minimum wall requirement). It is not the intent of this request to require manufacturing and quality requirements of ASME Section III such as impact testing and NDE. These are considered permissible reductions in special treatment requirements consistent with the low risk classification. In addition, STP will conduct post-installation pressure tests in accordance with ASME Section XI. Other additional requirements of ASME Section III, including Subsection NCA, General Requirements, would not apply.

For example, the replacement of a LSS or NRS ASME valve would consist of the following steps:

1. **Initiate a design change package to implement the replacement of a code valve with a non-code valve.**
2. **Identify the proposed ANSI B16.34 valve**
3. **Ensure and document that the material specifications of the replacement valve are consistent with the requirements of ASME Section II or equivalent standard material (e.g., ASTM).**
4. **Ensure that the wall thickness exceeds the original design minimum wall thickness. (Note: this step, along with step 3 above, will ensure that the stresses are within allowables).**
5. **Prepare a Work Package that identifies installation requirements required by ANSI B31.1 such as welding, NDE, etc.**
6. **Install the valve and conduct post-installation testing in accordance with ASME Section XI and the requirements of the work package.**
7. **Maintain associated records for the life of the plant.**

STP considers that these measures are sufficient to provide reasonable assurance, commensurate with the low risk significance, that the technical requirements of a recognized code are satisfied and that replacement items would remain functional under all design conditions.

3. *The July 13, 1999, submittal stated that an exemption to General Design Criterion (GDC) 4, which includes qualification for dynamic effects, was requested. During the meetings with the staff on August 31, September 1, and October 5, 1999, the licensee stated that an exemption was not requested for GDC 4 in its entirety and that dynamic qualification of electrical and mechanical components was out of the scope of the exemption request.*

   (a) *The staff requests that the licensee clarify the scope of the proposed exemption request under GDC 4, including whether dynamic qualification is considered in the scope of the exemption request.*

   (b) *In addition, indicate whether or not the dynamic qualification of the piping, and cable raceways and conduits are also included in the exemption request.*

RESPONSE: (a) and (b)

General Design Criterion 4 addresses both the environmental and dynamic effects design bases. GDC 4 states that structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including the loss-of-coolant accidents. The structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the commission demonstrate that the probability of fluid systems piping rupture is extremely low under conditions consistent with the design basis for the piping.

To clarify STP's position, the exemption request with respect to GDC 4 is only for the environmental effects design bases for LSS and NRS SSCs as described in the submittal paragraph 4.1.1. All safety related SSCs, (HSS, MSS, LSS, and NRS), including piping, cable trays, and conduit, will continue to be protected or otherwise designed to withstand the dynamic effects as described in GDC 4. The original draft exemption request addressed GDC 4 in its entirety. The exemption request has been modified to clarify relief from the environmental effects design bases of GDC 4 only for all LSS and NRS SSCs. The dynamic qualification of piping, cable raceways, and conduits is not included in this exemption request.

For example, if a safety-related LSS transmitter located in Containment (qualified for high temperature / high humidity operation following a loss-of-coolant accident) were to fail in normal operations, STP could replace this transmitter with a functionally equivalent, commercial grade non-safety related transmitter (like-for-like replacement) since dynamic qualifications are not affected. However, if this LSS transmitter were to be relocated to a different area inside of Containment, a design change package would be generated to specially address the dynamic qualification impacts to facilitate this change. The design change would conform to the GDC 4 dynamic effects in regards to the new mounting of the transmitter and the routing of the conduit to support transmitter operation.

4. *It is not clear from the licensee's submittal whether the request is for a one-time exemption from the 50.59 evaluation requirements (i.e. for assessing the impact of deleting special treatment requirements on a component by component basis) or whether the proposal is for a permanent and more global exemption from 50.59 evaluations for equipment categorized as LSS or NRS. For example, after these special treatment requirements are relaxed, is it the licensee's intention to continue to use 50.59 to evaluate subsequent changes to the LSS and NRS components (e.g. repair or replacement) to determine if an unreviewed safety question exists (i.e., and therefore requires prior staff review and approval) or is the licensee suggesting that components categorized as LSS and NRS are outside the scope of 50.59 entirely? Please either confirm that 50.59 will be used to evaluate subsequent changes to components categorized as LSS or NRS, or describe an alternate process for controlling those changes.*

RESPONSE:

To clarify our position, STP requests a permanent and global exemption from the 50.59 evaluation process for all LSS and NRS components, but only for special treatment requirement issues. The 50.59 evaluation process will still be used for LSS or NRS components for issues unrelated to special treatment requirements (e.g., functional changes, design changes, etc.) as applicable.

Changes in special treatment requirements for LSS or NRS components will be controlled through existing commercial treatment programs that provide reasonable assurance that design and functional requirements are met. These controls include, but are not limited to, the Corrective Action Program for identification and correction of deficiencies, engineering evaluations as needed to ensure that functional/design features are not affected, and appropriate post maintenance testing to validate that the functional requirements of the component are still satisfied. These commercial practices are addressed in detail in Section 3.3.3 of the Exemption Request.

For example, if STP were to replace a failed safety-related LSS/NRS component with a commercial-grade, non-safety-related component that meets existing design and functional requirements, then a 50.59 evaluation would not need to be performed. The 50.59 evaluation process would not be required in this situation since the change is associated solely with special treatment requirements within the scope of this exemption request. Alternatively, if STP were to replace a safety-related LSS/NRS component with a commercial-grade component that does not satisfy existing functional and/or design requirements, then a 50.59 evaluation would be performed. In this case, the 50.59 evaluation process would be required since a functional or design change is affected which is outside the scope of special treatment requirements.

5. *Existing controls regulating facility changes, such as 10 CFR 50.59, are intended to preserve the deterministic licensing and design basis. High safety significant (HSS) and medium safety significant (MSS) systems, structures, and components (SSCs) may be risk significant based on performance attributes derived from circumstances which are not within the bounds of the existing design basis. Therefore, existing change controls may not provide a sufficient mechanism to preserve these risk significant characteristics. Please identify those areas where risk-significant attributes are not addressed by current special treatment requirements. In addition, describe what additional controls will be implemented for HSS and MSS SSCs to ensure risk significant attributes are not changed inappropriately.*

RESPONSE:

Areas where risk significant attributes (critical attributes) may not be addressed by current special treatment requirements involve non-safety-related components that have been categorized as HSS or MSS. Prior to implementation of the risk significance determination process, these components were not required to adhere to any special treatment requirements or other controls other than normal commercial practices. With the advent of the risk-informed processes, STP has identified these components as deserving special attention and is implementing additional controls to provide increased assurance that the critical attributes are preserved. Examples of additional controls that may be applied include:

1. **Procurement process - Receipt inspection performed to verify that the critical attributes meet design/functional requirements.**
2. **Maintenance Activities –**
   - **Use of planned and fully documented maintenance work packages**
   - **Quality Control hold points**
   - **Additional post-maintenance testing**
3. **Maintenance Rule - Inclusion in the Maintenance Rule scope, if not already included.**
4. **Preventive Maintenance –Inspections and preventive/predictive maintenance activities that are targeted toward the critical attributes.**
5. **Other Plant Activities – Increased sensitivity to the critical attributes of these components whenever other plant activities may impact these attributes.**

For safety related HSS and MSS components, STP considers that these components and their critical attributes are within the bounds of the licensing and design basis (in other words, for safety-related components, STP does not take credit for any functions in the PRA that are not already part of the design and licensing basis). Of all of the functions that a component is designed to perform, those that are associated with its risk significance are identified as critical attributes. Therefore, critical attributes are a subset of the design functions of a component and are within the bounds of the licensing and design basis. While there may be risk significant events beyond the scope of the design basis where HSS and MSS components would be required to function, they would not be called upon to perform functions that have not already been identified as critical attributes. Therefore, existing controls regulating facility changes are considered to be adequate to preserve these risk significant critical attributes.

6. *The July 13, 1999, submittal indicates that the licensee is not requesting an exemption from 10 CFR 50.55a. However, the ASME Code as incorporated by reference in 50.55a establishes quality criteria for replacement parts. Section 4.2.1 of the submittal states that, "For safety-related LSS and NRS (not risk significant) components, the requested exemption would enable STP to replace an ASME component with a non-ASME component without the need to perform a detailed 50.59 evaluation or seek prior NRC approval." With respect to this point, provide information to address the following items:*

   (a) *How can this be accomplished without exemption from the 50.55a requirements or an approval of an acceptable alternative under 50.55a(a)(3)(i)?*

   (b) *Will individual systems contain both ASME and commercial grade parts? If yes, please describe how the licensee will ensure system safety functionality.*

   (c) *How will records be maintained, e.g., the N-5 data package for piping systems?*

RESPONSE:

To clarify the STP position, STPNOC does request an exemption from 10CFR50.55a with respect to ASME Code requirements for the repair or replacement of LSS and NRS components or associated parts and for in-service testing and inspection for such components. The exemption request has been updated to reflect this change.

For replacement activities under this exemption, ASME Code components could be replaced with non-Code components, as the need arises. Thus, individual systems could contain both Code and non-Code components. As described in the response to Question 2, the replacement process would be conducted in accordance with the requirements of STP's design change program and the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks), subject to ensuring that the material specifications and the design methodology are consistent with the original requirements. These measures will provide an appropriate level of assurance of system safety functionality and will provide proven and reasonable assurance, commensurate with the low risk significance, that the components will function as designed.

The N-5 data report provides records of the original construction and is not revised. Up to this point, subsequent changes or replacements have been performed in accordance with Section XI and result in a NIS-2 data report. The current configuration of a piping system, with regards to Code records, is ascertained by the review of the original N-5 data report and any subsequent NIS-2 data reports. Under the proposed exemption for the replacement of LSS and NRS components, NIS-2 data reports would not be compiled. Instead, the data typically contained in the NIS-2 reports would be documented in the design change packages and in the work packages, as described in the response to Question 2. These types of records are maintained for the life of the plant and provide adequate documentation to ascertain the configuration of replacement items.

7. *The licensee has indicated that in-service inspection (ISI) and testing (IST) are not included in the scope of the exemption request and stated that it would use RG 1.178 and RG 1.175 to risk-inform ISI and IST at a later time (see Attachment 3 to the licensee's July 13, 1999, submittal, in response to IST Question No. 1). It is not clear whether the licensee intends to take safety-related components categorized as LSS or NRS out of the scope of their ISI and IST program as part of the proposed exemption request. Section 4.1.2 of the licensee's proposed exemption request states, "For LSS and NRS components, South Texas Project (STP) seeks to reestablish ASME Code class boundaries at a component level basis rather than on a system level basis without prior NRC approval. If this exemption is granted, LSS and NRS ASME components may be replaced with non-ASME components without prior NRC approval."*

   (a) *Please verify that ASME code components will be inspected and tested in accordance with the code requirements until such time as alternative risk-informed ISI and IST programs are approved under a separate regulatory action.*

   (b) *Would the non-ASME replacement components continue to be tested and inspected in accordance with the ASME Code? If not, explain your rationale for not continuing ASME tests or inspections.*

RESPONSE:

To clarify STP's position, STP has re-evaluated the ISI and IST program interfaces and has concluded that they should be within the scope of this exemption request. ISI and IST programs are special treatment processes. In that regard, the risk significance evaluation process can be used as an input into establishing system functional and component importances within the ISI and IST programs. The decision to include ISI and IST in the exemption request is due to the fact that current risk informed code cases do not allow for scope changes. Under the current ASME O&M code cases, only test frequency changes or testing strategy changes are permitted. It is STP's position that future risk-informed ISI and IST approaches must include scope and testing strategy alternatives in order to be consistent with the intent of Options 2 and 3 of SECY 98-300 for risk-informing 10CFR50. Thus, it is determined that ISI and IST are within the scope of this exemption request.

STP requests that LSS and NRS components be exempted from ASME IST and ISI programs. This exemption accomplishes the scope change consistent with Option 2 of SECY 98-300. Exempting LSS and NRS components from ISI and IST programs will result in those ASME code components, as well as non-ASME replacement components, not being required to be tested or inspected in accordance with ASME Code requirements (e.g., reporting, trending, etc.). It should be noted, however, that industry accepted testing and inspection requirements may be applied as appropriate for currently installed or replacement components in order to provide reasonable assurance of component functional capability.

The rationale for this approach is that for LSS and NRS components, the rigor associated with complete compliance with ASME Code requirements is not necessary for reasonable assurance that components are capable of performing their intended function(s). Reasonable assurance for LSS and NRS components is achieved through other programs currently in effect. For example, the Maintenance Rule will monitor LSS/NRS SSC performance at the plant/system/train level, as appropriate. The Maintenance Rule monitoring of LSS/NRS SSCs consists of Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function, which provides a mechanism for regulatory oversight for equipment performance. Also, the Corrective Action Program is effective in identifying equipment nonconformances and deficiencies regardless of risk significance. Further support for this position is the result of a recent analysis requested by the Staff (meeting on April 10 and 11, 2000). At this meeting, the Staff requested a PRA sensitivity analysis in which all LSS component failure rates were increased by a factor of 10 which is well beyond Maintenance Rule and Corrective Action Program thresholds. The results of this study provide a bounding analysis showing the risk impact of elimination of special treatment requirements for LSS components. The study results indicated only a small increase in CDF, and the increase was well within the limits identified in Regulatory Guide 1.174. Thus, the rationale as described above provides the basis for STP's position that compliance with ASME Code ISI and IST requirements for LSS and NRS components is not necessary to provide reasonable assurance of component reliability and performance.

Additionally, future IST and ISI changes meeting the intent of Option 3 of SECY 98-300 are also envisioned for HSS and MSS components. It is STP's intent to work with industry institutions, such as ASME, to establish risk-informed methods to address ASME IST and ISI special treatment requirements. As alternative risk informed approaches are approved and test frequency changes and test strategy changes are permitted, changes to the STP's IST and ISI programs would also be amended. STP will also continue to pursue safety and cost beneficial changes not requiring regulatory approval.

8. *Important aspects regarding special treatment provisions may exist in various licensee commitments. Before the staff can entertain an approval of the proposed exemption, the staff needs to understand how the exemptions will affect those commitments, and what process will be used by the licensee to control changes to commitments. Please explain the process to control changes to any commitments involving special treatment activities, that could result from implementing the proposed exemptions. This includes changes to commitments that have been implemented in response to Generic Letters, Bulletins, Inspection Reports, commitments made to support licensing actions, etc.*

RESPONSE:

In general, changes to STP commitments related to special treatment requirements will be controlled using STP's commitment control process. STP's process is consistent with the guidance of the Nuclear Energy Institute (NEI) NEI-99-04, entitled *Guidelines for Managing NRC Commitment Changes*, which the NRC found acceptable in SECY-00-0045. However, granting of STP's exemption request will affect the manner in which STP's commitment control process will be implemented. In particular, based upon its commitment control process and the exemption request, STP will be taking the following actions to control changes in its commitments related to special treatment requirements:

- **<u>Changes in Technical Specifications, License Conditions, and Orders</u> – The technical specifications identify special treatment requirements. STP will not make a change in these requirements without applying for and receiving prior NRC approval of an amendment to its technical specifications. Similarly, STP will seek NRC approval prior to changing any special treatment requirements in an order or license condition.**

- **<u>Changes in the Updated Final Safety Analysis Report (UFSAR)</u> – The UFSAR for STP describes some of the special treatment requirements for STP. Normally, changes in the USFAR would require STP to perform an evaluation under 10 CFR 50.59, and to seek prior NRC approval for any changes that satisfy the criteria in that regulation. However, STP has requested an exemption from Section 50.59 to enable STP to change the special treatment requirements for LSS and NRS components as described in the UFSAR without performing a 50.59 evaluation or seeking prior NRC approval. Therefore, following grant of the exemption, STP will notify the NRC of any changes in the UFSAR special treatment requirements for LSS and NRS components in accordance with 10 CFR 50.71(e).**

- **<u>Changes in the Quality Assurance Program (QAP) Description</u> – The QAP for STP describes some of the special treatment requirements for STP, including requirements for LSS and NRS components. Normally, STP would need to evaluate changes in the QAP description under 10 CFR 50.54(a), and to seek prior NRC approval for any changes that involved a reduction in commitments. However, STP has requested an exemption from Section 50.54(a) to enable STP to change the special treatment requirements for LSS and NRS components as identified in the QAP description without seeking prior NRC approval. Therefore, following grant of the exemption, STP will notify the NRC of any changes in the special treatment requirements for LSS and NRS components in the QAP description in accordance with 10 CFR 50.54(a).**

- **<u>Changes in Other Commitments</u> – For changes in other commitments related to the special treatment requirements for LSS and NRS components, STP will implement its project procedure on licensing commitment management and administration. This procedure contains provisions that are similar to those in NEI 99-04. The exemption itself will serve as the bases for changing these commitments.**

In support of this exemption request, STP has not attempted to identify every commitment involving a special treatment requirement for an LSS or NRS component, nor does STP believe that such an exercise is necessary, warranted, or beneficial. First, until the NRC grants the exemption, STP categorizes the components, and STP applies the exemption and establishes new treatment requirements for the categorized components, it is not possible to identify which, if any, commitments will be affected. Second, the generic assessments that STP has provided in its exemption request envelope the impacts attributable to the changes in particular commitments. Therefore, there is no reason to evaluate the impact of each individual change in a commitment to the special treatment requirements for an LSS or NRS component.

9. *The licensee's July 13, 1999, exemption request did not adequately describe the process STPNOC will use to categorize and make subsequent changes to special treatment requirements for safety-related LSS and NRS equipment. As outlined in RG 1.174, the staff needs to have a clear description of the overall process. The staff additionally needs to establish an appropriate level of regulatory change control over that process before it can accept the proposed exemptions. Please provide an enhanced description (i.e., such as might be included in a revision to the licensee's Updated Final Safety Analysis Report (UFSAR), revision to the licensee's Operations Quality Assurance Plan (OQAP), or included as an exemption condition) of the exemption request and associated processes to be implemented by STPNOC. Also, propose whether this description should be placed in the UFSAR, the OQAP, or be included an as exemption condition (or be placed in some other location), and discuss the reasons for your desired location of this description. The enhanced description should address at a minimum:*

- *the specific aspects of each regulation for which an exemption is requested,*
- *the component categorization process (for components modeled and not modeled in the licensee's PRA),*
- *proposed implementation plans including a description of the treatment processes that will be applied to safety-related components categorized as HSS vs. MSS vs. LSS vs. NRS as well as to non-safety-related components categorized as HSS or MSS,*
- *the process used to assess the aggregate change in plant risk (CDF or LERF) associated with changes in special treatment for components,*
- *the integrated decision making process used by the licensee (including consideration of the defense-in-depth philosophy and safety margins),*
- *performance monitoring processes,*
- *feedback and corrective action processes,*
- *plans for periodic reassessment of the overall process and program,*
- *processes for controlling changes to the aforementioned plans and processes.*

RESPONSE:

Section 2 of the exemption request has been modified to provide an enhanced description of the process to be implemented by STP for categorization of components. STP further notes that additional details regarding the processes and programs referred to in this response can be found in the responses to other questions in the RAI.

STP proposes to include the attached information as a new Section 13.7 to the STP Updated Final Safety Analysis Report (UFSAR), which will be controlled in accordance with the requirements in 10 CFR 50.59. Additionally, as discussed in Section 13.7.5.3, STP is proposing additional criteria for controlling changes. The new UFSAR section is not all of the changes that are required for the UFSAR. For instance, Ch. 3 descriptions of safety classification will require complementary changes for consistency. STPNOC will incorporate the necessary consistency changes in the UFSAR update that documents the implementation of the exemption.

STP's proposed process represents the first-of-a-kind program, and it may be expected that changes will be needed or desirable as experience with implementation of the process is accumulated. The change criteria incorporated into the new Section 13.7 will provide flexibility for STP to make changes in the process based upon this experience, while ensuring that significant changes will be subject to prior NRC approval. As discussed in more detail in Attachment 6, STP believes that the USFAR is the appropriate location for this information. The OQAP is not the appropriate location, because the risk-informed process applies to more subjects than just quality assurance, and because the change control process for the OQAP would be unduly restrictive if applied to Section 13.7. Additionally, it would be inappropriate (and inconsistent with NRC's policy toward the technical specifications and other license conditions) to incorporate the attached information as a license condition. Such a condition would create an undue burden on both STP and the NRC, because it would require STP to apply for and the NRC to review and approve relatively minor changes in the process.

13.7 RISK-INFORMED SPECIAL TREATMENT REQUIREMENTS

13.7.1 Introduction

NRC regulations in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of components that are safety-related, important to safety, or otherwise come within the scope of the regulations. These special treatment requirements go beyond normal commercial and industrial practices, and include quality assurance (QA) requirements, qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. STP has been granted an exemption from the special treatment requirements. Table 13.7-1 identifies the regulations from which an exemption was granted and the scope of the exemption. This exemption only pertains to special treatment requirements; it does not change the design and functional requirements for components.

STP has a risk-informed process for categorizing the safety/risk significance of components. This process is described in Section 13.7.2. Components with no or low safety significance have been exempted from the scope of most of the NRC regulations that impose special treatment requirements, and instead are subject to normal industrial and commercial practices. Additionally, non-safety related components with medium or high safety significance are evaluated for enhanced treatment. Components retain their original regulatory requirements unless they have been recategorized using the process described below. The treatment for the various categories of components is described in Section 13.7.3. As part of this process, STP also performs continuing evaluations and assessments, which are described in Section 13.7.4. Finally, STP applies quality assurance to this process, and controls changes to the process, as described in Section 13.7.5.

13.7.2 Component Categorization Process

13.7.2.1 <u>Overview of Categorization Process</u>. The process utilized by STP in categorizing components consists of the following major tasks:

1. **Identification of functions performed by the subject plant system.**
2. **Determination of the risk significance of each system function.**
3. **Identification of the system function(s) supported by that component.**
4. **Determination of a risk categorization of the component based on probabilistic risk assessment (PRA) insights (where the component is modeled)**
5. **Development of a categorization of the component based on deterministic insights.**
6. **Designation of the overall categorization of the component, based upon the higher of the risk categorization and the deterministic categorization.**
7. **Identification of critical attributes for components determined to be safety/risk significant.**

The processes for determining the risk categorization and deterministic categorization of a component are described in more detail in Sections 13.7.2.3 and 13.7.2.4.

Based upon these processes, a component is placed into one of four categories: 1) high safety/risk significant (HSS), 2) medium safety/risk significant (MSS), 3) low safety/risk significant (LSS), and 4) non-risk significant (NRS). This categorization process does not, in and of itself, affect the other classifications of the component (e.g., safety, seismic, ASME classification).

The process is implemented by a Working Group comprised of individuals experienced in various facets of nuclear plant operation and reviewed by an Expert Panel. This integrated decision process is described in more detail in Section 13.7.2.2.

13.7.2.2 Comprehensive Risk Management Process. The integrated decision-making process used by STP is documented by procedure. The integrated decision-making process incorporates the use of an Expert Panel and Working Groups. The Expert Panel is comprised of qualified senior level individuals and is responsible for oversight of the program and for reviewing the activities and recommendations of the Working Group. The Working Group is comprised of experience individuals who apply risk insights and experience to categorize components in accordance with the process described in this Section and make recommendations to the Expert Panel.

Procedures control the composition of and processes used by the Expert Panel and Working Group. Procedures also identify training requirements for members of the Expert Panel and Working Group, including training on probabilistic risk assessment, risk ranking, and the graded quality assurance process. Finally, the procedures specify the requirements for a quorum of the Expert Panel and Working Group, meeting frequencies, the decision-making process for determining the categorization of components, the process for resolving differing opinions among the Expert Panel and Working Group, and periodic reviews of the appropriateness of the programmatic control and oversight provided to categorized components.

13.7.2.3 PRA Risk Categorization Process. A component's risk categorization is initially based upon its impact on the results of the PRA.

STP's PRA calculates both a core damage frequency (CDF) and a large early release frequency (LERF). The PRA models internal initiating events at full power and also accounts for the risk associated with external events.

The PRA configuration control program incorporates a feedback process to update the PRA Model. The updates are segregated into two categories:

- **The plant operating update incorporates plant design changes and procedure changes that affect PRA-modeled components, initiating event frequency updates, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months.**

- **The comprehensive data update incorporates changes to plant-specific failure rate distributions and human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months.**

The PRA model may be updated on a more frequent basis if an update would result in a significant increase in the CDF.

Only components that are modeled in the PRA are given an initial risk categorization. The PRA risk categorization of a component is based upon its Fussell-Vessely (FV) importance, which is the fraction of the CDF and LERF to which failure of the component contributes, and its risk achievement worth (RAW), which is the factor by which the CDF and LERF would increase if it were assumed that the component is guaranteed to fail. Specifically, PRA risk categorization is based upon the following:

PRA Ranking	Criteria
High	RAW ≥ 100.0 or FV ≥ 0.1 or FV ≥ 0.005 and RAW ≥ 2.0
Medium (Further Evaluation is Required)	FV < 0.005 and 100.0 > RAW ≥ 10.0
Medium	FV ≥ 0.005 and RAW < 2.0 or FV < 0.005 and 10.0 > RAW ≥ 2.0
Low	FV < 0.005 and RAW < 2.0

13.7.2.4 <u>Deterministic Categorization Process</u>. Components are subject to a deterministic categorization process, regardless of whether they are also subject to the risk categorization process using PRA insights. This deterministic categorization process can result in an increase, but not a decrease, in a component's categorization.

A component's deterministic categorization is directly attributable to the importance of the system function supported by the component. In cases, where a component supports more than one system function, the component is classified based on the highest safety classification of the function supported. In categorizing the functions of a system, the Working Group considers five critical questions regarding the function, each of which is given a different weight. These questions and their weight are as follows:

QUESTION	WEIGHT
Is the function used to mitigate accidents or transients?	5
Is the function specifically called out in the emergency operating procedures (EOPs) or Emergency Response Procedures (ERPs)?	5
Does the loss of the function directly fail another risk-significant system?	4
Is the loss of the function safety significant for shutdown or mode changes?	3
Does the loss of the function, in and of itself, directly cause an initiating event?	3

Based on the impact on safety if the function is unavailable and the frequency of loss of the function, each of the five questions is given a numerical answer ranging from 0 to 5. This grading scale is as follows:

"0" - Negative response

"1" - Positive response having an insignificant impact and/or occurring very rarely

"2" - Positive response having a minor impact and/or occurring infrequently

"3" - Positive response having a low impact and/or occurring occasionally

"4" - Positive response having a medium impact and/or occurring regularly

"5" - Positive response having a high impact and/or occurring frequently

The definitions for the terms used in this grading scale are as follows:

Frequency Definitions –

- **Occurring Frequently – continuously or always demanded**
- **Occurring Regularly – demanded > 5 times per year**
- **Occurring Occasionally – demanded 1-2 times per cycle**
- **Occurring Infrequently – demanded < once per cycle**
- **Occurring Very Rarely – demanded once per lifetime**

Impact Definitions –

- **High Impact – a system function is lost which likely could result in core damage and/or may have a negative impact on the health and safety of the public**
- **Medium Impact – a system function is lost which may, but is not likely to, result in core damage and/or is unlikely to have a negative impact on the health and safety of the public**
- **Low Impact – a system function is significantly degraded, but no core damage and/or negative impact on the health and safety of the public is expected**
- **Minor Impact – a system function has been moderately degraded, but no core damage or negative impact on the health and safety of the public**
- **Insignificant Impact – a system function has been challenged, but no core damage or negative impact on the health and safety of the public**

Although some of these definitions are quantitative, both of these sets of definitions are applied based on the collective judgment and experience of the Working Group.

The numerical values, after weighting, are summed; the maximum possible value is 100. Based on the sum, functions are categorized as follows:

SCORE RANGE	CATEGORY
0 – 20	NRS
21 – 40	LSS
41 – 70	MSS
71 – 100	HSS

A function with a low categorization due to a low sum can receive a higher risk classification if any one of their five questions received a high numerical answer. Specifically, a weighted score of 25 on any one question results in an HSS categorization; a weighted score of 15-20 on any one question results in a minimum categorization of MSS; and a weighted score of 9-12 on any one question results in a minimum categorization of LSS. This is done to ensure that a component with a significant risk in one area does not have that risk masked because of its low risk in other areas.

- **In general, a component is given the same categorization as the system function that the component supports. However, a component may be ranked lower than the associated system function**

13.7.2.5 Defense in Depth and Safety Margins. For the following reasons, the exemption and the categorization process maintain defense in depth and sufficient safety margins:

- **Functional requirements and the design configuration of systems are retained.**
- **No existing plant barriers are removed or altered.**
- **Design provisions for redundancy, diversity, and independence are maintained.**
- **The plant's response to transients or other initiators is not affected.**
- **Preventive or mitigative capability of components is preserved.**
- **There is no change in any of the safety analyses in the UFSAR.**
- **Existing safety-related LSS and NRS components will not be replaced, absent good cause (e.g., obsolescence or failure). Since the existing safety-related LSS and NRS components were designed, procured, manufactured, and installed in accordance with the existing special treatment requirements, these components have inherent design margins to perform their intended functions that will not be adversely affected by this exemption.**
- **Normal commercial and industrial practices provide an appropriate and acceptable level of assurance that safety-related LSS and NRS components will be able to perform their intended functions.**
- **The corrective action program is applied to safety-related LSS and NRS components. This program provides reasonable assurance that deficiencies involving safety-related LSS and NRS components will be identified, corrected, and necessary action taken to ensure acceptable performance levels are maintained.**

13.7.3 Treatment for Component Categories

13.7.3.1 Description of Treatment for Component Categories. The following treatment is provided for the various component categories:

- **Safety-Related HSS and MSS Components – These components continue to receive the treatment required by NRC regulations and STP's associated implementing programs.**

- **Non-Safety-Related HSS and MSS Components – These components will continue to receive any existing special treatment required by NRC regulations and STP's implementing programs. Additionally, the risk-significant functions of these components will receive consideration for enhanced treatment. This consideration is described in Section 13.7.3.2.**

- **Safety-Related LSS and NRS Components – These components receive normal commercial and industrial practices. These practices are described in Section 13.7.3.3.**

- **Non-Safety-Related LSS and NRS Components – The treatment of these components is not subject to regulatory control.**

- **Uncategorized Components – Until a component is categorized, it continues to receive the treatment required by NRC regulations and STP's associated implementing programs, as applicable.**

13.7.3.2 Enhanced Treatment for Non-Safety Related HSS and MSS Components. Non-safety-related HSS and MSS components may perform risk-significant functions that are not addressed by STP's current treatment programs.

Examples of process enhancements for non-safety-related HSS and MSS components may include:

- **Performing routine preventive maintenance (PM) tasks more frequently to ensure component reliability**
- **Ensuring that the component's critical attributes are functionally validated following maintenance activities**
- **Ensuring that replacement part controls are enhanced for the components (e.g., TARGETED components receive engineering evaluation for item equivalency replacements).**
- **Increasing the quality oversight of work activities and work documentation (e.g., audits, surveillances, and monitoring activities are performed for TARGETED components).**

These identified processes provide reasonable assurance that HSS and MSS components will be able to perform their safety significant functions.

13.7.3.3 Normal Commercial and Industrial Practices for Safety-Related LSS and NRS Components

A description of STP's commercial practices is provided below.

Plant activities are conducted in a manner that is consistent with a conservative, technically-sound approach to plant operations which properly values nuclear safety and public protection. Goals and performance criteria are established commensurate with the importance to safety and plant reliability of the service, SSC, process or function.

13.7.3.3.3 Design Control Process. The Station's Design Control Program is used for both safety related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications which are in the Configuration Management Program. The Design Control Program complies with 10 CFR Part 50, Appendix B and is described in the Operations Quality Assurance Plan (OQAP).

13.7.3.3.4 Procurement Process. Station procurement activities, including the identification, procurement, and receipt of replacement components or parts, are performed in accordance with approved procedures and processes. These processes include requirements for appropriate material and performance specifications, inspection, and special handling.

13.7.3.3.5 Installation Process. The Station's component and part installation activities are accomplished through the Design Change Programs and/or through the

Maintenance processes. Components procured under the commercial treatment program may be installed in plant systems that are governed by special treatment requirements, including but not limited to ASME Code, Class 1E, and seismic requirements. In such instances, proceduralized measures are taken to ensure that the most limiting requirements are applied at the interface between the commercial component and the rest of the system. Appropriate post-maintenance testing is performed.

13.7.3.3.6 Maintenance Process. The Station's maintenance process activities include corrective, preventive, and predictive maintenance. These maintenance activities are performed in accordance with approved procedures, processes, and guidelines.

13.7.3.3.7 Inspection, Test, and Surveillance Process. The Station's inspection and test process is primarily addressed and implemented through the Maintenance process. As stated above, the Maintenance process addresses inspections and tests through corrective, preventive, and predictive maintenance activities.

The Station's surveillance process is governed by approved test procedures to ensure that Technical Specification requirements are met and satisfied. These Technical Specification requirements apply to some safety-related and non-safety-related SSCs, including some components that are categorized as LSS. These surveillances will continue to be performed.

In addition to the surveillances required by Technical Specifications, additional monitoring is performed to validate the acceptable performance of SSCs. Examples of programs used to monitor SSC performance include Maintenance Rule, Operator Rounds, System Engineer reviews, and management assessments.

13.7.3.3.8 Corrective Action Program. The Station's Corrective Action Program is used for both safety-related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications. The Corrective Action Program complies with 10 CFR Part 50 Appendix B, and is described in the OQAP.

13.7.3.3.9 Management and Oversight Process. The Station's management and oversight process is accomplished through approved procedures and guidelines. This process includes independent oversight, line self-assessments, and Maintenance Rule implementation (system or train level for LSS and NRS). In addition, the Graded Quality Assurance Working Group periodically assesses SSC performance.

In addition, management of LSS and NRS components subject to commercial practices is also governed by technical and administrative procedures as described throughout Section 13.7.3.3.

13.7.3.3.10 Configuration Control Process. The Station's configuration control process is controlled through approved procedures and policies. The design control process ensures that the configuration of the Station is properly reflected in design documents and drawings. Changes to the Station are controlled through design change

packages (modifications) which require that control drawings and documents be updated prior to closeout of the modification package.

In addition, configuration control addresses the status of components day-to-day in the field. SSCs are tagged and are manipulated by qualified Operations personnel per procedure. The configuration control process manages and controls the physical changes (procedural and equipment) to the facility to assure that the plant configuration and practices correctly reflect the licensing bases. Non-ASME components installed in ASME Code systems are identified and tracked.

13.7.4 Continuing Evaluations and Assessments

13.7.4.1 Performance Monitoring. STP has performance monitoring processes for the changes in the special treatment. This monitoring includes the following:

- **Maintenance Rule Program – Specific performance criteria are identified at the plant, system, or train level. Regardless of their risk categorization, components that affect MSS or HSS functions will be monitored and assessed in accordance with plant, system and/or train performance criteria. Data used for monitoring is obtained from various sources, such as work orders, condition reports, and test results.**

- **Performance Reporting & Identification Database – This database collects both positive and negative indicators from the performance of plant activities, such as corrective maintenance, installation of modifications, and conduct of testing. The Quality organization provides oversight of this database. Corrective Action Program - Condition reports document degraded equipment performance or conditions, including conditions identified as a result of operator rounds, system engineer walk-downs, and corrective/preventive maintenance activities.**

13.7.4.2 Feedback and Corrective Action. STP has feedback and corrective action processes to ensure that equipment performance changes are evaluated for impact on the component risk categorization, the application of special treatment, and other corrective actions. At least once per cycle, performance data is compiled and presented to the Working Group for review, which is performed for each risk-categorized system. Performance and reliability data are generally obtained from sources such as the Maintenance Rule Program and Operating Experience Review.

This process ensures that any significant negative performance changes that are attributed to the relaxation of special treatment controls are addressed timely by the reinstatement of applicable controls up to and including the re-categorization of the component's risk significance, as appropriate.

13.7.4.3 Process for Assessing Aggregate Changes in Plant Risk. The Expert Panel is responsible for assessing and approving the aggregate effect on plant risk for risk-informed applications.

The process used to access the aggregate change in plant risk associated with changes in special treatment for components is based on periodic updates to the station's PRA and the associated PRA risk ranking sensitivity studies.

13.7.4.4 Periodic Assessments. STP's risk-informed programs and processes undergo periodic assessments as determined by the Expert Panel. These assessments are performed in accordance with station procedures, and may include Quality audits, external audits, and self-assessments. The results of the assessments, along with any efficiencies or recommendations identified, are addressed using the Corrective Action Program.

13.7.5 Quality Assurance and Change Control for the Risk-Informed Process

13.7.5.1 Quality Assurance for the PRA and Categorization Process. The provisions of the Operations Quality Assurance Plan (OQAP), Chapter 15.0, Quality Oversight Activities, govern the oversight of the risk categorization process. The program implemented by Chapter 15 provides for independent oversight activities (including audits, assessments, evaluations, performance monitoring, and surveillances) to ensure that the requirements of the OQAP are being properly implemented.

STP has a PRA Configuration and Control program, which is structured to ensure that changes in plant design and equipment performance are reflected in the PRA as appropriate. The PRA Configuration and Control process is administered by procedures and guidelines that ensure proper control of changes to the models. Changes are reviewed by persons independent from the person making the change and are approved by the PRA supervisor.

13.7.5.2 Internal Process for Controlling Changes. Upon approval, the implementation documents (e.g., procedures, Risk Significance Basis Documents, Master Equipment Databases) can not be changed without approval and oversight of the Working Group and the Expert Panel. Processes such as the assessment process, periodic feedback process, and the continuous feedback process provide input for consideration in making changes. The Working Group may recommend appropriate changes and forward these recommendations to the Expert Panel for approval.

Procedures delineate the responsibilities for implementing changes in the implementation documents. Condition reports are generated, as appropriate, to document and track changes. The Expert Panel is designated to maintain cognizance over the implementation of the Comprehensive Risk Management program and adjusting program criteria as appropriate.

13.7.5.3 Regulatory Process for Controlling Changes. Changes affecting Section 13.7 will be controlled in accordance with 10 CFR 50.59, as applicable. In addition, the proposed changes in Section 13.7 must satisfy the following criteria:

- **STP may make changes in the categorization process described in Section 13.7.2, without prior NRC approval, provided that the revised process:**

- **has both a PRA risk categorization process and a deterministic categorization process;**
- **designates the category of a component based upon the higher of the PRA risk categorization and deterministic categorization;**
- **provides a risk categorization of a component as LSS or NRS only if the component has little or no impact on CDF or LERF (using commonly accepted assessment techniques such as FV and RAW); and**
- **provides a deterministic categorization of a component as LSS or NRS only if the component has no more than a minor impact on the ability of the plant to prevent or mitigate accidents.**

- **STP may make changes in the treatment described in Section 13.7.3, without prior NRC approval, provided that the revised treatment provides reasonable assurance that components:**

  - **can perform their HSS and MSS functions, and**
  - **can perform their safety-related functions under design basis conditions.**

- **STP may make changes in the evaluations and assessments described in Section 13.7.4, without prior NRC approval, provided that the revised evaluations and assessments include the following elements:**

  - **Processes to monitor the effect of changes in special treatment on the reliability/availability of safety-significant and safety-related functions;**
  - **Processes to ensure that significant changes in reliability/availability of safety-significant and safety-related functions are evaluated for impact on component risk categorization, application of special treatment, and corrective action; and**
  - **Periodic assessments to update the PRA and to reassess the risk categorization of components.**

TABLE 13.7-1

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
10 CFR 21.3 – An exemption to exclude safety-related LSS and NRS components from the scope of the definition of "basic component."	The procurement, dedication, and reporting requirements in Part 21 are not applied to safety-related LSS and NRS components.
10 CFR 50.34(b)(6)(ii) - An exemption to the extent that it incorporates provisions from 10 CFR Part 50, Appendix B.	Refer to request for exemption from Appendix B.
10 CFR 50.34(b)(11) – An exemption to the extent that it incorporates seismic qualification requirements in Part 100.	Refer to request for exemption from Part 100.
10 CFR 50.49(b) – An exemption to exclude LSS and NRS components from the scope of electric equipment important to safety for the purposes of environmental qualification of electrical equipment.	• The qualification documentation and files specified in Section 50.49 are not applicable to LSS and NRS components. • LSS and NRS components are not required to be maintained in a qualified condition under Section 50.49. • LSS and NRS components may be replaced with components that are not qualified under Section 50.49. • LSS and NRS components, as applicable under Section 50.49, are designed to function in the installed environment. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.
10 CFR 50.54(a)(3) – An exemption from the requirement to seek prior NRC approval for reductions in the commitments in the QA program description involving LSS and NRS components.	STP does not need prior NRC approval for reductions in commitments in the QA program description related to LSS and NRS components.
10 CFR 50.55a(f) and (g) – An exemption from the requirements of ASME Section XI, for repair and replacement of safety-related LSS and NRS components, subject to ensuring that the material specifications and the design	Safety-related LSS and NRS components may be repaired and replaced with components that meet the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks), subject to ensuring that the material

Regulation	Scope of Exemption
(design loadings, design methodology, and stress allowables) are consistent with the original requirements.	specifications and the design are consistent with the original requirements and conducting post-installation pressure tests in accordance with Section XI of the ASME Code. Section 13.7.3.3 identifies the quality, design and procurement controls that are applied to safety-related LSS and NRS components that are repaired or replaced.
10 CFR 50.55a(f) – An exemption from meeting the requirements of ASME Section XI for testing of safety related LSS and NRS components.	Safety-related LSS and NRS components are not in the scope of component-specific inservice testing requirements. System-level testing requirements continue to be applied. Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.55a(g) – An exemption from meeting the requirements of ASME Section XI for inspection of safety related LSS and NRS components.	Safety-related LSS and NRS components are not in the scope of component-specific inservice inspection requirements. Section 13.7.3.3 identifies controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.55a(h) – An exemption to exclude safety related LSS and NRS components from the scope of components required to meet sections 4.3 and 4.4 of IEEE 279.	Sections 4.3 and 4.4 of IEEE 279 do not apply to safety related LSS and NRS components. The other requirements listed in IEEE 279, including functional and design requirements, are applicable. Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.59(a)(1), (a)(2) and (b)(1) (pre-1999 version); 10 CFR 50.59(c)(1), (c)(2), and (d)(1) (2000 version) – An exemption from the requirement to perform a written evaluation of changes in special treatment requirements for LSS and NRS components. Also an exemption from the requirement to seek prior NRC approval for such changes to the extent that they fall within the listed criteria in 50.59.	STP is not required to perform 50.59 evaluations for changes in the special treatment requirements for LSS and NRS components, and is not required to seek prior NRC approval for those changes.
10 CFR 50.65(b) – An exemption to exclude LSS and NRS components from the scope of SSCs covered by the Maintenance Rule.	• STP is required to monitor performance on a plant/system/train level, as applicable. As applicable, STP evaluates failures of LSS and NRS components to determine whether such failures affect MSS or HSS function(s) which then constitute a maintenance rule functional failure at the applicable plant/system/train level.

Regulation	Scope of Exemption
10 CFR Part 50 Appendix A, GDC 1 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 1.	• **LSS and NRS components are not required to satisfy the QA requirements in GDC 1 (except to the extent that they incorporate Criteria III, XV, and XVI of Appendix B to Part 50).** • **Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.**
10 CFR Part 50, Appendix A, GDC 2 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 2, to the extent that GDC 2 requires tests and inspections to demonstrate that SSCs are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.	• **LSS and NRS components are not required to be maintained in a qualified condition under GDC 2.** • **LSS and NRS components may be replaced with components that are not qualified under GDC 2.** • **LSS and NRS components, as applicable under GDC 2, are designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.**
10 CFR Part 50, Appendix A, GDC 4 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 4, to the extent that GDC 4 requires inspection and testing to demonstrate that SSCs are able to withstand environmental effects. GDC 4 requirements related to dynamic effects would not be exempted.	• **LSS and NRS components are not required to be maintained in a qualified condition under GDC 4.** • **LSS and NRS components may be replaced with components that are not qualified under GDC 4.** • **LSS and NRS components, as applicable under GDC 4, are designed to withstand applicable environmental effects. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.** • **GDC 4 requirements with respect to dynamic effects are applicable.**
10 CFR Part 50, Appendix A, GDC 18 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 18, to the extent that GDC 18 requires that inspections and testing be performed for individual features, such as wiring, insulation, connections, switchboards, relays, switches, and buses.	• **Individual LSS and NRS components within the electrical power systems are not required to be inspected or tested.** • **System functional tests are still required in accordance with GDC 18.** • **Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.**
10 CFR Part 50 Appendix B, Introduction – An exemption to exclude safety-related LSS and NRS components from the scope of safety-related SSCs covered by	• **Safety-related LSS and NRS components are not required to satisfy the QA requirements in Appendix B, except for design control, control of nonconformances, and corrective action.**

Regulation	Scope of Exemption
Appendix B (except for Criterion III pertaining to Design Control and Criteria XV and XVI governing non-conformances and corrective actions).	• **Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.**
10CFR Part 50, Appendix J, B.III – An exemption to exclude safety-related LSS and NRS components, subject to the additional limitations list under Scope of Exemption, from the scope of components requiring local leak rate tests and containment isolation valve leak rate tests.	• **Local leak rate tests of LSS containment isolation valves and other safety-related LSS or NRS components are not required. With respect to LSS containment isolation valves, this exemption only applies to valves that satisfy one or more of the following criteria:** - **The valve is not required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers).** - **The valve is normally closed and in a physically closed, water-filled system (e.g., containment isolation valves in the Demineralized Water system)** - **The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Main Feedwater system).** - **The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve.** - **The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).** • **Cumulative limits for containment leakage are based upon the tested components, with the assumption that the exempted components contribute zero leakage.**

Regulation	Scope of Exemption
	• Section 13.7.3.3 identifies controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR Part 100, Appendix A.VI(a)(1) and (2) – An exemption to exclude safety-related LSS and NRS components from the scope of SSCs covered by these sections, to the extent that these sections require testing and inspection to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.	• LSS and NRS components are not required to be maintained in a qualified condition under Part 100. • LSS and NRS components may be replaced with components that are not qualified under Part 100. • LSS and NRS components, as applicable under Part 100, are designed to withstand the effects of seismic events without loss of capability to perform their safety function. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.

10. *The licensee is proposing to downgrade the manual initiation of protective functions one lower level than the ranking of the controlled component. This will result in manual initiation functions being downgraded to LSS when the controlled component is categorized MSS and, thus, manual initiation will be exempted from the special treatments. However, manual initiation is required by IEEE-279 which is embedded in 10 CFR 50.55a(h).*

    (a) Therefore, explain why an exemption from 10 CFR 50.55a(h) has not been requested.

    (b If such an exemption request is proposed, provide the technical basis for the request.

RESPONSE (part a):

We agree with the NRC feedback. Sections 4.3 and 4.4 of IEEE Standard 279 do reference quality and environmental qualification requirements for protection systems and do not exclude the manual initiation portion of those systems from these requirements. Therefore, STP has requested an exemption from 10CFR50.55a(h) with respect to sections 4.3 and 4.4 of IEEE 279 in order to allow exemption of LSS and NRS components from these special treatment requirements. STP would continue to meet the other requirements listed in IEEE 279, including functional and design requirements.

RESPONSE (part b):

Manual initiation components included in the scope of IEEE 279 that have been risk ranked by STP consist of handswitches. STP is using the convention of risk ranking control room handswitches one level lower than the controlled component, except that if the controlled component is LSS, the handswitch must also be LSS. The basis for this convention is contained in a set of general notes, which have been approved by the Expert Panel. For control room handswitches, the general notes provide the following justifications:

> **"Reliability of handswitches has been very good. Local and/or Auxiliary Shutdown Panel redundant switch available. Most time sensitive operations are automatic, do not require switch manipulation, and rely only on handswitch circuit continuity for success. The probability of a circuit continuity failure in a static role is very low and is clearly less than the probability of failure for the controlled component. Automatic safety systems are periodically tested. These tests include the automatic initiation circuitry and the manipulation of these handswitches. In addition, handswitches are manipulated on a regular basis as part of routine operations. Any failures in the handswitch or its associated electrical circuitry would manifest themselves during these operations."**

Under this convention, handswitches used for the manual initiation of protective systems could be ranked LSS if the controlled component is MSS. These handswitches would be exempt from the special treatment requirements in IEEE 279. The technical basis for this is as follows:

1. **The handswitches would continue to meet all other requirements of IEEE 279, including design requirements.**
2. **The experience of STP and the industry with handswitches has shown them to be very reliable. Comparisons of failure rates for safety related vs. non-safety related handswitches both at STP and in the industry have been performed. Results show that the failure frequency for non-safety related handswitches is no greater than that for safety related handswitches. Details on this review can be found at the end of the response to this question.**
3. **A handswitch is a typically rugged component that is unlikely to be affected by seismic conditions.**
4. **All of the handswitches within the scope of IEEE 279 are located in a mild environment and therefore would not be subject to specific environmental qualification requirements.**
5. **Plant systems are periodically tested. The scope of these tests includes the operation of handswitches, such as these. If any malfunction occurred, it would be captured in the performance and feedback process and evaluated for impact on risk significance.**
6. **The primary method of actuating protective systems is through automatic means. Handswitches are provided only as backup. If both the automatic initiation and the main backup control room handswitch failed, redundancy would be available via redundant handswitches located in one or more of the following locations: Control Room, Auxiliary Shutdown Panel, or Transfer Panels.**

As stated earlier, the STP convention for risk ranking handswitches is contained in a set of general notes that promote consistency in the risk ranking process for similar components. However, where appropriate, the Working Group can recommend and the Expert Panel can approve risk rankings that are more conservative than those provided for in the general notes. For example, in the Residual Heat Removal system, some control room handswitches were ranked the same as the controlled component due to their support of the manual start and/or alignment of the system. The table at the end of the response to this RAI question lists handswitches that have been categorized as HSS or MSS. As can be seen from the last column, most of these handswitches have been categorized more conservatively than the general note guidelines.

<u>Results Of Reviews To Compare Reliability Of Safety Related Versus Non-Safety Related Handswitches</u>

STPNOC asserts that, for components within the scope of this exemption request, non-safety-related component failure rates are not appreciably greater than corresponding safety-related component failure rates for similar component types. To support this assertion, STPNOC has performed a data analysis of Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX)

data. Nuclear industry data reporting to the Nuclear Plant Reliability Data System (NPRDS) spans the time period from 1977 through 1996. The EPIX Maintenance Rule and Reliability Information (MRRI) database includes component failure data since 1996. NPRDS component engineering data includes indication of safety class, thus enabling a distinction between safety-related component and non-safety-related component failure rates. While the MRRI database does not include a safety-class distinction, INPO was able to provide STPNOC an MRRI database file for 1997-1999 data that is "back-linked" to NPRDS, thus providing indication of safety class. The NPRDS data and MRRI data were first analyzed separately then merged to provide a large-scope analysis to support responses for the STPNOC GQA RAIs.

The scope of this merged NPRDS-MRRI analysis included consideration of over 670,000 component records and over 166,000 component failure records for those components. For RAI Item 10, this analysis included consideration of the circuit breaker (NPRDS/MRRI component ID code CKTBRK), which, for this analysis, is assumed to subsume all safety-related and non-safety-related hand switches included in the NPRDS and MRRI databases. Analysis shows that the calculated safety-related CKTBRK failure frequency, 8.36E-07 functional failures per calendar hour, is actually greater than the non-safety-related CKTBRK failure frequency, 7.57E-07 functional failures per calendar hour, based on historical merged NPRDS-MRRI data. The relative difference between these two values is well within the normal range factor (approximately 3) for this type of failure frequency parameter and is not significant.

The results of this analysis have shown that, in general, nuclear power plant non-safety related equipment failure frequencies are no greater than or roughly equivalent to those for corresponding types of safety-related equipment. The failure data contained in EPIX and NPRDS cannot be construed to be a complete data set for non-safety related nuclear power plant components because there has been no requirement to supply this failure data. However, given the volume of information available, the overall conclusions of the data analysis task are considered to be valid.

In addition to the analysis of the data contained in the EPIX database, STPNOC has performed limited data collection in support of an on-going Balance-of-Plant (BOP) model. The data collected covers active equipment necessary to support power production (e.g., feedwater and condensate pumps). While not directly applicable to handswitches, the collected data indicates no apparent difference in the failure rates for normally operating motors between safety and non-safety related equipment. These results support the conclusions of the data analysis of the EPIX data.

HANDSWITCHES RANKED HIGH OR MEDIUM (U1 TRAIN A ONLY)

SYSTEM	COMP_ID	COMP_DESC	RISK	HIGHER THAN GEN NOTE
AUX FW	A1AFHS0048A	(AFW 11) OCIV MOV-0048	HIGH	YES
AUX FW	A1AFHS7506A	AFW PUMP 11	HIGH	YES
AUX FW	A1AFHS7525	S/G 1A AFW FLOW CONTROL	HIGH	YES
AUX FW	A1AFHS7525C	AFW FLOW CONT FV-7525 (RESET)	HIGH	YES
AUX FW	D1AFHS0019A	(AFW 14) OCIV MOV-0019	HIGH	YES
AUX FW	D1AFHS0143A	MN STM ISOL MOV-0143	HIGH	YES
AUX FW	D1AFHS0514A	TURB TRIP/THROT MOV-0514	HIGH	YES
AUX FW	D1AFHS7526	S/G 1D AFW FLOW CONTROL	HIGH	YES
AUX FW	D1AFHS7526C	AFW AUTO FLOW CONT FV-7526 (RESET)	HIGH	YES
ESS CHILLED WATER	A1CHHS0710	EAB CHILLER VCH001 AUX OIL PUMP	HIGH	YES
DIESEL GEN	A1DGHS5499E	DIESEL GENERATOR #11ILOCAL/REMOTE HAND SWITCH	HIGH	YES
DIESEL GEN	A1DGHS5499L	DIESEL GENERATOR #11IIN PARALLEL MODEIHAND SWITCH	HIGH	YES
DIESEL GEN	A1DGHS5499N	DIESEL GENERATOR #11IEMERGENCY MODE RELEASEIHAND SWITCH	HIGH	YES
MAIN STEAM	A1MSHS7411	S G 1A POWER-OPERATED RELIEF VALVE TRANSFER SW	HIGH	YES
MAIN STEAM	D1MSHS7441	S G 1D POWER OPERATED RELIEF VALVE	HIGH	YES
RC	A1RCHS0001A	RC PZR POWER OPERATED RELIEFIVALVE ISOLATION VALVE MOV-0001A	HIGH	YES
RC	A1RCHS0001D	RC PRZR 1A POWER-OPERATEDIRELIEF VLV BLOCK VLVIMOV-0001A XFER SW	HIGH	YES
RC	A1RCHS0655A	RCS PRZR 1A POWER-OPERATED RELIEF VALVE PCV-0655A CONTROL ROOM HAND SWITCH	HIGH	YES
RC	A1RCHS0655C	RCS PRZR 1A POWER-OPERATEDIRELIEF VALVE PCV-0655A TRANSFIHSW	HIGH	YES
RHR	A1RHHS0060A	PMP 1A SUCT ISOL MOV-0060A	HIGH	YES
RHR	A1RHHS0061I	RHR SUCTION ISOLATION VALVE HANDSWITCH TRAIN A	HIGH	YES
RHR	A1RHHS0867	RHR PUMP 1A	HIGH	YES
AUX FW	A1AFHS0048B	AFW PUMP NO 11 ISOLATION VALVE TRANSFER SWITCH	MEDIUM	YES
AUX FW	A1AFHS7506B	AFW PUMP 11 TRANSFER SWITCH	MEDIUM	YES
AUX FW	A1AFHS7517A	X CONN FV-7517	MEDIUM	YES
AUX FW	A1AFHS7525A	AFW REG FV-7525	MEDIUM	YES
AUX FW	A1AFHS7525B	S/G 1A AFW FLOW CONTROL TRANSFER SWITCH	MEDIUM	YES
AUX FW	D1AFHS0019B	AFW PUMP NO 14 ISOLATION VALVE TRANSFER SWITCH	MEDIUM	YES
AUX FW	D1AFHS0019C	AFW PUMP NO 14 ISOLATION VALVE LOCAL CONTROL	MEDIUM	YES
AUX FW	D1AFHS0143B	AFWP 14 TURBINE ISOLATION VALVE TRANSFER SWITCH	MEDIUM	YES
AUX FW	D1AFHS0143C	AFWP 14 TURBINE ISOLATION VALVE LOCAL CONTROL	MEDIUM	YES
AUX FW	D1AFHS0514B	AUX FW PUMP 14 TURBINE THROTTLE VALVE OPEN/CLOSE	MEDIUM	YES
AUX FW	D1AFHS0514C	AFWP 14 TURBINE THROTTLE VALVE TRANSFER SWITCH	MEDIUM	YES
AUX FW	D1AFHS7518A	X CONN-7518	MEDIUM	YES
AUX FW	D1AFHS7526A	AFW REG FV-7526	MEDIUM	YES
AUX FW	D1AFHS7526B	S/G 1D AFW FLOW CONTROL TRANSFER SWITCH	MEDIUM	YES
AUX FW	D1AFHS7537A	AFW TURB MOV-514 (TRIP)	MEDIUM	YES
CCW	A1CCHS4509	COMP CLG WTR PUMP 1A	MEDIUM	
CVCS	A1CVHS0025	CVCS CHG FLW MOV-0025 SEL HSW	MEDIUM	YES
CVCS	A1CVHS0082	RCS EXCESS LETDOWN HEAT EXCHANGEI1A INL MOV-0082 SEL HSW	MEDIUM	YES
CVCS	A1CVHS0465	RCS LETDOWN STOP LCV-0465 SELIHSW	MEDIUM	YES
CVCS	A1CVHS0465B	RCS LETDOWN STOP LCV-0465 RMTITRANS SW	MEDIUM	YES
CVCS	N1CVHS0286	POS DISP CHG PUMP 1A SEL SW	MEDIUM	
DIESEL GEN	A1DGHS5499M	DIESEL GENERATOR #11IEMERGENCY STARTIDUAL SWITCH	MEDIUM	
ECW	A1EWHS6880	ECW PUMP A HAND SWITCH TRAIN A	MEDIUM	
ECW	A1EWHS6880A	ECW PUMP A HAND SWITCH TRAIN A	MEDIUM	YES
ECW	A1EWHS6880B	ECW PUMP A HAND SWITCH TRAIN A	MEDIUM	YES
HVAC-CONT.	A1HCHS9744	(AFW PUMP AREAS) PUMP 11 SPLY FAN 11A HC-VFN001	MEDIUM	
HVAC-FHB	A1HFHS9517	(LHSI/HHSI/CCS PUMP CUBICLES) TRN A / RM 006 SUPP CLR 11A HF-VAH004	MEDIUM	
HVAC-MAB	A1HMHS9406	(ESSEN CHILLER AREA) TRN A/RM 067 SUPP CLR 11A HM-VAH019	MEDIUM	
HVAC-MAB	A1HMHS9406B	ESSEN CHLR AREA ROOM 067 SUPPL CLR 11A CONT TRANSFER	MEDIUM	YES
HVAC-MAB	A1HMHS9409	(CCW PUMP CUBICLES) PUMP 1A/RM 067 SUPP CLR 11A HM-VAH001	MEDIUM	
HVAC-MAB	A1HMHS9409A	CCW PUMP 1A ROOM 067 SUPPL CLR 11A	MEDIUM	YES
HVAC-MAB	N1HMHS9419	TIE DAMPER FV-9419	MEDIUM	YES

HANDSWITCHES RANKED HIGH OR MEDIUM (U1 TRAIN A ONLY)

SYSTEM	COMP_ID	COMP_DESC	RISK	HIGHER THAN GEN NOTE
RC	A1RCHS0001C	RC PRZR 1A POWER-OPERATEDIRELIEF VLV BLOCK VLVIMOV-0001A LOCAL SW	MEDIUM	YES
RC	A1RCHS0601	RCS RX HD VENT THROT VALVEIHCV-0601 TRANSF SW	MEDIUM	YES
RC	A1RCHS0655	RCS COLD OVER PRESSURE MITIGATION ACTUATION CONTROL ROOM HAND SWITCH	MEDIUM	
RC	A1RCHS0655B	RCS PRZR 1A POWER-OPERATEDIRELIEF VALVE PCV-0655A LOC SW	MEDIUM	YES
RC	A1RCHS3657A	RX VESSEL HEAD VENT ISOLATION VALVE HV-3657A CONTROL ROOM HAND SWITCH	MEDIUM	
RC	A1RCHS3658A	RX VESSEL HEAD VENT ISOL VLV HV-3658A CONTROL ROOM HAND SWITCH	MEDIUM	
RHR	A1RHHS0060D	RHR SUCTION ISOLATION VALVE HANDSWITCH TRAIN A	MEDIUM	YES
RHR	A1RHHS0060G	RHR SUCTION ISOLATION VALVE HANDSWITCH TRAIN A	MEDIUM	YES
RHR	A1RHHS0061F	RHR SUCTION ISOLATION VALVE HANDSWITCH TRAIN A	MEDIUM	YES
RHR	A1RHHS0867A	RHR PUMP 1A REMOTE/LOCAL HANDSWITCH	MEDIUM	YES
RHR	A1RHHS0867B	RHR PUMP 1A SWITCHGEAR ROOM HANDSWITCH	MEDIUM	YES
SI	A1SIHS0011A	MINI FLOW ISOL MOV-0011A	MEDIUM	
SI	A1SIHS0012A	MINI FLOW ISOL MOV-0012A	MEDIUM	
SI	A1SIHS0013A	MINI FLOW ISOL MOV-0013A	MEDIUM	
SI	A1SIHS0014A	MINI FLOW ISOL MOV-0014A	MEDIUM	
SI	A1SIHS0016A	CNTMT SUMP TO SI SUCT HDR ISOL MOV-0016A	MEDIUM	
SI	A1SIHS0031A	LOOP A TC INJ MOV-0031A	MEDIUM	
SI	A1SIHS0957	HHSI PUMP 1A	MEDIUM	
SI	A1SIHS0967	LHSI PUMP 1A	MEDIUM	

11. The licensee's exemption request states that "LSS components generally include piping, locked open valves, hand switches, and outside containment isolation valves sized 3" and under [emphasis added]." Please describe the process for categorizing containment isolation valves (CIVs). Describe what special treatment will be applied to these LSS valves to ensure that they remain functional. Alternatively, the licensee could provide an analysis of the effect of degraded containment isolation valve performance/reliability on the probability of an inter-system loss of coolant accident (ISLOCA) and large early release frequency (LERF) as was done by the licensee to extend inservice test intervals for CCW and SI system CIVs (Reference NRC Safety Evaluation dated July 23, 1999).

RESPONSE:

A clarification is in order concerning the risk categorization of inboard and outboard containment isolation valves. STP assigns the same risk categorization to both the inboard and outboard containment isolation valves at a particular system location. Furthermore, although the size of the line is a consideration, there is no size threshold that automatically dictates a particular risk significance. Any indication to the contrary was erroneous. The process for categorizing containment isolation valves is given below.

Containment isolation valves are typically categorized as LSS and meet one or more of the following criteria:

a. **The valve is required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers).**
b. **The valve is normally closed and in a physically closed, water-filled system. (e.g., containment isolation valves in the Demineralized Water system)**
c. **The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Component Cooling Water system and in the Instrument Air system).**
d. **The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve.**
e. **The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).**

As an atypical system, the Reactor Containment Building HVAC system did not meet the above criteria and its containment isolation valves were categorized as MSS. It is an air-to-air system. The line (duct) size is large and failure to isolate concurrent with a purging operation could lead to a radiation release. For additional information, see the response to RAI question 16.

For details regarding the treatment of LSS containment isolation valves, please refer to the response to question 34 (c).

With regard to PRA analyses on this issue, STP notes that our PRA model for an inter-system loss of coolant accident (ISLOCA) includes analysis of 48 valves. The definition of this event is a failure of the isolation valves between the reactor coolant system and a lower pressure interfacing system that leads to a primary coolant leak that bypasses containment. ISLOCA does not involve containment penetrations 3" and under.

The systems and penetrations associated with the ISLOCA analysis are the high and low head safety injection discharge lines to containment, and the component cooling water inlet and outlet from the residual heat removal heat exchanger. Of the 48 valves that contribute to the ISLOCA frequency, only 18 are containment isolation valves. Only 9 of the 18 containment isolation valves are ranked low by the STP risk categorization process. The following table represents a breakdown of the risk ranking for containment isolation valves whose failure could lead to an ISLOCA.

UNIT 1 TAG/TPNS	SERVICE_DESC	PRA Rank	GQA Rank
2N121XSI0018A	LHSI PUMP 1A DISCHARGE MOV (SI-MOV-0018A)	Medium	Medium
2N121XSI0018B	LHSI PUMP 1B DISCHARGE MOV	Medium	Medium
2N121XSI0018C	LHSI PUMP 1C DISCHARGE MOV (SI-MOV-0018C)	Medium	Medium
2N121XSI0030A	(IRC) LHSI PUMP 1A DISCH CHECK VALVE	Medium*	High
2N121XSI0030B	(IRC) LHSI PUMP 1B DISCH CHECK VALVE	Medium*	High
2N121XSI0030C	(IRC) LHSI PUMP 1C DISCH CHECK VALVE	Medium*	High
2R201TCC0012	CC-MOV-0012 (CCW SUPPLY TO RHR "A" MOV)	Low	Low
2R201TCC0013	M-33 CHECK VALVE (CCW TO RHR PUMP SEAL COOLERS AND RHR HEAT EXCHANGERS)	Medium*	Medium
2R201TCC0049	(IRC) CCW FROM "A" RHR ISOLATION (LOCAL HANDWHEEL)	Low	Low
2R201TCC0050	(OCIV) CC-MOV-0050 CCW FROM TRAIN "A" RHR OCIV	Low	Low
2R201TCC0122	(OCIV) CC-MOV-0122 (CCW B SUPPLY TO RHR OCIV)	Low	Low
2R201TCC0123	(IRC) ICIV FOR CCW TO B TRAIN RHR COMPONENTS	Medium*	Medium
2R201TCC0129	(ICIV) CC-MOV-0129 (CCW B FROM RHR HEADER ICIV)	Low	Low
2R201TCC0130	CC-MOV-0130 (CCW B FROM RHR CONTAINMENT ISO MOV)	Low	Low
2R201TCC0182	CC-MOV-0182 (CCW C SUPPLY TO RHR MOV)	Low	Low
2R201TCC0183	(IRC) CCW RHR HX 1C INLET CHECK VALVE	Medium*	Medium
2R201TCC0189	(IRC) CC-MOV-0189 (CCW C FROM RHR HEADER ISO MOV)	Low	Low
2R201TCC0190	CC-MOV-0190 (CCW C SUPPLY FROM RHR)	Low	Low
Medium* represent components with RAW between 10 and 100.			

In response to RAI question number 21, a sensitivity study was performed to show the impact of postulating increased failure rates (i.e., increased by a factor of 10) for low ranked components to the CDF and LERF. Components analyzed in this study encompass the 9 LSS containment isolation valves analyzed for contributing to an ISLOCA. The impact to the annual average CDF and LERF of increased failure rates for all LSS components are as follows:

	Current Average (events/reactor year)	Sensitivity Study λ_{LSS}*10 (events/reactor year)	Increase	% Increase
CDF	9.0781E-6	9.3232E-6	2.4510E-7	2.7%
LERF	1.3742E-7	1.3911E-7	1.6900E-9	1.2%

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF). These results show that the aggregate effects of increased failure rates are well within acceptance guidelines and are consistent with the Commission's Safety Goal Policy.

The above discussion demonstrates that the overall effects from increased failure rates for LSS components do not impose adverse safety conditions. In no way is the above discussion intended to imply that STP would find it acceptable to allow failure rates to increase by a factor of 10. It is STP's intent to maintain all equipment functional regardless of risk category. The risk categorization process enables different maintenance and testing strategies to be employed depending on a component's risk significance. Components with low safety significance are maintained with a repair or replace as needed philosophy (See also the response to RAI #34). Thus, for the 9 LSS containment isolation valves, these components will be maintained when degradation is identified (i.e., when leakage is noted during routine valve isolation activities in support of maintenance tasks). For LSS components, this represents an appropriate maintenance strategy which is commensurate with their safety significance level.

Other deterministic factors are also important to note with regard to ISLOCA. In order for an ISLOCA to occur, multiple failures of equipment and components must happen at nearly the same times. Normally closed valves must fail in addition to piping failures, heat exchanger failures, etc.

The ISLOCA event at STP is not a significant contributor to CDF or LERF. Combined with the testing and maintenance strategy described above, this provides a proper risk informed approach for these containment isolation valves.

12. *On page 8 of Attachment 1 to the July 13, 1999, submittal, the licensee stated that LSS and NRS Class 1E electrical equipment could be replaced with non-Class 1E equipment. Based on information conveyed by licensee representatives during an August 31, 1999, meeting, it is our understanding that electrical systems would be excluded at this time from the risk-informed treatment process due to cost/benefit considerations. Please clarify the extent to which electrical SSCs are proposed to be exempted from the special treatment regulations, and modify the exemption request as appropriate.*

RESPONSE:

STP is proposing to exempt LSS and NRS components from the scope of the special treatment regulations.

Upon grant of the exemption, STP will not apply the special treatment requirements to currently installed LSS and NRS components. For replacement activities, safety-related electrical equipment will be evaluated to determine the extent that special treatment requirements can be eliminated. Additionally, even though we will apply our commercial practices to procurement of replacement components, we will treat the replacement commercial device as Class IE by performing an engineering evaluation and procuring it with the required design and functional attributes commensurate with an IE classification. Where this cannot be achieved, the component will be procured as a fully qualified, safety related item.

The following justifications are provided to support our position:

- **Loss of the function of an LSS or NRS electrical component (e.g., failure of a motor operator to provide motive force to its associated valve when demanded) has been determined through the risk categorization process to have little or no impact on protecting the core, mitigating the consequences of an accident, or protecting the public health and safety.**

- **For handswitches, where the associated end device has been risk categorized as MSS and the handswitch has been ranked at a lower risk, STP will confirm and document that credible failure(s) of the handswitch could not prevent the associated end device from performing its MSS function(s) or that cause the loss of other critical required attributes. In the case where this cannot be confirmed, STP will revise the risk categorization of the handswitch to match that of the end device. For additional discussions on handswitches, please refer to the response to Question 10.**

- **When the need arises, STP may replace an existing LSS or NRS Class IE component with a commercial grade component that meets the existing functional design requirements, including environmental parameters. Procuring a commercial component is accomplished by performing an engineering evaluation to provide reasonable assurance that the replacement component satisfies the required form, fit, and function, will not degrade other IE equipment, and is capable of meeting the design functional requirements (including seismic and environmental considerations); however, the component will not be specifically qualified. If a replacement commercial**

grade component does not satisfy all of the design functional requirements applicable to a Class IE component, it will be classified as Non IE, and the replacement device will be properly isolated from the Class IE circuitry. The circuit downstream of the required isolation device(s) would be also classified as Non IE and would properly separated from Class 1E circuitry. Where the above is not feasible, the component will be procured as a fully qualified, safety related item.

- **Upon grant of the exemption, by purchasing functionally equivalent replacement components from commercial grade vendors, reasonable assurance is provided that the new component will perform is design function and will not challenge the IE distribution system.**

- **The above measures will provide reasonable assurance, commensurate with the risk significance, that LSS and NRS electrical components will be able to satisfactorily perform their design functions and will not electrically degrade other components or challenge the IE distribution system. Therefore, STP intends to continue to identify safety related LSS and NRS electrical components as Class IE, similar to maintaining the safety related classification for LSS and NRS safety related mechanical components.**

- **For cabling, STP intends to maintain current separation requirements. Therefore, non-IE circuit cabling will be properly separated from IE circuits.**

- **In order for a failure of the LSS and NRS component to have a risk significant impact on the plant, the following must occur at the same time:**

  - **The replacement component must fail in such a way as to affect the upstream electrical circuitry.**

  - **Any fully qualified Class IE isolation device(s) must fail to protect the upstream circuits.**

  - **The above two failures occur during a design basis event.**

  STP does not consider this scenario to be credible.

- **Finally, STP will perform monitoring to provide an appropriate level of assurance that LSS and NRS electrical components will perform in accordance with the assumptions in the PRA. All equipment necessary to mitigate the consequences of initiating events is included in the plant PRA. Changes to the risk significance of components included in the PRA will not result in removal of the equipment from the model. As component replacements are made, changes in equipment failure rates, if they occur, will be identified by the Corrective Action Program or the Maintenance Rule Program (if failure of the LSS or NRS SSC could affect an HSS/MSS function) and the new failure rates incorporated into the PRA model during the cycle updates. Requantification of the model with the changed failure rates may result in a change to the components' risk ranking. In addition, any change in equipment performance will be deterministically**

evaluated during the periodic reviews by the GQA Working Group. These evaluations may also result in a change to the components' risk ranking.

Based on industry and STP data, the failure rates for most equipment whose special treatment requirements are relaxed are not expected to change significantly. Therefore, we expect no impact on plant risk in terms of core damage frequency or large early release frequency as a result of the exemption. Notwithstanding this conclusion, any change to core damage frequency or large early release frequency is expected to be gradual and detectable before a significant impact to core damage

Additional Justification

A sensitivity study was performed to show the impact of postulating increased failure rates for low ranked components to the CDF and LERF. The approach of the study was to increase the component failure rate by a factor of 10 for all components ranked LSS. Note, by definition all components credited in the plant specific PRA for accident/transient mitigation are ranked at least LSS. There are 431 component categorized as LSS and which are modeled in the PRA. The results are as follows:

	Current Average (events/reactor year)	Sensitivity Study λ_{LSS}*10 (events/reactor year)	Increase	% Increase
CDF	9.0781E-6	9.3232E-6	2.4510E-7	2.7%
LERF	1.3742E-7	1.3911E-7	1.6900E-9	1.2%

In all cases, increasing the failure rates of LSS components by a factor of 10 was greater than the 95th percentile for each of the LSS component failure rate distributions.

The above increases in CDF and LERF are within the acceptance guidelines for very small changes as outlined in Regulatory Guide 1.174. The acceptance guidelines are 1E-6 delta CDF and 1E-7 delta LERF. Results from this study are small and consistent with the intention of the Commission's Safety Goal Policy Statement.

The sensitivity study incorporating a factor of 10 increase in failure rates for all LSS equipment (including Class 1E equipment) modeled in the PRA is believed to represent a conservative "bounding case" because there is evidence that component failure characteristics for non-safety-related equipment do not differ significantly from those for safety-related equipment. STPNOC asserts that, for components within the scope of the STPEGS Graded QA Program, non-safety-related component failure frequencies are not appreciably greater than corresponding safety-related (including Class 1E) component failure frequencies for similar component types. To support this assertion, STPNOC has performed a data analysis of Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX) data. Nuclear industry data reporting to the Nuclear Plant Reliability Data System (NPRDS) spans the time period

from 1977 through 1996. The EPIX Maintenance Rule and Reliability Information (MRRI) database includes component failure data since 1996. NPRDS component engineering data includes indication of safety class, thus enabling a distinction between safety-related component and non-safety-related component failure frequencies. While the MRRI database does not include a safety-class distinction, INPO was able to provide STPNOC an MRRI database file for 1997-1999 data that is "back-linked" to NPRDS, thus providing indication of safety class. The NPRDS data and MRRI data were first analyzed separately then merged to provide a large-scope analysis to support responses for the STPEGS exemption request RAIs. STPNOC has developed a report, entitled "Safety-Related Versus Non-Safety-Related Equipment Failure Frequency Data Analysis for Nuclear Power Plants in the United States" dated April 6, 2000, describing this NPRDS-MRRI data analysis. This report is available upon request.

The scope of this merged NPRDS-MRRI analysis included consideration of over 670,000 component records and over 166,000 component failure records for those components. The historical data analyzed consisted of over 74 billion component-hours of experience. Tables 1 and 2 (attached) from the response to RAI #42 provide analysis results information for all 33 component type data categories contained in the merged NPRDS-MRRI database. These tables show that the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS-MRRI data. This analysis shows that, of 33 component type categories investigated, 21 had higher safety-related failure frequency values than corresponding non-safety-related categories. Non-safety-related failure frequency values were significantly higher than corresponding safety-related failure frequencies in only one of the 33 categories (the "containment penetration" component type category). The analysis shows that, for most component types, the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data.

An argument often made in this type of comparison is that there is more safety-related component experience in the database than non-safety-related component experience. This is valid. However, the failure frequency parameters, calculated simply in terms of reported failures per component-hour of experience in this analysis, are being compared on a consistent basis. For example, in the circuit breaker component type category, the following failure frequencies are determined with respect to safety classification:

For safety-related circuit breakers

- **Number of failures: X_{SR} = 6,457**
- **Number of operating hours, Y_{SR} = 7,723,785,888**

This yields a failure frequency of:

$$X_{SR}/Y_{SR} = 6{,}457/7{,}723{,}785{,}888 = 8.36E-7 \text{ failures per operating hour}$$

Similarly, for non-safety-related circuit breakers

- **Number of failures: X_{NSR} = 1,345**
- **Number of operating hours: Y_{NSR} = 1,777,678,176**

This yields a failure frequency of:

$$X_{NSR}/Y_{NSR} = 1{,}345/1{,}777{,}678{,}176 = 7.57E-7 \text{ failures per operating hour}$$

One can conclude that we have a greater degree of confidence that the historical failure frequency for safety-related circuit breakers represents the "true" failure frequency (calculated for infinite experience), than we do for the non-safety-related circuit breakers. However, in this case, there are large numbers of component-hours of experience for both safety-related and non-safety-related components, indicating that we have relatively high confidence in both results.

Another way of looking at this is that, if we were to "scale" the safety-related experience down to the non-safety-related experience level, we would multiply both the component-hours of experience and the reported failure count by the ratio of non-safety-related to safety-related component-hours of experience i.e.,

$$Scale = 1{,}777{,}678{,}176/7{,}723{,}785{,}888 = 0.23$$

If we do this, we get the similar results as with the actual experience numbers, i.e.,

$$Scaled\ Number\ of\ NSR\ Failures = Scale * X_{SR} = 0.23 * 6{,}457 = 1486 \approx 1345$$

Likewise, we would get the similar results if we were to scale the non-safety-related experience up to the safety-related experience. That is, if we increase or decrease the component-hours of experience for a component type category of interest in the database by some factor, we would expect to have a higher or lower number of reported failures by the same factor.

Another simple conservative example will show how low the projected frequency of these types of fault back-propagation events are at STPEGS. First, there are several types of circuits supplying Class 1E loads that are LSS based on the PRA and on the Working Group/Expert Panel evaluations. Most of these types of circuits have multiple protection devices (i.e., circuit breakers, fuses, isolation transformers, etc.) between the endpoint load (assumed to be the source point of the electrical fault being propagated in our example)

and an MCC or load center where multiple cascading failures could be experienced as a result of the initial fault. In the case of some MOVs and fans at STPEGS, there is only one breaker between the endpoint motor and an MCC where multiple faults could conceivably be experienced as a result of a fault generated in the MOV motor or associated local power or control circuitry. We conservatively choose this case for our example here. In our example, a plant "initiating event" from the PRA must occur to cause a demand of the target LSS MOV in our example. From the current STPEGS PRA database, each unit at STPEGS can expect to, on average, experience 3.98 initiating events per year (4.54E-04 events per hour). Only a portion of these events may actually require a demand of our target MOV, but we will conservatively assume that all of them do. In the PRA model, an MOV usually only has to change state once in the event sequence, therefore, only a fraction of the total failure modes possible for the component will be modeled, but we will conservatively assume that all failure modes quantified in the PRA database for MOVs apply in our example (i.e., could result in an event leading to a cascading electrical fault). Studies performed for "hot short" type failures show that only a fraction of the total failure mode events for an MOV would be expected to result in the type of cascading electrical fault of interest here. We will conservatively assume, in this example, that all the associated failure mode events of interest will also result in hot short conditions that could be propagated as a cascading electrical fault (this, again, is very conservative). The current STPEGS PRA database shows the following mean values for MOV failure rates for failure modes that could be associated with initiation of a cascading electrical fault:

MOV failure to open on demand = 1.25E-03
MOV failure to close on demand = 1.07E-04
MOV transfers open (or closed) per hour = 2.99E-07

In our conservative example, there is only one protective device (a Class 1E circuit breaker that will remain Class 1E under the proposed STPNOC Exemption Request) between the potential source of the cascading electrical failure and other power or control circuitry for other components that could be safety-related. The current STPEGS PRA database shows the following mean values for circuit breaker failure rates for failure modes that could be associated with propagation of a cascading electrical fault:

Breaker failure to open on demand = 2.16E-04
Breaker failure to close on demand = 1.06E-03
Breaker transfers open (or closed) per hour = 7.25E-07

Based on this information, a very conservative upper bound of the total frequency of generation of a cascading electrical fault from our MOV circuit of interest back to a point in STPEGS circuitry that could result in a potential cascading failure of additional equipment supplied by the MOV's MCC can be estimated as follows:

Frequency of fault propagation

$$(3.98/8766 * (1.25E-3 + 1.07E-4) + 2.99E-7) * (2.16E-4 + 1.06E-3) + 7.25E-7$$
$$= 7.26E-7 \; events \; per \; hour$$
$$= 6.37E-3 \; events \; per \; year$$

This frequency is strongly dominated by the local failure of the breaker itself, not from faults caused by demands based on the MOV failure or the PRA initiating event. If we conservatively assume that all the MOV failure rates will increase by a factor of 10 after being replaced by commercial grade equipment, the new total frequency of fault propagation can be estimated as follows:

Frequency of fault propagation

$$(3.98/8766 * (1.25E-3 + 1.07E-4) + 2.99E-7) * 10 * (2.16E-4 + 1.06E-3) + 7.25E-7$$
$$= 7.37E-7 \; events \; per \; hour$$
$$= 6.46E-3 \; events \; per \; year$$

This represents a "delta" frequency of approximately 1E-08 events per hour, or about 9E-05 events per year for our conservative example. Recall that, from the safety-data versus non-safety-related failure frequency discussion above (and in RAI item 42), assuming a factor of 10 increase in failure rate for commercial grade equipment over current Class 1E equipment is not justified (i.e., is grossly conservative) based on available objective evidence from the INPO EPIX database, which indicates that failure frequencies for all the components of interest in our example here are within a factor of less than 2. Thus, the impact of implementing risk categorization at STPEGS could be expected to increase the "back-propagation" of electrical faults by approximately 1.4%, at the most. This conservative example shows that implementing risk categorization at STPEGS is expected to have an insignificant impact on back-propagation of electrical faults at the station.

In conclusion, STP proposes to exempt LSS and NRS Class 1E electrical components from the scope of special treatment requirements subject to the bulleted items provided at the beginning of this response (e.g., component has little or no impact on mitigating accidents/transients, etc.). STP has performed the following sensitivity studies in support of this question:

1. **The overall impact of postulated failure rate increase is within the acceptance guidelines as outlined in Regulatory Guideline 1.174;**
2. **An evaluation of safety-related versus non-safety-related equipment failure frequencies demonstrates, for most component types, that the failure frequencies are higher for safety-related components;**
3. **It has been demonstrated that if we "scale" down safety-related experience to non-safety-related experience level, the number of equipment failures do not appreciably change; and**

4. **It was conservatively demonstrated that the frequency of occurrence of "back-propagating" electrical faults could result in an increase of only 1.4% which is not an appreciable change.**

Therefore, there is confidence that removing special treatment requirements for LSS and NRS Class 1E equipment would not unduly hinder the ability of the plant to prevent or mitigate the consequences associated with accidents or transients.

13. *With respect to the proposed Appendix B exemptions:*

    (a) *Provide an amplified description of the proposed commercial quality practices that will be used by the licensee (and by the licensee's vendors) to serve as an alternative to each of the 15 Appendix B criteria for which an exemption is requested.*

    (b) *Provide an expanded discussion about how these commercial quality practices will provide reasonable assurance that safety-related LSS/NRS equipment will reliably perform their design functions.*

    (c) *Appendix B, Criterion IV specifies that measures shall be established to assure that applicable design requirements are suitably included in procurement documents. Licensees rely on purchase orders to convey design requirements to vendors so that replacement parts will continue to reliably function under design conditions. Please justify why a complete exemption from Criterion IV is appropriate, given the importance of procurement documents in ensuring conformance of procured equipment with applicable design requirements. Describe in detail what measures will be imposed to ensure that design requirements are met.*

RESPONSE:

In the exemption request, STP requested exemption from 15 of the 18 Appendix B criteria. The three criteria for which exemption was not requested included:

- **Criterion III – Design Control**
- **Criterion XV – Nonconforming Materials, Parts, or Components, and**
- **Criterion XVI – Corrective Action**

For the remaining 15 criteria, STP proposes the following commercial quality practices as an alternative to that specified in Appendix B. It should be remembered that exemption is only requested for components that are risk categorized as LSS and NRS. LSS and NRS components, by definition, serve little, if any, function in mitigating the consequences of an accident or protecting the health and safety of the public during a design basis event or any other credible event. With this in mind, the following commercial practices for NRS and LSS components will be followed:

- **Criterion I (Organization) – the Quality organization will focus on HSS/MSS components, including non-safety related HSS/MSS components and will not be required to provide oversight for LSS/NRS components or activities. The Comprehensive Risk Management Expert Panel provides organizational oversight for the categorization of SSCs and for the implementation of risk-informed activities at STP. The GQA Working Group provides oversight for the categorization of SSCs and monitors the implementation feedback for potential adjustments in controls or categorization.**

- **Criterion II (Quality Assurance Program) – the Operations Quality Assurance Program (OQAP) will be modified to focus on HSS/MSS SSCs. Commercial programs, procedures, and practices (i.e., Balance of Plant) are in place to provide appropriate controls over activities affecting LSS and NRS components. These processes have been proven to provide satisfactory controls to provide an appropriate level of assurance that Balance-of-Plant equipment operates safely and reliably. Likewise, these processes will provide reasonable assurance that LSS/NRS SSCs can perform their design basis functions. The implementation of these activities will be under the oversight of the Expert Panel, who will receive input from the GQA Working Group, other Working Groups, and plant staff.**

- **Criterion IV (Procurement Document Control) – Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.**

- **Criterion V (Instructions, Procedures, and Drawings) – appropriate procedures, instructions, and drawings are in place, and will be used, as appropriate, in support of activities affecting LSS/NRS components. The use of these instructions, procedures, and drawings will follow good business practices to provide reasonable assurance that LSS/NRS components will operate reliably and can satisfy their design functional requirements.**

- **Criterion VI (Document Control) – appropriate commercial practices will be followed to properly control documents affecting LSS/NRS components and activities. These practices will be governed by administrative procedures which will provide reasonable oversight over the LSS/NRS activities.**

- **Criterion VII (Control of Purchased Material, Equipment, and Services) – appropriate commercial practices will be followed to provide reasonable assurance that purchased material and equipment for LSS/NRS components conform to the procurement documentation. STP will continue to procure LSS/NRS components from reputable vendors. These practices will be governed by administrative procedures.**

- **Criterion VIII (Identification and Control of Materials, Parts, and Components) – appropriate commercial practices will be followed to provide reasonable assurance that incorrect or defective material, parts, and components are not used in LSS/NRS component applications. These practices will be governed by administrative procedures.**

- **Criterion IX (Control of Special Processes) – special processes will follow good commercial practices, and will be administratively controlled using existing processes and programs. Appropriate measures will be followed to ensure the reliability of LSS/NRS components, and to provide reasonable assurance that these components can perform their design functional requirements.**

- **Criterion X (Inspection) – commercial practices will be followed to ensure the reliability of LSS/NRS components, and to provide reasonable assurance that these components can perform their design functional requirements. Supervisor oversight or peer observations may be used to provide additional assurance that activities are completed in a safe and effective manner.**

- **Criterion XI (Test Control) – commercial practices will be followed to provide reasonable assurance that LSS/NRS components can satisfy their design functional requirements. Appropriate post-maintenance testing will be performed as well as operational checks to provide reasonable assurance that components will function.**

- **Criterion XII (Control of Measuring and Test Equipment) – commercial maintenance procedures, work instructions, and practices will be followed to use tools, gauges, instruments, and other measuring and testing equipment (M&TE). It is expected that this equipment will continue to be controlled and calibrated as it is currently, however, if a post-calibration check of the M&TE fails, evaluation of impact on LSS and NRS components will not be required, nor will any rework be required on LSS/NRS components.**

- **Criterion XIII (Handling, Storage, and Shipping) – appropriate commercial practices will be used to ensure that LSS/NRS components are properly handled, stored, shipped, cleaned, and preserved to ensure that replacement components retain their design functional requirements.**

- **Criterion XIV (Inspection, Test, and Operating Status) – LSS and NRS components will continue to remain within the existing configuration control program at STP. This includes the appropriate tagging to identify operational or maintenance status. Commercial practices will be used to identify the status of inspections or tests (normally contained within procedural guidance).**

- **Criterion XVII (Quality Assurance Records) – administrative controls will specify appropriate records and documentation for LSS and NRS components. Records that are administratively required to be retained will be controlled through the existing document control process.**

- **Criterion XVIII (Audits) – LSS and NRS components will be monitored under the Maintenance Rule program at the system/train/plant level if their failure affects an HSS or MSS function. In addition, periodic reviews performed by the GQA Working Group will assess the appropriateness of the controls placed on LSS/NRS components and the risk categorization for these components. Furthermore, the Quality Organization has**

and will continue to assess the overall GQA program and provide findings/recommendations to STP Management. Other assessments may be pursued based on good business practices or as directed by the Comprehensive Risk Management Expert Panel.

The above commercial practices have provided effective oversight and control for balance-of-plant components and activities. It is reasonable to expect that these same controls can effectively be used on LSS/NRS components, which serve little, if any, function in mitigating the consequences of an accident or protecting the health and safety of the public during a design-basis event or any other credible event.

RESPONSE: (part b)

Commercial practices will be followed for LSS/NRS components. STP asserts that for similar components, the failure rate of non-safety related SSCs does not differ appreciably from the failure rate of safety-related SSCs. See the response to Question 42 for additional insights into this assertion. Therefore, commercial practices have been demonstrated to provide reasonable assurance that LSS/NRS components will reliably perform their design functions.

From a procurement perspective, purchases of safety related LSS/NRS parts using commercial practices would require an engineering evaluation. This evaluation will compare the form, fit, and function between the existing LSS/NRS safety related part and the proposed part using commercial non-Appendix B suppliers. These parts will be evaluated and justified in accordance with site procedure OPGP03-ZE-0072 "Replacement Item Equivalency Evaluation (IEE) Program". This procedure requires that critical characteristics for design functions are selected and a comparison of existing and proposed critical characteristics is made and justified. Specifications that describe parts are available from manufacturers that provide the necessary data to make these comparisons. Typically, this information is contained in what is called a "catalog cut sheet". The Item Equivalency procedure uses the guidance of EPRI Report NP- 6406, "Technical Evaluation of Replacement Items Guideline". This evaluation will ensure that the proposed replacement will perform the design function. The Equivalency Evaluation is used for part number changes to ensure that the replacement part will function and fit, and to maintain configuration control. The IEE does not dedicate a part and STPNOC is not proposing that a safety related LSS/NRS part or component would require a dedication. Where the proposed change is a component (assembly of parts), a Design Change will be utilized in most cases. A design change is required if design basis documents such as Piping and Instrumentation Diagrams (P&IDs) require change due to the component change. The existing Design Change process at STP will be utilized for these changes. These processes are currently the practice at STPNOC for all components regardless of quality classification. These practices are used for parts and components in the Turbine-Generator Building and in the Reactor Containment Building.

After acceptability is demonstrated, a purchase order (PO) will be issued to a commercial vendor. Part number, model number, size, and description, as appropriate, will identify the required component. The purpose of this identification is to ensure that the part or component ordered, and evaluated, is the same as that received. The PO may also contain drawing references and specification references where this information is needed to describe the part or component. There will be a requirement in each PO that substitutions are not allowed without written permission. Engineering will evaluate each request for a substitution. Upon receipt of the part or component, an inspection will be performed for damage and to ensure that the part or component ordered is the component received.

Commercial manufacturers and suppliers are motivated to provide high quality and reliable products, which meet published specifications. These purchases will start with a comparison of the existing part or components design characteristics to the proposed component characteristics as described above. In many cases, the only difference will be the certifications and traceability to an Appendix B program. In other cases, materials may be different. Where functional differences are identified, the engineering evaluation mentioned above will ensure that the part or component will function as designed. Some differences may require a design change to implement the change.

Normal commercial procurement practices specified above will provide reasonable assurance that safety-related LSS and NRS components will be able to satisfy their intended design functional requirements. If the evaluations discussed above do not provide the necessary assurance that the component can satisfy its design functional requirements, additional testing, up to and including qualification-type testing, will be pursued to ensure that reasonable assurance is appropriately provided.

Attached to this response is an example of an equivalency evaluation that could be prepared for the replacement of a safety related LSS/NRS component with a commercial grade component. This example is provided for additional insight only.

RESPONSE: (part c)

An exemption is needed from Criterion IV for the following reasons:

1) **For LSS/NRS components, the procurement documents will not necessarily specify regulatory requirements, design bases, or other requirements. Instead, as discussed in part b of this response, the procurement documents may simply reference a part number.**

2) **For LSS/NRS components, the vendor/supplier will not be required to have a quality assurance program that is consistent with Appendix B. Instead, as discussed in part b of this response, STP may procure LSS and NRS components from normal commercial vendors and suppliers.**

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Replacement Item Equivalency Evaluation Program

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Item Equivalency Evaluation (IEE) No.: 99-1566-2 REV. NO.: 0 RMS FILE NO.: Z31.02

SECTION I **ITEM IDENTIFICATION**

Existing Condition		Proposed Condition	
Item Description:	Temp. indicator 0-2,000 °F	Item Description:	Temp. indicator 0-1,999 °F
Mfr. Name:	Honeywell	Mfr. Name:	Chromalox
Mfr. Model:	R7351	Mfr. Model:	N/A
Mfr. Part No.:	R7351A1080	Mfr. Part No.:	3901-1-1-1-12
Vendor Name:	Wilson-Mohr, Inc.	Vendor Name:	N/A
Vendor Model:	R7351	Vendor Model:	N/A
Vendor Part No.:	R7351A1080	Vendor Part No.:	N/A

SECTION II **CHANGE DESCRIPTION / SCREENING**

Type of Change:	Administrative or Equivalent Change (completion of Sections III through VI is **not required**) ☐	Alternate Replacement (completion of sections III through VI is **required** to evaluate acceptance) **Note:** IEE cannot be performed for replacement items which impact the design basis ☒

Description of Change(s):
The Honeywell temperature indicator P/N R7351A1080 is obsolete with no direct replacement. This IEE identifies the Chromalox temperature indicator P/N 3901-1-1-1-12 as an alternate replacement

Check all that apply:

DCP required For installation?	Safety Related?	Equipment Qualification Requirements Applicable?	Seismic Requirements Applicable ?
☒	☒	☐	☒

ENVIRONMENTAL QUAL PKG NO: N/A SEISMIC QUAL PKG NO: TO BE ASSIGNED
ENVIRONMENTAL REQUIREMENTS: N/A SEISMIC REQUIREMENTS:

Comparison results: (check box for item evaluation result)
[X] Alternate Item is **Acceptable**
[ ] Alternate Item is **NOT Acceptable**

This ITEM EQUIVALENCY EVALUATION applies to:

- [ ] Bulk commodity for plant wide use without restriction.
- [ ] Bulk commodity for use in mild environment.
- [X] Specific TPNS tags or applications. Item may not be used in applications not evaluated in this package. Contact Procurement Engineering if item may be required in other applications. (Specific TPNS are listed in Section III)

DISTRIBUTION
DED: [X] NPMM: [X] OTHER: [ ] RMS: [X]

IEE Originator:	Date:
Interdisciplinary Review:	Date:
Reviewed By:	Date:
Engineering Supervisor:	Date:

Revision	Description of Changes(s)

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SECTION III	**APPLICATION IDENTIFICATION**		
Parent Component Description / Function: (Item to be linked to this parent(s))	N/A	Parent TPNS and/or MPL(s)	N/A
Item Description / Function:	Provides an indication of hydrogen recombiner temp.	ItemTPNS and/or MPL(s)	B1(2)CGTI9975 C1(2)CGTI9980
Description of specific applications or limitations (if applicable):	N/A		

QC Class:	4	GQA:	LRS	Class Bin (s)	501-59523	P.O. (s)	4000 / 8000

SECTION IV **ITEM COMPARISON**

Critical Characteristics	Existing	Proposed	Results	Justification (if applicable)
1) Power	1) 120 VAC	1) 120 VAC	Same - Acceptable	
2) Input signal	2) Type K thermocouple	2) Type K thermocouple	Same - Acceptable	
3) Indicator Type	3) Analog	3) Analog	Same – Acceptable	
4) Temperature Range	4) 0 – 2,000 °F	4) 0 – 1,999 °F	Different - Acceptable	The 1 degree temperature difference is negligible. Recombination temperature is assured by maintaining temperature above 1225 °F and the unit is not used at it's max. range.
5) Accuracy	5) +/- 1% nominal	5) +/- 0.5% of span over mid-80% of scale	Different - Acceptable	The proposed has a better accuracy and is acceptable.
6) Dimensions Hole Length Width Depth Face Length Width Mounting	6) 5.43" 5.43" 7" 6" 6" Two pressure type mounting bracket and screws.	6) 3.6" 3.6" 3.2" 3.8" 3.8" Two pressure type screw in mounting tabs.	Different - Acceptable	A DCP is needed to provide guidance for fabrication and installation of a cover plate.
7) Indicator display	7) Needle	7) LED	Different – Acceptable	Needle can be read to nearest 20 °F mark, the LED can be read to the nearest °F.
8) Power consumption	8) 6.5 watts maximum	8) 10 watts	Different - Acceptable	The proposed will draw approx. 30 mA more than the existing. This is increase is negligible.

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SECTION V **RELATED DOCUMENTATION UPDATE / DATABASE UPDATE**

New Documents / Drawings: Required prior to Installation? Yes ☐ No ☒

Doc. Type	New Document/ Drawing Number	Rev	Title / Description
VTD	VTD-C332-0026	0	3901 Chromalox Overtemperature Controller Users Manual

Revised Documents:

Doc. Type	Document/ Drawing Number	Rev	Title / Description	DCP or CR # for rev. as applicable:

Affected Databases: Change Required? Yes ☒ No ☐

MPL	Sequence No.	Rev	Description of Changes Required
To be created			Add a new MPL for the Chromalox temperature indicator as indicated on pages 1 and 2 of this IEE.

SECTION VI **GENERAL COMMENTS / REFERENCES**

Comments:

Note to NPMM: THIS ITEM IS TO BE PLACED ON RESTRICTED ISSUE. A DCP IS REQUIRED FOR INSTALLATION. A cover plate must be fabricated to secure the units in the panel as they are physically smaller than the existing ones. These indicators are in the plant scaling manual and the main control panel equipment list. Additionally, wiring labels on several plant drawings must be updated.
*This item must be dedicated to be safety related, Class 1E and seismically qualified. Select an AVL Appendix B Supplier to dedicate and supply a seismic qualification report.

General Note: At the time of issue of IEE 99-1566-2 all the temperature indicators were still operational and there was no immediate need. There is a 30 day T.S. associated with these indicators. The purpose of these documents was to facilitate purchase, dedication, and availability of this item in the warehouse in the event one fails. Replacement of all the indicators was considered. However, IAW FC-990025 to 0PGP04-ZE-0309, rev. 6, changes which impact the simulator are a minor Mod. As there was no immediate need, it was decided not to pursue a minor Mod. The restriction notes in the IEE identify design activities needed to install this item.

References:

VTD-W351-0042, Electric Hydrogen Recombiners
VTD-C332-0026, 3901 Chromalox Overtemperature Controller User's Manual
ES 953426, Westinghouse Electrical Specification for Recombination Unit
3Z349ZS0120, Specification for Main Control Panels
DWG 14926-4304-00250, Electrical Wiring Diagram for Control Panel CP002-WD1
DWG 4304-00188-MMD, Front Arrangement CP002
DWG 14926-0917-1(2)-00004, H2 Recombiner Power Schematic
DWG 14926-0917-1(2)-00002-AWN, H2 Recombiner BOM
DWG 9-E-CG10-1 #1(#2), Elementary Master Block Diagram Elec. Hydrogen Recombiner
DWG 5-Z-01-9-Z-47511 #1(#2), Main Control Panel Equipment List
DWG 5-Z-34-9-Z-44520, Main Control Panel Plates & Cut Out Details

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(IEE Continuation Page)

IEE Section:	ITEM COMPARISON			
Critical Characteristics	Existing	Proposed	Results	Justification (if applicable)
9) Weight	9) 8 lbs. shipping	9) 614 g or 1.35 lbs. (Field measured at the warehouse as no weight was provided in vendor literature)	Different – Acceptable	The proposed weighs less than the existing.
10) Indicating lights	10) Red and green LED indicate the process condition ie. above or below setpoint.	10) No red/green LEDs	Different – Acceptable	This instrument is not used for automatic control and is not programmed for a setpoint so indication above or below is not needed.

14. *Please clarify the following. As written, the licensee's exemption request for 10 CFR 50.65 implies that the exemption applies only to safety-related LSS and NRS components and not to any nonsafety-related SSCs. The maintenance rule scope specified in 50.65(b) applies to safety-related and nonsafety-related SSCs.*

    (a) Is the licensee requesting exemption from 50.65 for any nonsafety-related SSCs?

    (b) If so, please provide a more specific request that addresses how the exemption will apply to all of the scoping requirements in 50.65(b) and the resulting changes to the maintenance rule (MR) program and monitoring.

    (c) How will LSS and NRS safety-related and nonsafety-related components be treated under the scope of 10 CFR 50.65(a)(4) when this new rule becomes effective?

RESPONSE (part a):

To clarify the STP position, STP is requesting an exemption from the requirements of 10CFR50.65 for <u>all</u> LSS and NRS components. The exemption request has been modified to clarify that the exemption applies to both safety-related and non-safety-related components that are classified as LSS or NRS.

RESPONSE (part b):

The exemption request has been clarified to state that the scoping requirements of 10CFR50.65(b) are no longer applicable for any component that has been categorized as LSS or NRS by the STP GQA process. However, all components (safety related and non-safety-related) that have been classified as HSS or MSS will be within the scope and requirements of 10CFR50.65. All components that have been classified as LSS or NRS are outside of the scope of the Maintenance Rule, and the requirements of 10CFR50.65 will not apply to them. Components that have yet to be categorized by the STP GQA process will remain the same as they have previously been scoped in accordance with 10CFR50.65 requirements unless and until they are categorized by the GQA process. This is depicted in the table below:

	HSS/MSS	LSS/NRS	Not yet ranked by GQA
Components	In MR Scope	Out of MR Scope *	As currently scoped by MR
Functions	In MR Scope*	Out of MR Scope	As currently scoped by MR

***LSS/NRS component failures that cause a HSS/MSS function to be lost will be counted as a Maintenance Rule Functional Failure (MRFF).**

Scoping for the Maintenance Rule is done at the function level. The Maintenance Rule program would be modified so that functions supported by HSS or MSS components are designated as being within the scope of the Maintenance Rule. Functions supported solely by LSS and NRS components would be designated as being outside of the scope of the Maintenance Rule. The monitoring of component performance will not be required for components outside the scope of the Maintenance Rule. The failure of LSS and NRS components are not expected to cause the exceedance of performance criteria used to monitor SSCs within the scope of the Maintenance Rule. If the failure of a LSS or NRS component affects an existing Maintenance Rule performance criteria (e.g., the failure of an Instrument Air isolation valve affects the unavailability performance criteria of Instrument Air System compressors), then the failure would be counted against the Maintenance Rule performance criteria. If the Maintenance Rule scoped system/train/component then exceeded its performance criteria, it would be evaluated for reclassification to category (a)(1). Systems classified as (a)(1) must have a Condition Report written to determine the cause of exceeding the performance criteria and to develop a plan of action to prevent recurrence.

Work packages for components not in the scope of the Maintenance Rule (i.e., LSS or NRS components) are not reviewed for Functional Failures, so they would not be counted against the performance criteria for reliability. The Corrective Action Program would address any failures of LSS or NRS components that do not affect an existing Maintenance Rule performance criteria. If the failures of LSS/NRS components were significant due to the consequences or the number of failures, then these failures would be evaluated as part of the periodic Graded Quality Assurance (GQA) review. The GQA Working Group would evaluate whether the components should have additional controls applied to them, or possibly a higher risk classification. If the component were reclassified to MSS or HSS, then the component would be added to the scope of the Maintenance Rule.

If any plant level performance criteria were exceeded, we would develop a plan of action to address the main contributors to the exceedance (whether the contributor was in the Maintenance Rule scope or not). If LSS/NRS components were some of the main contributors to exceeding the plant level performance criteria, then these components would be evaluated for application of additional controls to improve performance, or possibly for component risk reclassification as part of the periodic GQA Working Group review.

RESPONSE (part c):

We do not intend to make any changes to our current risk assessment process (cumulative risk profiles) due to the removal of LSS and NRS components from the scope of the Maintenance Rule. If a LSS or NRS component is taken out of service and affects the overall risk, then its risk impact will continue to be assessed and managed.

15. *What is the mechanism and time frame to identify any changes in risk categorization of components from LSS/NRS to MSS or HSS that may be a result from operating experience or plant facility modifications? What is the time frame that these components will then return to the scope of the appropriate special treatment and how will a demonstration be made that shows the performance or condition of the components are being effectively controlled through the performance of appropriate special treatment?*

RESPONSE:

The mechanism for identifying potential changes to component risk categorization resulting from both in-house and industry operating experience utilizes the Corrective Action Program (CAP) and the GQA six-month review process.

The Corrective Action Program is controlled by procedure 0PGP03-ZX-0002 and permits anyone at the plant site who identifies a deficiency to document that condition for correction. These documented deficiencies are available for review each day by Station personnel, and are acted upon to implement appropriate remedial and/or corrective actions.

The GQA six month review process is governed by procedure 0PGP02-ZA-0003, Comprehensive Risk Management. On a once-per-six-month frequency, the Operating Experience Group performs a comprehensive evaluation of conditions generated within the previous six months against each specific risk-categorized system designator, and reports the results to the Working Group. This report includes information for the current reporting period, as well as the two previous reporting periods. The Working Group is tasked with determining if any risk categorization revisions are warranted based on:

- **a degradation of equipment performance,**
- **System Engineer input,**
- **Maintenance Rule input,**
- **Licensing, Quality, or Operations organization input,**
- **Design changes, or**
- **Probabilistic Risk Assessment input.**

Whenever degraded performance is attributed to the reduction or relaxation of special treatment controls, the Working Group will recommend the appropriate remedial action up to and including the reinstatement of the subject special treatment control(s) and the potential re-categorization of the component's risk significance to a higher level. Any proposed risk categorization changes are submitted to the Expert Panel for approval. Once approved, the risk categorization change is reflected electronically in the controlled Master Equipment Database and through a revision to the Risk Significance Basis Document for that system. In addition, if the risk categorization was changed from LSS/NRS to MSS or HSS, or if a special treatment control was reinstated, a new condition report would be generated to assess the impact of returning the subject component to the scope of the appropriate special treatments. This assessment would include an evaluation

of activities performed on, with, or for the component during the time that the component was excluded from the scope of special treatment requirements.

While no specific timeframe is identified for reinstatement of the special treatment controls, it is expected that these controls will be reinstated in a timely manner (generally within the normal 12 week Functional Equipment Group (FEG) Work Week windows, if possible. If operational conditions necessitate that these additional controls be applied sooner, appropriate action will be taken to incorporate the controls.). The generated condition report remains open until all corrective actions, if any, are implemented as appropriate. These corrective actions may include, but are not limited to, an evaluation of the component's impact on current operating conditions and the Technical Specifications. The component's performance would continue to be monitored as part of future periodic reviews to provide an appropriate level of assurance that the applied controls are effective. It should be noted that the component's impact on current operating conditions is assessed in accordance with the standard operability review that is performed following initiation of the condition report.

Potential risk categorization changes resulting from plant modifications are identified either during the development of the modification or during the periodic six-month review performed by the Working Group on the associated system. Currently, potential impacts to component categorization identified during the modification development phase are documented on a condition report and forwarded to the Working Group for evaluation. While the existing modification process procedure does not explicitly require an evaluation for risk categorization impacts, this procedure will be revised to include the requirement for an impact evaluation on system function/component risk categorizations when modifications are proposed. Any risk categorization changes resulting from plant modifications are implemented as described in the six-month review process discussed above.

It should also be noted that the above process does not preclude the Working Group from acting upon condition reports associated with potential risk categorization changes more frequently than every six months.

16. *STP is a 10 CFR Part 50, Appendix J, Option B, plant. Please provide more specificity about what portions (specific sections) of Appendix J, Option B, are to be exempted.*

RESPONSE:

STP is requesting an exemption from the Type C test requirements in Appendix J, Option B, Section III.B to 10 CFR Part 50, to the extent that those requirements pertain to containment isolation valves that meet the following criteria:

1. **The valve has been categorized as LSS or NRS; and**

2. **The valve meets one or more of the following criteria:**

   a. **The valve is required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers).**
   b. **The valve is normally closed and in a physically closed, water-filled system. (e.g., containment isolation valves in the Demineralized Water system)**
   c. **The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Component Cooling Water system and in the Instrument Air system).**
   e. **The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve.**
   e. **The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).**

The above criteria describe a set of penetrations where leakage paths which would threaten public health and safety are not credible. The penetrations meeting criterion 2.a are in a closed system which is under duty during accident conditions and, therefore, represent pathways for mass and inventory to enter containment and, if exiting containment (e.g., recirculation cooling post LOCA), represent mass and inventory which is contained in a closed system. Criterion 2.b penetrations, which are normally water-filled and closed, are eliminated since, in addition to the physical barriers of piping and water inventory, the CIVs are already in a closed position (i.e., the containment isolation function is already satisfied), thus providing an additional physical barrier to prevent leakage. Criterion 2.c represents penetrations where leakage is not possible due to the physical barriers of piping (which is rated higher than containment design pressure), existing water inventories, and actuated valve barriers of various types contained within these closed systems. Criterion 2.d represents containment isolation valves in a closed system where the rating exceeds the containment design pressure rating. This criterion applies to valves connected to the

reactor coolant boundary where the process line between the CIV and the reactor coolant boundary is non-nuclear safety. Criterion 2.e is for valves of 1 inch or less in size whose failure will not, by definition, lead to a large early release.

The following table is a listing of all containment isolation valves. The last column indicates whether the valve is in the scope of the exemption based on satisfying the above criteria.

Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-01(3)	2S141TMS0143	TERRY TURBINE MAIN STEAM SUPPLY ORC ISOLATION MOV	4	HIGH	57	NO	N/A	
M-01(3)	A1MSFV7442	SG 1D MN STM ISOLATION BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-01(3)	D1AFFV0143	TERRY TURBINE WARM-UP VALVE	1	LOW	57	NO	N/A	
M-01(3, 5)	A1MSFSV7444	STEAM GENERATOR 1D MAIN STEAM ORC ISOLATION VALVE	30	MEDIUM	57	NO	N/A	
M-01(4)	2S101TMS0546	MAIN STEAM ISOL VALVE 1D ABOVE SEAT DRAIN ISOL	2	LOW	57	NO	N/A	
M-02(3)	A1MSFV7412	SG 1A MN STM ISOLATION BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-02(3, 5)	A1MSFSV7414	STEAM GENERATOR 1A MAIN STEAM ORC ISOLATION VALVE	30	MEDIUM	57	NO	N/A	
M-02(4)	2S101TMS0543	MAIN STEAM ISOL VALVE 1A ABOVE SEAT DRAIN ISOL	2	LOW	57	NO	N/A	
M-03(3)	A1MSFV7422	SG 1B MN STM ISOLATION BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-03(3, 5)	A1MSFSV7424	STEAM GENERATOR 1B MAIN STEAM ORC ISOLATION VALVE	30	MEDIUM	57	NO	N/A	
M-03(4)	2S101TMS0544	MAIN STEAM ISOL VALVE 1B ABOVE SEAT DRAIN ISOL	2	LOW	57	NO	N/A	
M-04(3)	A1MSFV7432	SG 1C MN STM ISOLATION BYPASS VALVE		MEDIUM	57	NO	N/A	
M-04(3, 5)	A1MSFSV7434	STEAM GENERATOR 1C MAIN STEAM ORC ISOLATION VALVE	30	MEDIUM	57	NO	N/A	
M-04(4)	2S101TMS0545	MAIN STEAM ISOL VALVE 1C ABOVE SEAT DRAIN ISOL	2	LOW	57	NO	N/A	
M-05(3)	B1FWFV7145A	STEAM GENERATOR 1D FEEDWATER INLET ORC ISOLATION VALVE BYPASS	3	MEDIUM	57	NO	N/A	
M-05(3, 7)	A1FWFV7144	STEAM GENERATOR 1D ORC FEEDWATER ISOLATION VALVE	18	MEDIUM	57	NO	N/A	
M-06(3)	A1FWFV7148A	STEAM GENERATOR 1A FEEDWATER INLET ORC ISOLATION VALVE BYPASS	3	MEDIUM	57	NO	N/A	
M-06(3, 7)	A1FWFV7141	STEAM GENERATOR 1A ORC FEEDWATER ISOLATION VALVE	18	MEDIUM	57	NO	N/A	
M-07(3)	A1FWFV7147A	STEAM GENERATOR 1B FEEDWATER INLET ORC ISOLATION VALVE BYPASS	3	MEDIUM	57	NO	N/A	
M-07(3, 7)	A1FWFV7142	STEAM GENERATOR 1B ORC FEEDWATER ISOLATION VALVE	18	MEDIUM	57	NO	N/A	
M-08(3)	B1FWFV7146A	STEAM GENERATOR 1C FEEDWATER INLET ORC ISOLATION VALVE BYPASS	3	MEDIUM	57	NO	N/A	
M-08(3, 7)	A1FWFV7143	STEAM GENERATOR 1C ORC FEEDWATER ISOLATION VALVE	18	MEDIUM	57	NO	N/A	
M-09	2N101XCS0006	CONTAINMENT SPRAY HEADER	8	*	56	YES	*	
M-09(6)	2N101XCS0001C	CONTAINMENT SPRAY TO RING HEADER	8	*	56	YES	*	
M-10	2N121XSI0004C	HI HEAD SAFETY INJECTION PUMP 1C DISCHARGE MOV (ORC)	6	MEDIUM	55	YES	NO	
M-10	2N121XSI0005C	HI HEAD SAFETY INJECTION PUMP 1C DISCHARGE CHECK VALVE (IRC)	6	HIGH	55	YES	NO	
M-11	2N121XSI0018C	LO HEAD SAFETY INJECTION PUMP 1C DISCHARGE MOV (ORC)	8	MEDIUM	55	YES	NO	
M-11	2N121XSI0030C	LO HEAD SAFETY INJECTION PUMP 1C DISCHARGE CHECK VALVE (IRC)	8	HIGH	55	YES	NO	
M-12	2S201TSL0027	CHEMICAL CLEANING RETURN	6	*	56	YES	*	
M-12	2S201TSL0029	CHEMICAL CLEANING RETURN	6	*	56	YES	*	
M-13	2N101XCS0004	CONTAINMENT SPRAY HEADER	8	*	56	YES	*	
M-13	2N101XCS0005	CONTAINMENT SPRAY HEADER	8	*	56	YES	*	

*Component has not gone through the risk ranking process.

Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-13(6)	2N101XCS0001B	CONTAINMENT SPRAY TO RING HEADER	8	*	56	YES	*	
M-14	2N121XSI0004B	HI HEAD SAFETY INJECTION PUMP 1B DISCHARGE MOV (ORC)	6	MEDIUM	55	YES	NO	
M-14	2N121XSI0005B	HI HEAD SAFETY INJECTION PUMP 1B DISCHARGE CHECK VALVE (IRC)	6	HIGH	55	YES	NO	
M-15	2N121XSI0018B	LO HEAD SAFETY INJECTION PUMP 1B DISCHARGE MOV (ORC)	8	MEDIUM	55	YES	NO	
M-15	2N121XSI0030B	LO HEAD SAFETY INJECTION PUMP 1B DISCHARGE CHECK VALVE (IRC)	8	HIGH	55	YES	NO	
M-16	2S201TSL0002	SLUDGE LANCING HIGH PRESSURE	2	*	56	YES	*	
M-16	2S201TSL0004	SLUDGE LANCING HIGH PRESSURE	2	*	56	YES	*	
M-17	2N101XCS0002	CONTAINMENT SPRAY HEADER	8	*	56	YES	*	
M-17(6)	2N101XCS0001A	CONTAINMENT SPRAY TO RING HEADER	8	*	56	YES	*	
M-18	2N121XSI0004A	HI HEAD SAFETY INJECTION PUMP 1A DISCHARGE MOV (ORC)	6	MEDIUM	55	YES	NO	
M-18	2N121XSI0005A	HI HEAD SAFETY INJECTION PUMP 1A DISCHARGE CHECK VALVE (IRC)	6	HIGH	55	YES	NO	
M-19	2N121XSI0018A	LO HEAD SAFETY INJECTION PUMP 1A DISCHARGE MOV (ORC)	8	MEDIUM	55	YES	NO	
M-19	2N121XSI0030A	LO HEAD SAFETY INJECTION PUMP 1A DISCHARGE CHECK VALVE (IRC)	8	HIGH	55	YES	NO	
M-20	2N121XSI0016C	CONTAINMENT EMERGENCY SUMP 1C TO SI TRAIN C PUMPS SUCTION ORC ISOLATION M	16	HIGH	56	YES	NO	
M-21	2N121XSI0016B	CONTAINMENT EMERGENCY SUMP 1B TO SI TRAIN B PUMPS SUCTION ORC ISOLATION M	16	HIGH	56	YES	NO	
M-22	2N121XSI0016A	CONTAINMENT EMERGENCY SUMP 1A TO SI TRAIN A PUMPS SUCTION ORC ISOLATION M	16	HIGH	56	YES	NO	
M-23	2R201TCC0208	(IRC) CC-MOV-0208(TR C CCW FROM RCFC'S CNTMNT ISO)	14	LOW	56	YES	YES	1, 2.c
M-23	2R201TCC0209	CC-MOV-0209 (CHILL H2O RETURN FROM "C" RCFC'S)	8	LOW	56	YES	YES	1, 2.c
M-23	2R201TCC0210	CC-MOV-0210 (TRAIN C CCW FROM 11/12 C RCFCS MOV)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-23	A1CCFV0864	RCFC CHILLED WATER RETURN	8	NRS	56	YES	YES	1, 2.c
M-24	2R201TCC0197	CC-MOV-0197 (TRAIN C CCW SUPPLY TO RCFC'S MOV)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-24	2R201TCC0198	(IRC) CCW INLET TO RCFC CHECK VALVE	14	LOW	56	YES	YES	1, 2.a, 2.c
M-24	2R201TCC0199	CC-MOV-0199 (CHILL H2O SUPPLY TO "C" RCFC'S MOV)	8	LOW	56	YES	YES	1, 2.c
M-25	2R201TCC0057	CC-MOV-0057 (A CCW TO RCFC UPSTREAM MOV)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-25	2R201TCC0058	(IRC) A TRAIN CCW TO RCFC'S ICIV CHECK VALVE	14	LOW	56	YES	YES	1, 2.a, 2.c
M-25	2R201TCC0059	CC-MOV-0059 (CHILL H2O INLET TO TRAIN A RCFC'S)	8	LOW	56	YES	YES	1, 2.c
M-26	2R201TCC0068	(IRC) CC-MOV-0068 (A CCW FROM RCFC'S CNTMNT ISOL)	14	LOW	56	YES	YES	1, 2.c
M-26	2R201TCC0069	CC-MOV-0069 (CCW TRAIN A DISCHARGE FROM RCFC'S)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-26	2R201TCC0070	CC-MOV-0070 (CHILL H2O RETURN FROM A TRAIN RCFC'S)	8	LOW	56	YES	YES	1, 2.c
M-26	B1CCFV0862	RCFC CHILLED WATER RETURN	8	NRS	56	YES	YES	1, 2.c
M-27	2R201TCC0136	CC-MOV-0136 (CCW TRAIN B INLET TO RCFC'S MOV)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-27	2R201TCC0137	CC-MOV-0137 (CHILL H2O SUPPLY TO "B" RCFC'S MOV)	8	LOW	56	YES	YES	1, 2.c
M-27	2R201TCC0138	(IRC) B TRAIN CCW TO RCFC'S ICIV CHECK VALVE	14	LOW	56	YES	YES	1, 2.a, 2.c

*Component has not gone through the risk ranking process.

Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-28	2R201TCC0147	(ICIV) CC-MOV-0147 (B CCW FROM RCFC'S ICIV)	14	LOW	56	YES	YES	1, 2.c
M-28	2R201TCC0148	CC-MOV-0148 (B CCW FROM RCFC'S MOV)	14	LOW	56	YES	YES	1, 2.a, 2.c
M-28	2R201TCC0149	CC-MOV-0149 (CHILL H2O DISCHARGE FROM "B" RCFC'S)	8	LOW	56	YES	YES	1, 2.c
M-28	C1CCFV0863	RCFC CHILLED WATER RETURN	8	NRS	56	YES	YES	1, 2.c
M-29(1)	B1PSFV4466	SI ACCUMULATORS SAMPLE HEADER ORC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-29(1)	C1PSFV4824	SI ACCUMULATOR SAMPLE HEADER IRC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-30	A1WLFV4920	REACTOR COOLANT DRAIN TANK VENT TO GASEOUS WASTE PROCESSING IRC ISOLATION	1	LOW	56	YES	YES	1, 2.c, 2.e
M-30	B1WLFV4919	REACTOR COOLANT DRAIN TANK VENT TO GASEOUS WASTE PROCESSING ORC ISOLATION	1	LOW	56	YES	YES	1, 2.c, 2.e
M-32		ILRT PENETRATION TEST CONNECTION		*	56	YES	*	
M-33	2R201TCC0012	CC-MOV-0012 (CCW SUPPLY TO RHR "A" MOV)	16	LOW	56	YES	YES	1, 2.c
M-33	2R201TCC0013	M-33 CHECK VALVE	16	MEDIUM	56	YES	NO	
M-34	2R201TCC0049	(IRC) CCW FROM "A" RHR ISOLATION (LOCAL HANDWHEEL)	16	LOW	56	YES	YES	1, 2.c
M-34	2R201TCC0050	(OCIV) CC-MOV-0050 CCW FROM TRAIN "A" RHR OCIV	16	LOW	56	YES	YES	1, 2.c
M-35	2R201TCC0122	(OCIV) CC-MOV-0122 (CCW B SUPPLY TO RHR OCIV)	16	LOW	56	YES	YES	1, 2.c
M-35	2R201TCC0123	(IRC) ICIV FOR CCW TO B TRAIN RHR COMPONENTS	16	MEDIUM	56	YES	NO	
M-36	2R201TCC0129	(ICIV) CC-MOV-0129 (CCW B FROM RHR HEADER ICIV)	16	LOW	56	YES	YES	1, 2.c
M-36	2R201TCC0130	CC-MOV-0130 (CCW B FROM RHR CONTAINMENT ISO MOV)	16	LOW	56	YES	YES	1, 2.c
M-37	2R201TCC0182	CC-MOV-0182 (CCW C SUPPLY TO RHR MOV)	16	LOW	56	YES	YES	1, 2.c
M-37	2R201TCC0183	(IRC) CCW RHR HX 1C INLET CHECK VALVE	16	MEDIUM	56	YES	NO	
M-38	2R201TCC0189	(IRC) CC-MOV-0189 (CCW C FROM RHR HEADER ISO MOV)	16	LOW	56	YES	YES	1, 2.c
M-38	2R201TCC0190	CC-MOV-0190 (CCW C SUPPLY FROM RHR)	16	LOW	56	YES	YES	1, 2.c
M-39	2R201TCC0291	(OCIV) CC-MOV-0291 (CCW TO RCP'S OCIV MOV)	12	LOW	56	YES	YES	1, 2.c
M-39	2R201TCC0318	(OCIV) CC-MOV-0318 (CCW TO RCP'S OCIV MOV)	12	LOW	56	YES	YES	1, 2.c
M-39	2R201TCC0319	(IRC) CCW TO RCP'S INLET CHECK VALVE	12	LOW	56	YES	YES	1, 2.c
M-40	2R201TCC0403	(ICIV) CCW RETURN FROM RCP ICIV (CC-MOV-0403)	12	LOW	56	YES	YES	1, 2.c
M-40	2R201TCC0404	CC-MOV-0404 (CCW FROM RCP'S ISOLATION MOV)	12	LOW	56	YES	YES	1, 2.c
M-40	2R201TCC0446	(IRC) CCW FROM RCP CHECK AROUND CC-0542	1	LOW	56	YES	YES	1, 2.e
M-40	2R201TCC0542	(IRC) CC-MOV-0542 (CCW FROM RCP ISOLATION MOV)	12	LOW	56	YES	YES	1, 2.c
M-40	D1CCFV4493	RCP CCW RETURN OCIV (CC-FV-4493)	8	LOW	56	YES	YES	1, 2.c
M-41	2V141ZHC0009	REACTOR CONTAINMENT BUILDING NORMAL PURGE EXHAUST (IRC) ISOLATION DAMPER	48	MEDIUM	56	YES	NO	
M-41	2V141ZHC0010	REACTOR CONTAINMENT BUILDING NORMAL PURGE EXHAUST (ORC) ISOLATION DAMPER	48	MEDIUM	56	YES	NO	
M-42	2V141ZHC0007	REACTOR CONTAINMENT BUILDING NORMAL PURGE SUPPLY (ORC) ISOLATION DAMPER	48	MEDIUM	56	YES	NO	

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Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-42	2V141ZHC0008	REACTOR CONTAINMENT BUILDING NORMAL PURGE SUPPLY (IRC) ISOLATION DAMPER	48	MEDIUM	56	YES	NO	
M-43(2)	2V141THC0003	REACTOR CONTAINMENT BUILDING SUPPLEMENTARY PURGE SUPPLY (IRC) ISOLATION D	18	MEDIUM	56	YES	NO	
M-43(2)	A1HCFV9776	REACTOR CONTAINMENT BUILDING SUPPLEMENTARY PURGE SUPPLY ISOLATION VALVE O	18	MEDIUM	56	YES	NO	
M-44(2)	2V141THC0005	REACTOR CONTAINMENT BUILDING SUPPLEMENTARY PURGE EXHAUST (IRC) ISOLATION	18	MEDIUM	56	YES	NO	
M-44(2)	A1HCFV9777	REACTOR CONTAINMENT BUILDING SUPPLEMENTARY PURGE EXHAUST ISOLATION VALVE	18	MEDIUM	56	YES	NO	
M-45	2R141XRC0046	(ICIV) RMW TO PRT INLET CHECK VALVE	3	LOW	55	YES	YES	1, 2.d
M-45	B1RCFV3651	PRT SPRAY ISOL VALVE (OCIV)	3	LOW	55	YES	YES	1, 2.d
M-45(1)	C1APFV2458	LIQUID POST ACCIDENT SAMPLE DISCHARGE	1	*	55	YES	*	
M-46	2R171TCV0022	(IRC) OVERPRESS PROT FOR CVCS LETDOWN AT PEN M46	0.75	LOW	55	YES	YES	1, 2.e
M-46	2R171XCV0023	(IRC) CVCS LETDOWN ISOLATION (CV-MOV-0023)	4	LOW	55	YES	NO	
M-46	2R171XCV0024	OCIV FOR CVCS LETDOWN (CV-MOV-0024)	4	LOW	55	YES	NO	
M-47	2R171TCV0077	(IRC) SEAL WATER RETURN ISOLATION (CV-MOV-0077)	2	LOW	55	YES	NO	
M-47	2R171TCV0078	RCP SEAL WATER I IRC BYPASS CHECK VALVE	0.75	LOW	55	YES	YES	1, 2.e
M-47	2R171TCV0079	(OCIV) SEAL WATER RETURN ISOLATION (CV-MOV-0079)	2	LOW	55	YES	NO	
M-48	2R171XCV0025	CHARGING ISOLATION (CV-MOV-0025)	4	MEDIUM	55	YES	NO	
M-48	2R171XCV0026	(IRC) CHARGING LINE IRC CHECK VALVE	4	MEDIUM	55	YES	NO	
M-51	2R171TCV0033A	(OCIV) CV-MOV-0033A, 1A RCP SEAL INJECTION OCIV	2	MEDIUM	55	YES	NO	
M-51	2R171TCV0033B	(OCIV) CV-MOV-0033B, 1B RCP SEAL INJECTION OCIV	2	MEDIUM	55	YES	NO	
M-51	2R171TCV0034A	(IRC) RCP 1A SEAL INJECTION LINE ICIV CHECK VLV	2	MEDIUM	55	YES	NO	
M-51	2R171TCV0034B	(IRC) RCP 1B SEAL INJECTION LINE ICIV CHECK VLV	2	MEDIUM	55	YES	NO	
M-52	2R171TCV0033C	(OCIV) CV-MOV-0033C, 1C RCP SEAL INJECTION OCIV	2	MEDIUM	55	YES	NO	
M-52	2R171TCV0033D	(OCIV) CV-MOV-0033D, 1D RCP SEAL INJECTION OCIV	2	MEDIUM	55	YES	NO	
M-52	2R171TCV0034C	(IRC) RCP 1C SEAL INJECTION LINE ICIV CHECK VLV	2	MEDIUM	55	YES	NO	
M-52	2R171TCV0034D	(IRC) RCP 1D SEAL INJECTION LINE ICIV CHECK VLV	2	MEDIUM	55	YES	NO	
M-53	2R171XCV0157	(OCIV) LOW PRESS LETDOWN TO RHR OCIV	4	LOW	55	YES	NO	
M-53	2R171XCV0158	(IRC) CVCS LETDOWN TO RHR PUMP ICIV CHECK VALVE	4	LOW	55	YES	NO	
M-55	2R161XRH0063B	RHR LOOP 1B RETURN TO RWST IRC ISOLATION VALVE	8	MEDIUM	55	YES	NO	
M-55	2R161XRH0064B	RHR LOOP 1B RETURN TO RWST ORC ISOLATION VALVE	8	MEDIUM	55	YES	NO	
M-56	2R301TWL0312	RCDT HEAT EXCHANGER OUTLET LINE ISOLATION MOV (IRC)	3	LOW	56	YES	YES	1, 2.c
M-56	B1WLFV4913	RCDT HEAT EXCHANGER OUTLET ISOLATION VALVE (ORC)	3	LOW	56	YES	YES	1, 2.c
M-57	2Q101TSA0505	SERVICE AIR	2	*	56	YES	*	
M-57(4)	2Q101TSA0504	SERVICE AIR	2	*	56	YES	*	
M-58	2Q111TIA0541	RCB INSTRUMENT AIR HEADER IRC CHECK VALVE	2	LOW	56	YES	YES	1, 2.c

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Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-58	B1IAFV8565	INSTRUMENT AIR SUPPLY TO REACTOR CONTAINMENT BUILDING ORC ISOLATION VALVE	4	LOW	56	YES	YES	1, 2.c
M-61	2S191TDW0502	DEMINERALIZED WATER SUPPLY	2	*	56	YES	*	
M-61(4)	2S191TDW0502	DEMINERALIZED WATER SUPPLY	2	*	56	YES	*	
M-62(3)	A1SBFV4150	STEAM GENERATOR 1D BLOWDOWN ORC ISOLATION VALVE	4	MEDIUM	57	NO	N/A	
M-63(3)	A1SBFV4153	STEAM GENERATOR 1A BLOWDOWN ORC ISOLATION VALVE	4	MEDIUM	57	NO	N/A	
M-64(3)	B1SBFV4152	STEAM GENERATOR 1B BLOWDOWN ORC ISOLATION VALVE	4	MEDIUM	57	NO	N/A	
M-65(3)	C1SBFV4151	STEAM GENERATOR 1C BLOWDOWN ORC ISOLATION VALVE	4	MEDIUM	57	NO	N/A	
M-68	2N121TSI0058	SAFETY INJECTION ACCUMULATORS IRC NITROGEN SUPPLY CHECK VALVE	1	NRS	55	YES	YES	1, 2.d, 2.e
M-68	A1RCFV3653	(IRC) PRT NITROGEN ISOL VALVE (ICIV) FLOW VALVE	1	LOW	55	YES	YES	1, 2.d, 2.e
M-68	A1SIFV3971	SIS CHECK VALVE TEST LINE ORC ISOLATION VALVE	0.75	LOW	55	YES	YES	1, 2.d, 2.e
M-68	A1SIFV3983	SAFETY INJECTION ACCUMULATORS ORC NITROGEN SUPPLY VALVE	1	LOW	55	YES	YES	1, 2.d, 2.e
M-68	B1RCFV3652	PRT NITROGEN ISOL VALVE (OCIV) FLOW VALVE	1	LOW	55	YES	YES	1, 2.d, 2.e
M-68	B1SIFV3970	SIS CHECK VALVE TEST LINE IRC ISOLATION VALVE	0.75	LOW	55	YES	YES	1, 2.d, 2.e
M-69	2R211XFC0013E	SFP COOLING & PURIFICATION DISCH TO RX CAVITY OR ICSA ORC ISOLATION VALVE	10	LOW	56	YES	YES	1, 2.c
M-69	2R211XFC0013F	SFP COOLING & PURIFICATION DISCHARGE TO ICSA IRC ISOLATION VALVE	10	LOW	56	YES	YES	1, 2.c
M-69	2R211XFC0050	SFP COOLING & PURIFICATION TO RX CAVITY IRC ISOLATION VALVE	3	LOW	56	YES	YES	1, 2.c
M-70	2R211XFC0006C	SFP COOLING PUMPS 1A&B SUCTION FROM ICSA IRC ISOLATION VALVE	10	LOW	56	YES	YES	1, 2.c
M-70	2R211XFC0007C	SFP COOLING PUMPS 1A&B SUCTION FROM ICSA ORC ISOLATION VALVE	10	LOW	56	YES	YES	1, 2.c
M-72	2Q061TED0064	CONTAINMENT NORMAL SUMP DISCHARGE	3	*	56	YES	*	
M-72	2Q061TED7800	CONTAINMENT NORMAL SUMP DISCHARGE	3	*	56	YES	*	
M-72(1)	A1APFV2453	CONTAINMENT SUMP POST ACCIDENT SAMPLE	1	*	56	YES	*	
M-75	2R371TPO0217	RCP OIL RETURN	2	*	56	YES	*	
M-75	2R371TPO0218	RCP OIL RETURN	2	*	56	YES	*	
M-76	2R161XRH0063C	RHR LOOP 1C RETURN TO RWST IRC ISOLATION VALVE	8	MEDIUM	55	YES	NO	
M-76	2R161XRH0064C	RHR LOOP 1C RETURN TO RWST ORC ISOLATION VALVE	8	MEDIUM	55	YES	NO	
M-77	2Q271TFP0756	CONTAINMENT FIRE PROTECTION	6	*	56	YES	*	
M-77	2Q271TFP0943	CONTAINMENT FIRE PROTECTION	6	*	56	YES	*	
M-79	2S201TSL0012	SLUDGE LANCING LOW PRESSURE	6	*	56	YES	*	
M-79	2S201TSL0014	SLUDGE LANCING LOW PRESSURE	6	*	56	YES	*	
M-80	2V141TRA0001	RCB EXHAUST | RT-8011 SUPPLY | IRC ISOLATION MOV	1	LOW	56	YES	YES	1, 2.e
M-80	2V141TRA0003	RCB EXHAUST | RT-8011 RETURN | IRC ISOLATION MOV	1	LOW	56	YES	YES	1, 2.e
M-80	2V141TRA0004	RCB EXHAUST | RT-8011 SUPPLY | ORC ISOLATION MOV	1	LOW	56	YES	YES	1, 2.e

*Component has not gone through the risk ranking process.

Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-80	2V141TRA0006	RCB EXHAUST I RT-8011 RETURN I ORC ISOLATION MOV	1	LOW	56	YES	YES	1, 2.e
M-80(1)	A1CMFV4101	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-80(1)	A1CMFV4127	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-80(1)	A1CMFV4128	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-80(1)	A1CMFV4135	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-81	2R371TPO0203	RCP OIL SUPPLY	2	*	56	YES	*	
M-81	2R371TPO0204	RCP OIL SUPPLY	2	*	56	YES	*	
M-82	2Q121TBA0006	BREATHING AIR TO CONTAINMENT	1	*	56	YES	*	
M-82(1)	A1APFV2456	CONTAINMENT ATMOSPHERE POST ACCIDENT SAMPLE	1	*	56	YES	*	
M-82(1)	A1APFV2457	CONTAINMENT ATMOSPHERE POST ACCIDENT SAMPLE	1	*	56	YES	*	
M-82(1)	C1CMFV4104	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-82(1)	C1CMFV4133	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-82(1)	C1CMFV4134	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-82(1)	C1CMFV4136	CONTAINMENT HYDROGEN MONITORING	1	*	56	YES	*	
M-82(4)	2Q121TBA0004	BREATHING AIR TO CONTAINMENT	1	*	56	YES	*	
M-83(3)	2S141TAF0019	(OCIV) AFW TO SG 1D OCIV (AF-MOV-0019)	4	HIGH	57	NO	N/A	
M-83(3)	A1FWFV7192	STEAM GENERATOR 1D PREHEATER ORC BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-84(3)	2S141TAF0085	STEAM GENERATOR 1C ORC AFW ISOLATION MOV	4	HIGH	57	NO	N/A	
M-84(3)	A1FWFV7191	STEAM GENERATOR 1C PREHEATER ORC BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-85(1)	A1APFV2455	RCS POST ACCIDENT SAMPLE	1	*	55	YES	*	
M-85(1)	A1APFV2455A	RCS POST ACCIDENT SAMPLE	1	*	55	YES	*	
M-85(1)	B1PSFV4450	PRESSURIZER VAPOR SAMPLE LINE IRC SOLENOID CHECK VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	B1PSFV4451	PRESSURIZER LIQUID SAMPLE LINE IRC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	B1PSFV4456	RCS HOT LEG SAMPLE HEADER ORC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	C1PSFV4451B	PRESSURIZER LIQUID SAMPLE LINE ORC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	C1PSFV4452	PRESSURIZER VAPOR SAMPLE LINE ORC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	C1PSFV4454	RCS HOT LEG 1A SAMPLE LINE IRC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-85(1)	C1PSFV4455	RCS HOT LEG 1C SAMPLE LINE IRC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-86(1)	A1APFV2454	RHR POST ACCIDENT SAMPLE	1	*	55	YES	*	
M-86(1)	B1PSFV4823	RHR SAMPLE HEADER IRC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-86(1)	C1PSFV4461	RHR SAMPLE HEADER ORC CONTAINMENT ISOLATION VALVE	1	LOW	55	YES	YES	1, 2.b, 2.e
M-86(3)	A1SBFV4186	STEAM GENERATOR 1D SAMPLE LINE ORC FLOW CONTROL VALVE	0.38	LOW	57	NO	N/A	
M-86(3)	A1SBFV4189	STEAM GENERATOR 1A SAMPLE LINE ORC FLOW CONTROL VALVE	0.38	LOW	57	NO	N/A	

*Component has not gone through the risk ranking process.

Penetration	Full Tag	Description	Valve Size	Risk	Current GDC	Current LLRT	Proposed Exemption	Basis
M-86(3)	B1SBFV4188	STEAM GENERATOR 1B SAMPLE LINE ORC FLOW CONTROL VALVE	0.38	LOW	57	NO	N/A	
M-86(3)	C1SBFV4187	STEAM GENERATOR 1C SAMPLE LINE ORC FLOW CONTROL VALVE	0.38	LOW	57	NO	N/A	
M-88	2R341TRD0008	REACTOR COOLANT VACUUM DEGASSING	3	*	55	YES	*	
M-88	2R341TRD0010	REACTOR COOLANT VACUUM DEGASSING	3	*	55	YES	*	
M-94(3)	2S141TAF0048	(OCIV) AFW TO SG 1A OCIV (AF-MOV-0048)	4	HIGH	57	NO	N/A	
M-94(3)	A1FWFV7189	STEAM GENERATOR 1A PREHEATER ORC BYPASS VALVE	4	MEDIUM	57	NO	N/A	
M-95(3)	2S141TAF0065	(OCIV) AFW TO SG 1B OCIV (AF-MOV-0065)	4	HIGH	57	NO	N/A	
M-95(3)	A1FWFV7190	STEAM GENERATOR 1B PREHEATER ORC BYPASS VALVE	4	MEDIUM	57	NO	N/A	
N.A.	A1XCFV1025	PERSONNEL AIR LOCK AIR SUPPLY		*	57	YES	*	
N.A.	A1XCFV1026	PERSONNEL AIR LOCK AIR SUPPLY		*	57	YES	*	
N.A.	A1XCFV1027	PERSONNEL AIR LOCK AUTO LEAK RATE MONITORING		*	57	YES	*	
N.A.	A1XCFV1028	PERSONNEL AIR LOCK AUTO LEAK RATE MONITORING		*	57	YES	*	

*Component has not gone through the risk ranking process.

17. *Appendix J, Option B, stipulates cumulative limits for containment leakage. If certain containment isolation valves (CIVs) are not to be leak tested at all, how will these leakage limits be verified? (The staff notes that since STP is an Option B plant, any changes to the cumulative leakage limit for STP will also require a TS change.)*

RESPONSE:

STP does not plan to revise the allowable leakage values contained in the Technical Specifications. Assuming that the scope of Appendix J is amended as described in Question 16, it is STP's intent to maintain the current Type B and C testing acceptance criteria for those penetrations remaining within the scope of Appendix J. In this way, the penetrations within the revised scope of Appendix J will retain continuity in testing, trending, and schedule. Verification that the Technical Specification allowable leakage values are satisfied will be performed using the individual values for those penetrations remaining within the scope of Appendix J. Those penetrations which have been removed from Appendix J scope by this exemption request will be assumed to contribute zero leakage value. The basis for assuming zero leakage value is that the penetrations removed from the scope of Appendix J have no real contribution to leakage rates which could result in a threat to public health and safety in terms of CDF or LERF. These subject penetrations meet the criteria described in the response to RAI #16.

18. *Please identify which configurations of CIVs will not be leak tested under the Appendix J exemption? An example of a configuration is a closed system with a single isolation valve at each penetration.*

RESPONSE:

The response to RAI #16 and its accompanying table describe the configurations of CIVs which will not be leak tested under the Appendix J exemption.

19. *STPNOC states that its snubber testing program will be modified to remove safety-related LSS and NRS snubbers from the scope of the program.*

    (a) *Please explain the process and criteria for categorizing safety-related snubbers as LSS or NRS.*

    (b) *How will the snubbers' purpose of protecting the safety function of a system (and not necessarily the functions of a specific component) be considered?*

    (c) *Also, STPNOC should discuss what other activities will provide reasonable confidence that the safety-related piping system which contains the affected snubbers will be able to perform its intended safety function if the snubbers are removed from the testing program.*

RESPONSE:

a) **STP's process assigns snubbers the same risk categorization as the pressure boundary function for the portion of the system that the snubber is located on. This is a conservative convention because snubber failure leading to a piping system failure is a highly unlikely event, as discussed below:**

   1) **Even though the snubber is designed to protect the system during a seismic event, the more credible failure mode would be failure of a snubber to allow for thermal movement during normal operations. If such a failure were severe enough to cause overstressing, it would exhibit itself first through deformation of the snubber itself or to its supports. It is highly unlikely that the piping would be damaged and even if it were, it would be through plastic deformation and/or through a leak-before-break scenario. Piping leaks would become quickly evident during scheduled operator walkdowns, system engineer walkdowns, or other visual or system performance indication. The probability of such an unlikely event occurring at the same time as a safety system being demanded to support accident or transient mitigation is even more remote.**

   2) **The ASME piping is robustly designed that failure of a snubber is highly unlikely to lead to a failure of the piping/component.**

STP's position regarding the robustness of the ASME-designed piping and the unlikeliness of snubber failure leading to piping failure is consistent with the research results identified in EPRI report TR-110381, <u>Risk-Based Snubber Inspection and Testing Guidelines</u>. Relevant excerpts from this report are provided below:

- **"Internal initiating events are the primary source of initiating events, since a locked up (fail rigid) snubber creates a more severe transient for the pipe segment or component than the response to a dynamic event. Even though the external initiating event (seismic) is probably the more relevant dynamic event (for which the snubber was typically designed to protect the system), the robustness of the ASME design for**

pressure integrity (see Appendix A) causes this external (seismic) initiating event to be less severe than the internal event described above [at beginning of this paragraph]." - From section 4.2 of report.

- **"The typical failure modes for a snubber are to "fail rigid"(especially for a mechanical snubber) and to "fail free" (especially for a hydraulic snubber). Since the normal role for a snubber is to move to accommodate thermal movement of the system piping during the typical operating cycle, the "fail rigid" failure mode might impose additional loads on the system (depending on amount of normal thermal movement during the operating cycle). However, as [testing referenced in] (Appendix A) demonstrated, the piping system is so robustly designed per ASME design rules, that the "fail rigid" snubber is highly unlikely to cause a piping system failure. Additionally, the "fail free" failure mode of the snubber is likewise highly unlikely to cause piping system failure, because the ASME design rules create a substantially stiff system that can accommodate this failure mode with relative ease (Appendix A)." – From section 4.3 of report.**

Therefore, STP considers that a piping system failure resulting from failure of a snubber either from an external or from an internal event is highly unlikely.

b) **During a seismic event and under some water hammer conditions, snubbers are designed to prevent sudden movements of piping and components that could, if unchecked, result in excessive stresses and potential breach of the pressure boundary. During normal operations, snubbers allow the piping and components to move in order to accommodate thermal growth.**

As discussed in the EPRI report, the typical failure mechanism for a mechanical snubber is to "fail rigid". Such a failure would impact the function of the snubber to allow for thermal growth, but would not impact its ability to restrain the pipe segment or component during a seismic event or under water hammer conditions. The majority of snubbers at STP are of mechanical design. The only hydraulic snubbers are located on the steam generators and have already been risk ranked as MSS. Therefore, all of the LSS and NRS snubbers are mechanical. Any increases in failure rates for these snubbers would thus not impact the safety function of the system during a seismic event, but could potentially be a factor during normal operations. However, as concluded in the EPRI report, the ASME-designed piping is so robustly designed for pressure boundary integrity that a "fail rigid" snubber is highly unlikely to cause piping system failure. Thus, the safety function of the system would not be affected.

Snubbers categorized as LSS or NRS are located on sections of the piping system where the pressure boundary function has been risk categorized as LSS or NRS. Therefore, even assuming that the piping fails, such a failure would exhibit itself through a leak-before-break condition. The resulting pressure boundary loss would not significantly impact the Medium or High risk significant functions of the system, if any, since the pressure boundary function in the area of the snubber is LSS or NRS.

c) **STP notes that implementation of snubber risk results will generally be focused on snubber in-service testing. The risk-informed evaluation of snubber in-service testing will be performed by a separate Working Group, similar to the MOV Working group, that will consider other related factors such as snubber service environment, monitoring and testing data, testing methods, and other considerations highlighted in the EPRI report. The recommendations of the snubber Working Group would require approval by the Comprehensive Risk Management Expert Panel before any revised testing strategy would be implemented.**

As documented in the EPRI report discussed above, the more severe transient for the pipe segment results from a locked up (fail rigid) snubber preventing the thermal movement of the pipe. Because the piping system is so robustly designed per ASME design rules, the "fail rigid" snubber is highly unlikely to cause a piping system failure. However, assuming that an LSS or NRS snubber were to cause such a failure, it would affect only the LSS or NRS pressure boundary function. Such a failure would be captured under STP's corrective action and feedback process, which would ensure that the appropriateness of the snubber's testing strategy and/or its risk significance is re-evaluated in light of the failure.

20. (a) *Explain how the common cause failure (CCF) basic event importance measure is estimated for the proposed exemptions. Explain the difference between the current method and the method reported in STP's graded quality assurance (GQA) program submittal dated August 4, 1997. Provide the basis for the new estimation method.*

RESPONSE:

STP Nuclear Operating Company uses RISKMAN® to quantify the Probabilistic Risk Assessment (PRA) model. For each full scope model quantification used in the various sensitivity studies associated with the PRA risk categorization process, a basic event importance file is generated. A full scope model quantification for the STP PRA model is a Level 1 or 2 At-Power PRA quantification including external events, internal fires and internal floods. This information contains, among other parameters, Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance values for each basic event and common cause "event" or "term" in the model.

The previous methodology for determining the PRA component risk categorization as described in an RAI dated November 6, 1997 used the following process:

- **the basic event importance files were generated from each RISKMAN® sensitivity study, and**
- **the basic event importance measures were "rolled up" into component importance measures.**

The "roll up" is accomplished as follows:

- **The component FV importance is calculated as the sum of the basic event and associated common cause term FV importance values.**
- **The component RAW is calculated as follows:**

$$RAW_{Comp} = 1 + \sum_{i=1}^{n} (RAW_i - 1)$$

Where,

RAW_i is the RAW value of a basic event and/or common cause term associated with the component of interest, and
RAW_{comp} is the combined RAW value for the component as a whole, including all associated common cause failure term impacts.

The important issue here was including the complete common cause term importance value for each and every associated component in a common cause group. This approach is extremely conservative and greatly over-estimates the importance based on double counting the common cause terms.

For example, consider a common cause group which is represented by three similar components, (e.g., pumps) in a symmetrical functional alignment at the plant. If system success criteria requires one of three trains of the system to be successful, and the independent basic event failure modes for the three components are represented by A, B,

and C, then the minimal cut sets for this function can be represented as follows: AB, BC, AC, [AB], [BC], [AC], and [ABC] where the terms in brackets represent common cause failure terms. The previous method for "rolling up" the importance's of these terms to their respective components includes the importance terms for each of the following:

- **Component A: A, [AB], [AC] and [ABC].**
- **Component B: B, [AB], [BC] and [ABC].**
- **Component C: C, [AC], [BC] and [ABC].**

As can be seen in this example elements of [AB], [BC], [AC] and [ABC] are counted more than once which results in an overly conservative estimate.

Thus, over counting of the doublet and triplet importance terms occurs in the overall computation of component importance measures. When more than three terms are included in a common cause group cut set, this multiple counting of the importance is further exacerbated (i.e., quadruple counting of four term common cause events, quintuple counting of five term common cause events, and so on). In reality, the common cause failure terms or cut sets are separate events in the risk model, and therefore, it is difficult to define how the importance of these dependent events should be accounted for in individual component risk categorization processes. However, it is evident that multiple counting of the importances from these events common cause is overly conservative.

In order to eliminate some of the conservatism associated with the above process, STP now splits the importance of multiple term common cause failure events evenly among their constituent components. For example, considering the case above with a common cause group with three similar components, an individual component, A, importance includes the whole contribution of the independent failure and partial contribution of the common cause event. Mathematically, the Fussell-Vesely importance for component A is represented by:

$$FV_{Comp\ A} = FV_A + \tfrac{1}{2} * FV_{[AB]} + \tfrac{1}{2} * FV_{[AC]} + \tfrac{1}{3} * FV_{[ABC]}$$

Where,

$FV_{Comp\ A}$ represent the total FV importance of component A,
$FV_{[AB]}$ represents the FV importance of the common cause event between component A and component B, and
$FV_{[ABC]}$ represents the FV importance of the common cause event between components A, B and C.

The common cause event term (e.g., $FV_{[AB]}$) is multiplied by 1/3 to prevent triple counting. The generic equation for determining the FV component importance associated common cause events is:

$$FV_{Comp\ x} = FV_x + \tfrac{1}{2} * FV_{Doublet} + \tfrac{1}{3} * FV_{Triplet} + \tfrac{1}{4} * FV_{Quadruplet} + \cdots$$

Where,

$FV_{Comp\ x}$ represents the total FV importance of component x.

Mathematically, the Risk Achievement Worth for component A is represented by:

$$RAW_{Comp\ A} = 1 + RAW_A - 1 + \tfrac{1}{2} * RAW_{[AB]} - 1 + \tfrac{1}{2} * RAW_{[AC]} - 1 + \tfrac{1}{3} * RAW_{[ABC]} - 1$$

Where,

$RAW_{Comp\ A}$ represent the total Risk Achievement Worth of component A,
$RAW_{[AB]}$ represents the Risk Achievement Worth of the common cause event between component A and component B, and
$RAW_{[ABC]}$ represents the Risk Achievement Worth of the common cause event between components A, B and C.

The generic equation for determining the RAW component importance associated common cause events is:

$$RAW_{Comp\ x} = 1 + RAW_x + \tfrac{1}{2} * RAW_{Doublet} + \tfrac{1}{3} * RAW_{Triplet} + \tfrac{1}{4} * RAW_{Quadruplet} + \cdots - n$$

Where,

$RAW_{Comp\ x}$ represents the total RAW importance of component x, and
n represents the number elements, basic event and/or common cause events.

STP has also performed a sensitivity study to determine the impact of the previous overly conservative method of including the double, triple and even quadruple counting of common cause.

The following table represents the results of PRA rank categorization:

Category	No. of Changes
Medium-R to High	26
Medium to Medium-R	0
Low to High	0
Low to Medium	20
No change	1068
Total	1114

Notes:
- Medium-R represents components with RAW values between 10 and 100, and
- No components decreased in rank.

The following table represents the component type associated with those components that did change ranks:

Component Type	Number of Components
Circuit Breakers	3
Dampers	6
Valves	37

The above 46 components are encompassed by 7 systems. These systems are:

System Designator	System Description	# of Components
CC	Component Cooling Water	6
DG	Standby Diesel Generator	3
HE	Electrical Auxiliary Bldg HVAC	6
MS	Main Steam*	20
PK	4kV AC Class 1E Power	3
RH	Residual Heat Removal*	6
SI	Safety Injection*	2
*Ranking results from this sensitivity study equate to the final ranking.		

Using the approach from the previous overly conservative methodology would result in the re-categorization of only 15 components in the Component Cooling Water, Standby Diesel Generator, and Electrical Auxiliary Bldg HVAC systems. The final risk categorization from the three of the other four systems (MS, RH and SI) would have no impact since the components in these system are already deterministically evaluated to be equivalent to the sensitivity study results. The 4kV AC Class 1E Power system has not yet been evaluated by the risk ranking process.

There are two main advantages in using the current approach. First, each component's importance measure includes contributions from independent failures and common cause events with respect to both accident/transient initiation and mitigation. Second, the importance of an individual component is not overstated and more realistically represents the true importance to the overall plant. The current methodology has evolved since 1997 in order to remove of some of the conservatism associated with the previous approach.

(b) In Section 5.2.4.1 of the submittal, it is indicated that the same PRA tools used for the GQA program will be used for the proposed exemption. In addition to the method of estimating CCF, identify other changes made, if any, to the categorization process since the GQA submittal was approved on November 6, 1997.

RESPONSE:

As outlined in the response to part (a), the method for PRA risk categorization has evolved to more accurately reflect a component's true importance with respect to common cause factors, accident initiation, and mitigation. Another change in the risk categorization process, as outlined in the SER (*Graded Quality Assurance, Operations Quality Assurance Plan (Revision 13), South Texas Project, Units 1 and 2 (STP)(TAC Nos. M92450 and M92451)*, November 6, 1997), is a process outlined in section 3.2.3, <u>Qualitative Categorization Methodology</u>. The first sentence in the second paragraph states:

"To expand the categorization to SSCs not modeled in the PRA (and accept the appropriateness of reduced QA controls on safety-related MSS-2 and LSS SSCs modeled in the PRA), the WG identifies and documents every component attribute which supports any HSS system function."

Attributes for HSS safety related components are identified, and each of these attributes is considered a critical attribute. For MSS and LSS safety related components, only the critical attributes are identified and documented. For non-safety related components only the HSS and MSS components have critical attributes identified and documented. However, STP does not identify and document every component attribute that supports any HSS system function as stated in the GQA SER.

The final change in the risk categorization process is associated with determining the importance of system functions. See the response to question 31 for more details on this change.

21. *Regulatory Guide 1.174 states that "all safety impacts of the proposed change are [to be] evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis..."*

    (a) *Provide a discussion on the aggregate impact of the proposed exemptions on plant risk in terms of CDF and LERF.*

    In Section 5.2.4.1, pages 16 and 17 of the submittal, it is stated that "STP performed sensitivity studies in which unreliability was simultaneously increased for medium safety significant and low safety significant SSCs of a similar type within the scope of the PRA. These studies evaluated the impact of increasing the unreliability of the group of SSCs by as much as an order of magnitude. Based upon these studies, STP determined that increases in the failure rate by as much as an order of magnitude had little, or no, impact on the final SSC risk categorization."

RESPONSE:

All equipment necessary to mitigate the consequences of initiating events are included in the plant PRA. Changes to the risk significance of components included in the PRA will not result in removal of the equipment from the model. As the Exemption is fully implemented, changes in equipment failure rates, if they occur, will be identified by the Maintenance Rule Program (Maintenance Rule monitoring of LSS/NRS SSCs consists of Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function) or the Corrective Action Program and the new failure rates incorporated into the PRA model during the cycle updates. Requantification of the model with the changed failure rates may result in a change to the component's risk ranking. However, based on evidence being collected to support the Balance of Plant model, the failure rates for most equipment whose QA requirements are relaxed will not change significantly.

Therefore, we expect no impact on plant risk in terms of core damage frequency or large early release frequency. Any change to core damage frequency or large early release frequency is expected to be gradual and detectable before a significant impact to core damage frequency of large early release frequency occurs.

b) *Provide the details and the results of the sensitivity analyses. It is unclear to us whether unreliability of all groups of SSCs were increased by an order of magnitude. If you assumed that the increase in unreliability is varied for different groups of SSCs, explain the basis of your assumption.*

RESPONSE:

The first sensitivity study involved modifying the failure rate for check valves. Check valves were selected on the basis that most of the valves would have a low ranking in the PRA. Another factor was that check valves experience both a passive (transfer close/open) and active failure (fail to open/close on demand) mode. Check valves in general have low failure rates which is ideal for changing the failure frequency by factors of 2, 5 and 10. This study of check valve failure rates resulted in no re-categorization of components from low to high. Only one check valve would have changed from low to medium, however, other sensitivity studies had already re-categorized this valve in the medium category.

Another sensitivity study was performed to show the impact of postulating increased failure rates for low ranked components to the CDF and LERF. The approach of the study was to increase the component failure rate by a factor of 10 for all components ranked LSS. There are 431 components categorized as LSS and which are modeled in the PRA. The results are as follows:

	Current Average (events/reactor year)	Sensitivity Study λ_{LSS}*10 (events/reactor year)	Increase	% Increase
CDF	9.0781E-6	9.3232E-6	2.4510E-7	2.7%
LERF	1.3742E-7	1.3911E-7	1.6900E-9	1.2%

In all cases increasing the failure rates of LSS components by a factor of 10 was greater than the 95th percentile for each of the LSS component failure rate distributions.

The above increases in CDF and LERF are with the acceptance guidelines for very small changes as outlined in Regulatory Guide 1.174. The acceptance guidelines are 1E-6 delta CDF and 1E-7 delta LERF. Results from this study are small and consistent with the intention of the Commission's Safety Goal Policy Statement.

Additional sensitivity studies on other equipment groups have been performed for other plant applications. Analyses have been completed for solid state protection system relays that investigated the effects of increasing failure rates by factors of ten and one hundred. No significant change in core damage frequency was seen with an increase in relay failure rates of one hundred. This is primarily due to redundancy (two out of four relay logic).

c) Identify the "types" of SSC selected, and define how a "group" was chosen.

RESPONSE:

Check valves were the only group selected for this sensitivity case study. The failure rates for both passive and active failure modes were changed at the same time.

(d) Explain why you only increased the failure rates one group at a time. Discuss if any of these studies lead to any changes in the categorization.

RESPONSE:

For the only group, check valves, the component failure rates for both passive and active failure modes were increased by a factor of 2, 5 and 10. There was only one component that changed categories from low to medium. This component was just inside the low ranking boundaries and changed to medium ranking when the failure rate was increased by a factor of 10. However, the composite rank for the check valve in question had already been ranked medium due to the importance of the valve during several planned maintenance evolutions. Therefore, this sensitivity study, in and of itself, did indicate that the overall risk ranking for the check valves was not changed. This same result would be expected for other component types if evaluated.

(e) Discuss how these sensitivity studies account for potential common mode failure in diverse and redundant systems under postulated accident conditions.

RESPONSE:

Common cause failure in multiple train systems (e.g. ECW, CCW, etc.) is explicitly modeled in the RISKMAN systems analyses for all active components within a system. Any change in the underlying basic event probability of failure is automatically carried through the quantification of the system including common cause. Other dependent failures which could affect multiple components, such as single point failure (tanks, etc.) are explicitly considered in the system and event tree models. Also, external events (fires, floods, seismic, etc.) are explicitly included in a similar fashion as single point failures. Thus, increase in the underlying failure is included in the quantification.

Potential common mode failures in diverse systems are explicitly modeled in the RISKMAN system models for some basic events such as 4kV breakers. For these components any increase in the basic failure rate data is quantified as described above. For other types of equipment, such as MOVs, potential changes in the underlying basic event failure data are not carried across diverse systems (i.e., intra system effects). This is because of the unique operating conditions for the diverse systems which affects the failure rates of such components.

22. *During the review of the Safety Injection (SI) system at STP, the staff noted that the system binder contained a general note allowing the limit switches which are used in actuation of critical components to be rated as LSS. However, upon inquiry from the NRC staff, the licensee stated that this note has been revised by a new note and the new note does not generalize the categorization of limit switches used for actuation of other components. Upon review of the SI system binder, it was determined that the SI system review was done based on the original note in the binder and was not based on the revised note.*

    (a) *Describe the general quality assurance program that is being or will be applied by STPNOC, and what corrective actions are being taken, on its risk categorization process to avoid these types of errors.*

    (b) *The staff also requests that the licensee justify this discrepancy not only for the SI system, but for all other systems where the old note has been listed in the system binder.*

    (c) *Also, the licensee should provide assurance that any other general note which has been revised such that it can affect the categorization of components, has been evaluated for the affected systems and the categorization of the components has been corrected if needed.*

RESPONSE (part a):

The provisions of the Operations Quality Assurance Plan (OQAP), Chapter 15.0, Quality Oversight Activities, govern the oversight of the risk categorization process. The program implemented by Chapter 15 provides for independent oversight activities (including audits, assessments, evaluations, performance monitoring, and surveillances) to ensure that the requirements of the Operations Quality Assurance Program are being properly implemented.

STP has performed a focused assessment on application of General Notes affecting limit switches. The results of this assessment are provided in part (c) of this response. In addition, STP will perform a broader review of all General Notes to ensure consistency and appropriateness in the application of the General Notes. Procedural guidance will also be added to 0PEP02-ZA-0001, Graded Quality Assurance Process, to clarify the control, use, and revision of General Notes in the risk categorization process. The general notes have also been revised to provide additional justification and clarification. These revised general notes are provided as an attachment to this response.

As detailed in the additional responses that follow, a condition report has been initiated to specifically re-evaluate limit switches that support actuation of risk significant components. The Corrective Action Program (CAP) supports the implementation of the OQAP, Chapter 13.0, Control of Conditions Adverse to Quality. This process requires that conditions be evaluated and resolved, that generic implications be addressed, and that actions to prevent recurrence are implemented, as appropriate.

RESPONSE (part b):

As with the risk categorization methodology, the development of the existing set of General Notes was an evolutionary process. Initially, STP used General Notes as a means to more efficiently document the risk bases for large numbers of similar components, such as vent and drain valves and indication-only instruments. General Notes were developed each time a new system was evaluated for risk categorization, and the developed General Notes were specific to that system.

Over time, it became apparent that improved consistency, justification, and efficiency could be obtained if one set of General Notes, applicable to all systems, was developed. This set of "Generic Notes" was specifically approved by the Expert Panel, and use of Generic Notes began in mid-1999. The Safety Injection system was one of the last systems to utilize the old-format notes.

RESPONSE (part c):

As stated, STP is in the process of conducting a review of all General Notes to ensure consistency and appropriateness in the application of the General Notes. Specific for the categorization of limit switches, it has already been determined that none of the other systems' notes made reference to limit switches except for the Fuel Handling Building HVAC (HF) system. For the HF system, the limit switch note references indication-only switches. This General Note specifically excluded switches involved in the actuation of components.

STP has evaluated the noted discrepancy on the Safety Injection (SI) limit switch involved in the actuation of critical components. While the initial review had identified that the limit switch needed to be categorized higher than its current risk of LSS, a more detailed evaluation concluded that the LSS risk was appropriate because the safety function of the critical component would not be impacted (Refer to detailed discussion at the end of this response). The scope of the evaluation also included other systems that have been categorized. Although the evaluation has not yet been completed, results to date show that the great majority of the limit switches are appropriately categorized and that none require a categorization higher than LSS.

Recognizing that the Risk Significance Basis Document (RSBD) is a "living" document, STP had, prior to identification of this discrepancy, initiated a mechanism for identifying and capturing needed changes to the RSBDs, utilizing the Corrective Action Program. As part of this program, STP intends to revise the affected RSBDs to reflect the current general notes, among other updates, during the 6-month review process. The revision process will ensure that the risk categorization of previously evaluated components is consistent with the system's revised set of general notes, and, if not, that the risk rank is revised as needed or appropriate justification is provided.

Results of SI limit switch review:

***A1SIZSC0016A* – Containment Emergency Sump 1A to SI Train A pump suction ORC isolation MOV Stem Mounted Limit Switch. The purpose of this limit switch is to provide an interlock for the mini-flow recirc line MOVs on the HHSI pumps. Before these mini-flow recirc line MOVs can be opened, close indication for the containment emergency sump suction MOV (SI-MOV-0016A) must be obtained. See logic diagram 5N129Z42002 for reference. This is the only interlock function that ZSC0016A provides.**

During a site visit by the NRC in August 1999, this limit switch was mistakenly identified as providing an interlock for the switchover of injection from the RWST to recirculation from the containment sumps. Neither the logic diagram (5N129Z42001) nor the elementary drawing (9-E-SI04-01) supports ZSC0016A providing logic to switchover. However, there is a limit switch that is internal to SI-MOV-0016A that does provide an interlock signal for switchover. Internal limit switches do not have a separate TAG/TPNS indentifier and have not been risk categorized. They are considered part of the host component (in this case, MOV operator SI-MOV-0016A, which is categorized as HSS).

GENERAL NOTES

#	SUBJECT	SCOPE	RISK (see remarks)	BASIS
1	Vent , drain, test valves	1 inch or less in size	NRS	Normally closed and capped. Gross leakage not credible. Good reliability based on STP and industry experience. Operator rounds are conducted periodically and would quickly identify any leakage. The Configuration Management program, which includes initial valve lineups, the Equipment Clearance Order process, and independent/dual verifications, provides adequate controls of valve position and ensures that the valve is capped.
2	Normally open manual valves in main flow path	Does not include throttle valves	Same as pressure boundary risk	An open valve is essentially a piece of pipe. Valve disk failure in a manner which would impede flow is not considered a credible event. These valves are locked open or locked-in-place, where additional assurance is required. Gross leakage not credible. Good reliability based on STP and industry experience. Operation of the system and the monitoring of system parameters are other indicators of proper valve status. Operator rounds are conducted periodically and would quickly identify any leakage. The Configuration Management program, which includes initial valve lineups, the Equipment Clearance Order process, and independent/dual verifications, provides adequate controls of valve position and ensures that the valve is locked, if applicable.
3	Other valves not included in Notes 1 and 2 above, including instrument root valves and branch line valves	a. 1 inch or less in size b. Size of valve relative to main process piping is small	NRS for pressure boundary purposes only	Gross leakage not credible. Good reliability based on STP and industry experience. Operator rounds are conducted periodically and would quickly identify any leakage. The Configuration Management program, which includes initial valve lineups, the Equipment Clearance Order process, and independent/dual verifications, provides adequate controls of valve position.
4	Snubbers		Same as pressure boundary risk	Even though the snubber is designed to protect the system during a seismic event, the more credible failure mode would be failure of a snubber to allow for thermal movement during normal operations (fail rigid). If such a failure were severe enough to cause overstressing, it would exhibit itself first through deformation of the snubber itself or to its supports. It is highly unlikely that the piping would be damaged (EPRI report TR-110381) and even if it were, it would be through plastic deformation and/or through a leak-before-break scenario. Piping leaks would become quickly evident during routine operator rounds, system engineer walkdowns, or other visual or system performance indication. The probability of such an unlikely event occurring at the same time as a safety system being demanded to support accident or transient mitigation is even more remote. Piping failure during a seismic event from a "fail free" snubber is also very unlikely due to the robustness of the ASME-designed systems (EPRI report TR-110381). Snubber is conservatively assigned the same risk as the pressure boundary risk for the portion of piping that the snubber is located on.

Remarks: 1. Unless ranked higher by the PRA.

2. When a critical attribute is provided for a component, it is understood that the critical attribute must function sufficiently enough to meet the design functional requirements associated with that attribute. For example, the attribute "Permit Flow in normal direction", as given to a check valve is understood to mean that the check valve must not only open in the normal direction of flow, but must open sufficiently enough to meet design flow requirements.

3. For a valve, the critical attribute of "pressure boundary" means ability to contain the fluid if the valve is normally open and ability to contain the fluid and isolate the line if the valve is normally closed.

4. Closed and capped 1 inch or less test valves that are part of the containment isolation boundary fall under the scope of Note 1 and are NRS.

GENERAL NOTES

#	SUBJECT	SCOPE	RISK (see remarks)	BASIS
5	Instrument Indication and recorders, including supporting devices such as transmitters, etc.	a.Visual indication only. Not involved in the generation of alarms or actuation signals b. Not identified by Operations as being critical	NRS	Failure would not affect risk significant system functions. The majority of these are local indicators. Diverse indication is typically available.
6	Handswitches, Control Room	If controlled component has some risk significance, risk of switch cannot be NRS	1 Level lower than controlled component	Reliability of handswitches has been very good. Local/ASP redundant switch available. Most time sensitive operations are automatic, do not require switch manipulation, and rely only on handswitch circuit continuity for success. The probability of a circuit continuity failure in a static role is very low and is clearly less than the probability of failure for the controlled component, which must change state. Automatic safety systems are periodically tested and these tests include the automatic initiation circuitry. In addition, handswitches are manipulated on a regular basis as part of routine operations. Any failure in the handswitch or its associated electrical circuitry would manifest itself during these operations.
7	Handswitches, Transfer (between control room and local/ASP)	If controlled component has some risk significance, risk of switch cannot be NRS	2 Levels lower than controlled component	Reliability of handswitches has been very good. Preferred method is to use control room switch. Transfer switch is normally positioned for control room operations. Thus, transfer switch would not normally have to be manipulated. Only function is circuit continuity. The probability of a circuit continuity failure in a static role is very low and is clearly less than the probability of failure for the controlled component, which must change state. Automatic safety systems are periodically tested and these tests include the automatic initiation circuitry.
8	Handswitches. Local or on Aux Shutdown Panel	If controlled component has some risk significance, risk of switch cannot be NRS	2 levels lower than controlled component	Reliability of handswitches has been very good. The need to use this switch would mean failure of the automatic initiation, if applicable, and either a malfunction in the control room switch or a need to evacuate the control room, both highly unlikely events.
9	Pressure boundary	Low pressure and / or high volume system	LOW	Low pressure high volume characteristics of system mean that credible leakage would not have a significant impact on system operation. Typically, there are means for make-up to the system. Reliability in this area has been good

Remarks: 1. Unless ranked higher by the PRA.

2. When a critical attribute is provided for a component, it is understood that the critical attribute must function sufficiently enough to meet the design functional requirements associated with that attribute. For example, the attribute "Permit Flow in normal direction", as given to a check valve is understood to mean that the check valve must not only open in the normal direction of flow, but must open sufficiently enough to meet design flow requirements.

3. For a valve, the critical attribute of "pressure boundary" means ability to contain the fluid if the valve is normally open and ability to contain the fluid and isolate the line if the valve is normally closed.

4. Closed and capped 1 inch or less test valves that are part of the containment isolation boundary fall under the scope of Note 1 and are NRS.

GENERAL NOTES

#	SUBJECT	SCOPE	RISK (see remarks)	BASIS
10	Containment Isolation	Line penetrating containment is part of a water system	LOW	Leakage paths that would threaten public health and safety are not credible. Failure of a containment isolation valve that is normally closed or that closes upon receipt of a containment isolation signal would not lead to a radiation release to the outside environment unless multiple failures of equipment occur at nearly the same time. A loss of coolant accident must occur along with a piping break and failure of the redundant containment isolation valve to close. Containment isolation valves that are required to be open during accident conditions are in a closed water system which is under duty during accident conditions and, therefore, represent pathways for mass and inventory to enter containment and, if exiting containment, represent mass and inventory which is contained in a closed system. In addition, the piping systems have a much higher pressure rating than the containment building.
11	Alarm Instrumentation		No higher than LOW	Provides useful information to operator, but failure would not, in and of itself, fail a risk significant system function. Diversity of alarm indication and system parameter indication are typically available.
12	Panels, Enclosures, and Terminal boards		No higher than LOW	Ranked LOW if they contain risk significant components; otherwise ranked NRS. Passive and inherently reliable device, based on STP and industry experience.
13	Limit Switches	a. Indication only, i.e., does not provide actuation signal b. Not identified by Ops as being critical	NRS	Indication only. Failure would not, in and of itself, fail a risk significant system function. Diversity is available through other means, such as indication of flow, pressure, etc. In addition, valves and HVAC dampers are manipulated on a regular basis as part of routine operations. Any failure in the associated position limit switches or in the associated electrical circuitry would manifest itself during these operations.

Remarks: 1. Unless ranked higher by the PRA.

2. When a critical attribute is provided for a component, it is understood that the critical attribute must function sufficiently enough to meet the design functional requirements associated with that attribute. For example, the attribute "Permit Flow in normal direction", as given to a check valve is understood to mean that the check valve must not only open in the normal direction of flow, but must open sufficiently enough to meet design flow requirements.

3. For a valve, the critical attribute of "pressure boundary" means ability to contain the fluid if the valve is normally open and ability to contain the fluid and isolate the line if the valve is normally closed.

4. Closed and capped 1 inch or less test valves that are part of the containment isolation boundary fall under the scope of Note 1 and are NRS.

23. *During the August 31, 1999, meeting, the licensee informed the staff that certain electrical components may continue to be classified as HSS or MSS, while the attached mechanical components are classified as LSS or NRS. Also, during the same meeting, the licensee informed the staff that components which perform a support function for HSS and MSS systems or components, will have the same HSS or MSS classification as the supported systems or components. Therefore, please describe:*

    (a) The process criteria or rules for classifying inter-connected and supporting components (e.g., electro-mechanical components, supporting systems or components) including consideration of functional dependencies, and

    (b) The process criteria that will be implemented to ensure that HSS or MSS electrical components will remain functional including consideration of potential adverse spatial interactions between mechanical and electrical components.

RESPONSE (part a):

The process for classifying supporting components centers on the impact and probability of failure on the primary component. For a typical electro-mechanical device, the mechanical component is tasked with supporting one or more system functions and the associated electrical component provides the motive power to the mechanical component. For example, a motor operated valve may be ranked as MSS because its failure to change state would fail a system function ranked MSS. The motor operator would then be ranked MSS because its failure would prevent the valve from changing state and would therefore fail the MSS function. Another example illustrates differences in risk between the primary component and its support component. A pump may support two system functions. The first function, which is ranked LSS, is to move fluid through that part of the system. The second function is pressure boundary, which is ranked MSS. The pump is therefore ranked MSS because one of its failure mechanisms (loss of pressure boundary) would fail the MSS function. The pump motor, on the other hand, is ranked LSS because its failure would prevent the pump from moving fluid but would not affect its pressure boundary integrity. Thus, only the LSS function would be impacted by an electrical failure involving the pump.

For a component whose failure could cause the failure of electrically interconnected components, the classification process involves an evaluation of the potential failure modes, the probability of failure, the impact of failure on the interconnected components, and the risk significance of the interconnected components. Under this process for example, a breaker feeding a single LSS load may be ranked as MSS because the evaluation concludes that failure of the breaker could credibly fail the upstream motor control center that is ranked MSS. The electrical load would remain as LSS, however, since electrical failure of the load would not credibly cascade past the breaker.

RESPONSE (part b):

As noted above, an electrical component that is physically attached to and provides motive power for its mechanical counterpart provides a support function to the mechanical component and would not typically be ranked higher than the mechanical component. If it is shown that the electrical component could credibly fail in such a manner as to fail other electrically connected components, then the subject electrical component would be risk ranked the same as the highest-ranked electrically-connected component that is impacted. Under such a scenario, the risk of the electrical component could be higher than that of the attached mechanical component. If this case were to occur, the Working Group would assess the appropriateness of the mechanical component's risk categorization, factoring in the mechanical component's failure potential to also fail the attached electrical component. Where appropriate, based on credible failure mechanisms and Working Group insight, the risk categorization of the mechanical component would be revised to match that of its electrical counterpart.

For additional insights and considerations, please refer to the responses to questions 9, 12, and 24.

24. *Attachment 1, Section 4.1.2, of the July 13, 1999, exemption request, states " . . . LSS and NRS Class 1E electrical equipment could be replaced with non-Class 1E equipment. In such an event, STP will take actions as necessary to ensure that HSS and MSS Class 1E components are appropriately protected per the requirements in the UFSAR." Based on information conveyed by representatives from STPNOC during an August 31, 1999, meeting, it is our understanding that:*

- *Safety-related (Class 1E) electrical SSCs, which are not (or may not be) subjected to the STPNOC process for categorization and treatment, will be considered safety-related HSS SSCs;*

- *STPNOC will take actions as necessary to ensure that these safety-related HSS electrical SSCs (as well as other safety-related HSS and MSS SSCs) will be appropriately protected per the requirements in the UFSAR, and;*

- *Safety-related electrical SSCs classified to LSS or NRS by the STPNOC process (which no longer meet any one or more of the special treatment requirements for which an exemption is being requested) will be isolated from HSS and MSS electrical systems the same way non-safety-related SSCs are isolated per UFSAR design commitments. Safety-related LSS or NRS SSCs will be isolated using qualified safety-related (Class 1E) isolation devices defined in the UFSAR and cables from the isolation device to the safety-related LSS or NRS SSCs will be routed in raceways that are separated from raceways or electrical containment penetrations which contain cables which serve safety-related HSS and MSS SSCs.*

Please confirm this understanding. If protection may be different from our understanding described above, provide the licensee's design criteria for providing electrical isolation and the proposed regulatory process for implementing the design changes necessary to provide this isolation. We acknowledge that the licensee's response to this question is dependent upon its response to an earlier question (RAI Question 12) about the extent to which electrical systems will be exempted from special treatment provisions.

RESPONSE:

- **Safety related (Class 1E) electrical SSCs that are not risk categorized will not be in the scope of this exemption request and will retain current special treatment requirements.**

- **STP will ensure that the above SSCs as well as electrical SSCs categorized as HSS or MSS are appropriately protected per the requirements of the UFSAR.**

- **Currently installed, safety related electrical SSCs are fully qualified Class IE devices. Those that are categorized as LSS or NRS will, upon grant of the exemption, be removed from the scope of the special treatment requirements. However, they will continue to be classified as Class IE and would not need to be isolated from HSS or MSS Class IE components.**

- **When the need arises, STP may replace an existing LSS or NRS Class IE component with a commercial grade component that meets the existing functional design requirements, including environmental parameters. As described in the response to RAI Question 12, an engineering evaluation will be performed to provide reasonable assurance that the replacement component satisfies the required form, fit, and function, will not degrade other IE equipment, and is capable of meeting the design functional requirements (including seismic and environmental considerations); however, the component will not be specifically qualified. By performing this evaluation, the component can be classified as Class IE and would not need to be isolated from HSS or MSS Class IE SSCs.**

- **If a replacement component does not satisfy the above requirements, it will be classified as Non-IE and will be required to meet the UFSAR commitments concerning isolation from IE circuits. In addition, the circuitry downstream of the required isolation devices would be properly separated from Class IE circuitry, as described in the UFSAR.**

- **Alternatively, STP could replace the existing LSS or NRS Class IE component with a fully qualified, Class IE device.**

For additional justification, please refer to the response to RAI Question 12.

25. *To facilitate the staff's review, provide the risk-significance basis document for the emergency diesel generator system.*

RESPONSE:

The Risk Significance Basis Document for the Standby Diesel Generator and supporting Systems was mailed to the NRC on January 26, 2000 (NOC-AE-00000260).

26. *Please provide an explanation about how the safety-significance determination process was applied to the auxiliary feedwater system (AFWS) steam supply orifices for the AFWS pump turbine. How did the determination process account for the design modification which had replaced steam condensate traps with orifices as a result of operational problems (turbine overspeed had apparently resulted from the presence of steam condensate in the AFWS pump turbine steam supply when the steam condensate traps had overfilled)?*

RESPONSE:

General - The risk significance determination process included specific discussion on the design modification that replaced the steam condensate traps with orifices. The system engineer provided the Working Group with information on the modification to help the members understand the basis and scope of the modification. The Working Group then utilized this knowledge in reaching consensus on the risk of the condensate removal function and its supporting components.

Modification Basis and Scope - STP verified through operational experience that large amounts of condensate buildup in the main steam supply line to the Terry Turbine can lead to an overspeed when the turbine is started. Therefore, the automatic start function of the Terry Turbine is dependent on effective moisture removal from the steam supply system.

The problems with moisture removal were numerous and are stated as follows:

1) **The turbine steam admission valve was located approx. 150 feet from the turbine, which provided a large storage space for accumulated condensate.**
2) **The steam admission valve had a relatively fast open stroke. The fast open gave the turbine governor/governor valve very little time to take control of the turbine prior to overspeed.**
3) **The drain lines were insufficient in capacity.**
4) **The drain flow was controlled by steam traps, which had a tendency to fail closed.**
5) **No moisture detection/alarm was available to plant operators in the event that moisture did accumulate in the drain lines.**

The following modifications to the drain system/steam supply were installed to rectify the above mentioned problems.

1) **The steam traps were replaced with orifices.**
2) **The steam admission valve was moved to approx. 2 feet from the turbine to eliminate the large cool/dead space where condensate had previously accumulated.**
3) **The stroke time of the steam admission valve was doubled to give the turbine governor more responsiveness when handling steam/moisture mixtures on turbine start.**
4) **Additional drain lines were added to the turbine to ensure more complete removal of moisture.**

5) **A moisture detecting sensor and thermocouple, with control room alarms were added to the drain system in order to notify operators in the event that condensate does accumulate.**

Basis for Risk Ranking - The condensate removal function was ranked High because the automatic start function of the Terry Turbine, itself a High risk component, is dependent on effective moisture removal from the steam supply system.

The components involved with detecting and alarming excessive moisture buildup in the steam lines were ranked Medium. This was based on the fact that there are multiple and independent means to detect and alarm moisture buildup. Therefore, failure of any one component would not fail the function.

The orifices, which replaced the steam traps, support the condensate removal function. These components were ranked Low based on the following:

1) **An orifice is inherently a very reliable device, as it has no moving parts.**
2) **The primary failure mechanism attributable to the orifice itself is erosion. Erosion would increase the amount of condensate removed. Therefore, failure would be in a conservative direction.**
3) **There are multiple lines and orifices installed such that failure of any one line or orifice would not impact the condensate removal function.**

Given the fail-safe characteristic of orifices and the redundancy of the multiple means for condensate removal, moisture detection, and alarms described above, it has been determined that the possibility of an orifice failure leading to a turbine overspeed trip is extremely low.

Additional Considerations – The critical attribute of "allow condensate to drain" is specified for these orifices. STP's process provides for special considerations when plant activities, such as maintenance or procurement, may affect the critical attribute(s). Increased controls and documentation are required for such activities. For example, maintenance work on the orifice would include appropriate controls to provide an appropriate level of assurance that the ability of the orifice to properly drain condensate has not been negatively affected when the component is returned to service.

STP's monitoring and feedback process ensures that any changes in equipment performance are evaluated for impact on risk significance. Condition reports are initiated to document component failures or performance degradations and the resulting corrective actions. Condition reports are also used to initiate and document the results of Preventive Maintenance activities. For each system whose components have been risk ranked, the associated condition reports are reviewed and evaluated periodically for evidence of negative performance trends. Any such evidence is brought to the attention of the Working Group where it is evaluated for impact on the risk ranking of the associated components. The Working Group, with Expert Panel approval, then adjusts the risk ranking, as appropriate. This feedback loop ensures that any negative performance

changes, including those potentially attributable to the relaxation of special treatment controls, are reflected in a revised risk ranking, as appropriate. For the subject orifices, this process will provide an appropriate level of assurance that any performance degradation, however unlikely, will result in a re-evaluation of the risk rank to ensure continuing appropriateness.

27. *During the staff's recent visit to the STP plant site, a sample comparison was completed for risk rankings in the risk-significance basis documents for two heating, ventilation and air conditioning (HVAC) systems. These systems included the electrical auxiliary building (EAB) HVAC and fuel handling building (FHB) HVAC.*

A sample comparison of risk rankings for fire dampers for the EAB HVAC and FHB HVAC systems, respectively, showed that EAB HVAC system dampers were assigned a risk ranking of "Medium" while FHB HVAC system dampers were assigned a risk ranking of "Low." Provide the bases for the differences in risk rankings. [The licensee has frequently cited fire dampers as an example of components brought into scope to receive "special treatment."]

Compare the risk rankings of the filtration fans, HEPA filter and carbon filter in both the EAB HVAC and FHB HVAC systems (i.e., a comparison of components that are typically covered by Technical Specifications) and provide the bases for any differences. Select two other examples where the risk rankings differ and provide the bases for the differences.

RESPONSE:

The EAB HVAC (HE) system fire dampers were ranked MEDIUM due to the potential consequences of the spread of fire resulting from a failed fire damper being more severe in this system than they are in the Fuel Handling Building HVAC (HF) system. In the HE system, it could not be assured that failure of a fire damper in one train would not prevent the fire from spreading to another train (another risk significant area).

The design of the HF system is different than the HE system in that the functions with the highest risk (MEDIUM) are associated with providing cooling air to essentially self-contained rooms such as the Safety Injection (SI) and Containment Spray (CS) pump rooms. Each such room has its own air handling unit and there is no interconnecting ductwork or fire dampers. There are 3-hour rated fire barriers (walls) between the three trains of SI/CS pump rooms. The rest of the system, including the supply and exhaust of air to/from the Fuel Handling Building is categorized LOW or NRS. The fire dampers are located in this portion of the system. Thus, failure of a fire damper in the HF system could only affect a LOW or NRS area.

In addition, the number and percentage of HE components ranked HIGH/MEDIUM far exceed those for the HF system, as shown below:

Sys	High	Medium	Total (all risks)
HE	90 (4.7%)	92 (4.7%)	1,970
HF	0 (0%)	6 (0.8%)	755

A comparison of risk rankings between the two systems is provided in the following table for selected components.

Comparison of similar components between HE and HF. NOTE: The HF fans supplying the SI pump rooms, which are not shown here, are ranked High by the PRA.

Type	Sys	Component	PRA Risk	Determ Risk	Final Risk	Basis
FAN (See Note above)	**HE**	EAB MAIN AREA AHU SUPPLY FAN 11A FN014	High	Med.	High	Deterministic risk based on component's support of system functions ranked Medium, including the smoke purge function. PRA risk based on high Risk Achievement Worth (RAW) and/or Fussell-Vesely (FV) values. Refer to PRA analysis for further details. Final risk is highest of PRA or deterministic.
	HF	FUEL HANDLING BUILDING MAIN EXHAUST FAN 11A	N/A	Low	Low	Deterministic risk based on component's support of functions ranked Low, including exhausting Fuel Handling Building air to the main vent stack. The PRA does not credit these components for accident/transient mitigation.
HEPA FILTER	**HE**	EAB AHU FILTRATION UNIT 11A HIGH EFFICIENCY FILTER	Med*	Med.	Med.	Deterministic risk based on component's impact on system functions ranked Medium, including the potential to impede cooling airflow if the filter is clogged. PRA risk based on similar considerations, resulting in relatively high RAW values (100.0 > RAW ≥ 10.0). Note: the asterisk in the PRA risk indicates that the Full QA program is to be applied to those critical attributes of the component that are associated with the RAW value.
	HF	FUEL HANDLING BUILDING EXHAUST FILTRATION UNIT HEPA FILTER 11A	N/A	Low	Low	Deterministic risk based on component's support of functions ranked Low, including the filtering of exhaust air to remove radioactive particulate. The PRA does not credit this component for accident/transient mitigation.
CARBON FILTER	**HE**	CONTROL ROOM MAKE-UP FILTRATION UNIT CARBON FILTER	N/A	Low	Low	Component supports system function to remove radioactive iodine from the airstream. Function is ranked Medium and component is deterministically ranked Low based on redundancy. The PRA does not credit this component for accident/transient mitigation.
	HF	FUEL HANDLING BUILDING EXHAUST FILTRATION UNIT CHARCOAL FILTER 11A	N/A	Low	Low	Deterministic risk based on component's support of functions ranked Low, including filtering of exhaust air to remove radioactive iodine. The PRA does not credit this component for accident/transient mitigation.
HEATER	**HE**	BATTERY ROOM REHEAT COIL HX008	N/A	Med.	Med.	Deterministic risk based on component's impact on system functions ranked Medium, including the function to maintain room temperatures within the design range (areas containing risk significant equipment). This heater is required to remain operational during a LOOP. The PRA does not credit this component for accident/transient mitigation
	HF	FUEL HANDLING BUILDING EXHAUST FILTRATION UNIT HEATER 11A	N/A	Low	Low	Deterministic risk based on component's support of functions ranked Low including the function to provide heating of the exhaust air to reduce moisture which could impact the carbon filters. The PRA does not credit this component for accident/transient mitigation.
DAMPER	**HE**	EAB MAIN AIR HANDLING UNIT 11A OUTLET BACKDRAFT DAMPER	High	Med.	High	Deterministic risk based on component's impact on system functions ranked Medium, including the function to maintain room temperatures within the design range (areas containing risk significant equipment). PRA risk based on high Risk Achievement Worth (RAW) and/or Fussell-Vesely (FV) values. Refer to PRA analysis for further details. Final risk is highest of PRA or deterministic.
	HF	FHB MAIN EXHAUST FAN 11A DISCHARGE BACKDRAFT DAMPER	N/A	Low	Low	Deterministic risk based on component's impact on system functions ranked Low, including the function to exhaust FHB air to the main vent stack under accident conditions. The PRA does not credit this component for accident/transient mitigation.

As a result of telephone conversations between the NRC and STP on specific components in the HE system, it was noted that some of the answers to the critical questions at the component level are not fully consistent with the final risk categorization assigned to the components or the supported functions. STP considers the final risk assigned to the system functions and components to be correct, and attributes the identified discrepancies to administrative documentation errors. STP has initiated a condition report to document this discrepancy and to implement corrective action. As part of this corrective action, STP will not be using critical questions at the component level since experience has shown that there is little associated value. In addition, STP has identified a focused group of components (about 5% of the total components risk categorized to date) that will be specifically reviewed for adequacy of documentation. Additional documentation sampling of other risk categorized components is occurring to fully assess the overall documentation adequacy. The results of these corrective actions are available upon request.

28. *Please describe how the licensee's risk determination process evaluates the significance of all areas covered by the Maintenance Rule scope (50.65(b)(1), (b)(2)(i), (b)(2)(ii), and (b)(2)(iii), and associated industry guidance). If the risk determination process does not cover the Maintenance Rule scope, provide appropriate justification as the staff will need to fully understand and evaluate the differences.*

RESPONSE:

The risk significance determination process encompasses all structures, systems, and components (SSCs) covered by the Maintenance Rule scope as described in the referenced regulations and associated industry guidance. For each system that is reviewed under this process, all "tagged" components (refer to RAI question no. 1 response for additional discussion), whether safety related or non safety-related, are categorized via the risk significance determination process. Any SSC that has not yet been risk categorized (i.e., a component in a system that has not yet been reviewed) will not be subject to possible relaxation of applicable special treatment requirements until such time that the risk categorization is performed.

The risk significance determination process is detailed in STPNOC procedures 0PGP02-ZA-0003, Comprehensive Risk Management, and 0PEP02-ZA-0001, Graded Quality Assurance Working Group Process. Generally, the process consists of blending the PRA risk for a component with a deterministic evaluation to reach an overall risk significance categorization. The deterministic evaluation consists of answering a set of five critical questions similar to those identified in the referenced regulation. The answers to these questions are weighted to provide an appropriate degree of significance, depending upon the importance of each question. In order to provide a consistent and robust approach, the system functions are first risk categorized through this process, followed by the relationship identification between each component and the system function(s) it supports, and finally, by the risk categorization of the component itself. Additional details can be found in the above referenced procedures and in other responses elsewhere in this RAI. The table on the following page provides a comparison between the Maintenance Rule scope and the scope of the Risk Significance Determination Process.

Based on the above, STP's position is that the risk significance determination process fully covers, and in fact exceeds, the scope of the Maintenance rule.

MAINT. RULE SCOPE	EQUIVALENT SCOPE IN RISK SIGNIFICANCE DETERMINATION PROCESS	COMMENTS
50.65(b)(1) – safety related structures, systems, and components (SSCs)	Safety related SSCs that are "tagged"; i.e., that are part of the Total Plant Numbering System (TPNS). In addition, STP may categorize non-tagged SSCs on a case-by-case basis.	Any safety related SSCs that are not evaluated by the Risk Significance Determination Process remain conservatively under the "Full" QA program and are excluded from the scope of this exemption request
50.65(b)(2) – Only those non-safety related SSCs that: (see list below)	All non-safety related SSCs that are tagged. In addition, STP may categorize non-tagged SSCs on a case-by-case basis.	Any non-safety related SSCs that are not evaluated by the Risk Significance Determination Process are excluded from the scope of this exemption request
(b)(2)(i) – are relied upon to mitigate accidents or transients or are used in Emergency Operating Procedures (EOPs)	The following questions are evaluated to determine the risk significance of SSCs: - Used to mitigate accidents or transients? - Used in EOPs or in Emergency Response Procedures?	
(b)(2)(ii) – whose failure could prevent SSCs from fulfilling their safety related function	The following question is evaluated: - Could fail a risk significant system?	This question in the risk significance determination process is essentially asking the following: Could the failure result in loss or substantial degradation of another system's risk significant functions?
(b)(2)(iii) – whose failure could cause a reactor scram or actuation of a safety related system	The following question is evaluated: - Could directly cause or has caused an initiating event?	An initiating event is an occurrence that causes a challenge to the plant. Refer to the following table for a listing of initiating events.
	The following additional question is evaluated: - Is it safety significant during shutdown or mode change operations?	An example would be instrumentation that is used to support mid-loop operations.

Initiating Event Categories Selected for Quantification of the South Texas Project Risk Model

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Loss of Coolant Inventory	1. Excessive LOCA	ELOCA
	2. Large LOCA	LLOCA
	3. Medium LOCA	MLOCA
	4. Small LOCA a. Non-Isolable b. Isolable	 SLOCA ILOCA
	5. Interfacing Systems LOCA	VSEQ
	6. Steam Generator Tube Rupture	SGTR
Transients	7. Reactor Trip	RTRIP
	8. Turbine Trip	TTRIP
	9. Loss of Condenser Vacuum	LCV
	10. Closure of All MSIVs	AMSIV
	11. Steam Line Break Upstream of MSIVs a. Steam Line Break Inside Containment b. Main Steam Relief or Safety Valve Opening	 SLBI MSV
	12. Steam Line Break Downstream of MSIVs	SLBD
	13. Inadvertent Safety Injection	SI
	14. Miscellaneous Transients a. Total Main Feedwater Loss (includes feedwater line break outside containment) b. Partial Main Feedwater Loss c. Excessive Feedwater Flow d. Closure of One MSIV e. Core Power Excursion f. Loss of Primary Flow	 TLMFW PLMFW EXMFW IMSIV CPEXC LOPF

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Common Cause Initiating Events (Support System Faults)	15. Loss of Offsite Power (LOOP) a. Loss of 345kV Grid b. Loss of All Offsite Power c. Loss of the Main Transformer	 LOSP LOSPX LOMT
	16. Loss of One DC Bus a. Loss of DC Bus E1A11 b. Loss of DC Bus E1B11	 L1DCA L1DCB
	17. Loss of Instrument Air	LOIA
	18. Total Loss of Essential Cooling Water (ECW) a. Loss of ECW - Three Trains Available b. Loss of ECW - Two Trains Available c. Loss of ECW - One Train Available	 LOECW3 LOECW2 LOECW1
	19. Total Loss of Component Cooling Water (CCW) a. Loss of CCW - Three Trains Available b. Loss of CCW - Two Trains Available c. Loss of CCW - One Train Available	 LOCCW3 LOCCW2 LOCCW1
	20. Loss of Electrical Auxiliary Building (EAB) HVAC a. Loss of EAB HVAC - Three Trains Available b. Loss of EAB HVAC - Two Trains Available c. Loss of EAB HVAC - One Train Available	 LOEAB3 LOEAB2 LOEAB1
	21. Loss of Control Room (CR) HVAC a. Loss of CR HVAC - Three Trains Available b. Loss of CR HVAC - Two Trains Available c. Loss of CR HVAC - One Train Available	 LOCR3 LOCR2 LOCR1
Seismic Events	22. 0.1g Seismic Event	SEIS1
	23. 0.2g Seismic Event	SEIS2
	24. 0.4g Seismic Event	SEIS3
	25. 0.6g Seismic Event	SEIS4

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Plant Fires	26. Control Room - Loss of All Three Motor-Driven AFW Pumps	FR10
	27. Control Room - Loss of CR HVAC and EAB HVAC	FR18
	28. Control Room - Loss of All AFW Pump Trains	FR23
	29. Zone 31Z047 - Cable Spreading Room Train B, Area B - Affects Train B (AC, DC), RCFC A, Recirculation Cooling Train A, RCP Seal Injection and PORV 656A	IZ047B
	30. Zone 31Z047 - Cable Spreading Room Train B, Area BC - Affects Train B and Train C AC and DC, RCFC A, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A, and the CCPs and PDP	IZ47BC
	31. Zone 31Z047 - Cable Spreading Room Train B, Area X - Affects Train B and Train C AC and DC, RCFC A, Recirculation Cooling Train A, RCP Seal Injection, PORV 656A, MSIVs CCPs, the PDP, and RCP CCW supply	IZ047X
	32. Zone 07Z071 - Auxiliary Shutdown Area, Area X - Affects Train A, Train B, and Train C AC Power, AFW Train D, CI Trains A and C, and the PDP	IZ071X
	33. Zone 03Z147 - Corridor and Changing Area O - Affects DG A, DG C, CCW A, B, C, LHSI A, HHSI A, CS A, B, C, CI Train A, B, C, ECH C, CCPs, and Recirculation Cooling Train A	IZ147O

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Plant Flooding (External)	34. LOOP and Positive Displacement (PD) Charging Pump	FL1
	35. LOOP, PD Pump, and All Three Emergency Diesel Generators	FL26 (For Categories 35 to 40)
	36. LOOP, PD Pump, and Loss of All ECW	
	37. LOOP, PD Pump, and Loss of All CCW	
	38. LOOP, PD Pump, All CCW, and One Train (B) of Essential Chillers	
	39. LOOP, PD Pump, and One Train (B) of RCFCs	
	40. LOOP, PD Pump, One Train (C) of AC Power, and Main Control Room	
	41. Breech of the Main Cooling Reservoir - LOOP, Loss of TSC Diesel Generator, and Plugging of the ECW Pump Traveling Screens by Debris	FLECW
Plant Flooding (Internal)	None	
Other Initiators	42. Severe Wind (Tornado) - LOOP, Loss of TSC Diesel Generator, and Plugging of the ECW Pump Traveling Screens by Debris	FLECW

29. *Explain the risk basis for concluding that certain CIVs do not require leakage rate testing as specified in Appendix J. Please reference information already submitted on the docket, if appropriate.*

RESPONSE:

The response to RAI #16 specifies the risk basis for concluding that certain CIVs do not require leakage rate testing as currently specified in Appendix J.

30. *Explain the categorization scheme for risk ranking SSCs not in the licensee's PRA and for system functions. Provide the basis for the 6-point (0 to 5) rating scale used by the plant's Working Group to risk-rank SSCs. For example, explain how "insignificant" impact is different from "minor" impact in discriminating the two points on the scale. Other examples include: "minor" impact and "low" impact, "rarely" occurring event and "infrequently" occurring event, "infrequently" occurring event and "occasionally" occurring event, "regularly" occurring event and "frequently" occurring event. Unless there is an underlying basis associated with these words to meaningfully differentiate the adjacent points on the scale, we find that some of the adjacent points on the proposed scale do not convey any intrinsically meaningful difference. If, for example, a smaller scale, i.e., 3-point scale, is used to clearly distinguish the points in the scale, discuss how such a scale might impact the risk-ranking results. In other words, provide a discussion of how a robustness of a scale affects the sensitivity of the risk-ranking results. Include in the discussion the basis of the weighting factors (and the associated numerical values) and their impact on the risk-ranking. Also include the basis for the "score ranges" for final risk ranking categorization.*

RESPONSE:

The referenced rating scale is used in the deterministic input to the risk categorization process for both PRA-modeled and non-modeled components. <u>Deterministic input</u> is defined in procedure 0PEP02-ZA-0001, Graded Quality Assurance Process, as:

> **"An assessment of risk significance based on the collective input from a panel of individuals experienced with the pertinent aspects of managing and operating a nuclear generating facility (e.g., operations, maintenance, design, engineering, and risk analysis). Deterministic input is used to supplement PRA risk rankings, and/or to compensate for PRA limitations and assumptions. Deterministic input is also used for components not modeled in the PRA."**

The GQA Working Group membership, as defined in procedure 0PGP02-ZA-0003, Comprehensive Risk Management, is made up of experienced personnel with diverse knowledge and backgrounds. In order to provide the Working Group members with a mechanism to collect and categorize their deterministic input in a consistent and documented manner, a set of five critical questions related to risk categorization are answered. Initially, during the development portion of the risk categorization process, these critical questions were just answered either "Yes" or "No". It quickly became evident, as experience was gained, that this method did not permit enough flexibility to adequately capture the risk insights and technical bases between various system functions or components. For example, the initiating event for loss of Essential Cooling Water has much more impact than the initiating event for loss of Instrument Air. Under the old method, both cases would only have answered "Yes" for the initiating event question, even though the risk significance impact would be quite different. Thus, the current rating scale was developed.

With this scale, the Working Group has a consistent means to assign a positive response value that is commensurate with the relative impact on the public health and safety resulting from the loss of a system function or component. By definition, the deterministic process is a process based on the collective wisdom and experience of qualified individuals. The rating scale provides a consistent means to generate gradations in possible responses. The terminology used to define each gradation of the scale (having insignificant impact, minor impact, low impact, etc; or occurring very rarely, infrequently, occasionally, etc) serve as aids to the Working Group members in the selection of the proper scale value for each positive critical question response. While these terms (insignificant, minor, rarely, infrequently, etc) were not specifically defined up to this point, the terminology did provide adequate guidance to the Working Group members to arrive at consensus agreements in this portion of the categorization process, and to document a technical basis for each response. As the positive response value increases through the scale from "1" through "5", it denotes progressive increases in risk significance impact, which is reflected in the proceduralized guidelines provided for using the rating scale. Usage of a smaller scale range would result in less flexibility and therefore less accuracy allowed to the Working Group in its deterministic assessment. Considering the wide variety of system functions and components, the present rating scale provides a good balance between providing enough flexibility in the risk categorization process and the complexity associated with varying degrees of responses.

Considering the terminology used by the Working Group in the deterministic evaluations as specified in procedure 0PEP02-ZA-0001, the following general definitions for these terms can be stated:

<u>Frequency Definitions</u> –

- **Occurring Frequently – continuously or always demanded**
- **Occurring Regularly – demanded > 5 times per year**
- **Occurring Occasionally – demanded 1-2 times per cycle**
- **Occurring Infrequently – demanded < once per cycle**
- **Occurring Very Rarely – demanded once per lifetime**

<u>Impact Definitions</u> –

- **High Impact – a system function is lost which likely could result in core damage and/or may have a negative impact on the health and safety of the public**
- **Medium Impact – a system function is lost which may, but is not likely to, result in core damage and/or is unlikely to have a negative impact on the health and safety of the public**
- **Low Impact – a system function is significantly degraded, with very low likelihood of core damage and/no negative impact on the health and safety of the public is expected**
- **Minor Impact – a system function has been moderately degraded, but NO core damage or negative impact on the health and safety of the public**
- **Insignificant Impact – a system function has been challenged, but NO core damage or negative impact on the health and safety of the public**

To ensure consistent, future replication of the responses and categorization results developed by the Working Group, these definitions have been added to procedure 0PEP02-ZA-0001, Graded Quality Assurance Working Group Process, as a working aid in the categorization process.

The following table provides examples for how the deterministic definitions are used for various responses to the accident/transient mitigation question as applied to system functions:

SYS	FUNC	FUNC DESC	ANS	DETERMINISTIC INPUT
CV	1.2.6	SUPPLY COOLING WATER TO THE CVCS SEAL WATER COOLER	0	ACC: FUNCTION NOT REQUIRED FOR ACCIDENT/TRANSIENT MITIGATION
DG	1.4.2	CIRCULATE HEATED WATER THROUGH THE DIESEL ENGINE JACKET TO MAINTAIN TEMPERATURE WHEN THE ENGINE IS NOT OPERATING	1	ACC : LOSS OF THIS FUNCTION WOULD NOT BY ITSELF PREVENT THE DIESEL FROM STARTING/RUNNING AND WOULD BE INDICATED BY A GRADUAL DECREASE IN TEMPERATURE ALLOWING OPERATOR RECOVERY ACTIONS
RA	3.15	PROVIDE CONTAINMENT ATMOSPHERIC GASEOUS AND PARTICULATE RADIOACTIVITY INSTRUMENT SIGNALS TO DETECT REACTOR COOLANT LEAKAGE (RT-8011)	2	ACC : DOES NOT BY ITSELF MITIGATE ACCIDENT OR TRANSIENT BUT DOES PROVIDE INFORMATION WHICH AIDS IN IDENTIFYING THE OCCURENCE OF AN ACCIDENT SO THAT OTHER MITIGATION ACTIONS CAN BE TAKEN. MINOR IMPACT
HE	1.1.1	EAB MAIN AREA – MAINTAIN ROOM TEMPERATURES WITHIN THE DESIGN RANGE (AREAS CONTAINING RISK SIGNIFICANT EQUIPMENT)	3	ACC: THIS HAS A SIGNIFICANT IMPACT ON ABILITY TO MITIGATE SINCE IT IS A SERVICE SYSTEM, BUT PROGRESSION IS SLOW.
DG	1.3.1	INJECT SUFFICIENT CLEAN FUEL OIL INTO THE DIESEL ENGINE FOR ENGINE OPERATION AND RETURN THE FUEL OIL OVERAGE BACK TO THE FOST	4	ACC : LOSS OF THIS FUNCTION WOULD IMPACT THE ABILITY TO MITIGATE AN ACCIDENT/TRANSIENT BUT THE FREQUENCY OF A LOOP IS RELATIVELY LOW
CH	1.3	CIRCULATE CHILLED WATER THROUGH THE FUEL HANDLING BUILDING ESF PUMP CUBICLE AIR HANDLING UNIT COOLING COILS (HF, CS, AND SI SYSTEMS)	5	ACC: CHILLED WATER IS REQUIRED TO SUPPORT HEAT REMOVAL VIA AHUs DURING ACCIDENT/TRANSIENT MITIGATION.

In addition to the rating scale, weighting factors are used to account for the relative impacts among the five critical questions. For example, the accident mitigation question is considered to have more risk significance impact on public health and safety than the initiating event question, assuming both were answered with the same positive response value. The Working Group determined that the five questions could be categorized into three weighting groups as follows:

- **Accident/Transient, and EOPs** — **Weight multiplier of 5 (most important)**
- **Fails Risk Significant System** — **Weight multiplier of 4 (moderate importance)**
- **Initiating Event, and Shutdown/Mode Change** — **Weight multiplier of 3 (least important)**

Weighting factors of "3", "4", and "5" were used, as detailed in procedure 0PEP02-ZA-0001, Graded Quality Assurance Process. Thus, a maximum positive response of "5" to all five questions for a specific system function or component would result in a score of "100". The scale was then divided into four sections corresponding with the four risk significance categories. For conservatism, only the lower 40% of the scale was reserved for NRS/LSS components and the upper 60% for MSS/HSS components. In addition, special exceptions were incorporated into the process to account for a high positive response to any one question which might be masked by a low overall score due to low values for the other four questions. For example, a maximum value for "5" for initiating event would result in a minimum risk categorization of "MSS", even if all other questions were answered in the negative.

It should be emphasized that the above process is an iterative process where initial responses to the questions are discussed, challenged, justified, and revised where appropriate. These discussions occur during the Working Group meetings where the members' insight and varied experience ensure that the final result reflects a comprehensive and justifiable deterministic judgment. If during this iterative process a consensus agreement cannot be reached by the Working Group members, a 'Dissenting Opinion' is documented and forwarded to the Expert Panel for resolution prior to documenting a final risk categorization.

The Working Group developed the above process after extensive discussion. This proposed process was then presented to the Comprehensive Risk Management Expert Panel for approval prior to use. Use of the rating scale has provided risk significance categorizations that are consistent with both the Working Group members' overall sense and judgment and that of the Expert Panel members. It should also be noted that the rating scale is provided as a guideline, and the Working Group and Expert Panel can and have deviated, in a risk-conservative direction, from the guideline, based on special circumstances.

31. *(a) Explain the potential difference in the importance of an SSC for at-power and shutdown modes and how such difference is accounted for in risk-ranking. For example, if an SSC that might be judged by the Working Group to be important with a score of "5" for a shutdown/mode-change critical question (with low scores for other four critical questions) could result in a final score less than "40," would it be categorized as a non-risk significant or a low safety significant SSC?*

(b) Discuss if the weighted sum is always used as the sole guideline or if other constraints are applied.

(c) Similarly, provide a discussion and examples of how an SSC's importance during external events (i.e., seismic, fire, and tornadoes) might affect its overall importance as applied toward the risk-ranking. Identify the external phenomena that were addressed in order to determine what impact the proposed exemption from environmental and dynamic effects will have on CDF and LERF.

RESPONSE (part a):

The use of the weighting scale as described in Addendum 2 of 0PEP02-ZA-0001, Graded Quality Assurance working Group Process, includes the following guidelines:

Score Range	Risk
0 - 20	NRS (Not Risk Significant)
21 - 40	Low
41 - 70	Medium
71 – 100	High

Exceptions **Minimum Risk**

Weighted Score of 25 on any one question (ACC or EOP only).....High Risk

Weighted Score of 15-20 on any one question .....Med Risk

Weighted Score of 9-12 on any one question .....Low Risk

Thus, if a component were to receive a score of "5"on the shutdown/mode change (s/d) question and worst case scenario of "0" on all other questions, the weighted score for the s/d question would be "15" and "0" for all the other questions. The overall score would then be "15". This would initially put it in the NRS category, but as noted above under "exceptions", a score of "15" on any one question would result in a minimum risk of MEDIUM for this hypothetical component.

RESPONSE (part b):

The weighted sum is not the sole guideline. In addition to the exception rule noted above, the Working Group is guided by the following (excerpted from the referenced procedure addendum):

"The overall score is used to help the GQA Working Group deterministically evaluate the risk significance. The GQA Working Group can deviate from the guide as necessary to account for special circumstances or the group members' knowledge and insight; Deviations from the guide are to be the exception rather than the rule and are to be documented and highlighted to the CRM Expert Panel. In addition, the GQA Working Group should utilize conservative decision-making in deterministically evaluating risk significance."

An additional constraint is applied whenever the PRA risk is greater than the risk obtained through the use of the weighted scale. In that instance, as shown on Addendum 3 of 0PGP02-ZA-0003, Comprehensive Risk Management, the PRA risk is used as the final risk.

RESPONSE (part c):

Some of the external events that are addressed in the STP PRA are: External floods from main cooling reservoir breach; tornado that fails offsite power and the essential cooling pond; seismic events from 0.1 to 0.6g (Note: the SSE for South Texas is 0.1g); and internal fires. These external events are included in the STP PRA results and are implicitly included in all risk rankings that are based on the PRA. The PRA evaluates seismic events and other external events that are well beyond the design basis external events required to be analyzed.

The first two external initiating events guarantee failure of offsite power and the Essential Cooling Pond. Core damage is assumed under these conditions. Containment response depends upon the status of the On-Line purge system, but the LERF is several orders of magnitude lower than the CDF.

The proposed exemption from environmental effects does not affect any of the external events modeled in the PRA. In terms of dynamic effects, only the seismic external events have an effect on the proposed exemption. The contribution to CDF from seismic events is 7.1×10^{-08} per year and is dominated by loss of offsite power and seismic failure of the emergency diesel generators, seismic failure of the Class 1E 120V Inverters or seismic failure of the Class 1E DC Battery system. Equipment for which exemption to dynamic effects is being requested do not affect CDF or LERF.

32. *During the GQA evaluation, the staff did not emphasize the review of the environmental and seismic analyses in your PRA because the special treatment requirements were not being modified. Discuss how the quality of your PRA, and related analyses to support these exemptions are sufficient to give reliable results.*

RESPONSE:

STP's PRA includes equipment failure contributions due to environmental effects and seismic effects. Active components which are credited for accident mitigation during seismic events and under severe accident conditions are generally categorized as HSS or MSS, however, there are cases of LSS categorized components (e.g., the MOV for the CCW to RCFC cross tie). The environmental effects are found in the spatial interactions analysis for the fire PRA and previous studies submitted to the NRC. The seismic effects are explicitly modeled in the seismic PRA and reflect the ability of the station to achieve stable conditions from a range of seismic events. Quality of environmental and seismic analysis of STP's PRA is described and documented in the *Level 2 PRA and Individual Plant Examination* submitted to the NRC in August of 1992.

<u>Equipment Survivability Analysis</u>

As part of STP's *Individual Plant Examination* a containment performance analysis was performed to evaluate equipment survivability during severe accidents. STP performed a qualitative analysis of equipment survivability such that equipment failures under severe accident conditions would not create instances of unusually poor containment performance given a severe accident. It was limited to the evaluation of possible mitigation of the accident once core degradation has occurred.

The mitigation of a severe accident can be achieved by activating the plant capabilities to cool the damaged core debris and to remove energy and radioactive material from the containment atmosphere. This can be achieved through the containment spray, the reactor containment fan coolers, the low head safety injection with residual heat removal exchanger, and the auxiliary feedwater systems.

The analysis reviewed selected degraded core damage sequences with respect to equipment survivability necessary to mitigate the consequence of containment release. Containment and/or compartment pressure and temperature where overlaid on a graphical plot with the equipment qualification (EQ) temperature and pressure, as appropriate. The analysis estimated the likelihood of equipment survivability for conditions prior to vessel breach and post vessel breach scenarios. In all cases equipment was likely to survive with the exception of seismically induced loss of all AC and DC power with turbine driven AFW pump failure. For this case the EQ pressure is reached for both lower and upper compartments in ≈ 26 hours. EQ temperature is reached in ≈ 11 and ≈ 26 hours for the lower and upper compartments, respectively. These results are reasonable and valid to the conclusions reached in the *IDCOR Technical Report 17, Equipment Survivability in a Degraded Core Environment.*

In a letter from the NRC to STP titled, *Individual Plant Examination (IPE) – Internal Events, South Texas Project, Units 1 and 2 (TAC Nos. M74471 and M74472)* dated August 9, 1995, the staff evaluation report stated, "The staff found the approach used to be consistent with Generic Letter 88-20..." The review included examination of the methodology, documentation and input data.

Seismic Events

STP's seismic risk analysis consists of the following five steps:

1. **Seismic Hazard Analysis: Determination of the frequency of the ground motions of various sizes at the site.**
2. **Fragility Analysis: Determination of the seismically-initiated ground acceleration at which plant structures and components are predicted to fail.**
3. **Plant Logic Analysis: Development of a logic model that depicts the consequence of structure and component failures. The model includes the seismically-induced events that may cause one or more different classes of initiating events and one or more failures of components or systems needed to respond to the initiating event as well as the consideration of non-seismic failures that can combine with seismically-induced failures to produce an accident sequence.**
4. **Initial Assembly: Quantification and assembly of the seismic hazard, component fragility, and plant logic to obtain point estimates of the frequencies of core melt and various plant damage states might result from seismic initiating events.**
5. **Final Assembly: After comparison with point estimates of plant damage state frequencies from other initiators, for those seismically initiated scenarios that are major frequency contributors, calculation of the probability distribution of plant damage state and core damage frequencies ready for combining with the probability distribution of frequencies from other initiating events.**

In a letter from the NRC to STP titled, *Evaluation of Probabilistic Safety Analysis – External Events for South Texas Project, Units 1 and 2 (TAC Nos. M73009 and M73010)*, dated August 31, 1993, the following statement was made with respect to STP's seismic analysis: "The staff found that the fragility approach used by the licensee, which has been used in other PSAs, is acceptable, and that the analysis of seismic events identified no significant weaknesses."

Additional quality information of STP's PRA is described as follows:

Description

STP has a Level 1/Level 2 PRA and IPE including external events. The external events portion contains both a Fire PRA (with Spatial Interactions analysis) and Seismic PRA analysis. STP's PRA has been structured to have a comprehensive treatment of common cause failures and plant configurations. A detailed human reliability analysis is also included.

Previous Reviews

STP's PRA has undergone several extensive NRC reviews in support of license amendments. Specifically,

- **"A Review of the South Texas Probabilistic Safety Analysis for Accident Frequency Estimates and Containment Binning" contracted through Sandia National Laboratories. NUREG/CR 5606;**
- **"Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Probabilistic Safety Analysis Evaluation," sent to the Houston Lighting & Power Company under cover letter dated January 21, 1992;**
- **"Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Probabilistic Safety Assessment - External Events," sent to the Houston Lighting & Power Company under cover letter dated August 31, 1993;**
- **"Issuance of Amendment Nos. 59 and 47 to Facility Operating License Nos. NPF-76 and NPF-80 and Related Relief Requests – South Texas Project, Units 1 and 2 (TAC Nos. M76048 and M76049)" sent to Houston Lighting & Power Company February 17, 1994;**
- **"Individual Plant Examination (IPE) - Internal Events, South Texas Project, Units 1 And 2-(STP) (TAC Nos. M74471 and M74472)" dated August 9, 1995 (Included equipment survivability analysis);**
- **"South Texas Project, Units 1 and 2 – Amendment Nos. 85 and 72 to Facility Operating License Nos. NPF-76 and NPF-80 (TAC Nos. M92169 and M92170)" sent to Houston Lighting and Power Company under a cover letter dated October 31, 1996. This amendment allows extension of the standby diesel generator allowed outage time to 14 days, and extension of the essential cooling water and essential chilled water allowed outage time to 7 days;**
- **"Graded Quality Assurance, Operations Quality Assurance Plan (Revision 13), South Texas Project, Units 1 and 2 (STP)(TAC Nos. M92450 and M92451) dated November 6, 1997.**

PRA Maintenance

STP's PRA Configuration and Control program is structured to ensure changes in plant design and equipment performance are reflected in the PRA as appropriate. The PRA Configuration and Control process is administered by procedures and guidelines that ensure proper control of all changes to the models by persons independent from the person making the change and approved by the PRA supervisor. STP's PRA will undergo a PRA certification under the Westinghouse Owner's Group Peer Review Process and is expected to be in compliance with the ASME PRA standard for risk informed applications.

PRA Self-Assessment

An independent assessment of the overall control process has been performed using the guidance from the BWR Owner's Group Peer Certification Process. All findings from this self-assessment were documented in the corrective action program and have been corrected. The conclusions from the self-assessment indicate that the methods used to control the PRA satisfy the appropriate requirements of Appendix B to 10CFR50. Given the current state-of-the-art in PRA analyses and techniques, and the control of the processes used to make changes to the model, the quality of the PRA is sufficient to achieve reliable results for this exemption request.

The above information provides a statement of STP's PRA quality. Improvements to STP's PRA have continued to be incorporated. STP's PRA is robust and contains a comprehensive treatment of equipment failure mechanisms, equipment/system interactions, plant specific data, etc. to give reliable results relative to the risk significance of plant equipment. Additionally, sufficient detail is present to allow meaningful performance indicators on equipment trends resulting from changes in special treatment requirements.

> *The Advisory Committee for Reactor Safeguards (ACRS) has suggested, and we are considering, determining the importance of SSCs for seismic, fire, and other external events based on the specific analysis alone. For example, the importance of SSC's for seismic events should be determined by only using the seismic analysis. This reduces the shadowing effect between analyses of different precision. Please describe how importance measures are obtained for the seismic and other external event analyses, and how these measures are used together with the internal events results.*

RESPONSE:

The STP PRA is a fully integrated model of plant risk from all categories of initiators. This means that all initiating events are included in all model quantification. The resulting risk importance measures are determined from sequences that are representative of all the initiating events. Risk importance measures for specific classes of initiating events have not been routinely calculated.

A special evaluation was performed in response to this question that looked at the risk importance measures by class of external event (fires, seismic, external floods). This evaluation is described in the response to question number 36 of the Request for Additional Information. The overall conclusion from this evaluation is that there is no change in basic event importance ranking when looking at the external events in isolation. The main reason there is relatively no change to the component risk ranking is due to the low overall importance external events have on the PRA.

Have any SSC's been identified that are important only for external events?

RESPONSE: No

Also, since the PRA assumes that the equipment is fully qualified for the environment it must operate in, please explain how you intend to incorporate environmental and seismic effects into your PRA such that you can estimate or bound the aggregate impact of all your proposed special treatment changes.

RESPONSE:

For environmental qualification effects, the PRA models room cooling for most components which perform active functions over the mission time of the PRA. For example, component cooling water pump is required to run for 24 hours. In order to prevent the pump from failing due to environmental concerns (i.e., increasing room temperature), the PRA models the air handling units (AHU) for the pump room. If the AHU fails then it is assumed in the PRA that the pump will fail due to temperatures above pump EQ qualifications. Another example is a containment isolation valve, which performs its action early during an accident, does not require room cooling. Room cooling is not modeled based on the component performing its function early in the accident prior to hazardous environmental conditions being reached. In this case, the containment isolation valve is isolated upon receipt of a containment isolation signal.

In addition, for seismic events, all systems necessary to mitigate the consequences of the events are included in the PRA model. In the model, the response of the components (fragility) necessary to support operation of the various systems is determined based on discrete acceleration levels. All components of a similar type (e.g., batteries or diesel generators) are assumed to fail at the same acceleration level based on these fragilities. This process bounds the aggregate impacts for seismic events for equipment that is necessary to mitigate the consequences of seismic events.

The PRA risk ranking process includes analysis that estimates or bounds the aggregate and individual impact of possible changes. The risk associated with possible changes in equipment performance are addressed by increasing equipment failure rates by a factor of 10 and by the use of the Risk Achievement Worth as an importance measure. One of the PRA sensitivity studies performed for determining component importance is increasing the failure rates by a factor of 10. This increase in risk for the CDF and LERF are with the acceptance guidelines for very small changes as outlined in Regulatory Guide 1.174. This analysis is also addressed in the response to RAI question number 21.b.

The other analysis that bounds the impact of possible changes in equipment performance is the use of Risk Achievement Worth (RAW) in determining component importance. The RAW determines the impact to CDF or LERF given guaranteed failure of the component. The RAW is one of two importance measures used in conjunction to determine risk significance. All components subject to changes in special treatment requirements are ranked LSS or NRS. All LSS components by definition will have a RAW less than 2. All components credited in the PRA for accident/transient mitigation or initiation are at least LSS. Therefore, by definition NRS components are not modeled in the PRA analysis.

The above information estimates or bounds the effect of possible component performance changes associated with the proposed special treatment requirements. The aggregate effects are analyzed via a sensitivity study on increasing failure rates for LSS components. At a component level the effects of possible changes in component performance are bounded by use of the RAW importance measure. Therefore, for components subjected to proposed changes in special treatment requirements, i.e., LSS, analysis has shown that the possible performance changes to LSS components have a negligible impact on CDF or LERF.

33. *In the licensee's risk categorization process, the safety significance of all system functions are determined by critical question responses assigned by the expert panel - even system functions modeled in the PRA.*

(a) *Explain how the importance of a component in the system impacts the safety significance of that system.*

(b) *For example, the licensee's PRA indicates that the Chemical and Volume Control System (CVCS) positive displacement pump is high safety significant, but the Working Group categorized the corresponding system function as low safety significant. We anticipated that the functions supported by a high safety significant SSC should also be categorized as high safety significant. In particular, your new method of having the expert panel directly assign grades to each system function does not seem to fully comport with assigning a safety significance to each system function based on a combination of PRA insights and deterministic insights. Please explain the source of the apparent discrepancy in the categorization. That is, what characteristics of the PRA models led to the high safety significance categorization for the Chemical and Volume Control (CVCS) pump, and how do these contrast with the characteristics assumed by the expert panel in assigning the grades to eventually end up with a low safety significance designation for the corresponding system function? Moreover, explain how such a designation would impact the risk-ranking of a component in the CVCS.*

RESPONSE (part a):

Deterministically, a component's importance is directly attributable to the importance of the function supported by the component. However, a component's importance is based not only on deterministic insights, but also includes probabilistic insights if the component is credited in the plant specific PRA. Deterministically, a component's importance is based on the relative contribution that the component provides in support of the system functions. For example, if the function of a check valve is to prevent reverse flow through a centrifugal pump and is not required for containment isolation, then the valve's importance would be based on the function it supports (i.e., protect the pump) and not on the containment isolation function. Probabilistically, a component's importance is based on its function to mitigate an accident or to prevent an initiating event. This includes both the reliability and availability of the component, which impacts the risk categorization of the component.

Response (part b):

The functions of the Chemical and Volume Control system (CVCS) positive displacement pump (PDP) are to hydrotest the Reactor Coolant System (RCS), to add chemicals to the RCS for pH and oxygen control, and to provide seal injection flow if both centrifugal charging pumps become inoperable. The Probabilistic Risk Assessment (PRA) credits the PDP pump only when seal injection flow is not available from the centrifugal charging pumps. Use of the PDP pump requires operator action to start the PDP and to maintain flow to the individual RCP seal injection lines. For event sequences that include failure of

plant offsite power, success also requires that the Technical Support Center diesel generator be available to power the PDP.

The PRA categorizes the PDP pump as HIGH due to previous poor performance. Both availability and reliability have continued to improve, and it is expected that updated risk categorization studies will result in the PDP being reclassified. The PRA risk categorization process is a compilation of sensitivity studies. The sensitivity studies demonstrate the robustness of the risk categorization process by providing analysis of the following:

- **effects of scheduled maintenance,**
- **removal of operator recovery,**
- **removal of common cause failures,**
- **increased failure rates over multiple systems, and**
- **reduced steam generator tube rupture frequency on large early release frequency.**

The average At-Power Probabilistic Risk Assessment (PRA) risk categorization, along with the above sensitivity studies, are used to produce a final PRA component risk categorization.

The basis for the HIGH categorization of the PDP is due to its previous poor performance and because of its high importance in five of the twenty-one sensitivity studies. In all other studies (e.g., removal of operator recovery, removal of common cause failures, etc.), the PDP was ranked no higher than MEDIUM. These sensitivity studies also included the average CDF and LERF where the PDP was categorized LOW.

The importance calculation affecting the categorization for the PDP is the Fussell-Vesely (FV) importance. FV measures the fraction of the overall risk involving sequences in which the component (i.e., PDP) is postulated to fail.

- **FV is a better indicator of component reliability on the selected figure-of-merit (i.e., core damage frequency);**
- **FV doesn't emphasize those components with high reliability and low overall fractional importance even though the impact of removing these from service could have significant impact; and**
- **Conversely, FV does highlight those components with low reliability levels which result in high fractional importance although the associated reduction in risk, given component success, is small.**

Because the PRA categorizes the PDP pump as HIGH partially due to previous poor performance, it is also expected that with the PDP's recent improved reliability and availability, the PRA importance categorization may result in a lower classification. Consideration for the low reliability and availability of this component demonstrates the robustness of the GQA risk categorization process.

34. *Due to redundancy, inboard and outboard containment isolation valves tend to be ranked low. The licensee decided to categorize inboard valves as high safety significant/medium safety significant and certain outboard valves as low safety significant. It is our understanding such a designation could change without any basis since you stated (during our visit to STP in October 1999) that it was only a matter of choice.*

    (a) *If both inboard and outboard containment isolation valves were considered to be low safety significant, explain why one was categorized high safety significant. Moreover, explain what would prevent you from designating both inboard and outboard isolation valves as low safety significant in the next or future operating cycle(s).*

    (b) *Explain the guidance and the basis for the guidance in helping to determine safety significance for similar situations or configurations.*

    (c) *Provide your expectations regarding the differences in monitoring/surveillance, stroke testing, and leak testing that LSS and HSS containment isolation valves will receive. Describe the implications of reclassifying the isolation valves on the maintenance rule implementation and the containment leakage performance indicator. Confirm whether containment isolation performance will be monitored at the component or system level.*

RESPONSE:

(a) **A clarification is in order concerning the risk categorization of inboard and outboard containment isolation valves. STP assigns the same risk categorization to both the inboard and outboard containment isolation valves at a particular system location. Any indication to the contrary was erroneous. Similarly, for other configurations where credit is taken for redundancy, the redundant components are assigned the same risk categorization.**

(b) **Containment isolation valves are typically categorized as LSS and meet one or more of the following criteria:**

    a. **The valve is required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers).**
    b. **The valve is normally closed and in a physically closed, water-filled system. (e.g., containment isolation valves in the Demineralized Water system)**
    c. **The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Component Cooling Water system and in the Instrument Air system).**

d. The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve.

e. The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).

As an atypical system, the Reactor Containment Building HVAC system did not meet the above criteria and its containment isolation valves were categorized as MSS. It is an air-to-air system. The line (duct) size is large and failure to isolate concurrent with a purging operation could lead to a radiation release. For additional information, see the response to RAI question 16.

(c) The implications of the classification process are primarily associated with differences in strategy and approach for LSS/NRS and HSS/MSS containment isolation valves. The strategy centers on programmatic measures used to predict and prevent degradation (HSS/MSS) versus programmatic measures used to repair or restore degradation once discovered (LSS/NRS). For HSS and MSS components, all special treatment requirements used to prevent, predict, monitor, and restore component functions are in place in order to provide reasonable assurance that components will perform their intended functions. For LSS and NRS components, stroke testing and/or leak testing is not required. In addition, special treatment requirements are not necessary since the reliability strategy is to monitor and restore component functions once they are identified through the corrective action program or the periodic feedback process.

For the LSS/NRS components, this is not to imply a run-to-failure philosophy. However, it is the intent and anticipation that degradations will be identified as a part of normal plant usage of these components and programs currently in effect at STP. Proceduralized programs already exist which provide mechanisms for identifying adverse trends and implementing corrective actions. The corrective action program is the trigger for other trending mechanisms such as system or train level Maintenance Rule performance criteria monitoring (system or train level monitoring) and the periodic feedback process (component level monitoring). Maintenance Rule monitoring for LSS/NRS SSCs consists of Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. The Maintenance Rule contains prescriptive actions for performance criteria being exceeded at the plant, system, or train level that are then reflected as corrective actions in the corrective action program. The feedback process reviews all corrective action items that have occurred over the last reporting period for the system under review and provides the information necessary for the Working Group to recommend corrective actions to the Expert Panel. Once approved, the corrective actions are implemented

and performance monitoring continues for the next reporting cycle. The use of these program controls provide reasonable assurance that LSS and NRS components are capable of performing their intended functions.

Once degradation has been identified in LSS or NRS components, then the restoration activities could include all the testing necessary to provide an appropriate level of assurance that the component is fully restored and functionally capable. This could include as part of corrective actions the performance of any or all surveillance testing, stroke time testing, leakage testing, or other refurbishments as required to provide reasonable assurance that the component is capable of performing its intended functions.

Thus, the implications of the classification process of the approach described above is that the containment isolation function is monitored and strategies for verifying containment isolation component functions are structured based on a component's risk significance. The containment leakage parameters will continue to be monitored at the component level for penetrations remaining within the Appendix J scope. Overall, the basic approach to the containment leakage parameter indicator remains unchanged (See the response to question 17 for additional details).

A summarization of certain specific differences in treatment between LSS/NRS and HSS/MSS containment isolation valves is given below:

	LSS/NRS	**HSS/MSS**
Monitoring/ Surveillances	Not normally within the surveillance program.	Components fall within the surveillance program.
Stroke testing	Not in the scope of required stroke testing	In the scope of required stroke testing.
Leak testing	Not in the scope of Appendix J testing	In the scope of Appendix J testing
Maintenance Rule Implementation	Monitored via the Maintenance Rule at the system/train/plant level if the SSC failure affects an HSS/MSS function.	In the Maintenance Rule scope.

35. *In Section 5.2.4.1, page 17 of your submittal, it is stated that you have identified approximately 100 non-safety-related SSCs that have been categorized as high safety significant and medium safety significant. To help us better understand your categorization process, please provide a list of these SSCs and a summary description of why they are important. Explain how this categorization is reflected in the plant PRA. The staff needs to have an understanding about the extent to which the PRA models relatively more significant plant equipment. (It may help to group certain components, as appropriate, when describing their-risk significance).*

RESPONSE:

Currently, there are 374 non-safety related SSCs risk ranked MEDIUM or HIGH. Of these, 220 are fire dampers in the Mechanical Auxiliary Building HVAC (HM) system. Attachment 1 provides a representative sample by listing only the Unit 1, train A components. In accordance with our implementation process, these components are evaluated to determine what additional quality assurance controls are to be applied to them.

The Attachment 1 listing shows the PRA risk, where applicable and/or modeled and the final risk. In some cases, there is no PRA risk because the component is not explicitly or implicitly modeled (e.g., AF turbine steam inlet drain line water level sensing switch). In other cases, there is no PRA risk because the component is implicitly modeled as part of a larger component (e.g., the manual control station for the RHR heat exchanger flow control valve is implicitly modeled as part of the valve). In the remaining cases, the final risk is sometimes driven by the PRA risk (e.g., positive displacement pump motor) or by the deterministic risk.

As is the case with safety related components, the final risk is a blending of the PRA risk and the deterministic risk. Where the component is not explicitly modeled by the PRA, the deterministic risk becomes the final risk. Fire dampers are examples of these and make up a large percentage of the Attachment 1 components.

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
AF	IBISSW	N1AFLSH7600	TDAFWP #14 T&T VALVE STEAM INLET DRAIN LINE WATER LEVEL		MEDIUM	PART OF LOOP IS USED TO MONITOR LEVEL IN THE TURBINE DRIVEN AUXILIARY FEED WATER PUMP INLET STEAM DRAIN LINES. THE LEVEL SWITCH ACTUATES ON HIGH LEVEL TO PROVIDE AN INPUT SIGNAL (ALARM DATA POINT) ON HIGH LEVEL ABOVE SET POINT TO THE PROTEUS PLANT COMPUTER. AN UNDETECTED HIGH LEVEL COULD CAUSE AN OVERSPEED TRIP OF THE TURBINE ON START-UP. REFER TO FUNCTION 4.3 AND ITS BASIS.
AF	IXMITR	N1AFLE7600	TDAFWP #14 T&T VALVE STEAM INLET DRN LINE WATER LVL		MEDIUM	PART OF LOOP IS USED TO MONITOR LEVEL IN THE TURBINE DRIVEN AUXILIARY FEED WATER PUMP INLET STEAM DRAIN LINES. THE LEVEL SWITCH ACTUATES ON HIGH LEVEL TO PROVIDE AN INPUT SIGNAL (ALARM DATA POINT) ON HIGH LEVEL ABOVE SET POINT TO THE PROTEUS PLANT COMPUTER. AN UNDETECTED HIGH LEVEL COULD CAUSE AN OVERSPEED TRIP OF THE TURBINE ON START-UP. REFER TO FUNCTION 4.3 AND ITS BASIS.
AF	PIPE	N1AFFO7552	LUBE OIL PUMP 15 RECIRC FLOW ORIFICE		MEDIUM	USED TO MAINTAIN PROPER OIL FLOW AND PRESSURE. FAILURE COULD IMPACT OPERATION OF THE TURBINE
AF	PIPE	N1AFFO7553	TERRY TURBINE GOVERNOR END BRG LUBE OIL SUPPLY FLOW ORIFICE		MEDIUM	USED TO MAINTAIN PROPER OIL FLOW AND PRESSURE. FAILURE COULD IMPACT OPERATION OF THE TURBINE
CV	CKTBRK	N1CVHS0286	POS DISP CHG PUMP 1A SEL SW		MEDIUM	MANUALLY OPERATED TO START POSITIVE DISPLACEMENT PUMP. RISK IS ONE LEVEL LOWER THAN PUMP RISK
CV	MOTOR	N1CVPA102A	CVCS POSITIVE DISPLACEMENT|CHARGI NG PUMP MOTOR|TPNS: 2R171NPA102A	H	HIGH	PRIMARILY USED FOR HYDROTESTING THE RCS. PROVIDES A MEANS FOR ADDING CHEMICALS TO THE RCS FOR pH AND OXYGEN CONTROL. PROVIDES SEAL INJECTION FLOW IF BOTH CCPs ARE INOPERABLE
CV	VALVE	N1CVLY3119	CVCS AUXILIARY SPRAY|LV-3119 SOLENOID VALVE	L	MEDIUM	OPENS MAIN VALVE ONLY WHEN SUPPLYING AUX SPRAY TO PZR TO COLLAPSE STM BUBBLE/COOL PZR DURING COOLDOWN OR TO DEPRESSURIZE SG IN CASE OF TUBE RUPTURE. MAIN VALVE IS 2ND VALVE AFTER CV-0009 TO PROVIDE RCS PRESS BOUNDARY INTEGRITY. MAIN VALVE FAILS CLOSED
HE	DAMPER	7V101VFF078	MAB MAIN EXHAUST AIR FUSIBLE LINK FIRE DAMPER (Note: risk approved by EP, to be implemented @ 6-month review)		MEDIUM	FIRE DAMPERS PROVIDE CAPABILITY TO ISOLATE HVAC TRAINS, SUB-SYSTEMS OR DUCTS TO PROTECT REDUNDANT EQUIPMENT NEEDED FOR SAFE SHUTDOWN OF THE REACTOR IN THE EVENT OF A FIRE. FIRE DAMPERS, LOCATED INSIDE HVAC DUCT, ACTIVATE WHEN INTERNAL DUCT TEMPERATURE MELTS FUSIBLE LINK OR UPON RECEIPT OF ELECTRO-THERMAL SIGNAL FROM FIRE DETECTION SYSTEM
HE	IBISSW	N1HEXSH9583	EAB OUTSIDE AIR INTAKE HIGH SMOKE DETECTION SWITCH		MEDIUM	DETECTOR PROVIDES A SIGNAL TO ISOLATE MAIN CONTROL ROOM AND TSC INLET HVAC DAMPERS.

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
HE	IBISSW	N1HEXSH9601	CONTROL ROOM TRAIN A RETURN AIR HIGH SMOKE DETECTION SWITCH		MEDIUM	SMOKE DETECTOR IN THE RETURN AIR DUCT OF ONE OF THREE OF THE CONTROL ROOM ENVELOPE CLEAN-UP AIR HANDLING UNITS (AHU). ACTUATES UPON THE DETECTION OF SMOKE TO PROVIDE AN ANNUNCIATION (22M-3-05F) IN THE CONTROL ROOM (CR).
HE	IXMITR	N1HEXE9601	CONTROL ROOM TRAIN A RETURN AIR SMOKE DETECTOR		MEDIUM	SMOKE DETECTOR IN THE RETURN AIR DUCT OF ONE OF THREE OF THE CONTROL ROOM ENVELOPE CLEAN-UP AIR HANDLING UNITS (AHU). ACTUATES UPON THE DETECTION OF SMOKE TO PROVIDE AN ANNUNCIATION (22M-3-05F) IN THE CONTROL ROOM (CR).
HM	CKTBRK	N1HMHS9419	TIE DAMPER FV-9419		MEDIUM	REFER TO ASSOCIATED COMPONENT
HM	DAMPER	[VARIOUS]	[FIRE DAMPER, TYPICAL. TOTAL OF 220 RANKED MEDIUM]		MEDIUM	FIRE DAMPERS PROVIDE CAPABILITY TO ISOLATE HVAC TRAINS, SUB-SYSTEMS OR DUCTS TO PROTECT REDUNDANT EQUIPMENT NEEDED FOR SAFE SHUTDOWN OF THE REACTOR IN THE EVENT OF A FIRE. FIRE DAMPERS, LOCATED INSIDE HVAC DUCT, ACTIVATE WHEN INTERNAL DUCT TEMPERATURE MELTS FUSIBLE LINK OR UPON RECEIPT OF ELECTRO-THERMAL SIGNAL FROM FIRE DETECTION SYSTEM.
IA	BLOWER	8Q111MCO0106	INSTRUMENT AIR COMPRESSOR 11	M*	MEDIUM	PROVIDES CONTINUOUS SUPPLY OF FILTERED, DRY, OIL-FREE COMPRESSED AIR AT SUITABLE PRESSURE AND FLOWRATE FOR PNEUMATIC INSTRUMENT OPERATION AND CONTROL OF PNEUMATIC VALVE AND DAMPER ACTUATORS. DETERMINISTICALLY RANKED AS LOW. FINAL RISK BASED ON PRA.
IA	VALVE	8Q111TIA0027	INSTRUMENT AIR RECEIVER OUTLET CHECK VALVE	M*	MEDIUM	PREVENT BACKFLOW WHEN THE SERVICE AIR SYSTEM IS PROVIDING AIR TO THE INSTRUMENT AIR SYSTEM. DETERMINISTICALLY RANKED AS LOW. FINAL RISK BASED ON PRA.
IA	VESSEL	8Q111MTS0162	INSTRUMENT AIR RECEIVER	M*	MEDIUM	SUPPLIES COMPRESSED AIR FOR PNEUMATIC CONTROLS, ACTUATION OF VALVES, DAMPERS AND SIMILAR DEVICES. AIR RECEIVER VOLUME IS BASED ON 2 MINUTE NORMAL SUPPLY OF INSTRUMENT AIR IN THE EVENT OF COMPRESSOR TRIP. DETERMINISTICALLY RANKED AS LOW. FINAL RISK BASED ON PRA.
RC	IBISSW	N1RCPS0455Z	RCS PRZR 1A PRZR PRESS CONT SEL|SW		MEDIUM	ALLOWS OPERATOR TO SELECT ONE OF FOUR PRESSURIZER PRESSURE CHANNELS
RC	ICLOOP	N1RCP0655B	RCS PRZR 1A LOOP 4 SPRAY VALVE		MEDIUM	THIS LOOP SENSES PRESSURIZER PRESSURE AND PROVIDES A CONTROL SIGNAL TO THE PRESSURE SPRAY VALVES TO OPENTHE VALVE TO RELIEVE PRESSURE IN THE PRESSURIZER
RC	ICNTRL	N1RCPC0655A	RCS PRZR 1A LOOP 4 SPR VALVE|PCV-0655 CONTROLLER		MEDIUM	ACTS TO MODULATE PCV0655A
RC	ICNTRL	N1RCPC0655B	RCS PRZR 1A LOOP 4 SPR VALVE|PCV-0655B CONTR		MEDIUM	MODULATES PCV-0655B OPEN ON HIGH PRESSURE TO PREVENT THE PRESSURIZER PRESSURE FROMJ REACHING THE SETPOINT OF THE PORVs

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
RC	ICNTRL	N1RCPC0655C	RCS PRZR 1A LOOP 4 SPR VALVE|PCV-0655 CONTROLLER		MEDIUM	MODULATES PCV-0655C OPEN ON HIGH PRESSURE TO PREVENT THE PRESSURIZER PRESSURE FROMJ REACHING THE SETPOINT OF THE PORVs
RC	ICNTRL	N1RCPK0655A	PRESSURIZER 1A|PORV (PCV-655A) I/P CONVERTER		MEDIUM	THE THREE CONTROL STATIONS (PK0655A, B, AND C) LOCATED IN THE CONTROL ROOM PROVIDE THE OPERATOR MANUAL OR AUTOMATIC CONTROL OVER THE PRESSURIZER SPRAY VALVES. CONTROL OF THE PRESSURIZER SPRAY IS REQUIRED TO PREVENT THE PRESSURE OF THE PRESSURIZER FROM EXCEEDING THAT OF THE PRESSURIZER RELIEF VALVES. PK0655A IS AN NCB CARD IN 7300 CABINET
RC	ICNTRL	N1RCPK0655B	RCS PZR 1A LOOP 1D SPRAY|VLV (PCV-0655B) I/P CONVERTER		MEDIUM	THREE HAND CONTROL STATIONS (PK0655A, B, AND C) IN THE CONTROL ROOM ARE AVAILABLE TO PROVIDE THE OPERATOR CONTROL OVER THE PRESSURIZER SPRAY VALVES. CONTROL OF THE PRESSURIZER SPRAY IS REQUIRED TO PREVENT THE PRESSURE OF THE PRESSURIZER FROM EXCEEDING THAT OF THE PRESSURIZER RELIEF VALVES.
RC	ICNTRL	N1RCPK0655C	RCA PRZR 1A LOOP 1 SPRAY|PCV-0655C CONT STA		MEDIUM	FAILURE COULD CAUSE POSSIBLE LOSS OF EFFECTIVE OPERATOR CONTROL OF PRESSURIZER SPRAY.
RC	INDREC	N1RCLG3660	REACTOR COOLANT SYSTEM LOOP 1A|MID LOOP OPERATIONS LEVEL GAUGE		MEDIUM	PROVIDES LOCAL INDICATION, ERFDADS INFORMATION, CONTROL ROOM INDICATION, OF REACTOR VESSEL WATER LEVEL DURING MIDLOOP OPERATIONS.
RC	INDREC	N1RCLR3660	RCS LEVEL LOOP A AND C MID LOOP OPERATION (2-PEN)		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	INDREC	N1RCPI0407A	RCS LOOP 1 WR PRESS		MEDIUM	AUX SHUTDOWN PANEL INDICATION
RC	INTCPM	N1RCPY3656C	PRESSURIZER|LOOP 1A SPRAY VALVE PCV-0655C|I/P PRESSURE CONVERTER		MEDIUM	ONE OF 2 PRESSURIZER SPRAY CONTROL VALVES USED TO PROVIDE SPRAY TO THE PRESSURIZER TO ASSIST IN EQUALIZING THE BORON CONCENTRATION BETWEEN THE REACTOR COOLANT LOOPS AND THE PRESSURIZER. THESE VALVES ARE AUTOMATICALLY MODULATED OPEN ON HIGH PRESSURE TO PREVENT THE PRESSURIZER PRESSURE FROM REACHING THE OPERATING (SET) POINT OF THE POWER-OPERATED RELIEF VALVES FOLLOWING A STEP LOAD REDUCTION.
RC	IXMITR	N1RCLIT3662	RCS MID LOOP OPERATIONS|LEVEL INDICATING TRANSMITTER		MEDIUM	PROVIDES LOCAL INDICATION OF REACTOR VESSEL WATER LEVEL DURING MIDLOOP OPERATIONS.
RC	IXMITR	N1RCLT0675	PRESSURIZER|COLD CAL LEVEL TRANSMITTER		MEDIUM	RC-L-0675 IS A FIFTH NON-CLASS 1E PRESSURIZER LEVEL TRANSMITTER/INDICATOR, CALIBRATED FOR LOW TEMPERATURE CONDITIONS. IT PROVIDES SIGNALS FOR PRESSURIZER WATER LEVEL AND ERFDADS DURING STARTUP, SHUTDOWN, AND REFUELING OPERATIONS.

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
RC	IXMITR	N1RCLT3660	REACTOR COOLANT SYSTEM LOOP 1A|OPERATIONS LEVEL TRANSMITTER		MEDIUM	THIS LEVEL LOOP SENSES REACTOR COOLANT LEVEL AND PROVIDES A RECORDING OF THIS LEVEL AND LOW-LOW LEVEL ANNUNCIATION (01M2-1F) IN THE CONTROL ROOM DURING MID LOOP OPERATION. THIS INFORMATION PROVIDES THE OPERATOR INFORMATION TO ASSIST IN MAINTAIN LEVEL WITHIN THE MID LOOP OPERATING BAND.
RC	MECFUN	9C241NXN101	REACTOR VESSEL-TO-CAVITY SEAL RING		MEDIUM	USED DURING REFUELING OPERATIONS
RC	MECFUN	RC1014HL5003W	REACTOR COOLANT SYSTEM|MECHANICAL SNUBBER|MODEL NUMBER: AD5501		MEDIUM	LIMITS PIPE STRESS DURING SEISMIC EVENTS. RISK BASED ON LOW PROBABILITY AND VERY LOW MAGNITUDE OF SEISMIC EVENTS AT STP
RC	MECFUN	RC1014HL5005S	REACTOR COOLANT SYSTEM|MECHANICAL SNUBBER|MODEL NUMBER: AD5501		MEDIUM	LIMITS PIPE STRESS DURING SEISMIC EVENTS. RISK BASED ON LOW PROBABILITY AND VERY LOW MAGNITUDE OF SEISMIC EVENTS AT STP
RC	MECFUN	RC1014HL5009	REACTOR COOLANT SYSTEM|MECHANICAL SNUBBER|MODEL NUMBER: AD501		MEDIUM	LIMITS PIPE STRESS DURING SEISMIC EVENTS. RISK BASED ON LOW PROBABILITY AND VERY LOW MAGNITUDE OF SEISMIC EVENTS AT STP
RC	MECFUN	RC1014HL5026	REACTOR COOLANT SYSTEM|MECHANICAL SNUBBER|MODEL NUMBER: AD501		MEDIUM	LIMITS PIPE STRESS DURING SEISMIC EVENTS. RISK BASED ON LOW PROBABILITY AND VERY LOW MAGNITUDE OF SEISMIC EVENTS AT STP
RC	VALVE	7R141TRC0203	(IRC) RV HD FE 3659A ISOL BYPASS		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141TRC0518	(IMB) RCS LEVEL SIGHT GLASS LIT-3662 DRAIN VALVE		MEDIUM	USED DURING MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0208	(IRC) LOOP 1 LEVEL TRANSMITTER LT-3660 ISOL VLV		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141ZRC0210	(IMB) LOOP C LG-3661 UPPER ROOT VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0211	(IMB) LOOP 1 LEVEL GAGE LG-366O VENT VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
RC	VALVE	7R141ZRC0212	(IMB) LOOP A MID LOOP LEVEL GAGE, LG-3660 DRAIN VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0213	(IMB) LOOP A MID LOOP LEVEL GAGE, LG-3660 UPPER ISOL		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141ZRC0214	(IMB) LOOP A LG-3660 LOWER ROOT VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0215	(IMB) LOOP A LG-3660 LOWER ROOT VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0216	(IMB) LOOP A MID LOOP LEVEL SENSING LINE VENT		MEDIUM	USED DURING MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0217	(IMB) LOOP 3 LEVEL GAGE LG-3661 VENT VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0218	(IMB) LOOP 3 LEVEL GAUGE LG-3661 DRAIN VALVE		MEDIUM	SUPPORTS MID-LOOP OPERATIONS
RC	VALVE	7R141ZRC0219	(IMB) LOOP 3 LEVEL GAGE LG-3661 UPPER ISOLATION		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141ZRC0220	(IMB) LOOP 3 LEVEL GAUGE LG-3661 LOWER ISOLATION		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141ZRC0221	(IMB) LOOP 3 LEVEL GAGE LG-3661 LOWER ISOLATION		MEDIUM	NORMALLY OPEN ROOT VALVE CONNECTED TO RCS PRESSURE BOUNDARY. PRESSURE BOUNDARY FAILURE OF VALVE MITIGATED BY UPSTREAM FLOW RESTRICTOR
RC	VALVE	7R141ZRC0222	(IMB) LOOP 3 LEVEL TRANS LT-3661 VENT VALVE		MEDIUM	USED DURING MID-LOOP OPERATIONS

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
RH	ICNTRL	N1RHHC0864	RHR HEAT EXCHANGER 1A CONTROL		HIGH	THE MANUAL CONTROL STATION PROVIDES REMOTE MANUAL CONTROL OF THE TRAIN A RHR HEAT EXCHANGER FLOW CONTROL VALVE FROM THE CONTROL ROOM OR THE AUX SHUTDOWN PANEL. THIS VALVE DOES NOT PERFORM A SAFETY FUNCTION. HOWEVER, THE VALVE IS NORMALLY OPEN AND FAILS OPEN TO ENSURE CORRECT POSITIONING DURING SAFETY INJECTION AND SAFE SHUTDOWN OPERATION. THE VALVE IS PROVIDED TO MANUALLY CONTROL THE REACTOR COOLANT FLOW THROUGH THE RHR HEAT EXCHANGER AND, SUBSEQUENTLY, THE RATE OF COOLDOWN OF THE RCS SYSTEM.
RH	ICNTRL	N1RHHK0864	RHR HEAT EXCHANGER 1A CONTROL		HIGH	THE MANUAL CONTROL STATION PROVIDES REMOTE MANUAL FLOW CONTROL THROUGH ONE OF THREE TRAINED RHR HEAT EXCHANGERS FROM THE CONTROL ROOM. THE FLOW CONTROL VALVE DOES NOT PERFORM A SAFETY FUNCTION, HOWEVER, THE VALVE IS NORMALLY OPEN AND FAILS OPEN TO ENSURE CORRECT POSITIONING DURING SAFETY INJECTION AND SAFE SHUTDOWN OPERATION.
RH	RELAY	N1RHFY3860	RHR HEAT EXCHANGER 1A OUTLET VALVE FV-3860 CURRENT/PNEUMATIC CONVERTOR		HIGH	RHR HEAT EXCHANGER FLOW CONTROL: THE PNEUMATIC TRANSDUCER (FY) RECEIVES AN ANALOG ELECTRICAL SIGNAL FROM A HAND CONTROLLER IN THE CONTROL ROOM AND CONVERTS THE ELECTRICAL SIGNAL TO A PNEUMATIC SIGNAL TO PROVIDE FOR THE POSITIONING OF AN AIR OPERATEDBUTTERFLY VALVE (FV) TO CONTROL REACTOR COOLANT FLOW THROUGH THE RHR HEAT EXCHANGER AND, SUBSEQUENTLY, THE RATE OF RCS COOLDOWN. PERFORMS NO SAFETY-RELATED FUNCTION. NORMALLY OPEN AND FAILS OPEN TO ENSURE CORRECT POSITIONING DURING SAFETY INJECTION, POST ACCIDENT AND THE ABILITY TO REACH SAFE SHUTDOWN.
SI	INTCPM	N1SIFY3857	RHR HEAT EXCHANGER 1A FCV-0851 CURRENT/PNEUMATIC CONVERTER		MEDIUM	PROVIDES FOR THE CONVERSION FROM AN ELECTOMAGNETIC SIGNAL TO A PNEUMATIC PRESSURE TO CONTROL VALVE FCV0833 FROM A SIGNAL FROM THE OUTPUT OF THE REMAINDER OF THE LOOP.

36. *In estimating the importance measures, Fussell-Vesely (FV) and Risk Achievement Worth (RAW), you have used the mean values of the parameters in the ratios. This practice usually results in reasonable approximation; however, this may not be the case for parameters whose epistemic uncertainties are very large. Please explain if this problem applies to your proposal and discuss how you will resolve it.*

RESPONSE:

Per a telephone conversation with the NRC staff on March 6th, 2000, the question concerning epistemic uncertainty can be addressed by calculating component importance for different categories of external events. External events, in general, rarely occur and, therefore, have large uncertainties. Sensitivity studies were performed to determine component importance associated with the following categories of external events: fires, floods, and seismic initiating events. A full quantification of the PRA model is performed for each sensitivity study of the external event category. Each category contains more than one initiator to describe the event. For example, the STP PRA analyzes seismic initiating events using four initiators. These are as follows:

Initiator	Description	Frequency
SEISM1	SEISMIC EVENT - G LEVEL 0.1	3.02E-05
SEISM2	SEISMIC EVENT - G LEVEL 0.2	2.89E-06
SEISM3	SEISMIC EVENT - G LEVEL 0.4	7.74E-07
SEISM4	SEISMIC EVENT - G LEVEL 0.6	6.14E-08

The sensitivity studies for fire and flood have similar classifications containing similar initiating events.

The same PRA ranking methodology used to calculate component importance was used for these sensitivity studies. In each case, the component's risk rank resulting from the sensitivity study was never more conservative than the current composite PRA risk rank. The following table represents changes from the composite PRA risk ranking to the sensitivity study component risk rankings:

	External Initiating Events		
	Fires	Floods	Seismic
No. of Components Remaining High	8	0	1
Change from High to Medium	38	13	8
Change from High to Low	251	281	288
Change from High to Medium-R	0	3	0
No. Remaining Medium-R	0	0	0
Change from Medium-R to Medium	3	0	0
Change from Medium-R to Low	134	137	137
No. Remaining Medium	62	0	0
Change from Medium to Low	170	232	232
No. Remaining Low	448	448	448
Total	1114	1114	1114
Note, there were no increases in the PRA ranking associated with this study			

The above results for the sensitivity studies demonstrate that no component increased in risk rank when analyzing only for the external event categories. For example, if the PRA rank were based only on fire initiators, there would be 289 fewer components in the high rank category, and 170 fewer components in the medium rank category.

The main reason component importance has decreased or stayed the same is due to the overall importance that external events have on the PRA model. For the most part, fires, floods, and seismic events guarantee failure of affected components. Those components that are affected by external events and are guaranteed failed will generally have a low risk ranking since the reliability and availability of the component does not impact the mitigation of accident/transient events. Note that all components in the PRA model are ranked at least low.

As shown by this analysis, the STP PRA risk ranking process is not susceptible to the influence of external events and their epistemic uncertainties. These sensitivity studies provided no new information to the PRA risk ranking process. Therefore, the STP risk rank process appropriately factors in the impacts of external events, and STP has no plans to change the current PRA risk ranking process based on these findings.

37. *You have taken the "Graded Quality Assurance" addendum from the "Comprehensive Risk Management" procedure (Rev. 2 dated 01/02/97) and issued a new procedure on "Graded Quality Assurance Working Group Process" (Rev. 0 dated 8/12/98). The new procedure has added explicit guidance for assigning components a lower significance than the safety significance of the function they support. The licensee's current guidance is as follows:*

If the component failure will fail the function or if credit for component reliability cannot be taken, then the component is ranked at the same risk as the highest risk function it supports.

As a general rule of thumb, if redundancy or backup is available and the reliability of the associated components has been good, the critical questions for the component can be answered at a lower value than given for the highest risk function supported by the component. However, the WG [working group] should use conservative judgement when taking credit for component redundancy

You use five "critical questions" to determine risk of a system function or component ranking. These questions are related to the impact on initiating event, risk significant system, accident/transient, emergency operating procedures, and shutdown/mode change. The response to these questions is one of any points ranging from "0" to "5." For example a score of "1" denotes "positive response having insignificant impact and/or occurring very rarely" and a score of "5" denotes "positive response having high impact and/or occurring frequently."

If this procedure is to be used for the proposed exemption request, explain how many points lower the "critical question" score can be assigned to a redundant component relative to the function's critical question score. For example, if a critical question score is "5" for a particular function, discuss whether a score of "4" or lower should be assigned for the relevant redundant components. Discuss whether all five (or all non-zero) critical question scores for all redundant components are scored lower than the scores for their function. If only "selected" redundant components are scored lower, provide the basis for such a decision. If only selected critical questions are scored lower, provide the basis for such a decision. If a component is placed in a lower safety significance category as a result of being assigned a lower critical question score, discuss how a justification (including a description of how a component is judged to be highly reliable) is developed.

RESPONSE:

Note: a clarification is necessary in support of this Question's response. As stated in the response to Question #27, STP is modifying the documentation requirements in the Risk Significance Basis Documents (RSBDs) to eliminate individual responses to each critical question at the component level. Therefore, while redundancy is considered at a component level, individual component critical question scores are not lowered a specific amount; the component's overall risk categorization is generally considered for lowering by one level. With this consideration in mind, additional insights to this question are provided below.

A component's categorization may be considered for one level lower than the most limiting system function supported when there are diverse means of satisfying the system function. In addition, if there are multiple, independent means of satisfying the system function, a reduction in categorization may be considered. Merely having multiple trains of a component available in a system does not automatically result in a lower risk categorization for a component.

When considering whether component redundancy or diversity is a factor, the Working Group evaluates redundancy based on system operating configuration, reliability history, recovery time available, and other factors. The Working Group examines the effect of the component failure on each system function supported by that component. The primary consideration is whether failure of the component will fail or severely degrade the function. If the answer is no, then component redundancy may be factored in, as long as the component's reliability and that of its redundant counterpart have been satisfactory. Component reliability is evaluated through reviews of Condition Reports, System Health Reports, inputs from the System Engineer, and input from the Operations representative on the Working Group. A component could be considered reliable when the component demonstrates strong operating performance with few deficiencies, the component has no open concerns based on industry operating experience, and site operating experience reflects no negative reliability trends or concerns. The final risk of the component cannot be "NRS" if the system function is "LSS", and generally cannot be more than one risk level lower than the system function.

STP's risk significance determination process requires that the justification used to support a risk categorization be documented. At the system function level, this is done by answering the critical questions and documenting the basis for each question response. For components, the first step in the risk categorization process consists of identifying the system function(s) that each component supports. This is documented in Attachment H of the Risk Significance Basis Document. Next, the component is initially assigned the same risk as the most limiting system function that it supports. If the Working Group reaches consensus that the initial risk is satisfactory, no additional documentation is required since the justification has already been provided within the function categorization. Only in the case where component redundancy or other insight is used to rank the component risk lower than the risk of its most limiting supported system function would additional documentation be required. This additional documentation is typically provided in Attachment I of the RSBD, under the General Notes or specific to the subject component under the "Additional Deterministic Input" column.

STP has concluded that answering the critical questions at the component level provides little value and introduces a greater potential for administrative documentation errors. The majority of components are risk categorized the same as the most limiting system function(s) that they support. Therefore, they were automatically assigned the same responses to the critical questions as the most limiting response from the supported system function(s). Clearly, there is no value gained in answering the critical questions for these components. For the other components where the risk is lower than the most limiting system function(s), STP had been providing additional documentation and relying on general notes to support the risk categorization. The responses to the critical questions were also adjusted downward to correspond with the risk categorization agreed to by the Working Group. In some cases, the linkage between the lower risk categorization and the documented justification was not clear and the numerical responses to the critical questions did not provide the clarification needed.

Based on the above, STP has decided to forego answering the critical questions at the component level and to bolster the documentation that justifies the risk categorization for those components whose risk is lower than the most limiting system function. As discussed in the response to Question 27, STP has initiated a condition report to review this issue and to strengthen the documentation, where necessary.

It is emphasized that the risk categorizations assigned to these components are considered satisfactory as they are the result of Working Group discussions and consensus, and do adhere to procedural guidance concerning component redundancy. As noted in the procedure, the Working Group utilizes conservative judgment when taking credit for component redundancy. The risk categorization recommendations and their bases are not finalized until presentation to and approval by the Comprehensive Risk Management Expert Panel.

38. *In order to understand the licensee's special treatment process, the staff needs more information on the following example. Section 7.2.1 of the licensee's proposed exemption request states, "In addition, as appropriate, STP will modify various programs (e.g., provisions for motor-operated valve (MOV) program, air-operated valve (AOV) program, snubber testing program, molded case circuit breaker program) to remove safety-related LSS and NRS components from the scope of these programs."*

    (a) Does this mean, for example, that LSS and NRS MOVs will be taken out of the scope of the GL 89-10 and GL 96-05 programs?

    (b) If it is your proposal to remove safety-related MOVs and AOVs from the scope of the current programs, please explain how it will be adequately demonstrated that the valves will continue to be capable of performing their safety-related functions.

RESPONSE:

(a) LSS and NRS safety-related MOVs will be taken out of the scope of the GL 89-10 and GL 96-05 programs. Although an exemption is not required to modify these programs, the modifications will be processed in accordance with STP's Commitment Change Process.

(b) For LSS and NRS safety-related MOVs that are removed from the scope of the GL 89-10 and GL 96-05 programs, STP will apply appropriate commercial treatments to provide reasonable assurance that the subject valves will be capable of satisfying their functional requirements. These commercial treatments (refer to the response to RAI 9, Section 13.7.3.3) will include the continuation of normal preventive maintenance (PM) activities to properly lubricate and inspect the MOVs. If deficiencies are noted during normal PM activities, the Corrective Action Program (CAP) will require generation of a Condition Report to document the deficiency and schedule it for correction. If correction of a LSS MOV affects the critical attribute for the MOV, an appropriate post-maintenance test (PMT) will be performed to provide assurance that the functional requirements of the valve can be validated. This will normally require an exercise stroke of the valve.

In addition, MOVs are periodically exercised during normal routine operations. These exercise operations also provide assurance that the valves can perform their function. If a deficiency is noted during these valve strokes, again, the Corrective Action Program will be used. Items that are entered into the Corrective Action Program are evaluated on a periodic basis through the monitoring and feedback program. If a decline in performance is noted, the Working Group will evaluate any additional controls which should be applied to the valve to return its performance back to expectations. If the enhancement of controls does not provide satisfactory results, the Working Group can re-evaluate the MOV for adjustment to its risk categorization.

MOVs will remain within the STP configuration and design control program. If an alteration to the valve is desired, the configuration and design control program and the 50.59 process will be used if functional or design features are affected.

While Air Operated Valves (AOVs) are not currently covered by a Generic Letter like MOVs, the practices applied to LSS and NRS safety-related AOVs would be similar to MOVs. AOVs will be subject to appropriate commercial treatment programs that will include performance of preventive maintenance and appropriate post-maintenance tests following corrective maintenance. AOVs also are periodically stroked during normal operations, and the AOVs will be subject to the Corrective Action Program and to the monitoring and feedback programs. The good business practices that are in use today, such as the use of diagnostic equipment to set up and troubleshoot AOVs, will be available for continued use. AOVs will remain in the STP configuration and design control program, and the 50.59 process will be used if functional or design features of the valves are affected.

39. (a) *Identify the process that will be used to select codes, standards, and plant procedures that describe "normal commercial and industrial practices" that will be used to procure, install, inspect, test, and maintain plant equipment that is removed from the scope of the special treatment controls. Please describe how the codes and standards will be evaluated to consider their use in lieu of the current special treatment requirements. Please provide some representative examples of the codes and standards that will utilized for the LSS and NRS equipment.*

(b) *Explain how these standards and procedures will provide adequate assurance that these components will remain functional under design-basis conditions (following a seismic or other external event and under design-basis environmental conditions). For example, the licensee could provide specific examples that demonstrate (or an analysis of data which supports the assertion) that certain commercial-grade components will remain functional under design-basis accident-like conditions.*

(c) *Similarly, for non-safety related SSCs that have been categorized as HSS or MSS, how will the licensee identify the conditions under which these components must function and how will the licensee identify the practices that need to be applied to these components in order to ensure their functionality.*

(d) *How will the licensee's process address the EQ qualified lifetime for safety-related components categorized LSS when those lifetimes are reached?*

RESPONSE (part a):

The large majority of components at STP are non-safety related, and are commercially procured and maintained. It is in the best interest of the utility (both from a safety perspective and from an economic perspective) to ensure that these components operate reliably and within their design functional requirements. In order to ensure that these components meet our safety and performance objectives, appropriate codes, standards, and site procedures are currently used to provide this assurance. Some of the codes and standards that are currently in effect on the balance-of-plant include ANSI B31.1 piping specifications, IEEE electrical specifications, etc. These specifications are used world-wide in a number of varied industrial applications. The systems and components that these specifications are applied against operate reliably to meet the safety and competitive challenges that exist.

For safety-related LSS and NRS components that are removed from the scope of special treatment requirements, STP proposes to impose sound commercial practices that are currently in place on the balance-of-plant and have a proven track record of demonstrated reliability and functionality (refer to the response to RAI 9, Section 13.7.3.3). Since these codes and standards are already in place and proven, STP does not intend to perform additional evaluations of the suitability of these codes and standards. STP considers that the use of these codes and standards is sufficient to provide an appropriate level of assurance of the satisfactory procurement, installation, testing, and maintenance of LSS

and NRS components. Additional evaluations are not necessary to provide assurance that these LSS and NRS components can reasonably perform their design functional requirements.

In addition, STP has programmatic procedures in place which will apply appropriate administrative controls over the LSS and NRS components that have been removed from the scope of special treatment. It is not anticipated that significant revisions to these procedures will be necessary; rather, clarification of the system/component scope to which the procedures apply will be updated.

RESPONSE (part b):

The components in question are risk categorized as LSS or NRS. As such, these components serve little, if any, function in mitigating the consequences of an accident or protecting the health and safety of the public during a design-basis event or any other credible event. However, commercial treatment, as specified in the response to part (a), will be applied to these components and will provide reasonable assurance that the design functional requirements are met. For components that must change state, the majority of these signals will be generated early in the accident sequence which is generally well before the environmental conditions have changed significantly. For components that must remain in service, it is reasonable to conclude that a commercial component that has similar design features and identical functional characteristics can and will perform its function when demanded. STP further notes that, except for normally open valves and other components that do not change state, very few, if any, components that must remain in service during a design basis event are risk categorized as either LSS or NRS.

For example, if an LSS transmitter that is both EQ and seismically qualified fails in normal service, the failed component will be replaced with a commercially available transmitter that satisfies the components functional requirements. This replacement transmitter will be procured with appropriate seals to meet the temperature rating and humidity rating that the transmitter is reasonably expected to see. The transmitter housing would be similarly robust compared to the failed transmitter's housing. If a design-basis event were to occur while the replacement transmitter was in service, it is reasonable to conclude that the replacement transmitter would perform its function. However, if the transmitter did unexpectedly fail, a component categorized as LSS is not relied on to mitigate the consequences of the accident.

RESPONSE (part c):

During the risk categorization process, non-safety related HSS and MSS components have critical attributes identified and documented in the Risk Significance Basis Document (RSBD). These critical attributes specify the risk significant functions that the component must perform. These critical attributes are also entered into the electronic Master Equipment Database which is available for query by Station personnel. These critical attributes are referenced during the procurement process (purchasing and receipt inspection), maintenance process (corrective maintenance, preventive maintenance, post-maintenance testing, etc), design process (engineering evaluations, design changes, etc), and others to provide appropriate insight into the administrative processes which ensure reliable component operation.

Non-safety related HSS and MSS components are placed under the TARGET program at STP. Once these components are identified as safety-significant through the risk categorization process, a Condition Report is generated to evaluate the existing controls that are placed on these components, and to identify what, if any, additional controls are needed to provide reasonable assurance that the component can satisfy its risk significant functional requirements. In addition, these components will be placed under the Maintenance Rule monitoring program. The reliability of these components will be assessed on a periodic basis through the GQA feedback process. This process will evaluate the reliability of the component, the adequacy of the existing controls and risk categorization, and the need for any changes.

These identified processes provide the appropriate assurance that the non-safety related, safety-significant SSCs would be properly monitored and their risk significant functionality ensured.

RESPONSE (part d):

Equipment Qualification (EQ) is a special treatment requirement that is not needed for LSS and NRS components to provide reasonable assurance that the plant can perform its risk significant functional requirements. It is expected that when a safety-related LSS or NRS EQ-qualified component fails, the replacement commercial component will be of a similar robust design as the component being replaced, and can reasonably be expected to operate under the design environments that are expected.

For safety-related LSS/NRS components that are currently EQ-qualified, these components will not be replaced once the EQ qualified end-of-life is reached. It is expected that these components will continue to perform their function with reasonable assurance. These components will not be replaced unless another circumstance (i.e., failed component, design change, etc) justifies the replacement.

40. *During the meeting on August 31, 1999, the licensee indicated that LSS/NRS equipment would be tested (post-maintenance or modification, and surveillance testing) to some degree to demonstrate the functional capability of the equipment.*

    (a) *Please provide an expanded discussion on this testing and the associated acceptance criteria.*

    (b) *Also, since no critical attributes are designated for NRS equipment in the system bases documents, how would the test acceptance criteria be determined?*

RESPONSE (part a and b):

The basic intent of post-maintenance testing (PMT) following corrective or preventive maintenance activities is to verify that the SSC can perform its intended function, to ensure that the original deficiency has been corrected, and to verify that no new deficiency has been introduced during the performance of the maintenance activity. In addition, post-maintenance testing requirements should be commensurate with the work performed and (based upon existing programs/procedures at the specific site) designed to make the program effective and efficient.

The current STP post-maintenance testing program includes a set of testing matrices, based upon component type and work completed, which detail the recommended testing to be performed. This approach, when utilized as a "standard", invokes consistent PMT requirements for components regardless of classification (excluding specific PMT requirements invoked by Technical Specifications, Industry Codes, Special Engineering testing, commitments, etc.).

The incorporation of risk insights into the post-maintenance/post-modification testing matrices is intended to mirror the tiered or graded maintenance approach of Full, Basic, and Target (see the response to Question 47 for an expanded discussion on this). The risk categorization bases provides the PMT planner with additional data (including component critical attributes) to utilize in evaluating and specifying PMT requirements. In addition, the PMT planner evaluates additional factors in determining specific PMT requirements such as component impact to overall plant/system reliability, economic impact, resource requirements, system health, technical experience, etc. in finalizing the post-maintenance testing scope. Factoring in risk insights into the PMT process permits resources to be focused on those components which are most important.

The grant of the Exemption would remove any mandatory PMT requirements for SSCs which are LSS or NRS. This will allow STP to apply PMT controls commensurate with the safety significance in a "tiered" or "graded" approach combined with additional factors to produce the most effective process.

The procedure governing PMT activities at STP is 0PGP03-ZM-0025. This procedure applies to maintenance requirements ONLY rather than requirements specified in Technical Specifications, Industry Codes and Standards, or Special Engineering testing.

As stated in the subject procedure, the following draft directions have been proposed when determining PMTs for maintenance activities (Note: this procedure is currently undergoing revision, and the revision is not finalized):

- **NON RISK SIGNIFICANT (NRS) components - Post-Maintenance testing is not required. The Planner/Supervisor has the option to prescribe PMT activities as deemed necessary based upon commercial business practices factoring in the significance of the work performed. Acceptance criteria for testing NRS components will be based upon proper component functioning using standard commercial practices.**

- **LOW SAFETY SIGNIFICANT (LSS) components - Post Maintenance Testing activities may be performed to ensure the original deficiency was corrected. If the component's critical attributes were affected, post maintenance testing will be performed to ensure the component can satisfactorily perform its critical functions. Acceptance criteria for testing will be based upon proper functioning of affected critical attributes using standard commercial practices.**

- **HIGH and MEDIUM SAFETY SIGNIFICANT (HSS/MSS) components - Post Maintenance Testing is required to ensure the original deficiency was corrected and to ensure that the component can perform its critical functions. Appropriate acceptance criteria will be selected based upon required design/functional requirements.**

The application of risk insights is not intended to wholesale delete the requirements for PMT for SSCs which are not safety significant. Other factors must also be considered to ensure SSC functionality. Risk insights provide the opportunity to streamline PMTs based upon sound business/maintenance practices, thus making PMTs optional for NRS/LSS components should all other data indicate that this is prudent. The majority of applications when PMTs will not be performed will include lube/inspect, inspect, or test-types of maintenance activities. Validation of proper system operation will generally provide assurance of proper component function. Surveillance testing may also be used, as appropriate, as a PMT for LSS/NRS SSCs.

The Design Change Process procedure (0PGP04-ZE-0309) provides specific requirements for post installation testing of design changes, and includes two specific requirements when determining testing. These requirements state: 1)Testing shall assure that the modification accomplishes the desired intent of the design, and 2)Testing shall assure that the installed modification functions properly with interfacing plant systems. Adequate verification of these two requirements generally demands a more complex approach to post-modification testing than that required for routine maintenance activities. Applying risk insights, which provide reasonable assurance that SSCs affected by design changes can perform their design/functional requirements, is appropriate for post-modification testing.

41. *The July 13, 1999, submittal describes (Attachment 3, pages 5 and 6) that the licensee's procurement requirements would specify environmental parameters that LSS/NRS equipment must withstand. However, during the site visit on October 5, 1999, the licensee indicated that purchase order requirements pertaining to environmental qualification aspects would not be imposed for LSS/NRS equipment. Please clarify the approach that the licensee intends to implement to provide confidence that LSS/NRS components will remain functional if they are exposed to a harsh environment.*

RESPONSE:

Components that are risk categorized as LSS or NRS, by definition, serve little, if any, function in mitigating the consequences of an accident or protecting the health and safety of the public during a design basis event or any other credible event. As LSS or NRS components (which are currently EQ-qualified) require replacement, a commercial replacement component will be procured. Procuring a commercial component is accomplished by performing an engineering evaluation to ensure that the replacement component satisfies the required form, fit, and function (including the ability to operate under specified environmental conditions). The replacement component will be capable of meeting the design functional requirements (including environmental considerations), however, the component will not be environmentally qualified. Additional details and insight on the procurement process of commercial components is provided in the response to Question #13.

By purchasing functionally equivalent replacement components, and by procuring from reputable vendors, reasonable assurance is provided that a quality product (which will meet the various challenges of service operation) is received. Upon receipt, an inspection is performed to validate that the received component satisfies the purchase requisition. If discrepancies are noted during this receipt inspection, the discrepancy will be resolved or the component will be returned to the vendor.

By purchasing functionally equivalent replacement components which meet the environmental considerations of the original component, and by performing a receipt inspection to validate that what was received is what was ordered, reasonable assurance is provided that the component can, and will, perform to its risk significant design functional requirements. Even if the subject component is exposed to a harsh environment, as long as these harsh environmental parameters are within the stated design functional requirements for the component, it is reasonable to expect that LSS and NRS components will be able to satisfactorily perform.

42. *During the staff's October 5 and 6, 1999, site visit to STP, the licensee stated that it sees no difference between the reliability of safety-related and commercial-grade components. Provide your analysis of the data to support the assumed failure probability and reliability of safety-related components categorized as LSS, which have been presumably designed, procured, tested and inspected to commercial standards, for operation of these components under normal operating conditions and under all design-basis conditions.*

RESPONSE:

STPNOC asserts that, for components within the scope of the STPEGS Graded QA Program, non-safety-related component failure rates are not appreciably greater than corresponding safety-related component failure rates for similar component types. To support this assertion, STPNOC has performed a data analysis of Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX) data. Nuclear industry data reporting to the Nuclear Plant Reliability Data System (NPRDS) spans the time period from 1977 through 1996. The EPIX Maintenance Rule and Reliability Information (MRRI) database includes component failure data since 1996. NPRDS component engineering data includes indication of safety class, thus enabling a distinction between safety-related component and non-safety-related component failure rates. While the MRRI database does not include a safety-class distinction, INPO was able to provide STPNOC an MRRI database file for 1997-1999 data that is "back-linked" to NPRDS, thus providing indication of safety class. The NPRDS data and MRRI data were first analyzed separately and then merged to provide a large-scope analysis to support responses for the STPEGS GQA RAIs. STPNOC has developed a report, entitled "Safety-Related Versus Non-Safety-Related Equipment Failure Frequency Data Analysis for Nuclear Power Plants in the United States" dated April 6, 2000, describing this NPRDS-MRRI data analysis. This report is available upon request.

The scope of this merged NPRDS-MRRI analysis included consideration of over 670,000 component records and over 166,000 component failure records for those components. The historical data analyzed consisted of over 74 billion component-hours of experience. GQA RAI 42 Tables 1 and 2 (attached) provide analysis results information for all 33 component type data categories contained in the merged NPRDS-MRRI database. These tables show that the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS-MRRI data. This analysis shows that, of 33 component type categories investigated, 21 had higher safety-related failure frequency values than corresponding non-safety-related categories. Non-safety-related failure frequency values were significantly higher than corresponding safety-related failure frequencies in only one of the 33 categories (the "containment penetration" component type category). The analysis shows that, for most component types, the calculated safety-related failure frequencies are generally greater than or roughly equivalent to those for corresponding types of non-safety-related components, based on historical NPRDS and MRRI data.

An argument often made in this type of comparison is that there is more safety-related component experience in the database than non-safety-related component experience. This is valid. However, the failure frequency parameters, calculated simply in terms of reported failures per component-hour of experience in this analysis, are being compared on a consistent basis. For example, in the circuit breaker component type category, there are 7,723,785,888 component-hours of safety-related circuit breaker experience. During that experience base, 6,457 failures of safety-related circuit breakers were reported, yielding a failure frequency of 8.36E-07 (=6,457/7,723,785,888) failures per component-hour. Similarly, there are 1,777,678,176 component-hours of non-safety-related circuit breaker experience in the database. During that experience base, 1,345 failures of non-safety-related circuit breakers were reported, yielding a failure frequency of 7.57E-07 (=1,345/1,777,678,176) failures per component-hour. The failure frequency parameters are calculated and compared on the same basis. One can conclude that we have a greater degree of confidence that the historical failure frequency for safety-related circuit breakers represents the "true" failure frequency (calculated for infinite experience), than we do for the non-safety-related circuit breakers. However, in this case, there are large numbers of component-hours of experience for both safety-related and non-safety-related components, indicating that we have relatively high confidence in both results.

Another way of looking at this is that, if we were to "scale" the safety-related experience down to the non-safety-related experience level, we would multiply both the component-hours of experience and the reported failure count by the ratio of non-safety-related to safety-related component-hours of experience (1,777,678,176/6,457/7,723,785,888). If we do this, we get the same results as with the actual experience numbers. Likewise, we would get the same results if we were to scale the non-safety-related experience up to the safety-related experience. That is, if we increase or decrease the component-hours of experience for a component type category of interest in the database by some factor, we would expect to have a higher or lower number of reported failures by the same factor.

GQA RAI 42 TABLE 1. SUMMARY OF MERGED NPRDS-MRRI COMPONENT TYPE CATEGORY SAFETY-RELATED VERSUS NON-SAFETY-RELATED FAILURE FREQUENCY COMPARISON RESULTS

COMPONENT DATA CHARACTERISTIC DESCRIPTION	NUMBER IN CATEGORY
TOTAL COMPONENT CATEGORIES ANALYZED:	33
NUMBER OF CATEGORIES WITH SAFETY-RELATED DEMAND FAILURE RATE GREATER THAN NON-SAFETY-RELATED FAILURE FREQUENCY:	21
NUMBER OF CATEGORIES WITH NON-SAFETY-RELATED DEMAND FAILURE RATE GREATER THAN SAFETY-RELATED FAILURE FREQUENCY:	12
CATEGORIES WHERE SAFETY-RELATED DEMAND FAILURE RATE IS MORE THAN A FACTOR OF 2 LESS THAN NON-SAFETY-RELATED FAILURE FREQUENCY:	3
CATEGORIES WHERE SAFETY-RELATED DEMAND FAILURE RATE IS MORE THAN A FACTOR OF 3 LESS THAN NON-SAFETY-RELATED FAILURE FREQUENCY:	1
TOTAL COMPONENT-HOURS OF EXPERIENCE DATA:	74,615,379,120
TOTAL FAILURE EVENT RECORDS ANALYZED:	116,413
TOTAL FUNCTIONAL FAILURES IN RECORD SET:	116,413
SAFETY-RELATED COMPONENT-HOURS OF EXPERIENCE:	60,968,091,504
NON-SAFETY-RELATED COMPONENT-HOURS OF EXPERIENCE:	13,647,287,616
SAFETY-RELATED FUNCTIONAL FAILURES IN RECORD SET:	93,697
NON-SAFETY-RELATED FUNCTIONAL FAILURES IN RECORD SET:	22,716

GQA RAI 42 TABLE 2. MERGED NPRDS-MRRI COMPONENT TYPE CATEGORY DATA ANALYSIS RESULTS

COMPONENT TYPE ID	COMPONENT DESCRIPTION	SAFETY-RELATED COMPONENT -HOURS	SAFETY-RELATED COMPONENT FAILURES	NON-SAFETY-RELATED COMPONENT -HOURS	NON-SAFETY-RELATED COMPONENT FAILURES	SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT -HOUR)	NON-SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT -HOUR)	NON-SAFETY-RELATED > SAFETY-RELATED FREQUENCY	NON-SAFETY-RELATED > 2*SAFETY-RELATED FREQUENCY	NON-SAFETY-RELATED > 3*SAFETY-RELATED FREQUENCY
ACCUMU	Accumulators, tanks, air receivers	320,096,904	286	51,778,080	9	8.93E-07	1.74E-07	NO	NO	NO
AIRDRY	Air dryers, dehumidifiers	20,415,504	149	26,830,248	168	7.30E-06	6.26E-06	NO	NO	NO
ANNUNC	Annunciator modules, alarms	21,289,632	9	50,028,864	4	4.23E-07	8.00E-08	NO	NO	NO
BATTRY	Batteries, battery chargers	188,054,640	1,109	34,188,936	170	5.90E-06	4.97E-06	NO	NO	NO
BLOWER	Blowers, compressors, fans, vacuum pumps, cooling units	327,993,024	1,601	106,903,032	808	4.88E-06	7.56E-06	YES	NO	NO
CKTBRK	Circuit breakers, contactors, controllers	7,723,785,888	6,457	1,777,678,176	1,345	8.36E-07	7.57E-07	NO	NO	NO
CRDRVE	Rod drive mechanism, hydraulic control unit	2,386,497,960	3,049	84,631,656	13	1.28E-06	1.54E-07	NO	NO	NO
DEMIN	Demineralizers, ion exchangers	44,136,024	72	72,290,016	255	1.63E-06	3.53E-06	YES	YES	NO
ELECON	Electrical conductors, bus, cable, wire	47,311,920	229	2,645,688	9	4.84E-06	3.40E-06	NO	NO	NO
ENGINE	Engines (gas, diesel)	42,954,168	1,364	3,009,408	45	3.18E-05	1.50E-05	NO	NO	NO
FILTER	Filters, strainers, screens	194,277,624	492	48,874,176	90	2.53E-06	1.84E-06	NO	NO	NO
GENERA	Generators, inverters, motor generators	155,717,880	1,618	41,882,208	400	1.04E-05	9.55E-06	NO	NO	NO
HEATER	Electric heaters	66,201,648	215	6,761,136	12	3.25E-06	1.77E-06	NO	NO	NO
HTEXCH	Heat exchanger, condenser, steam generator	414,941,280	1,468	356,166,816	1,105	3.54E-06	3.10E-06	NO	NO	NO
IBISSW	Bistable, switch (mechanical, electronic)	4,583,711,328	7,309	1,168,451,712	1,367	1.59E-06	1.17E-06	NO	NO	NO
ICNTRL	Instrument controllers	898,170,120	2,617	754,194,216	2,054	2.91E-06	2.72E-06	NO	NO	NO
INDREC	Indicators, recorders,	1,165,607,472	1,572	467,257,680	452	1.35E-06	9.67E-07	NO	NO	NO

COMPONENT TYPE ID	COMPONENT DESCRIPTION	SAFETY-RELATED COMPONENT -HOURS	SAFETY-RELATED COMPONENT FAILURES	NON-SAFETY-RELATED COMPONENT -HOURS	NON-SAFETY-RELATED COMPONENT FAILURES	SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT -HOUR)	NON-SAFETY-RELATED COMPONENT FAILURE FREQUENCY (FAILURES / COMPONENT -HOUR)	NON-SAFETY-RELATED > SAFETY-RELATED FREQUENCY	NON-SAFETY-RELATED > 2*SAFETY-RELATED FREQUENCY	NON-SAFETY-RELATED > 3*SAFETY-RELATED FREQUENCY
	gauges									
INTCPM	Integrator/computation module	5,147,811,144	6,485	1,254,243,600	1,619	1.26E-06	1.29E-06	YES	NO	NO
IPWSUP	Electronic power supply	2,421,707,832	2,710	307,631,568	421	1.12E-06	1.37E-06	YES	NO	NO
ISODEV	Isolation devices	1,331,855,808	774	158,385,984	96	5.81E-07	6.06E-07	YES	NO	NO
IXMITR	Transmitters, detectors, elements	4,019,348,664	9,775	950,110,272	1,298	2.43E-06	1.37E-06	NO	NO	NO
MECFUN	Governors, couplings, gear boxes	145,165,920	790	64,157,760	346	5.44E-06	5.39E-06	NO	NO	NO
MOTOR	Motors (electric, hydraulic, pneumatic)	894,689,184	1,212	217,592,112	450	1.35E-06	2.07E-06	YES	NO	NO
PENETR	Containment penetrations, air locks, hatches	562,056,384	922	2,977,224	121	1.64E-06	4.06E-05	YES	YES	YES
PIPE	Pipes, fittings, rupture discs	127,431,000	415	22,303,536	104	3.26E-06	4.66E-06	YES	NO	NO
PUMP	Pumps, eductors	745,949,736	4,797	160,325,160	1,136	6.43E-06	7.09E-06	YES	NO	NO
RELAY	Relays	8,447,729,424	2,922	348,630,792	275	3.46E-07	7.89E-07	YES	YES	NO
SUPORT	Supports, hangers, snubbers	899,955,000	908	38,081,304	44	1.01E-06	1.16E-06	YES	NO	NO
TRANSF	Transformers, shunt reactors	259,542,552	161	194,772,312	150	6.20E-07	7.70E-07	YES	NO	NO
TURBIN	Turbines (steam, gas)	28,295,040	363	48,378,888	380	1.28E-05	7.85E-06	NO	NO	NO
VALVE	Valves, dampers	13,192,044,024	20,420	3,375,651,384	4,061	1.55E-06	1.20E-06	NO	NO	NO
VALVOP	Valve operators	4,112,662,464	11,279	1,450,059,720	3,909	2.74E-06	2.70E-06	NO	NO	NO
VESSEL	Pressure vessel, reactor vessel, pressurizer	30,684,312	148	413,952	0					
	TOTAL:	60,968,091,504	93,697	13,647,287,616	22,716			12	3	1

43. *Section 4.1.2 of the licensee's application states that " . . . STP will utilize purchase requirements or other evaluations to ensure the availability of replacement components to function under design conditions, without performing qualification tests." Also, during the site visit on October 5 and 6, 1999, the licensee indicated that non-safety-related components that are categorized as either MSS or HSS will have special treatment applied as necessary to ensure that their critical attributes are satisfied. These critical attributes, as documented in the licensee's system categorization notebooks, were derived from the PRA failure modes but they were not very specific. For example, a system categorization notebook would only indicate that a particular valve should open to provide flow to a particular heat exchanger. The critical attribute did not specify the design-basis conditions under which the flow needs to be provided.*

How will the licensee's process identify and ensure that the each component's specific critical attributes will be satisfied (i.e., for safety-related components categorized as LSS and NRS and non-safety-related components categorized as MSS or HSS) so there will be adequate assurance that these components will be functional under design-basis conditions?

RESPONSE:

For safety-related components that are risk categorized as NRS, it should be noted that these components have no critical attributes. The PRA does not take credit for NRS components functioning during a design basis event; however, it is likely that these components will perform their function during these challenges. Replacement NRS components are procured to be capable of meeting their design functional requirements (including environmental considerations). These components receive a receipt inspection, and are appropriately post-maintenance tested to validate that they satisfy their design functional requirements. This provides reasonable assurance that NRS components can perform their function.

For safety-related LSS components, replacement components will be procured commercial grade. Similar to NRS components, replacement LSS components are procured to be capable of meeting their design functional requirements (including environmental conditions). These functional requirements will envelope the credible design basis conditions that the components can be expected to see. However, these components will not be subjected to qualification testing. Since LSS components have assigned critical attributes, these attributes will be factored into the engineering evaluation that is performed during the procurement process. See the response to Question 13 for additional insight into the procurement process. In addition, the post-maintenance testing for LSS replacement components will focus on ensuring that the critical attributes are properly demonstrated.

While it is true that the critical attributes stated in the Risk Significance Basis Document are not specific in nature (i.e., the stated critical attribute does not specify the design basis conditions under which the attribute must be accomplished), the critical attributes as stated do provide adequate insight to ensure that the component is properly procured, installed, and tested, and the design documents provide the details regarding the attributes. For the example specified in the question, if the critical attribute for a valve was to open to provide flow to a heat exchanger, the safety-related, qualified valve that is currently installed is expected to meet this critical attribute through appropriate design features and testing. If this valve requires replacement and is replaced by a commercial valve, the procured valve will be capable of meeting the design functional requirements (including environmental considerations). The received replacement valve will be similarly robust in design features and function compared to the installed valve. Upon installation, the critical attributes will be validated through post-maintenance testing to provide an appropriate level of assurance that the function can be performed. However, the valve may not be tested at design basis conditions, but it is designed to meet these conditions. When in operation, and in the unlikely event that design basis conditions challenge the valve, there is reasonable assurance that the valve will operate under conditions that it is designed to meet. In any case, the component is not relied on to mitigate the accident or to protect the health and safety of the public. Through this process, reasonable assurance is demonstrated that the valve can satisfy its critical attributes when called upon.

For non-safety related components that are risk categorized as MSS/HSS, similar procurement, installation, and testing processes will be followed as specified for the safety-related LSS components above. Additional quality inspection points may be designated during component receipt, and quality hold points may be designated during installation. During critical installation steps, independent/dual verification points may be used to provide additional confidence. In addition, these components are included in the Maintenance Rule monitoring program. This program evaluates failures and provides solutions to any problems that may occur in the future.

44. *In its July 13, 1999, request, STPNOC states " . . . the change in the special treatment requirements for LSS components is not expected to impact system performance levels, because STP will continue to monitor system performance under the Maintenance Rule program and take appropriate corrective actions as necessary to maintain system performance." The STPNOC request also states that "the effect on equipment availability of reduced or eliminated special treatment requirements will be seen based on future PRA performance data updates and the periodic GQA performance evaluation and feedback process."*

    (a) *The staff judged that licensee's graded quality assurance program would have a minimal impact on the reliability of the equipment, and it was recognized that the operability of equipment under adverse transient conditions would still be ensured by the other special treatment requirements. For the proposed exemption request, you have stated that any widespread and larger deviations in reliability should be detectable through the sophisticated monitoring and feedback procedure. Please describe how section 5.3.11 of the GQA program (which describes the GQA performance feedback loop and considerations for adjusting GQA controls) will be implemented for the proposed exemptions. Provide an explanation as to how your monitoring and feedback procedure will assure that changes in SSC reliability (in excess of those assumed during the safety significance determination process) under adverse conditions will be detected.*

    (b) *Explain how the use of 1) station performance indicators, 2) periodic updates of the PRA with respect to performance data and 3) maintenance rule 50.65(a)(3) periodic evaluations will quantitatively assess the SSC reliability under off-normal operating conditions.*

    (c) *Please describe how the licensee's corrective action program will consider the reestablishment of selected "special treatment requirements" when component performance suggests the need for such controls.*

    (d) *The licensee states that the Maintenance Rule (MR) will be used for monitoring and feedback but also says that LSS components will be removed from the scope of the MR (thereby deleting all component-level feedback). Please provide a description of a component-level monitoring program that feeds information back to the licensee's corrective action program.*

    (e) *From information conveyed during the August and October meetings, the licensee indicated that the corrective action requirements of the MR would continue to apply if LSS or NRS component failures or performance problems result in exceeding the established MR performance measures or criteria for plant/system/train level functions of systems comprised of a mix of HSS, MSS, LSS and NRS components. Please confirm this position. In addition, please explain the process for making repetitive maintenance preventable functional failures (RMPFF) determinations for HSS and MSS equipment. Will these RMPFF determinations consider previous failures of LSS and NRS components where there could be common maintenance*

practices or similar equipment failures? The staff will need to understand how RMPFF determinations integrate relevant information from LSS or NRS equipment problems.

RESPONSE (part a):

The monitoring and feedback process described in the GQA program will ensure the identification and evaluation of changes in component reliability, regardless of the cause. Thus, changes in reliability resulting from the reduction or elimination of special treatment requirements would be captured under the feedback and monitoring process. This would apply to normal and off-normal operating conditions. No reliability data is available or can be gathered under adverse transient conditions, such as a seismic event or a LOCA.

STP's monitoring and feedback process ensures that changes in equipment performance are evaluated for impact on risk significance. Condition reports are initiated to document component failures or performance degradations and the resulting corrective actions. Condition reports are also used to initiate and document the results of Preventive Maintenance activities. For each system whose components have been risk ranked, the associated condition reports are reviewed and evaluated periodically for evidence of negative performance trends. Any such evidence is brought to the attention of the Working Group where it is evaluated for impact on the risk ranking of the associated components. The Working Group, with Expert Panel approval, then adjusts the risk ranking, as appropriate. This feedback loop ensures that any negative performance changes that are attributed to the relaxation of special treatment controls, are addressed by the reinstatement of applicable controls up to and including the re-categorization of the component's risk significance, as appropriate.

STP notes that these components will still be required to meet functional and design requirements. In addition, these components were ranked LSS and NRS specifically by assuming their failure and assessing the associated consequences on the safety of the plant. Therefore, even if it is assumed that the removal or reduction of special treatment requirements such as documentation, inspection, and testing will degrade the performance of the component during adverse transient conditions, the risk significance determination process has already shown that the impact on plant safety is minimal or non-existent.

RESPONSE (part b):

The reliability of SSC's under off-normal conditions is the same as the reliability determined during normal plant conditions. Off-normal conditions (also known as anticipated operational occurrences or anticipated transient conditions) do not impose unique operating conditions for SSC's necessary to mitigate the consequences of anticipated transients and therefore do not affect the reliability of the SSC.

Reliability data for SSC's is obtained from data collected during plant operation to support the Maintenance Rule Program, information collected from the On-line Maintenance Program, review and discussion of system status with System Engineers and review of operating logs. All of these sources are used to provide the information necessary to perform periodic updates of the PRA. The PRA is used to establish performance criteria of SSC reliability under the Maintenance Rule Program 50.65(a)(2). Insights gained from this data are used to quantitatively assess SSC reliability under both normal and off-normal conditions.

RESPONSE (part c):

Degraded component performance is identified to and evaluated by the Working Group, as described previously in the performance monitoring and feedback loop. The main components of this process are the Condition Reporting Process, the Operating Experience program, and the PRA model updates that include component reliability data. The Working Group periodically evaluates component data identified by these programs. If the evaluation shows that component performance has degraded as a result of the removal or reduction of controls, including those identified in special treatment requirements, then the Working Group will recommend that the appropriate controls be re-established, up to and including changing the risk categorization of the component, if necessary. For example, a component previously ranked LSS may be revised to MSS. Such a component would then become subject to additional special treatment requirements, as necessary. A condition report would be initiated to facilitate this change and to evaluate the impact of the change. The impact evaluation would include any activities previously performed under the exemption from special treatment requirements. Timely and appropriate action would then be taken to administratively return the subject component to the special treatment controls.

RESPONSE (part d):

The Maintenance Rule program as implemented by the South Texas Project will continue to provide component-level feedback for HSS and MSS components. LSS and NRS components whose failure affects any Maintenance Rule performance criteria (whether at the train, system or plant level) will be an input to the periodic reviews. There are several methods by which degrading performance of LSS and NRS components affecting the Maintenance Rule is fed back into the risk ranking process. First, the Maintenance Rule

Coordinator participates in the periodic review meetings. The Maintenance Rule Coordinator, along with the System Engineer, will provide input if the negative performance from LSS and NRS components has any significant effect on the health of the system. Second, as stated in the response to question 14, if a Maintenance Rule performance criteria is exceeded due to failures of LSS or NRS components, then the affected system would be evaluated for reclassification to Maintenance Rule category (a)(1). All (a)(1) classifications must be evaluated to determine the cause and develop a plan of corrective actions to prevent recurrence. Third, condition reports generated against the components are reviewed by the Operating Experience Group to identify any adverse component performance trends. The results of this review, which would include condition reports generated as a result of operator rounds and system engineer walkdowns, is then provided to the Working Group during the six-month review process. Therefore, significant adverse component performance is captured in both the Maintenance Rule program and the condition reporting process, and both provide feedback to the Working Group.

RESPONSE (part e):

STP confirms that corrective action requirements of the Maintenance Rule would continue to apply if LSS or NRS component failures or performance problems result in exceeding the established Maintenance Rule performance measures or criteria for plant/system/train level functions of systems comprised of a mix of HSS, MSS, LSS, and NRS components. STP does not intend to explicitly monitor the performance of SSCs that are outside of the scope of the Maintenance Rule (i.e., LSS and NRS SSCs) but we do intend to continue to follow the existing corrective action requirements of the Maintenance Rule for any performance criteria exceedance. Please see the response to previous RAI question 14.b for clarifying details.

STP's current process for identifying repetitive maintenance preventable functional failures (RMPFF) consists of comparing the subject Maintenance Rule Functional Failure (MRFF) with previous similar MRFFs. The comparison focuses on failure modes and failure causes of components that perform identical functions in the same system on both units. We intend to follow this same process for HSS and MSS SSCs. If a Maintenance Rule performance criteria is exceeded, the affected system would be evaluated for reclassification to Maintenance Rule category (a)(1). As is the existing practice, we will not consider previous failures of SSCs that are outside of the scope of the Maintenance Rule as part of the RMPFF determination process.

45. *Please describe how the licensee's overall process considers spatial relationships such as seismic interactions or fires. Describe the evaluations and processes that will provide reasonable assurance that LSS and/or NRS equipment will maintain functionality and conformance with design provisions which should preclude adverse interactions (such as spraying, flooding, seismic interaction, electrical separation, and electrical isolation) with HSS and/or MSS equipment. The staff expects that STP will maintain robust provisions that will preclude these adverse interactions.*

RESPONSE:

The STPEGS process for considering spatial relationships such as seismic interaction or fire is based on sound engineering insight and practices. In general, spatial relationships are considered to be a functional requirement of the SSC, providing the component is expected to survive the spatial event. When considering fire spatial events, relatively few, if any, components at STPEGS are expected to survive the fire events. Likewise, relatively few components are qualified for the high energy line-break / pipe whip impact or pipe jet impingement regardless of their current risk categorization. In addition, relatively few components are currently qualified for flooding, and therefore are not expected to survive the flood event. With this in mind, no LSS/NRS components are given credit for mitigating spatial events, and therefore, these LSS/NRS components are not qualified to survive the event.

In contrast, seismic interaction is a standard which is currently maintained between commercial equipment and equipment subject to special treatment requirements regardless of safety classification. Seismic interaction would be considered a functional requirement and would be maintained. STPEGS would maintain provisions that would preclude adverse interactions between HSS/MSS components and LSS/NRS components where required. Where STPEGS deviates from current methodologies to achieve this functional objective, STPEGS would provide a justification with the new methodology, again based on sound engineering insight and practices.

For example, if a Seismic II/I LSS component fails in normal service and requires replacement, a commercial replacement component that is non-seismically-qualified would be procured. The replacement component would be evaluated to determine if the component size, weight, orientation, and mounting are equivalent to the failed component. The following conclusions could be drawn:

- **If equivalency exists, the component would continue to be mounted in its previously analyzed II/I configuration.**

- **If the replacement component varies in relative size, weight, orientation, or mounting, a further evaluation would occur that factors in spatial interactions. If the only components that the replacement component could credibly interact with during a spatial event are categorized as LSS or NRS, then the evaluation would conclude that spatial interaction has no safety significance.**

- **If the replacement component varies in relative size, weight, orientation, or mounting, and the evaluation concludes that there exists either HSS or MSS components that the replacement component could credibly interact with during a spatial event, then an engineering evaluation would be performed to ensure safe shutdown capability. If the plant cannot be safely shut down with the interaction, a design change package would be created to fully evaluate and ensure that the Seismic II/I separation criteria are satisfied.**

46. (a) *Clarify how systems that are comprised entirely of safety-related LSS and NRS components or systems that are comprised of a mixture of safety-related LSS and NRS and nonsafety-related LSS and NRS components will be treated under the maintenance rule.*

(b) *Provide examples of systems where this situation occurs (i.e. radiation monitoring system, emergency lighting system, plant communication system).*

(c) *How will performance monitoring at the plant/system/train function level against established criteria continue for these systems as stated in the exemption request?*

RESPONSE (part a):

To clarify the focus of the question, STP has no systems that are comprised entirely of safety-related LSS and NRS components. All systems that contain only LSS and NRS components include both safety-related and non-safety-related components.

Systems that are comprised entirely of LSS and NRS components will be outside of the scope of the Maintenance Rule, and the requirements of 10CFR50.65 will not apply to them. See the response to RAI questions 14b and 44 for more details on how we intend to address non-scoped components whose failures affect existing Maintenance Rule performance criteria.

RESPONSE (part b):

Of the 29 systems that have undergone the GQA categorization process so far, the following six systems are composed entirely of LSS and NRS components:

RA – Radiation Monitoring System
PS – Primary Sampling System
WL – Liquid Waste Processing System
DI – Standby Diesel Combustion Air Intake System
DX – Standby Diesel Generator Exhaust System
XG – Diesel Generator Building

It is likely that, when categorized, all of the Emergency Lighting System and the Plant Communication System components would be ranked as LSS or NRS. It is expected that as STP continues with the categorization process, that additional systems will be identified that contain only LSS and NRS components.

RESPONSE (part c):

The performance of LSS and NRS components would not be explicitly monitored under the Maintenance Rule. The performance of these components would primarily be monitored through the Corrective Action Process and through the Working Group periodic reviews. See the responses to RAI questions 14b and 44d for more discussion concerning the monitoring of the performance of LSS and NRS SSCs.

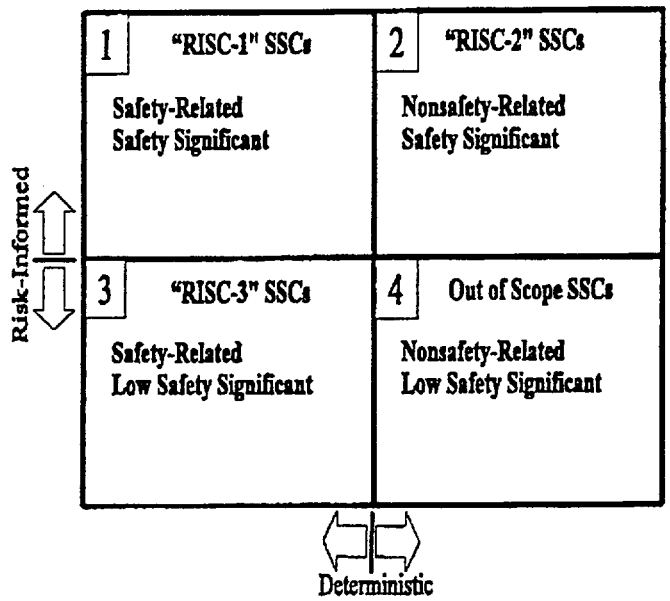
SOUTH TEXAS PROJECT

REQUEST FOR INFORMATION TO SUPPORT INITIATIVES TO INCORPORATE RISK-INFORMED INSIGHTS INTO 10 CFR PART 50 REGULATIONS

47) *In SECY-99-256, "Rulemaking Plan for Risk-Informing Special Treatment Requirements," the NRC staff describes a scheme for categorizing SSCs according to their safety significance and status under the deterministic safety-related regime. This scheme divides SSCs into 4 bins. (See Figure 1.) Risk-informed safety class 1, or RISC-1 SSCs are presently safety-related and are determined to be safety significant by a risk-informed categorization process. RISC-2 SSCs are not presently safety-related, but have been determined to be safety significant by a risk-informed categorization process. RISC-3 SSCs are presently safety-related, but have been found to be of low safety significance by a risk-informed categorization process. Remaining SSCs are expected to be out of the scope of special treatment requirements, though other regulatory controls may still apply.*

In an effort to equate current Risk-informed Rulemaking efforts with your exemption request, please describe how the STP risk categorizations compare to these classifications.

Figure 1: Diagram of Categorization and Treatment



RESPONSE:

The STP NRC-approved risk categorization process also classifies components into four categories as follows:

- **HIGH (HSS),**
- **MEDIUM (MSS),**
- **LOW (LSS), and**
- **NON-RISK SIGNIFICANT (NRS)**

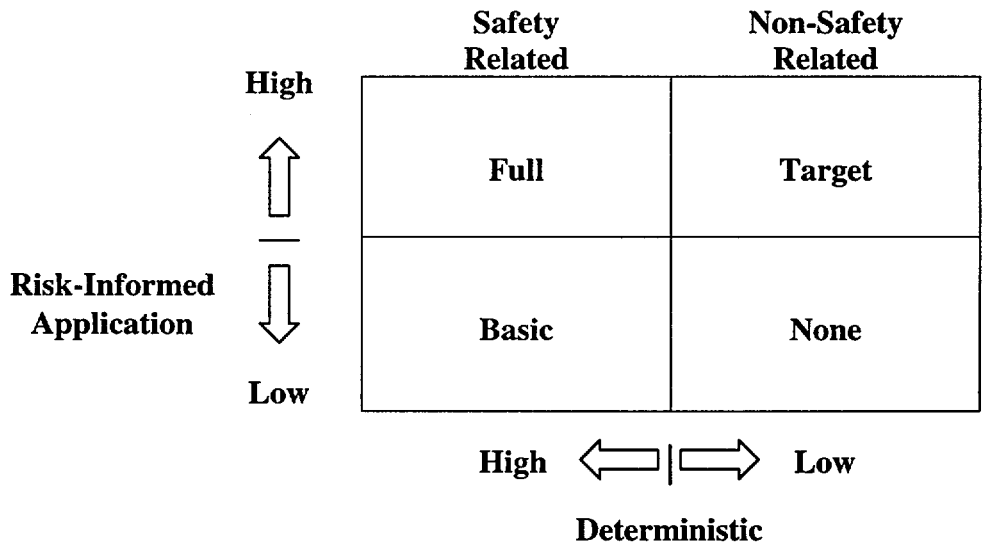
These risk categorizations are procedurally covered in 0PEP02-ZA-0001, Graded Quality Assurance Working Group Process.

As discussed in procedure 0PGP02-ZA-0003, Comprehensive Risk Management, STP further defines programmatic controls of the categorized components as follows:

- **FULL,**
- **BASIC,**
- **TARGET, and**
- **NONE.**

These programmatic controls define the level of both regulatory and administrative treatment that individual components will receive.

Figure 2: STP GQA Programmatic Controls



FULL controls apply to safety significant, safety-related SSCs that have been risk-categorized as HSS or MSS. These components currently receive full regulatory controls and special treatment applications. These full controls will continue once the Exemption request is granted. FULL controls would generally equate to the controls that are recommended for RISC-1 SSCs.

BASIC controls apply to non-safety significant, safety-related components that have been risk-categorized as LSS or NRS. BASIC controls are defined as the minimum required regulatory practices which provide reasonable assurance that these SSCs can satisfy their design/functional requirements. These controls reflect the most economical and efficient means of conducting business on non-safety significant components. Upon grant of the Exemption, LSS and NRS components would be exempt from the regulatory special treatment requirements, however, since LSS components also have critical attributes defined for them, these critical attributes would be factored into the administrative treatment for LSS components. LSS and NRS components would be procured commercial, and the Corrective Action Program (CAP) would still fully apply to these components. In addition, these components would be monitored on a system/train/plant level per the Maintenance Rule (Maintenance Rule monitoring of LSS/NRS SSCs consists of Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function), and the feedback process would evaluate the satisfaction of performance for possible reinstatement of special treatment requirements or possible component recategorization. Comparing these controls to the proposed 'four boxes', BASIC controls would apply to the proposed RISC-3 box (See Figure 3).

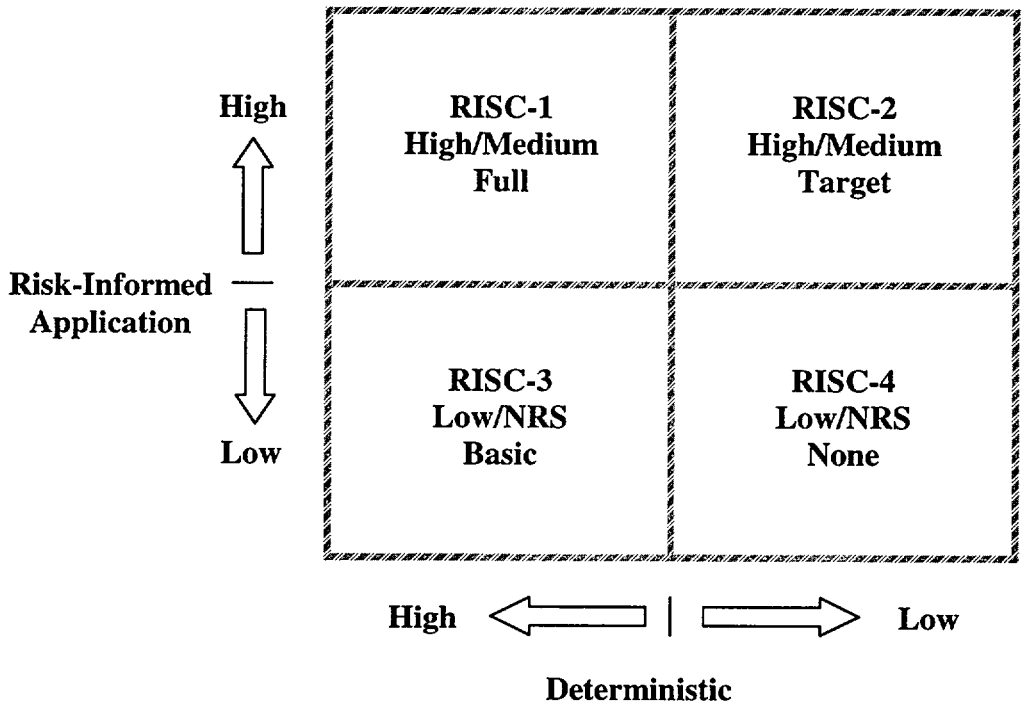
In addition, the proposed RISC-1 box defines these components as 'safety-related, safety significant'. The STP approach would place HSS and MSS safety-related components into the RISC-1 box. The RISC-3 box is defined as 'safety-related, low safety significant'. STP would place LSS and NRS safety-related components into the RISC-3 box.

TARGET controls apply to non-safety-related, safety significant components that have been risk-categorized as HSS or MSS. TARGET controls are subject to specific regulatory special treatment requirements and to additional administrative controls. These controls will be specifically 'targeted' to the critical attributes that resulted in the component being categorized as HSS or MSS. Components under these controls will remain non-safety-related and will be procured commercial, but the special treatments will be appropriately applied to give additional assurance that the component will be able to perform its function when demanded. TARGET controls directly equate to the RISC-2 proposed box of 'non-safety-related, safety significant'.

Currently, STP specifies non-safety-related, non-safety-significant, augmented quality SSCs (seismic II/I, fire protection, radwaste, post-accident monitoring) as being under a Targeted program. The attributes which makes these SSCs important from a quality perspective currently receive enhanced treatment controls. However, upon grant of the exemption, these non-safety-significant SSCs will be removed from the scope of augmented quality control. These SSCs will have standard commercial treatment applied to them which provide reasonable assurance that these non-safety-significant SSCs can satisfy all of their functional requirements. These SSCs would be non-safety-related, non-safety-significant and would equate to the proposed RISC-4 box.

NO regulatory controls are currently applied to the non-safety significant, non-safety-related LSS and NRS categorized components, and upon grant of the Exemption, no regulatory controls would be added. Components in this category would still receive appropriate administrative controls to give reasonable assurance that these components will perform their design function efficiently and reliably. The components (whose additional controls are NONE) equate directly to the proposed RISC-4 box, 'non-safety-related, low safety significant'.

Figure 3: Comparison of the Proposed STP Programmatic Controls to NRC Proposed Categorization and Treatment



ATTACHMENT 5

ASSURANCE OF FUNCTIONALITY

OF LSS AND NRS COMPONENTS

ASSURANCE OF FUNCTIONALITY OF LSS AND NRS COMPONENTS

1.0 Introduction

South Texas Project (STP) has requested an exemption from the special treatment requirements in NRC's regulations for components that have been categorized as low safety significant (LSS) or non-risk significant (NRS). In lieu of the special treatment requirements, STP is proposing to provide normal commercial treatment for such components. However, STP is not proposing to make any changes in the design or design functions of the LSS and NRS components.

The NRC staff has questioned whether commercial treatment will provide reasonable assurance that the LSS and NRS components will be able to perform their safety functions. In particular, NRC staff has questioned whether there is a need for measures that specifically verify that LSS and NRS components can perform their safety functions under design basis conditions.

As discussed below:

- Commercial treatment will provide reasonable assurance that LSS and NRS components can perform their safety functions under design basis conditions.

- Even if it were assumed that commercial treatment would result in a substantial decrease in the reliability of LSS and NRS components, there would be no significant impact on safety.

- It would be inconsistent with the intent of the Commission to require additional measures to verify that LSS and NRS can perform their safety functions under design basis conditions.

2.0 Commercial Treatment Provides Reasonable Assurance of Functionality

A description of STP's commercial treatment is contained in a separate document provided to the NRC, entitled *Elements of the South Texas Project Commercial Treatment for LSS and NRS SSCs*. These elements include:

- Controls over Procurement and Installation - STP will continue to use the LSS and NRS components that are currently installed, and will replace them only as the need arises. Since the existing LSS and NRS components were procured and installed using the special treatment programs, there is reasonable assurance that these components can perform their safety functions under design basis conditions. The procurement of replacement LSS and NRS components will be accomplished through an engineering evaluation. STP will provide an appropriate level of assurance that the form, fit, and function (including capability to withstand environmental conditions) of the replacement component is equivalent to that of the original component. Receipt inspections will verify that the component received is the component ordered, and post-modification/maintenance tests will be performed as appropriate to verify the function of the component.

- Maintenance - LSS and NRS components will be removed from service, if required, in logical groupings approximately four times per year to support corrective, preventive, and predictive maintenance activities. Preventive maintenance will include routine maintenance checks, inspections, replacements, tests, adjustments, calibrations, and post-maintenance testing, as appropriate. Predictive maintenance activities will typically include periodic lube oil analyses on large motors and pumps, vibration analyses of rotating equipment, thermographic analyses of mechanical and electrical components to identify improper temperature conditions or electrical hot-spots, acoustic analysis for valve leak-by or component leakage, and motor potential diagnostic testing, as appropriate.

- Monitoring and Corrective Action - Under STP's Maintenance Rule Program, LSS and NRS safety-related components will be monitored, as appropriate, at the plant/system/train level. Monitoring LSS/NRS SSCs consists of tracking Maintenance Rule Functional Failures (MRFFs) whose failure results in loss of an HSS/MSS function. In addition, LSS and NRS components will be monitored by routine Operator rounds, System Engineer walkdowns and resulting System Health Reports, GQA Working Group periodic system reviews, and Corrective Action Program performance thresholds. When deficiencies are found in LSS and NRS components, the deficiencies will be subject to the existing Corrective Action Program, which satisfies the requirements in Appendix B to CFR Part 50. Furthermore, if a component fails to satisfy performance expectations, the commercial controls will be evaluated and adjusted as necessary to provide appropriate treatment to the component.

Available data indicate that commercial treatment programs are effective in ensuring the functions of compoents. STP has performed an analysis of Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX) data. The scope of this analysis includes over 670,000 component records, 166,000 component failure records, and 74 billion component-hours of experience. This analysis shows that, of the 33 component categories investigated, 21 had higher safety-related failure frequencies than non-safety-related failure frequencies. Non-safety-related failure frequency values were significantly higher than corresponding safety-related failure frequencies in only one of the 33 categories. In addition, STP has collected data on active equipment necessary to support power production at STP (e.g., feedwater and condensate pumps). The collected data indicate no apparent difference in the failure rates for normally operating motors between safety and non-safety related equipment. Based upon this data, STP concludes that the changes in the special treatment requirements for LSS and NRS components will not significantly impact their failure rates.

In summary, the commercial treatment provisions include 1) controls to provide an appropriate level of assurance that LSS and NRS components will be procured and installed such that they are capable of performing their design basis functions, 2) maintenance activities to provide an appropriate level of assurance that LSS and NRS components will continue to satisfy their design basis functions, and 3) monitoring of LSS and NRS components to provide an appropriate level of assurance that deficiencies in the performance of LSS and NRS components are promptly corrected. Additionally, industry and STP data indicate that the failure rates for components subject to commercial treatment are not significantly different than the failure rates for components subject to special treatment. Therefore, STP concludes that commercial treatment

provides reasonable, risk-commensurate assurance that LSS and NRS components will be able to perform their safety functions under design basis conditions.

3.0 A Substantial Decrease in Reliability of LSS and NRS Components Would Not Significantly Impact the Safety of the Plant

As discussed above, industry and STP data indicate that the failure rates for components subject to commercial treatment are not significantly different than the failure rates for components subject to special treatment. Nevertheless, even if it were unrealistically assumed that the change from special treatment to commercial treatment would result in a substantial decrease in the reliability of LSS and NRS components, there would be no significant impact on risk.

NRS components are not sufficiently significant to warrant modeling in the PRA. Furthermore, the vast majority (approximately 90%) of the safety-related LSS components were so categorized solely for deterministic reasons; i.e., they were not sufficiently significant to warrant modeling in the PRA. Therefore, the exemption to exclude these components from the scope of the special treatment requirements will not affect risk levels as determined by STP's probabilistic risk assessment (PRA).

STP performed a sensitivity study to determine the impact on core damage frequency (CDF) and large early release frequency (LERF) from postulating a factor of 10 increase in the failure rates of all PRA-modeled LSS components. In all cases, increasing the failure rates of these LSS components by a factor of 10 resulted in a failure rate that was greater than the 95th percentile for each of the LSS component failure rate distributions. The cumulative impact to the annual average CDF and LERF of the increased failure rates for all PRA-modeled LSS components categorized to date is shown below:

	Current Average (events/yr)	**Sensitivity Study λ_{LSS}*10 (events/reactor year)**	**Increase**	**Increase**
CDF	9.0781E-6	9.3232E-6	.4510 E-7	2.7%
LERF	1.3742E-7	1.3911E-7	.6900 E-9	1.2%

The above increases in CDF and LERF are within the acceptance guidelines for changes as outlined in Regulatory Guide 1.174 (i.e., 1E-6 delta CDF and 1E-7 delta LERF). Based upon the above, STP concludes that, even if it is unrealistically assumed that the change in the special treatment requirements were to result in significant degradation of NRS and LSS components, there would be little or no increase in risk.

4.0 It Would Be Inconsistent with the Intent of the Commission to Require Additional Treatment for LSS and NRS Components

As discussed above, commercial treatment provides reasonable, risk-commensurate assurance that LSS and NRS components will be able to perform their safety functions under design basis conditions. It would be inconsistent with the intent of the Commission to require additional treatment or evaluations to provide further assurance of the functionality of LSS and NRS components.

The STP exemption request is a prototype for risk-informing the special treatment requirements in the Commission's regulations. Specifically, STP is a prototype for rulemaking Option 2 in SECY-98-300, as approved by the Commission. With respect to Option 2, SECY-98-300 states:

> Under this option, SSCs of low safety significance (from a risk-informed assessment) would move from "special treatment" to normal industrial (sometimes called "commercial") treatment, but would remain in the plant and be expected to perform their design function but without additional margin, assurance, or documentation associated with high safety significant SSCs.

Similar statements are provided in the Advanced Notice of Proposed Rulemaking entitled *Risk-Informing Special Treatment Requirements* (65 Fed. Reg. 11488, March 3, 2000). Thus, a requirement for additional treatment for low safety significant components, above and beyond that provided by commercial treatment, would be inconsistent with Option 2.

Furthermore, the current special treatment requirements in NRC's regulations are the cumulative result of prescriptive, deterministic regulations promulgated over the years. Imposing a requirement for additional treatment, beyond that provided by commercial treatment, for low safety significant components would be inconsistent with NRC's intent to move toward risk-informed, performance-based regulation. In particular, for low safety significant components, licensees should have flexibility to identify the treatment to be afforded to such components (i.e., commercial treatment), subject to monitoring to provide an appropriate level of assurance that the treatment is effective in maintaining an acceptable level of risk. As is stated in SECY-98-300:

> Since changes to [special treatment] requirements in the revised regulations would apply to those SSCs of low risk importance, it is anticipated that such an approach could be accomplished with no significant safety impact. However, as part of this process, the staff would have to ensure that the licensee had appropriate assessment and feedback programs in place to reflect SSC performance degradation back into the PRA and to modify SSC risk importance as necessary.

It would be inconsistent with this principle to replace the current special treatment requirements with a different set of treatment requirements for low safety significant components. In essence, the NRC would only be exchanging one set of prescriptive requirements for another set of prescriptive requirements, without truly making its regulations risk-informed and performance-

based. Furthermore, the cost of identifying additional treatment for each low safety significant function would be prohibitive and would outweigh any benefit from the STP exemption (or to Option 2 in general). Therefore, imposing a requirement for additional treatment, beyond STP's normal commercial practices, is not a viable option.

5.0 Conclusions

STP's commercial treatment includes provisions for control of procurement, maintenance, and monitoring and corrective action. These provisions provide reasonable assurance that LSS and NRS components will be able to perform their safety functions under design basis conditions. This conclusion is supported by industry and STP data, which indicate that commercial treatment is effective in ensuring the function of components. Furthermore, STP's sensitivity studies demonstrate that, even if it is unrealistically assumed that the change in the special treatment requirements were to result in significant degradation of NRS and LSS components, there would be little or no increase in risk. Finally, it would be inconsistent with the intent of the Commission and the concept of risk-informed performance-based regulation to require STP to provide additional treatment, beyond its normal commercial practices, for LSS and NRS components. Therefore, STP's commercial treatment program provides an acceptable basis for ensuring the safety functions of LSS and NRS components, and the NRC should not impose a requirement for any additional treatment.

ATTACHMENT 6

LOCATION OF STP'S COMMITMENTS

RELATED TO THE EXEMPTION REQUEST

LOCATION OF STP'S COMMITMENTS RELATED TO THE EXEMPTION REQUEST

The NRC staff has raised a question regarding the proper location for STP's commitments related to its exemption request to exclude low safety significant (LSS) and non-risk significant (NRS) components from the scope of the special treatment requirements in NRC's regulations. For the reasons discussed below, STP believes that the proper location is the STP Updated Final Safety Analysis Report (UFSAR).

- The UFSAR Already Encompasses Similar Information – The UFSAR already includes a discussion of topics similar to those addressed in the exemption request, including the conduct of operations, quality assurance (QA), testing and inspections, maintenance, and qualification.

- The Commitments Are Suitable for Inclusion in the UFSAR – Under 10 CFR 50.34(b)(6), UFSARs are required to include information on organization and authorities of personnel, QA, maintenance, surveillance, testing, and the conduct of operations. This information encompasses the type of information that will be the subject of STP's commitment related to the exemption request. STP considers the commitments related to the exemption request to be integral to the information required to be in the UFSAR under Section 50.34(b)(6). Accordingly, STP would not be able to remove those commitments from the UFSAR under NEI 98-03, *Guidelines for Updating Final Safety Analysis Reports*, except in accordance with the change control process described below.

- The UFSAR Is a Controlled Document – The UFSAR is a controlled document and must be periodically updated under 10 CFR 50.71(e). This will ensure that STP keeps the NRC informed of changes in the commitments related to the exemption. Additionally, by placing these commitments in the UFSAR, the commitments will have high visibility to STP personnel, the NRC, and the public.

- The UFSAR Is Subject to an Appropriate Change Control Process – Under 10 CFR 50.59, a licensee must prepare written evaluations of changes in its UFSAR and must seek prior NRC approval of changes that meet the criteria in Section 50.59. These criteria will ensure that NRC approval will be obtained prior to making any change in STP's commitments that could have more than a minor impact on safety. The NRC staff has questioned whether the criteria in Section 50.59 may be too flexible as applied to STP's commitments related to the exemption. STP believes that the criteria in Section 50.59 are suitable, and that STP should be allowed to make changes in its commitments that would have little or no impact on safety per the criteria in Section 50.59. However, to address the concerns of the NRC staff, STP has provided supplemental change control criteria for its commitments (refer to Attachment 1, Section 3.4), which provides STP with governing controls, yet allows sufficient flexibility to make changes in the details of its processes without prior NRC approval.

There are several other possible licensing basis documents in which STP's commitments related to the exemption could be located, including the Operations Quality Assurance Plan (OQAP), the license, and a separate licensing basis document. STP believes that each of these locations has drawbacks. In particular:

- OQAP – It would be inappropriate to place the commitments in the OQAP because the scope of the exemption request is broader than the scope of the OQAP. Furthermore, changes in the OQAP are subject to the stringent controls in 10 CFR 50.54(a), which requires prior NRC approval for changes that involve a reduction in commitments. Since the STP exemption request is a prototype, and since it may be expected that there will be numerous minor changes in STP's processes as experience is gained, the restrictions in Section 50.54(a) would be unduly burdensome to both STP and the NRC if applied to STP's commitments related to the exemption request.

- License Condition - It would be inappropriate to make the commitments into a license condition. The license should be reserved for matters that have the most significance to safety, such as the technical specifications. STP's commitments related to the exemption do not rise to that level of importance, and the license should not be cluttered with such detailed information. Moreover, the license cannot be changed with prior NRC approval, and it would be unduly burdensome to both STP and the NRC to require a license amendment for every change in STP's commitments related to the exemption request.

- Separate Licensing Basis Document – From STP's perspective, it would be acceptable to place STP's commitments in a separate licensing basis document, and to control changes to that document using the commitment control process in NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, which was accepted by the NRC in SECY-00-0045. However, NEI 99-04 provides licensees with substantial flexibility to make changes in commitments. STP understands that the NRC staff desires that a more stringent change control process be applied to STP's commitments related to the exemption request.

Therefore, for the reasons discussed above, STP believes that the UFSAR is the most appropriate location for its commitments related to the exemption request.

ATTACHMENT 7

STPEGS UFSAR

STPEGS UFSAR

13.7 RISK-INFORMED SPECIAL TREATMENT REQUIREMENTS

13.7.1 Introduction

NRC regulations in 10 CFR Parts 21, 50, and 100 contain special treatment requirements that impose controls to ensure the quality of components that are safety-related, important to safety, or otherwise come within the scope of the regulations. These special treatment requirements go beyond normal commercial and industrial practices, and include quality assurance (QA) requirements, qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. STP has been granted an exemption from the special treatment requirements. Table 13.7-1 identifies the regulations from which an exemption was granted and the scope of the exemption. This exemption only pertains to special treatment requirements; it does not change the design and functional requirements for components.

STP has a risk-informed process for categorizing the safety/risk significance of components. This process is described in Section 13.7.2. Components with no or low safety significance have been exempted from the scope of most of the NRC regulations that impose special treatment requirements, and instead are subject to normal industrial and commercial practices. Additionally, components with medium or high safety significance are evaluated for enhanced treatment. Components retain their original regulatory requirements unless they have been recategorized using the process described below. The treatment for the various categories of components is described in Section 13.7.3. As part of this process, STP also performs continuing evaluations and assessments, which are described in Section 13.7.4. Finally, STP applies quality assurance to this process, and controls changes to the process, as described in Section 13.7.5.

13.7.2 Component Categorization Process

13.7.2.1 Overview of Categorization Process. The process utilized by STP in categorizing components consists of the following major tasks:

1. Identification of functions performed by the subject plant system.
2. Determination of the risk significance of each system function.
3. Identification of the system function(s) supported by that component.
4. Determination of a risk categorization of the component based on probabilistic risk assessment (PRA) insights (where the component is modeled)
5. Development of a risk categorization of the component based on deterministic insights.
6. Designation of the overall categorization of the component, based upon the higher of the PRA categorization and the deterministic categorization.
7. Identification of critical attributes for components determined to be safety/risk significant.

The processes for determining the risk categorization and deterministic categorization of a component are described in more detail in Sections 13.7.2.3 and 13.7.2.4.

Based upon these processes, a component is placed into one of four categories: 1) high safety/risk significant (HSS), 2) medium safety/risk significant (MSS), 3) low safety/risk significant (LSS), and 4) non-risk significant (NRS). This categorization process does not, in and of itself, affect the other classifications of the component (e.g., safety, seismic, ASME classification).

The process is implemented by a Working Group comprised of individuals experienced in various facets of nuclear plant operation and reviewed by an Expert Panel. This integrated decision process is described in more detail in Section 13.7.2.2.

13.7.2.2 Comprehensive Risk Management Process. The integrated decision-making process used by STP is documented by procedure. The integrated decision-making process incorporates the use of an Expert Panel and Working Groups. The Expert Panel is comprised of qualified senior level individuals and is responsible for oversight of the program and for reviewing the activities and recommendations of the Working Group. The Working Group is comprised of experience individuals who apply risk insights and experience to categorize components in accordance with the process described in this Section and make recommendations to the Expert Panel.

Procedures control the composition of and processes used by the Expert Panel and Working Group. Procedures also identify training requirements for members of the Expert Panel and Working Group, including training on probabilistic risk assessment, risk ranking, and the graded quality assurance process. Finally, the procedures specify the requirements for a quorum of the Expert Panel and Working Group, meeting frequencies, the decision-making process for determining the categorization of components, the process for resolving differing opinions among the Expert Panel and Working Group, and periodic reviews of the appropriateness of the programmatic control and oversight provided to categorized components.

13.7.2.3 PRA Risk Categorization Process. A component's risk categorization is initially based upon its impact on the results of the PRA.

STP's PRA calculates both a core damage frequency (CDF) and a large early release frequency (LERF). The PRA models internal initiating events at full power, and also accounts for the risk associated with external events.

The PRA configuration control program incorporates a feedback process to update the PRA Model. The updates are segregated into two categories:

- **The plant operating update incorporates plant design changes and procedure changes that affect PRA-modeled components, initiating event frequency updates, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a period not to exceed 36 months.**

- **The comprehensive data update incorporates changes to plant-specific failure rate distributions and human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a period not to exceed 60 months.**

The PRA model may be updated on a more frequent basis if an update would result in a significant increase in the CDF.

Only components that are modeled in the PRA are given an initial risk categorization. The PRA risk categorization of a component is based upon its Fussell-Vessely (FV) importance, which is the fraction of the CDF and LERF to which failure of the component contributes, and its risk achievement worth (RAW), which is the factor by which the CDF and LERF would increase if it were assumed that the component is guaranteed to fail. Specifically, PRA risk categorization is based upon the following:

PRA Ranking	Criteria
High	RAW ≥ 100.0 or FV ≥ 0.1 or FV ≥ 0.005 and RAW ≥ 2.0
Medium (Further Evaluation is Required)	FV < 0.005 and 100.0 > RAW ≥ 10.0
Medium	FV ≥ 0.005 and RAW < 2.0 or FV < 0.005 and 10.0 > RAW ≥ 2.0
Low	FV < 0.005 and RAW < 2.0

13.7.2.4 <u>Deterministic Categorization Process</u>. Components are subject to a deterministic categorization process, regardless of whether they are also subject to the risk categorization process using PRA insights. This deterministic categorization process can result in an increase, but not a decrease (from the PRA risk), in a component's categorization.

A component's deterministic categorization is directly attributable to the importance of the system function supported by the component. In cases, where a component supports more than one system function, the component is classified based on the highest safety classification of the function supported. In categorizing the functions of a system, the Working Group considers five critical questions regarding the function, each of which is given a different weight. These questions and their weight are as follows:

QUESTION	WEIGHT
Is the function used to mitigate accidents or transients?	5
Is the function specifically called out in the emergency operating procedures (EOPs) or Emergency Response Procedures (ERPs)?	5
Does the loss of the function directly fail another risk-significant system?	4
Is the loss of the function safety significant for shutdown or mode changes?	3
Does the loss of the function, in and of itself, directly cause an initiating event?	3

Based on the impact on safety if the function is unavailable and the frequency of loss of the function, each of the five questions is given a numerical answer ranging from 0 to 5. This grading scale is as follows:

"0" - Negative response

"1" - Positive response having an insignificant impact and/or occurring very rarely

"2" - Positive response having a minor impact and/or occurring infrequently

"3" - Positive response having a low impact and/or occurring occasionally

"4" - Positive response having a medium impact and/or occurring regularly

"5" - Positive response having a high impact and/or occurring frequently

The definitions for the terms used in this grading scale are as follows:

Frequency Definitions –

- **Occurring Frequently – continuously or always demanded**
- **Occurring Regularly – demanded > 5 times per year**
- **Occurring Occasionally – demanded 1-2 times per cycle**
- **Occurring Infrequently – demanded < once per cycle**
- **Occurring Very Rarely – demanded once per lifetime**

Impact Definitions –

- **High Impact – a system function is lost which likely could result in core damage and/or may have a negative impact on the health and safety of the public**

- **Medium Impact – a system function is lost which may, but is not likely to, result in core damage and/or is unlikely to have a negative impact on the health and safety of the public**

- **Low Impact – a system function is significantly degraded, but no core damage and/or negative impact on the health and safety of the public is expected**

- **Minor Impact – a system function has been moderately degraded, but no core damage or negative impact on the health and safety of the public**

- **Insignificant Impact – a system function has been challenged, but no core damage or negative impact on the health and safety of the public**

Although some of these definitions are quantitative, both of these sets of definitions are applied based on the collective judgment and experience of the Working Group.

The numerical values, after weighting, are summed; the maximum possible value is 100. Based on the sum, functions are categorized as follows:

SCORE RANGE	CATEGORY
0 – 20	NRS
21 – 40	LSS
41 – 70	MSS
71 – 100	HSS

A function with a low categorization due to a low sum can receive a higher risk classification if any one of their five questions received a high numerical answer. Specifically, a weighted score of 25 on any one question results in an HSS categorization; a weighted score of 15-20 on any one question results in a minimum categorization of MSS; and a weighted score of 9-12 on any one question results in a minimum categorization of LSS. This is done to ensure that a component with a significant risk in one area does not have that risk masked because of its low risk in other areas.

- **In general, a component is given the same categorization as the system function that the component supports. However, a component may be ranked lower than the associated system function.**

13.7.2.5 Defense in Depth and Safety Margins. For the following reasons, the exemption and the categorization process maintain defense in depth and sufficient safety margins:

- **Functional requirements and the design configuration of systems are retained.**
- **No existing plant barriers are removed or altered.**
- **Design provisions for redundancy, diversity, and independence are maintained.**
- **The plant's response to transients or other initiators is not affected.**
- **Preventive or mitigative capability of components is preserved.**
- **There is no change in any of the safety analyses in the UFSAR.**
- **Existing safety-related LSS and NRS components will not be replaced, absent good cause (e.g., obsolescence or failure). Since the existing safety-related LSS and NRS components were designed, procured, manufactured, and installed in accordance with the existing special treatment requirements, these components have inherent design margins to perform their intended functions that will not be adversely affected by this exemption.**
- **Normal commercial and industrial practices provide an appropriate and acceptable level of assurance that safety-related LSS and NRS components will be able to perform their intended functions.**
- **The corrective action program is applied to safety-related LSS and NRS components. This program provides reasonable assurance that deficiencies involving safety-related LSS and NRS components will be identified, corrected, and necessary action taken to ensure acceptable performance levels are maintained.**

13.7.3 Treatment for Component Categories

13.7.3.1 Description of Treatment for Component Categories. The following treatment is provided for the various component categories:

- **Safety-Related HSS and MSS Components – These components continue to receive the treatment required by NRC regulations and STP's associated implementing programs. Additionally, as discussed in Section 13.7.3.2, STP evaluates the risk-significant functions performed by these components to determine whether there are any functions that are not being treated under STP's current programs, and provides enhanced treatment for such functions.**

- **Non-Safety-Related HSS and MSS Components – These components will continue to receive any existing special treatment required by NRC regulations and STP's implementing programs. Additionally, the risk-significant functions of these components will receive consideration for enhanced treatment. This consideration is described in Section 13.7.3.2.**

- **Safety-Related LSS and NRS Components – These components receive normal commercial and industrial practices. These practices are described in Section 13.7.3.3.**

- **Non-Safety-Related LSS and NRS Components – The treatment of these components is not subject to regulatory control.**

- **Uncategorized Components – Until a component is categorized, it continues to receive the treatment required by NRC regulations and STP's associated implementing programs, as applicable.**

13.7.3.2 Enhanced Treatment for Non-Safety Related Components. Non-safety-related HSS and MSS components may perform risk-significant functions that are not addressed by STP's current treatment programs.

Examples of process enhancements for non-safety-related HSS and MSS components may include:

- **Performing routine preventive maintenance (PM) tasks more frequently to ensure component reliability**
- **Ensuring that the component's critical attributes are functionally validated following maintenance activities**
- **Ensuring that replacement part controls are enhanced for the components (e.g., TARGETED components receive engineering evaluation for item equivalency replacements).**
- **Increasing the quality oversight of work activities and work documentation (e.g., audits, surveillances, and monitoring activities are performed for TARGETED components).**

These identified processes provide reasonable assurance that HSS and MSS components will be able to perform their safety significant functions.

13.7.3.3 Normal Commercial and Industrial Practices for Safety-Related LSS and NRS Components

A description of STP's commercial practices is provided below.

Plant activities are conducted in a manner that is consistent with a conservative, technically-sound approach to plant operations which properly values nuclear safety and public protection. Goals and performance criteria are established commensurate with the importance to safety and plant reliability of the service, SSC, process or function.

13.7.3.3.3 Design Control Process. The Station's Design Control Program is used for both safety related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications which are in the Configuration Management Program. The Design Control Program complies with 10 CFR Part 50, Appendix B and is described in the Operations Quality Assurance Plan (OQAP).

13.7.3.3.4 Procurement Process. Station procurement activities, including the identification, procurement, and receipt of replacement components or parts, are

performed in accordance with approved procedures and processes. These processes include requirements for appropriate material and performance specifications, inspection, and special handling.

13.7.3.3.5 Installation Process. The Station's component and part installation activities are accomplished through the Design Change Programs and/or through the Maintenance processes. Components procured under the commercial treatment program may be installed in plant systems that are governed by special treatment requirements, including but not limited to ASME Code, Class 1E, and seismic requirements. In such instances, proceduralized measures are taken to ensure that the most limiting requirements are applied at the interface between the commercial component and the rest of the system. Appropriate post-maintenance testing is performed.

13.7.3.3.6 Maintenance Process. The Station's maintenance process activities include corrective, preventive, and predictive maintenance. These maintenance activities are performed in accordance with approved procedures, processes, and guidelines.

13.7.3.3.7 Inspection, Test, and Surveillance Process. The Station's inspection and test process is primarily addressed and implemented through the Maintenance process. As stated above, the Maintenance process addresses inspections and tests through corrective, preventive, and predictive maintenance activities.

The Station's surveillance process is governed by approved test procedures to ensure that Technical Specification requirements are met and satisfied. These Technical Specification requirements apply to some safety-related and non-safety-related SSCs, including some components that are categorized as LSS. These surveillances will continue to be performed.

In addition to the surveillances required by Technical Specifications, additional monitoring is performed to validate the acceptable performance of SSCs. Examples of programs used to monitor SSC performance include Maintenance Rule, Operator Rounds, System Engineer reviews, and management assessments.

13.7.3.3.8 Corrective Action Program. The Station's Corrective Action Program is used for both safety-related (LSS and NRS as well as HSS and MSS SSCs) and non-safety-related applications. The Corrective Action Program complies with 10 CFR Part 50 Appendix B, and is described in the OQAP.

13.7.3.3.9 Management and Oversight Process. The Station's management and oversight process is accomplished through approved procedures and guidelines. This process includes independent oversight, line self-assessments, and Maintenance Rule implementation (system or train level for LSS and NRS). In addition, the Graded Quality Assurance Working Group periodically assesses SSC performance.

In addition, management of LSS and NRS components subject to commercial practices is also governed by technical and administrative procedures as described throughout Section 13.7.3.3.

13.7.3.3.10 Configuration Control Process. The Station's configuration control process is controlled through approved procedures and policies. The design control process ensures that the configuration of the Station is properly reflected in design documents and drawings. Changes to the Station are controlled through design change packages (modifications) which require that control drawings and documents be updated prior to closeout of the modification package.

In addition, configuration control addresses the status of components day-to-day in the field. SSCs are tagged and are manipulated by qualified Operations personnel per procedure. The configuration control process manages and controls the physical changes (procedural and equipment) to the facility to assure that the plant configuration and practices correctly reflect the licensing bases. Non-ASME components installed in ASME Code systems are identified and tracked.

13.7.4 Continuing Evaluations and Assessments

13.7.4.1 Performance Monitoring. STP has performance monitoring processes for the changes in the special treatment. This monitoring includes the following:

- **Maintenance Rule Program – Specific performance criteria are identified at the plant, system, or train level. Regardless of their risk categorization, components that affect MSS or HSS functions will be monitored and assessed in accordance with plant, system and/or train performance criteria. Data used for monitoring is obtained from various sources, such as work orders, condition reports, and test results.**

- **Performance Reporting & Identification Database – This database collects both positive and negative indicators from the performance of plant activities, such as corrective maintenance, installation of modifications, and conduct of testing. The Quality organization provides oversight of this database. Corrective Action Program - Condition reports document degraded equipment performance or conditions, including conditions identified as a result of operator rounds, system engineer walk-downs, and corrective/preventive maintenance activities.**

13.7.4.2 Feedback and Corrective Action. STP has feedback and corrective action processes to ensure that equipment performance changes are evaluated for impact on the component risk categorization, the application of special treatment, and other corrective actions. At least once per cycle, performance data is compiled and presented to the Working Group for review, which is performed for each risk-categorized system. Performance and reliability data are generally obtained from sources such as the Maintenance Rule Program and Operating Experience Review.

This process provides an appropriate level of assurances that any significant negative performance changes that are attributed to the relaxation of special treatment controls are addressed timely by the reinstatement of applicable controls up to and including the re-categorization of the component's risk significance, as appropriate.

13.7.4.3 Process for Assessing Aggregate Changes in Plant Risk. The Expert Panel is responsible for assessing and approving the aggregate effect on plant risk for risk-informed applications.

The process used to access the aggregate change in plant risk associated with changes in special treatment for components is based on periodic updates to the station's PRA and the associated PRA risk ranking sensitivity studies.

13.7.4.4 Periodic Assessments. STP's risk-informed programs and processes undergo periodic assessments as determined by the Expert Panel. These assessments are performed in accordance with station procedures, and may include Quality audits, external audits, and self-assessments. The results of the assessments, along with any efficiencies or recommendations identified, are addressed using the Corrective Action Program.

13.7.5 Quality Assurance and Change Control for the Risk-Informed Process

13.7.5.1 Quality Assurance for the PRA and Categorization Process. The provisions of the Operations Quality Assurance Plan (OQAP), Chapter 15.0, Quality Oversight Activities, govern the oversight of the risk categorization process. The program implemented by Chapter 15 provides for independent oversight activities (including audits, assessments, evaluations, performance monitoring, and surveillances) to ensure that the requirements of the OQAP are being properly implemented.

STP has a PRA Configuration and Control program, which is structured to ensure that changes in plant design and equipment performance are reflected in the PRA as appropriate. The PRA Configuration and Control process is administered by procedures and guidelines that ensure proper control of changes to the models. Changes are reviewed by persons independent from the person making the change and are approved by the PRA supervisor.

13.7.5.2 Internal Process for Controlling Changes. Upon approval, the implementation documents (e.g., procedures, Risk Significance Basis Documents, Master Equipment Databases) can not be changed without approval and oversight of the Working Group and the Expert Panel. Processes such as the assessment process, periodic feedback process, and the continuous feedback process provide input for consideration in making changes. The Working Group may recommend appropriate changes and forward these recommendations to the Expert Panel for approval.

Procedures delineate the responsibilities for implementing changes in the implementation documents. Condition reports are generated, as appropriate, to document and track changes. The Expert Panel is designated to maintain cognizance over the implementation

of the Comprehensive Risk Management program and adjusting program criteria as appropriate.

13.7.5.3 Regulatory Process for Controlling Changes. Changes affecting Section 13.7 will be controlled in accordance with 10 CFR 50.59, as applicable. In addition, the proposed changes in Section 13.7 must satisfy the following criteria:

- **STP may make changes in the categorization process described in Section 13.7.2, without prior NRC approval, provided that the revised process:**

  - **has both a PRA risk categorization process and a deterministic categorization process;**
  - **designates the category of a component based upon the higher of the PRA risk categorization and deterministic categorization;**
  - **provides a risk categorization of a component as LSS or NRS only if the component has little or no impact on CDF or LERF (using commonly accepted assessment techniques such as FV and RAW); and**
  - **provides a deterministic categorization of a component as LSS or NRS only if the component has no more than a minor impact on the ability of the plant to prevent or mitigate accidents.**

- **STP may make changes in the treatment described in Section 13.7.3, without prior NRC approval, provided that the revised treatment provides reasonable assurance that components:**

  - **can perform their HSS and MSS functions, and**
  - **can perform their safety-related functions under design basis conditions.**

- **STP may make changes in the evaluations and assessments described in Section 13.7.4, without prior NRC approval, provided that the revised evaluations and assessments include the following elements:**

  - **Processes to monitor the effect of changes in special treatment on the reliability/availability of safety-significant and safety-related functions;**
  - **Processes to ensure that significant changes in reliability/availability of safety-significant and safety-related functions are evaluated for impact on component risk categorization, application of special treatment, and corrective action; and**
  - **Periodic assessments to update the PRA and to reassess the risk categorization of components.**

TABLE 13.7-1

EXEMPTIONS FROM SPECIAL TREATMENT REQUIREMENTS

Regulation	Scope of Exemption
10 CFR 21.3 – An exemption to exclude safety-related LSS and NRS components from the scope of the definition of "basic component."	The procurement, dedication, and reporting requirements in Part 21 are not applied to safety-related LSS and NRS components.
10 CFR 50.34(b)(6)(ii) - An exemption to the extent that it incorporates provisions from 10 CFR Part 50, Appendix B.	Refer to request for exemption from Appendix B.
10 CFR 50.34(b)(11) – An exemption to the extent that it incorporates seismic qualification requirements in Part 100.	Refer to request for exemption from Part 100.
10 CFR 50.49(b) – An exemption to exclude LSS and NRS components from the scope of electric equipment important to safety for the purposes of environmental qualification of electrical equipment.	• The qualification documentation and files specified in Section 50.49 are not applicable to LSS and NRS components. • LSS and NRS components are not required to be maintained in a qualified condition under Section 50.49. • LSS and NRS components may be replaced with components that are not qualified under Section 50.49. • LSS and NRS components, as applicable under Section 50.49, are designed to function in the installed environment. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.
10 CFR 50.54(a)(3) – An exemption from the requirement to seek prior NRC approval for reductions in the commitments in the QA program description involving LSS and NRS components.	STP does not need prior NRC approval for reductions in commitments in the QA program description related to LSS and NRS components.
10 CFR 50.55a(f) and (g) – An exemption from the requirements of ASME Section XI, for repair and replacement of safety-related LSS and NRS components, subject to ensuring that the material specifications and the design	Safety-related LSS and NRS components may be repaired or replaced with components that meet the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks), subject to ensuring that the material

Regulation	Scope of Exemption
(design loadings, design methodology, and stress allowables) are consistent with the original requirements.	specifications and the design are consistent with the original requirements and conducting post-installation pressure tests in accordance with Section XI of the ASME Code. Section 13.7.3.3 identifies the quality, design and procurement controls that are applied to safety-related LSS and NRS components that are repaired or replaced.
10 CFR 50.55a(f) – An exemption from meeting the requirements of ASME Section XI for testing of safety related LSS and NRS components.	Safety-related LSS and NRS components are not in the scope of component-specific inservice testing requirements. System-level testing requirements continue to be applied. Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.55a(g) – An exemption from meeting the requirements of ASME Section XI for inspection of safety related LSS and NRS components.	Safety-related LSS and NRS components are not in the scope of component-specific inservice inspection requirements. Section 13.7.3.3 identifies controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.55a(h) – An exemption to exclude safety related LSS and NRS components from the scope of components required to meet sections 4.3 and 4.4 of IEEE 279.	Sections 4.3 and 4.4 of IEEE 279 do not apply to safety related LSS and NRS components. The other requirements listed in IEEE 279, including functional and design requirements, are applicable. Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR 50.59(a)(1), (a)(2) and (b)(1) (pre-1999 version); 10 CFR 50.59(c)(1), (c)(2), and (d)(1) (2000 version) – An exemption from the requirement to perform a written evaluation of changes in special treatment requirements for LSS and NRS components. Also an exemption from the requirement to seek prior NRC approval for such changes to the extent that they fall within the listed criteria in 50.59.	STP is not required to perform 50.59 evaluations for changes in the special treatment requirements for LSS and NRS components, and is not required to seek prior NRC approval for those changes.
10 CFR 50.65(b) – An exemption to exclude LSS and NRS components from the scope of SSCs covered by the Maintenance Rule.	• STP is required to monitor performance on a plant/system/train level, as applicable. As applicable, STP evaluates failures of LSS and NRS components to determine whether such failures affect MSS or HSS function(s) which then constitute a maintenance rule functional failure at the applicable plant/system/train level.
10 CFR Part 50 Appendix A, GDC	• LSS and NRS components are not required to satisfy

Regulation	Scope of Exemption
1 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 1.	the QA requirements in GDC 1 (except to the extent that they incorporate Criteria III, XV, and XVI of Appendix B to Part 50). • Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR Part 50, Appendix A, GDC 2 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 2, to the extent that GDC 2 requires tests and inspections to demonstrate that SSCs are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.	• LSS and NRS components are not required to be maintained in a qualified condition under GDC 2. • LSS and NRS components may be replaced with components that are not qualified under GDC 2. • LSS and NRS components, as applicable under GDC 2, are designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.
10 CFR Part 50, Appendix A, GDC 4 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 4, to the extent that GDC 4 requires inspection and testing to demonstrate that SSCs are able to withstand environmental effects. GDC 4 requirements related to dynamic effects would not be exempted.	• LSS and NRS components are not required to be maintained in a qualified condition under GDC 4. • LSS and NRS components may be replaced with components that are not qualified under GDC 4. • LSS and NRS components, as applicable under GDC 4, are designed to withstand applicable environmental effects. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement. • GDC 4 requirements with respect to dynamic effects are applicable.
10 CFR Part 50, Appendix A, GDC 18 – An exemption to exclude LSS and NRS components from the scope of SSCs important to safety under GDC 18, to the extent that GDC 18 requires that inspections and testing be performed for individual features, such as wiring, insulation, connections, switchboards, relays, switches, and buses.	• Individual LSS and NRS components within the electrical power systems are not required to be inspected or tested. • System functional tests are still required in accordance with GDC 18. • Additionally, Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related LSS and NRS components.
10 CFR Part 50 Appendix B, Introduction – An exemption to exclude safety-related LSS and NRS components from the scope of safety-related SSCs covered by Appendix B (except for Criterion	• Safety-related LSS and NRS components are not required to satisfy the QA requirements in Appendix B, except for design control, control of nonconformances, and corrective action. • Section 13.7.3.3 identifies other controls that are applied to ensure the functionality of safety-related

Regulation	Scope of Exemption
III pertaining to Design Control and Criteria XV and XVI governing non-conformances and corrective actions).	LSS and NRS components.
10CFR Part 50, Appendix J, B.III – An exemption to exclude safety-related LSS and NRS components, subject to the additional limitations list under Scope of Exemption, from the scope of components requiring local leak rate tests and containment isolation valve leak rate tests.	• Local leak rate tests of LSS containment isolation valves and other safety-related LSS or NRS components are not required. With respect to LSS containment isolation valves, this exemption only applies to valves that satisfy one or more of the following criteria: - The valve is not required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers). - The valve is normally closed and in a physically closed, water-filled system (e.g., containment isolation valves in the Demineralized Water system) - The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Main Feedwater system). - The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is non-nuclear safety (i.e., the valve itself would have been classified as non-nuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve. - The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release). • Cumulative limits for containment leakage are based upon the tested components, with the assumption that the exempted components contribute zero leakage. • Section 13.7.3.3 identifies controls that are applied to ensure the functionality of safety-related LSS and NRS components.

Regulation	Scope of Exemption
10 CFR Part 100, Appendix A.VI(a)(1) and (2) – An exemption to exclude safety-related LSS and NRS components from the scope of SSCs covered by these sections, to the extent that these sections require testing and inspection to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.	• **LSS and NRS components are not required to be maintained in a qualified condition under Part 100.** • **LSS and NRS components may be replaced with components that are not qualified under Part 100.** • **LSS and NRS components, as applicable under Part 100, are designed to withstand the effects of seismic events without loss of capability to perform their safety function. Section 13.7.3.3 identifies the design and procurement controls that are applied to LSS and NRS components to achieve this requirement.**

ATTACHMENT 8

ELEMENTS OF THE SOUTH TEXAS PROJECT COMMERCIAL TREATMENT FOR SAFETY-RELATED LSS AND NRS SSCs

Elements of the South Texas Project
Commercial Treatment for Safety-Related LSS and NRS SSCs

Introduction

The South Texas Project (STP) does not have a specific procedure or program labeled as a "Commercial Treatment Program". Rather, the commercial treatment elements are contained in numerous plant programs and procedures (most of which are applicable to both safety-related and non-safety related equipment and processes). These programs and procedures have been successfully implemented on the non-nuclear Balance of Plant portion of the Station and have been effective in maintaining its availability and reliability. In addition, commercial treatment has been proven to provide reasonable assurance that the Balance of Plant structures, systems, and components (SSCs) can satisfy their design functional requirements. These same commercial treatment elements have historically been used to provide an appropriate level of assurance that high safety significant (HSS) and medium safety significant (MSS) non-safety related SSCs can satisfy their design functional requirements. Since these commercial treatment elements have successfully ensured the design functional requirements of HSS and MSS non-safety related SSCs, these commercial treatment practices will be adequate to provide an appropriate level of assurance that low safety significant (LSS) and non-risk significant (NRS) safety-related SSCs can satisfy their design functional requirements.

The paper entitled *Assurance of Functionality of LSS and NRS Components* provides further support for the conclusion that commercial treatment will provide reasonable assurance that safety-related LSS and NRS SSCs will be able to perform their design functional requirements.

The following summarizes the Commercial Treatment elements currently in place at the South Texas Project, as they will be implemented on LSS and NRS safety-related SSCs when the exemption request is granted. The first two sections address personnel requirements under the commercial treatment program.

Note: The description of STP's commercial treatment practices lists, by number and title, examples of relevant procedures. The listing of such procedures should not be construed as incorporating those procedures by reference.

Accountability, Responsibility, and Organization

The general authorities, responsibilities, and accountabilities for STP personnel engaged in power plant activities (both safety-related and non-safety related) are defined and described in approved Station procedures and policies. Examples of the subjects addressed by these procedures and policies include:

- Conduct of Operations (OQAP 3.0)
- Professionalism and Standards of Performance (NGP0110)
- Goals and Objectives Program (NGP0114)
- Qualification of Plant Staff Personnel (0PGP03-ZA-0065)
- Radiation Protection (0PGP03-ZR-0050)
- General Employee Training (0PGP03-ZT-0133)
- Indoctrination and Training of Licensing Personnel (0PLP01-ZA-0003)

Organizational responsibilities are established to assure the accomplishment of corporate, site, and performance goals. To support Station goals and objectives, technical and administrative procedures are established and implemented. Procedural guidance is provided to evaluate whether the performance of a particular activity requires a procedure. Factors that are considered in determining whether a procedure is warranted include the complexity of the work and its impact on the SSC safety functions. Procedural controls include the provision to allow personnel to stop the process, and through the corrective action program, to change the controls or processes to ensure safety functions are satisfied. The process for the development, review, and approval of new procedures, procedure revisions, procedure changes and procedure deletions as well as the performance of periodic procedure reviews is also established and controlled.

Plant activities are conducted in a manner that is consistent with a conservative, technically-sound approach to plant operations which properly values nuclear safety and public protection. Goals and performance criteria are established commensurate with the importance to safety and plant reliability of the service, structure, system, component, process or function. Examples of the subjects addressed by policies and procedures which govern goals and performance criteria include:

- Goals and Objectives Program (NGP0114)
- System Performance Monitoring (0PGP03-ZE-0038)
- Performance Indication Tracking Guide (0PGP05-ZV-0013)
- Quality Surveillance/Performance Monitoring (0PQP02-ZA-0003)
- Tests/Evaluations Requiring Additional Controls (0PGP03-ZA-0506)

Personnel are responsible for assuring that goals and performance criteria are met, and are responsible for recommending changes that support the accomplishment of company and site goals and performance criteria.

Performance Expectations for Personnel

Performance expectations and standards for personnel are specified in approved procedures, policies, and guidelines. These expectations and standards are satisfied through the use of performance criteria. Examples of the subjects covered by these procedures, policies, and guidelines include:

- Performing/Verifying Station Activities (0PGP03-ZA-0010)
- Minor Maintenance Guideline (MG-0002)
- Design Change Implementation (0PGP04-ZE-0312)
- Work Direction (0PGP03-ZA-0113)

Personnel are responsible for performing their activities consistent with these expectations, as well with established company standards. It is the responsibility of personnel to propose recommendations (when improvements are recognized) and to refine work processes to achieve expectations and resolve deficiencies.

These next sections address the specific treatment aspects of the Commercial Treatment program.

1. Design Control Process

The Station's Design Control Program is used for both safety related (LSS and NRS as well as HSS and MSS SSCs) and non-safety related applications which are in the Configuration Management Program. The Design Control Program complies with 10CFR50, Appendix B and is used for maintaining design inputs and functional requirements, for repair, replacement, or modification of pressure-retaining capability of ASME systems, and for the accomplishment of design activities. For example, LSS/NRS ASME Class 2 and 3 substitutions will be performed using design control procedures and comply with Quality Group D codes and standards per RG 1.26. The Design Control Program conforms to Regulatory Guide 1.64 (Quality Assurance Requirements for the Design of Nuclear Power Plants) and ANSI N45.2.11-1974 (Quality Assurance Requirements for the Design of Nuclear Power Plants). The STP Design Control Program is described in the Operations Quality Assurance Plan (OQAP), which has been reviewed and accepted by the NRC. Examples of subjects addressed in the procedures that implement the Design Control Program include:

- Design Change Implementation (0PGP04-ZE-0312)
- Plant Modifications (0PGP04-ZE-0310)
- Design Change Package (0PGP04-ZE-0309)
- Control of Configuration Changes (0PGP03-ZM-0021)
- Design & Modification Control (OQAP 6.0)
- 10CFR50.59 Evaluations (0PGP05-ZA-0002)

Upon grant of the exemption, STP may replace LSS and NRS ASME components with non-ASME components by ensuring that the material specifications and the design (design loadings, design methodology, and stress allowables) are consistent with the original requirements. The replacement component remains safety-related and 10CFR50 Appendix B design controls still apply.

For LSS and NRS ASME items that are replaced with non-ASME items, STP will manufacture or procure, install, and perform examinations of the replacement items in accordance with the technical requirements of a nationally recognized non-nuclear Code, Standard or Specification suitable for that item (e.g., B31.1 series for piping, B16.34 for valves, API 620 for 0-15 psi atmospheric storage tanks or API 650 for other tanks). However, consistency with the original material specification and design will be maintained by using the original ASME Section II material or equivalent standard material (e.g., ASTM), applying the original ASME Section III Design Specification loadings and load combinations, using the ASME Section III calculation methodology and allowables. (Note: The use of ASME Section III allowables is justified because it is consistent with the ASME Section III design methodology. Also, the use of identical or equivalent material and original design methodology will, in general, eliminate the need for reanalysis provided the wall thickness exceeds original design minimum wall requirement). It is not the intent of this request to require manufacturing and quality requirements of ASME Section III such as impact testing and NDE. These are considered permissible reductions in special treatment requirements consistent with the low risk classification. In addition, STP will conduct post-installation pressure tests in accordance with ASME Section XI. Other additional requirements of ASME Section III, including Subsection NCA, General Requirements, would not apply.

As an alternative to the above preferred approach, STP would have the option to design, manufacture and test a replacement component in accordance with the technical requirements of the applicable nationally recognized non-nuclear Code, Standard or specification, including design methodology and allowables, except that the design loadings of the original design specification will apply. In this case a recalculation may be required.

For example, the replacement of an LSS or NRS ASME valve would consist of the following steps:

1. Initiate a design change package to implement the replacement of a code valve with a non-code valve.
2. Identify the proposed ANSI B16.34 valve
3. Ensure and document that the material specifications of the replacement valve are consistent with the requirements of ASME Section II or equivalent standard material (e.g., ASTM).
4. Ensure that the wall thickness exceeds the original design minimum wall thickness. (Note: this step, along with step 3 above, will ensure that the stresses are within allowables).
5. Prepare a Work Package that identifies installation requirements required by ANSI B31.1 such as welding, NDE, etc.

6. Install the valve and conduct post-installation testing in accordance with ASME Section XI and the requirements of the work package.
7. Maintain associated records for the life of the plant.

These measures will be sufficient to provide reasonable assurance, commensurate with the low risk significance, that the technical requirements of the Code will be satisfied for LSS and NRS replacement items, and that the replacement items will remain functional under design basis conditions.

1. Procurement Process

Station procurement activities, including the identification, procurement, and receipt of replacement components or parts, are performed in accordance with approved procedures and processes. These processes include the following:

- Identification – The need for replacement components, parts, or material is identified and documented through either the Corrective Action Program (CAP) or through the Procurement and Inventory Control System (PICS). CAP is used to identify needed replacement parts when deficiencies are noted or when improvements are planned, and PICS will identify the need to procure replacement parts when inventory levels fall below specified criteria.

- Procurement Evaluation – A procurement request form is prepared and evaluated to ensure that technical requirements and quality requirements have been adequately described and specified. This evaluation provides an appropriate level of assurances that the original design inputs and assumptions for the SSC have been properly factored in. Detailed procurement information, catalog identifications, and specifications, as applicable, are documented in a formal Purchase Order to the supplier. This documented information supports the procurement of the desired replacement component or part.

- Engineering Evaluation - Whenever conditions exist such that the proposed replacement components or parts deviate from the design specifications (technical requirements) of the original item, an engineering evaluation is performed to assess the design adequacy of the proposed replacement. Such an evaluation compares the form, fit, and function of the replacement component to the original component. The comparison of form, fit, and function includes the capability of the replacement component to perform its function under design basis conditions, including applicable seismic and environmental conditions. This evaluation is documented and performed in accordance with either the Item Equivalency Evaluation procedure, the Commercial Dedication Process, or the Condition Report Engineering Evaluation (CREE). These processes are consistent with the guidance provided by EPRI report NP-6406, 'Technical Evaluation of Replacement Items Guideline'. Alternatively, a design change may be developed (in accordance with the Design Control process described above) to validate the acceptability of the replacement item. Procurement specifications are developed to delineate specific technical or quality requirements when a detailed, formal Purchase Order is developed.

- Purchase Order – Procurement requirements are specified on a Purchase Order which includes design specifications, quality requirements, conformance with nationally recognized consensus standards (if required by the design specifications), and shipping requirements. Substitutions are not allowed unless written authorization is granted by Engineering. In addition, catalog identifications may be placed on a Purchase Order in support of a replacement part procurement.

- Receipt Inspection – At the time of receipt, the received item is inspected by warehouse personnel to ensure that the item was not damaged in the process of shipping, and that the item received is the item ordered. In addition, when the item is delivered to the field for use, a similar inspection is performed by Maintenance personnel.

- Special Handling & Storage – The procurement program provides for the identification and implementation of special handling and storage requirements (if required) to ensure that the item is not damaged or degraded during shipment to the site or during storage on site. These handling and storage requirements consider available recommendations from the vendor. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

Examples of the subjects addressed in approved procedures which govern the procurement process includes the following:

- Procurement of Materials (0PGP03-ZP-0011)
- Replacement Item Equivalency Evaluation (0PGP03-ZE-0072)
- Purchase Order/Control Management (0PGP03-ZP-0013)
- Receiving Material (0PNP01-ZP-0026)
- Marking, Handling, Storage, Maintenance of Materials (0PNP01-ZP-0033)

2. Installation Process

The Station's component and part installation activities are accomplished through the Design Change Programs and/or through the Maintenance processes. Components procured under the commercial treatment program may be installed in plant systems that are governed by special treatment requirements, including but not limited to ASME code, Class 1E, and seismic requirements. In such instances, proceduralized measures are taken to ensure that the most limiting requirements are applied at the interface between the commercial component and the rest of the system. Examples of such requirements include design, installation, welding, non-destructive examination, inspection, and testing requirements.

Upon completion of installation activities, a post-maintenance test (PMT) (or other tests as specified by the Work Planner, Work Supervisor, System Engineer, or Plant Operations) is performed, if the installation could affect an SSCs design function. Post-maintenance testing is performed to provide an appropriate level of assurance that the SSC is performing within expected parameters and component functionality is verified prior to being returned to service. The post-maintenance testing may necessitate that the SSC be placed in service to validate the acceptance of its performance. Post-maintenance testing is not necessarily performed under design basis conditions. Examples of post-maintenance attributes which may be checked include:

- no visible leaks of process fluid
- smooth and expected operation of stroked valves
- expected fluid pressure and flow from pumps
- rotating equipment operating with no abnormal noises
- expected circuit continuity and indication
- expected vibration responses on rotating equipment

The component is only returned to service after Operations accepts the results of the maintenance activities and/or the post-maintenance or operability tests. If expected performance is not achieved, the work order document remains open to continue replacement/repair/rework until the activity is satisfactorily completed.

Examples of subjects addressed by approved procedures and policies used to support the installation process include:

- Work Process (0PGP03-ZA-0090)
- Post Maintenance Testing Program (0PGP03-ZM-0025)
- Minor Maintenance Guidelines (MG002)
- Tool Pouch Maintenance (MG001)
- Condition Reporting Process (0PGP03-ZX-0002)
- Design Change Implementation (0PGP04-ZE-0312)
- Control of Configuration Changes (0PGP03-ZA-0021)

3. Maintenance Process

The Station's maintenance process activities include corrective, preventive, and predictive maintenance. These maintenance activities are performed in accordance with approved procedures, processes, and guidelines. Each of these types of maintenance activities is discussed below:

Corrective Maintenance

- Identification – An SSC deficiency is identified, documented, and tracked through the Corrective Action Program (i.e., a condition report is written). All personnel on site are encouraged to generate condition reports as soon as a deficiency is identified.

- Evaluation - The deficiency is evaluated by a senior reactor operator (SRO)-qualified individual to determine the appropriate corrective maintenance to be performed. Troubleshooting actions may be required if the cause of the deficiency cannot be readily ascertained. Engineering support may also be required to identify the cause of the deficiency. The evaluation includes a determination of the level of planning, work instructions, and documentation needed to perform the work activity. Examples of factors that are considered include the complexity of the work, impact on SSC critical attributes, potential impact on operating equipment, and testing requirements.

- Scheduling –Based on the results of the above evaluation, the deficiency is determined to be acceptable as-is, or corrective maintenance activities are scheduled. This schedule is based upon a consideration of restraints such as material, engineering, plant conditions, craft resources, and the relative impact of the deficiency upon plant safety.

- Planning –Based on the results of the above evaluations, the approach to work package development is determined.

    - If the work activity is straightforward and within the scope of the skill of the craft, no planned package is necessary and the work activity does not require scheduling. Maintenance personnel will coordinate work performance with Operations personnel. Only electronic documentation of the work completed is required. This approach is known as Tool Pouch Maintenance.

    - If the work activity is straightforward, however, procedure support is needed to aid in work performance and documentation, then a streamlined planned package is developed by either the work planner or work supervisor. The work activity is normally scheduled, however, scheduling is not required. If scheduling is not required, then Maintenance personnel will coordinate work performance with Operations personnel. This approach is known as Minor Maintenance.

    - If the work activity is complex, a detailed planned work package is developed. This package describes the necessary work instructions, measuring and test equipment requirements (M&TE), initiation of necessary permits, and identification of any hold points, post-maintenance testing requirements, and documentation requirements. The work activity is scheduled to coordinate work performance with Operations personnel. This approach is known as a detailed, planned package.

- Implementation – Maintenance work is performed as planned to correct the deficient condition.

- Closure – Maintenance actions taken to correct the deficient condition, as well as the results of any required post-maintenance tests, are documented. Hard-copy documentation is archived as required per procedure and is available for retrieval. Input into the Corrective Action Program, Maintenance Rule Program, and component history is provided as applicable.

Preventive Maintenance (PM)

- Identification – Preventive maintenance is an action taken to maintain a piece of equipment within design operating condition and to extend its life, and is performed for the purpose of preventing equipment failure. Preventive maintenance tasks and work packages are developed for active structures, systems, or components factoring in vendor recommendations and sound technical insights. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

- Evaluation –Engineering and Maintenance personnel perform an evaluation to determine an appropriate periodic and/or condition-based PM for the specific component. This evaluation includes a consideration of vendor recommendations, industry experience, site experience, service conditions, safety significance, INPO and EPRI guidelines, commitments, NRC and industry notices, Qualification Maintenance Database, codes, standards, and technical insight.

- Scheduling – Preventive Maintenance tasks are scheduled based upon a consideration of factors such as the ability to perform the PM activity on-line, as low as reasonably achievable (ALARA) dose considerations, and minimization of individual equipment outages. In addition, PMs may be utilized to accomplish certain corrective maintenance activities based upon plant requirements and efficiency of performance.

- Planning – Preventive Maintenance work instructions are developed in accordance with the PM Planners Guide. Work instructions are written to provide an appropriate level of assurance that the PM can be performed by qualified craftsmen. Work instructions include necessary permits, PMT requirements, M&TE requirements, radiation work permit (RWP) requirements, and documentation requirements.

- Implementation – The Preventive Maintenance task is performed to maintain equipment reliability. If a deficiency is noted during PM task performance which is not within the scope or intent of the PM, the deficiency may be corrected by either making a "one-time change" to the PM instructions, or if conditions do not permit immediate correction, issuing a condition report. Both methods are administratively controlled by approved procedures. Post maintenance testing, if required by the work instructions, is completed and the equipment is released to Operations for return to service.

- Closure – Preventive Maintenance actions as well as post maintenance test results are documented. The completed package is turned-over to Operations for review and acceptance. The equipment is returned to service after Operations accepts the results of the task activity and the post maintenance testing. If required by the work package, an operability test and/or other desired testing is performed. The documentation is archived as required by procedure and is available for retrieval. Inputs into the Corrective Action Program, the Maintenance Rule Program and the component history are provided, as applicable.

Predictive Maintenance

Predictive maintenance activities are intended to anticipate, monitor, and/or preclude degradation mechanisms which, if left unattended, would result in component failure. The frequency and scope of these maintenance actions are established and documented based on various considerations such as vendor recommendations, environmental operating conditions, safety significance, and operating performance history. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

Examples of predictive maintenance activities include:

- periodic lube oil analyses on large motors and pumps
- vibration analyses of rotating equipment
- thermographic analysis of both mechanical and electrical SSCs to identify improper temperature conditions or electrical hot-spots
- motor potential diagnostic testing

Deficiencies that are discovered during the performance of predictive maintenance activities are either immediately corrected or identified for correction under the Corrective Action Program.

Post maintenance testing, if required by the work instructions, is performed to provide an appropriate level of assurance that the SSC is performing within expected parameters prior to being returned to service.

Examples of the subjects addressed by approved procedures and guidelines used to support the corrective, preventive, and predictive maintenance process includes:

- Work Process (0PGP03-ZA-0090)
- PM Program (0PGP03-ZM-0002)
- Work Direction (0PGP03-ZA-0113)
- Minor Maintenance Guidelines (MG001)
- Tool Pouch Maintenance Guidelines (MG002)
- Post Maintenance Testing (0PGP03-ZM-0025)
- Troubleshoot/Rework Process (0PMP01-ZA-0041)

4. Inspection, Test, and Surveillance Process

The Station's inspection and test process is primarily addressed and implemented through the Maintenance process. As stated above, the Maintenance process addresses inspections and tests through corrective, preventive, and predictive maintenance activities. These activities factor in vendor recommendations into the selected approach. STP may deviate from vendor recommendations based on specific circumstances and sound business practices. Such deviations are not required to be documented.

The Station's surveillance process is governed by approved test procedures to ensure that Technical Specification requirements are met and satisfied. These Technical Specification requirements apply to some safety-related and non-safety related SSCs, including some components that are categorized as LSS. These surveillances will continue to be performed in the same manner as they are currently. However, if future Technical Specification relief is granted, appropriate surveillances will be modified as needed.

In addition to the surveillances required by Technical Specifications, additional monitoring is performed to validate the acceptable performance of SSCs. Examples of programs used to monitor SSC performance include the following.

- Maintenance Rule - Monitoring will be performed at the system, train, or plant level to ensure SSC reliability levels are maintained. LSS and NRS component performance is not explicitly monitored, but degrading performance in these components, if significant, would be observed at the system or train level. Failures of LSS or NRS components that result in the system's performance criteria being exceeded are identified and resolved under a corrective plan of action, which includes actions to prevent recurrence.

- Operator Rounds – Routine inspections and observations of plant systems and equipment are performed on a per shift basis by both licensed and non-licensed plant operators.

- System Engineer Reviews – The operation of plant systems is monitored by the cognizant system engineer through physical walkdowns and reviews of condition reports and other factors that may impact the health of the system. System Engineer reviews (per the System Engineering Responsibilities and Expectations Guideline) include:

  - visual inspections
  - performance reviews
  - material deficiency reviews
  - feedback from Operations and Maintenance personnel to receive additional insights into the performance of SSCs

- Management Observations – Station management spends time in the field to monitor equipment and personnel performance. These field observations by experienced nuclear managers provide additional insight and assurance that SSCs are performing within expectations. In addition, a report of all Condition Reports (CRs) generated within the previous 24-hours is available to all managers for review.

- Graded Quality Assurance Working Group evaluations – Periodic reviews of SSC performance are conducted by the Graded Quality Assurance (GQA) Working Group to ensure that the expected level of SSC performance is achieved. These in-depth reviews are conducted at least once per cycle. The periodic system reviews performed by the GQA Working Group include:

  - comprehensive corrective action reviews
  - reviews of System Engineer system health status and insight on changes

- PRA model updates
- system modification reviews
- licensing, quality, and operational document reviews
- performance trend reviews

Examples of subjects addressed by approved procedures and guidelines used to support the inspection, test, and surveillance process include:

- Work Process (0PGP03-ZA-0090)
- PM Program (0PGP03-ZM-0002)
- Plant Surveillances (0PGP03-ZE-0004)
- Post Maintenance Testing (0PGP03-ZM-0025)
- Maintenance Rule Program (0PGP04-ZE-0313)
- Graded Quality Assurance Working Group Process (0PEP02-ZA-0001)

5. Corrective Action Program

The Station's Corrective Action Program is used for both safety related (LSS and NRS as well as HSS and MSS SSCs) and non-safety related applications. The Corrective Action Program complies with 10CFR50 Appendix B, and is used to identify and correct equipment deficiencies and nonconformances, to determine the cause of significant conditions adverse to quality, and to initiate actions to prevent recurrence. The STP Corrective Action Program is described in the Operations Quality Assurance Program (OQAP), which has been reviewed and accepted by the NRC.

Examples of subjects addressed in approved procedures and policies that implement the Corrective Action Program include:

- Condition Reporting Program (0PGP03-ZX-0002)
- Corrective Action Program (NGP0131)
- Control of Conditions Adverse to Quality (OQAP 13)
- CR Engineering Evaluation Program (0PGP04-ZA-0002)

6. Management and Oversight Process

The Station's management and oversight process is accomplished through approved procedures and guidelines. This process includes the following:

- Assessments - Assessments are performed to provide a systematic examination of activities, processes, and systems, structures, or components to evaluate the effectiveness of work practices and/or management controls. The type, frequency and degree of specificity of assessments are determined by the importance to the SSC's safety functions, by the performance history of structures, systems, and components, or by the work activity being evaluated. There are two categories of assessments:

- Independent Oversight – These assessments consists of monitoring, surveillance, audit, and inspection activities. These assessments are routinely performed, and the level of detail of the assessment is based on the safety significance of the SSC and its performance history.

- Line Self-Assessments – These assessments may be continuous or focused. Examples of continuous self-assessments include peer-checking activities, independent/dual verification activities, and field observation activities. Continuous self-assessments are performed daily based upon the task scope. Focused assessments evaluate specific programs, processes, or activities. Focused self-assessments evaluate compliance with Station commitments and requirements, and are performed at least once every two years. The frequency of performing focused self-assessments may be adjusted based on actual performance feedback.

Assessments are performed by line organizations or personnel, by management, or by independent internal or external organizations or groups. The importance to the safety function and performance history determines the degree of management and technical oversight. Personnel performing assessments are qualified through training, work experience, or certification.

Actions and findings identified during assessment activities are documented on a Condition Report in accordance with the Corrective Action Program. Action ownership and management oversight are provided to these open items to ensure proper follow-up and closure of identified issues.

- Maintenance Rule - Monitoring is performed at the system, train, or plant level to ensure SSC reliability levels are maintained. LSS and NRS component performance is not explicitly monitored, but degrading performance in these components, if significant, would be observed at the system or train level. Failures of LSS or NRS components that result in the system's performance criteria being exceeded are identified and resolved under a corrective plan of action, which includes actions to prevent recurrence.

- Graded Quality Assurance (GQA) Working Group - SSC performance is periodically assessed (at least once per cycle) by the GQA Working Group to ensure that the expected level of SSC performance is achieved. If an SSC fails to satisfy performance expectations, the commercial controls are evaluated to determine if a weakness in the controls is a cause of or contributing factor to the degraded performance. If the controls are the cause of or a contributor to the degraded performance, the controls are enhanced as necessary to address the weakness. Following the enhancement of the controls, SSC performance will continue to be assessed to ensure that the adjustment to controls has accomplished the desired level of reliability and availability.

- PRA Model Updates - The PRA configuration control program incorporates a feedback process to update the PRA Model. The updates are segregated into two categories:

- The first category, called a plant operating update, will incorporate plant design changes and procedure changes that affect PRA-modeled components, initiating event frequency updates, and changes in SSC unavailability that affect the PRA model. These changes will be incorporated into the model on a frequency not to exceed 36 months.

- The second category, called a comprehensive data update, will incorporate changes to plant-specific failure rate distributions, human reliability, and any other database distribution updates (examples would include equipment failure rates, recovery actions, and operator actions). This second category will be updated on a frequency not to exceed 60 months. The "comprehensive data update" will allow for the accumulation of enough data to properly update distributions without creating an undue burden to perform these time-consuming data updates.

The 36-month frequency specified above is based upon the proposed Nuclear Energy Institute's (NEI) frequency for PRA model updates. In addition, the 60-month frequency is based upon data self-assessments performed by STP personnel earlier this year. Any update will entail an engineering evaluation to determine if applicable 'state-of-the-art' changes relative to PRA technology should be incorporated.

The PRA model may be updated on a more frequent basis if an update would result in a significant increase in the Core Damage Frequency (CDF). All potential model changes are evaluated on a monthly basis with respect to the impact on CDF. If a potential change would result in an increase in CDF of greater than 10%, the model can be updated at the discretion of the PRA Administrator. Also, during the "plant operating update", each of the data variables updated normally during the "comprehensive data update" will be evaluated for changes in the data. If plant specific history demonstrates an approximate order of magnitude change, then that PRA variable can be updated during the "plant operating update" at the discretion of the PRA Administrator.

In addition, technical and administrative procedures are established and implemented as described throughout this document. Administrative procedures identify the factors to be considered in determining whether a procedure is warranted; such factors include the complexity of the work and its impact on the SSC safety functions. Procedural controls include the provision to allow personnel to stop the process, and through the corrective action program, to change the controls or processes to ensure safety functions are satisfied. The process for the development, review, and approval of new procedures, procedure revisions, procedure changes, and procedure deletions, as well as the performance of periodic procedure reviews, is also established and controlled.

Procedures provide for the qualification, training, and certification of personnel. The procedures provide for General Employee Training, indoctrination/nuclear training, certification when required by applicable commercial code or standard incorporated in a design specification, and continuing training and retraining. Personnel are qualified, trained, and certified commensurate with the functions they perform. Experienced personnel may be exempted from prerequisite training.

Documentation, review, and retention requirements of the completed work activities are governed by administrative procedures and work instructions. Examples of documentation include:

- Design drawings and specifications
- Procurement documents, such as procurement request forms, engineering evaluations of item equivalency, and purchase orders or catalog orders
- Documentation of completed installations and the results of applicable post-modification tests
- Documentation of completed maintenance activities and the results of applicable post-maintenance tests
- Documentation of Technical Specification surveillances, documentation required by the Maintenance Rule program, reports of periodic evaluations by the Graded Quality Assurance Working Group, and documentation of deficiencies identified through operator rounds, system engineer reviews, and management observations.
- Condition reports of conditions adverse to quality
- Documentation of actions and findings resulting from assessments.

Procedures identify the types of inspection, test, and surveillance equipment requiring control and calibration, and the interval of calibration. When calibration is required, each organization is responsible for assuring that the M&TE that it uses has been calibrated to the accuracy required for its intended use. Equipment that is in error or defective is removed from service or properly tagged to indicate the error or defect.

Examples of subjects addressed in approved procedures and guidelines used to support the management and oversight process includes:

- Plant Audits (0PQP01-ZA-0001)
- Oversight Planning and Scheduling Process (0PQP01-ZA-0015)
- Quality Surveillance/Monitoring (0PQP02-ZA-0003)
- Maintenance Rule Program (0PGP04-ZE-0313)
- Graded Quality Assurance Working Group Process (0PEP02-ZA-0001)
- Probabilistic Risk Assessment Program (0PGP04-ZA-0604)
- Independent Plant Assessments (0PQP01-ZA-0006)
- Self-Assessment Process (STP-0423)

7. Configuration Control Process

The Station's configuration control process is controlled through approved procedures and policies. The Design Control Process ensures that the configuration of the Station is properly reflected in design documents and drawings. Changes to the Station are controlled through Design Change packages (modifications) which require that control drawings and documents be updated prior to closeout of the modification package.

In addition, configuration control addresses the status of components day-to-day in the field. SSCs are tagged and are manipulated by qualified Operations personnel per procedure. The Configuration Control process manages and controls the physical changes (procedural and equipment) to the facility to assure that the plant configuration and practices correctly reflect the licensing bases. Non-code (ASME) components installed in code systems are identified and tracked.

Examples of subjects addressed in approved procedures and policies that implement the configuration control process include:

- Design Change Implementation (0PGP03-ZE-0031)
- Operations Configuration Management (0PGP03-ZO-0039)
- Performing/Verifying Station Activities (0PGP03-ZA-0010)
- Configuration Management Program (0PGP03-ZA-0109)
- Control of Configuration Changes (0PGP03-ZA-0021)
- Plant Modifications (0PGP04-ZE-0310)