UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 18, 2001

LICENSEE: STP Nuclear Operating Company

FACILITY: South Texas Project, Units 1 and 2

SUBJECT: SUMMARY OF MEETING WITH STP NUCLEAR OPERATING COMPANY TO DISCUSS DRAFT SAFETY EVALUATION OPEN ITEMS ON SOUTH TEXAS PROJECT, UNITS 1 AND 2, MULTIPART EXEMPTION FROM THE SPECIAL TREATMENT RULES OF 10 CFR PART 50

On December 6 and 8, 2000, the U.S. Nuclear Regulatory Commission (NRC) and STP Nuclear Operating Company (STPNOC) met in Rockville, Maryland, to discuss open items identified in the draft safety evaluation issued on November 15, 2000, related to STPNOC's request for exemption from special treatment requirements of 10 CFR Parts 21, 50, and 100. The purpose of the meeting was to facilitate communication between the staff and the licensee to allow the effective resolution of the open items in the draft safety evaluation.

Enclosure 1 provides a list of attendees at the 2-day meeting. Enclosure 2 provides the meeting agenda (including revision 1 to the agenda). Enclosure 3 provides a copy of the information used by the staff to discuss issues related to the use of common cause failure in determination of the risk significance of components (this information was emailed to the licensee in advance of this meeting to facilitate discussions between the staff and the licensee). Enclosure 4 provides an excerpt from Regulatory Guide (RG) 1.176, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," used by the staff during the meeting. Enclosure 5 provides the information used by the licensee in the discussion on its proposed resolution of issues.

During the meetings on December 6 and 8, 2000, the following areas were discussed:

Engineering Evaluations:

STPNOC provided a rewrite of the proposed Final Safety Analysis Report (FSAR) section on procurement that clarified its processes for the use of engineering evaluations when obtaining replacement structures, systems, or components (SSCs). The rewrite described when an engineering evaluation would be used, when an equivalency evaluation would be used, and when an engineering analysis would be needed. The proposal appeared reasonable to the staff, but requires further staff evaluation to determine whether it is acceptable. From an environmental qualification aspect, it was still not clear to the staff that an exemption is required from 10 CFR 50.49 based on the STPNOC proposal. Assuming the proposed approach is technically acceptable, the staff must conclude whether the licensee proposal meets the requirements of 10 CFR 50.49 (in which case no exemption would be required) or that the proposed approach does not meet the regulation, but satisfies the underlying purpose of the regulation (in which case an exemption would be required). Similar logic could be used for addressing the seismic requirements of 10 CFR Part 100.

National Consensus Standards:

STPNOC provided a rewrite of the proposed FSAR section that provided criteria for using national consensus standards. These included the provisions to apply consensus standards required by the State of Texas, apply those consensus standards in existence at the time of the exemption, and for adoption of future standards as STPNOC determines appropriate. The staff provided feedback that rather than the standards in existence at the time of the exemption, the licensee should rely on the standards currently being used at the facility for its balance of plant SSCs. Otherwise, the licensee could be required to review all existing national consensus standards to determine whether they applied to the balance of plant processes. The staff's initial impression was that this appeared to be a reasonable approach, with the request that STPNOC provide the staff with some indication regarding the standards that would be followed as required by the State of Texas.

Common Cause Failure:

STPNOC has indicated that it will most likely return to the calculational method for risk importance measures used under graded quality assurance to consider common cause failure in the categorization of SSCs. Both STPNOC and the staff recognize that the graded quality assurance approach is conservative in its calculation of the common cause contribution of individual components to risk importance measures. The licensee indicated that there would be a small impact on the overall categorization (at each unit, approximately 23 SSCs would transition from low safety significant (LSS) to medium safety significant (MSS) and therefore come out of the scope of the exemption, and about 20 SSCs would transition from MSS to high safety significant (HSS) with no change in exemption scope). During the briefing of the Advisory Committee on Reactor Safeguards (ACRS) on December 7, 2000, an issue was raised by the ACRS regarding what is the correct method for calculating the risk achievement worth. The licensee's proposal to return to the graded quality assurance calculational method may not resolve the issue raised at ACRS, but may be sufficient as a basis for the requested exemptions.

Change control:

STPNOC proposed a modification to the change control process that would allow them to make beneficial changes or changes that do not adversely impact the basis for the NRC's granting of any exemption. The approach proposed by STPNOC is similar to the controls used in changes to security plans, operating quality assurance plans, or emergency plans. The approach proposed by STPNOC is for the licensee to inform the NRC of all changes to the processes, however, only those that the licensee determined would reduce the effectiveness of the processes would be submitted to the NRC for prior review and approval. This approach could be acceptable to the staff provided appropriate criteria can be developed that establishes the threshold for determining when prior NRC review and approval would be necessary. The emphasis by the staff was to impress upon STPNOC the fact that the NRC needs to have confidence that the basis for any exemption it may grant remain valid going forward under any proposed change control process.

Beyond Design Basis HSS/MSS Functions:

STPNOC proposed alternate wording to address the treatment of beyond design basis functions for non-safety related HSS and MSS SSCs. The licensee has focused on applying the processes that would be imposed by the graded quality assurance program for Targeted Treatment. The determination would include ensuring that if existing control were not sufficient to maintain the reliability and availability of the SSC consistent with its categorization, STPNOC would identify what additional special treatment controls should be imposed to provide the required reliability and availability consistent with its categorization. For safety-related HSS and MSS SSCs STPNOC stated that it has not identified any HSS or MSS function that is not already covered by current special treatment requirements. However, STPNOC indicated that if it does, it would apply the same controls applied to non-safety-related HSS and MSS SSC functions. The staff noted that this appeared to be a reasonable method to resolve this issue.

Limitations on CDF and LERF:

The staff acknowledged that a quantitative metric (probabilistic risk assessment (PRA) based) other than core damage frequency (CDF) or large early release frequency (LERF) may not be available for assessing the importance of SSCs that do not have a significant role in preventing core damage or large early releases, but do have a role in mitigating the consequence of core damage accidents. However, the staff pointed out to STPNOC that in RG 1.176, the NRC's expectation is that "functions credited in the PRA for long-term overpressure protection, but which do not contain any SSCs with CDF or LERF based importance measures above the guideline values, should be identified and the safety significance explicitly assigned." (See Enclosure 4 for an excerpt from RG 1.176 (section 2.2.2), referenced by the staff.) The staff felt that while its categorization process can be used to explicitly assign safety significance, STPNOC does not necessarily ask the question regarding the mitigation of the consequences of accidents in categorizing SSCs. STPNOC indicated they had a better understanding of the basis for the staff's position and believed that the categorization process had taken that into consideration. However, STPNOC agreed that it would be reasonable to look at the SSCs that have been categorized with this question in mind to see what impact it would have on the categorization of those SSCs and that this question could be factored into its categorization of SSCs going forward.

Testing of Pumps and Valves:

Discussions regarding the testing of pumps and valves focused on the need to provide confidence of functionality on a continuing basis. The licensee identified an area where it was confused regarding inspection, tests, and surveillance as discussed in section 4.3.3.5 of the November 15, 2000, draft safety evaluation. The staff clarified that the discussion in the draft safety evaluation in this area was primarily targeted at the testing of pumps and valves and is most directly related to the exemption request from the inservice testing (IST) requirements imposed by 10 CFR 50.55a(f). STPNOC also sought clarification on the nature of the testing expected by the staff and indicated that it would prefer to conduct periodic pass/fail type tests for LSS and non-risk significant (NRS) safety-related pumps and valves that were formerly included within the scope of the IST program. STPNOC indicated that the test methodologies used in the IST program would be used, but without all the provisions required by the IST program (i.e., restrictions on test instrumentation, alert ranges, actions, etc). The staff and

STPNOC agreed that there are circumstances when normal operational activities (such as switching between operating trains, normal system operation, periodic technical specification testing, etc.) could be used as the basis for confidence in the ongoing functionality of pumps and valves when there existed a relationship to operation under design basis conditions. These circumstances generally would apply to systems that are used in the normal modes of plant operation (full power operation, plant heatup or cooldown, or shutdown operation), but generally would not apply to systems that remain in standby conditions during normal plant operation. Many of the safety-related standby systems were designed to address design basis accidents. Most of these systems would be categorized as HSS or MSS and therefore are outside the scope of the exemption requests. However, for those safety-related standby systems that are categorized as LSS or NRS, the staff indicated it would need predictive/diagnostic type tests (similar to IST tests) to provide confidence in the ongoing functionality of these SSCs. Also, the staff indicated that the testing that is conducted would require monitoring parameters that give indication of component degradation. The staff clarified its intent that testing of LSS and NRS SSCs would in no circumstances be beyond what would be required by the IST program required by 10 CFR 50.55a(f).

At the end of the meeting, both STPNOC and the staff agreed to continue working to resolve the open items identified in the draft safety evaluation. However, STPNOC and the staff recognized that the target completion date of April 15, 2001, may be challenged due to the complexity of a number of the open items and the followup from the December 7, 2000, meeting with the Advisory Committee on Reactor Safeguards on the resolution of the open items in the draft safety evaluation. Overall, the meetings with the licensee were productive and both well attended and supported by the staff. More interactions will occur between the staff and the licensee as the licensee prepares responses to the draft safety evaluation open items. The staff expects to receive the responses in January 2001 to support the completion of open item resolution by mid-February 2001.

/RA/
John A. Nakoski, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
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Office of Nuclear Reactor Regulation

Docket Nos. 50-498 & 50-499

Enclosures:
1. List of Attendees
2. Meeting Agenda
3. NRC Handout
4. Excerpt from RG 1.176
5. Licensee Handout

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LIST OF ATTENDEES
DECEMBER 6 AND 8, 2000
MEETING BETWEEN NRC AND STPNOC
COMMERCIAL PRACTICES

NAME	TITLE/POSITION	ORGANIZATION	12/6	12/8
Dick Wessman	Deputy Director	DE		x
Richard Barrett	Branch Chief	NRR/DSSA/SPSB	x	x
Jose Calvo	Branch Chief	NRR/DE/EEIB	x	x
Gene Imbro	Branch Chief	NRR/DE/EMEB	x	
Stu Richards	Project Director	NRR/DLPM/PDIV	x	x
Goutam Bagchi	Senior Level Advisor	NRR/DE	x	x
Robert Hermann	Sr. Level Scientist - Materials	NRR/DE	x	
David Terao	Sr. Regional Coordinator	EDO	x	
David Fischer	Section Chief (A)	NRR/DE/EMEB	x	x
Bob Gramm	Section Chief	NRR/DLPM/PDIV-1	x	x
Cornelius Holden	Section Chief	NRR/DE/EEIB	x	x
Mark Rubin	Section Chief	NRR/DSSA/SPSB	x	
Steve West	Section Chief	NRR/DRIP/RGEB	x	
John Fair	Sr. Mechanical Engineer	NRR/DE/EMEB	x	
Hukam Garg	Sr. Electrical Engineer	NRR/DE/EEIB	x	x
John Knox	Sr. Electrical Engineer	NRR/DE/EEIB		x
Eileen McKenna	Sr. Reactor Engineer	NRR/DRIP/RGEB		x
John Nakoski	Sr. Project Manager (STP)	NRR/DLPM/PDIV-1	x	x
Tim Reed	Sr. Reactor Engineer (RIP50)	NRR/DRIP/RGEB	x	x
Tom Scarbrough	Sr. Mechanical Engineer	NRR/DE/EMEB	x	x
Harold Walker	Sr. Reactor System Engineer	NRR/DSSA/SPLB		x
Peter Balmain	Operations Engineer	NRR/DIPM/IQMB	x	x
Stephen Disnmore	Realiability and Risk Analyst	NRR/DSSA/SPSB	x	x
Joe Golla	Project Manager	NRR/DRIP/RGEB	x	
Ken Heck	Operations Engineer	NRR/DIPM/IQMB	x	
Samuel Lee	Realiability and Risk Analyst	NRR/DSSA/SPSB	x	x
Matthew A. Mitchell	Materials Engineer	NRR/DE/EMCB	x	
Ron Young	Reactor System Engineer	NRR/DSSA/SPLB		x
Asimios Malliakas	Risk Analyst	RES/DRAA/PRAB		x
Pete Prassinos	Risk Analyst	RES/DRAA	x	
Mark McBurnett	Director, Quality & Licensing	STPNOC	x	x
Steve Frantz	Attorney for STPNOC	Morgan, Lewis, & Bockius	x	x
Scott Head	Supervisor - Licensing	STPNOC	x	x
Rick Grantom	Risk Management	STPNOC	x	x
Glen Schinzel	GQA Project Manager	STPNOC	x	x
Roger Huston	Principal	Licensing Support	x	
Adrian Heymer	Project Manager	NEI	x	x

Enclosure 1

**RISK-INFORMED EXEMPTION TO
SPECIAL TREATMENT REQUIREMENTS
MEETING BETWEEN STPNOC AND NRC ON DSE OPEN ITEMS**

MEETING PURPOSE: To provide a forum for the exchange of information in support of the resolution of the Open Items identified in the November 15, 2000, Draft Safety Evaluation.

AGENDA:

WEDNESDAY, DECEMBER 6, 2000 - ROOM O-13B4

1:00pm to 2:30pm:	Broad discussion on Engineering Evaluation vs. Engineering Analysis - when does STPNOC use evaluations vs. when does STPNOC use analysis.
2:30pm to 2:45pm:	Break
2:45pm to 4:15pm:	Testing - baselining of replacement components, monitoring, and PMT.
4:15pm to 5:15pm:	Examples to provide PRA insights into how beyond design-basis conditions are considered for HSS/MSS SSCs

FRIDAY, DECEMBER 8, 2000 - ROOM O-13B4

8:00am to 8:30am:	Feedback from ACRS meeting
8:30am to 9:00am:	National Consensus Standards - What level of commitment is the staff looking for?
9:00am to 10:45am:	Change Control - level of detail in FSAR (Open item 5.1)
10:45am to 11:00am:	Break
11:00am to 12:00pm:	PRA issues (limitations on CDF/LERF and latent consequences, and Common Cause Failure issues)
12:00pm to 12:30pm:	Wrapup

RISK-INFORMED EXEMPTION TO
SPECIAL TREATMENT REQUIREMENTS
MEETING BETWEEN STPNOC AND NRC ON DSE OPEN ITEMS
(Revision 1)

MEETING PURPOSE: To provide a forum for the exchange of information in support of the resolution of the Open Items identified in the November 15, 2000, Draft Safety Evaluation.

AGENDA:

WEDNESDAY, DECEMBER 6, 2000 - ROOM O-13B4

1:00pm to 2:30pm:	Broad discussion on Engineering Evaluation vs. Engineering Analysis - when does STPNOC use evaluations vs. when does STPNOC use analysis.
2:30pm to 2:45pm:	Break
2:45pm to 3:30pm:	Testing - baselining of replacement components, monitoring, and PMT.
3:30pm to 4:00pm:	National Consensus Standards - What level of commitment is the staff looking for?
4:00pm to 5:15pm:	Examples to provide PRA insights into how beyond design-basis conditions are considered for HSS/MSS SSCs

FRIDAY, DECEMBER 8, 2000 - ROOM O-13B4

8:00am to 8:30am:	Feedback from ACRS meeting
8:30am to 10:00am:	PRA issues (limitations on CDF/LERF and latent consequences, and Common Cause Failure issues)
10:00am to 10:15am:	Break
10:15am to 11:30am:	Change Control - level of detail in FSAR (Open item 5.1)
11:30am to 12:30pm:	Discussion on STPNOC proposed resolution of issues
12:30pm to 1:00pm:	Wrapup

Below is a simplified example of an estimation of RAW contribution from common cause failures. This example depicts the RAW involving Component "X" from a 3-train system which has X, Y, Z components. Thus the Random failures include: X, Y, and Z and Common Cause failures include: [XY], [XZ], [YZ], and [XYZ]. Assume 0.1 to be the failure probability for each of these failures.

Case	Failure Type	Base Condition	Random Failure of X (Assumed failure of X)	Assumed failure of doublet common cause failures involving x: [xy] or [xz]	Assumed failure of triplet CCF	Assumed failure of all CCFs involving X	Assumed failure of X plus assumed failure of all CCFs involving X	Assumed failure of ccf [xy]	Assumed failure of ccf [xz]
1	XYZ	(.1)(.1)(.1) = 0.001	(1)(.1)(.1) = 0.01	(.1)(.1)(.1) = 0.001	(.1)(.1)(.1) = 0.001	(.1)(.1)(.1) = 0.001	(1)(.1)(.1) = 0.01	(.1)(.1)(.1) = 0.001	(.1)(.1)(.1) = 0.001
2	X[YZ]	(.1)(.1)= 0.01	(1)(.1) = 0.1	(.1)(.1)= 0.01	(.1)(.1)= 0.01	(.1)(.1)= 0.01	(1)(.1) = .1	(.1)(.1)= 0.01	(.1)(.1)= 0.01
3	Y[XZ]	(.1)(.1)= 0.01	(.1)(.1)= 0.01	(.1)(1)= 0.1	(.1)(.1)= 0.01	(.1)(1)= 0.1	(.1)(1)= 0.1	(.1)(.1)= 0.01	(.1)(1)= 0.1
4	Z[XY]	(.1)(.1)= 0.01	(.1)(.1)= 0.01	(.1)(1)= 0.1	(.1)(.1)= 0.01	(.1)(1)= 0.1	(.1)(1)= 0.1	(.1)(1)= 0.1	(.1)(.1)= 0.01
5	[XYZ]	.1	.1	.1	1.0	1.0	1.0	.1	.1
	TOTAL	0.131	0.23	0.311	1.031	1.211	1.31	0.221	0.221

Therefore,

(1) Using the STP proposed method.....

$RAW_{doublet}$ = ½ RAW[XY] + ½ RAW[XZ] = ½ (0.221) + ½ 0.221 = 0.221

(2) Using the above chart....

$RAW_{doublet}$ = 0.311

(3) Using the STP GQA method....

$RAW_{doublet}$ = RAW[XY] + RAW[XZ] = 0.221 + 0.221 = 0.442

This example demonstrates that the method used for GQA to account for the RAW from CCFs overestimates the "true" value. However, it also demonstrates that the proposed method underestimates the RAW contribution from CCFs.

EXCERPT FROM RG 1.176

Enclosure 4

2.2.1 Engineering Evaluation Guidelines

The engineering evaluation should assess whether the impact of the proposed change is consistent with the defense-in-depth philosophy. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent decision guidelines are acceptable.

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system and uncertainties (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in Appendix A to 10 CFR 50 is maintained.

The engineering evaluation should also assess whether the impact of the proposed change is consistent with the principle that sufficient safety margins are maintained. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent decision guidelines are acceptable.

- Codes and standards or alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the licensing basis (e.g., Final Safety Analysis Report (FSAR) and supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

2.2.2 Guidelines for Defense in Depth and Safety Margins

Defense in depth and safety margins are expected to be addressed generally by considering the following GQA program aspects.

- The GQA process will not result in changes to the plant configuration. Therefore, no existing plant barriers will be removed. Additionally, existing system redundancy, diversity, and independence will be maintained.
- The GQA process will not result in changes to the technical requirements (e.g., design bases or operational parameters) associated with SSCs.
- The resulting QA provisions will provide the necessary level of assurance that low safety-significant, safety-related and high safety-significant, non-safety-related SSCs remain capable of performing their safety function.

The core damage frequency (CDF) and large early release frequency (LERF) figures of merit do not fully cover long-term containment overpressure protection. Functions credited in the PRA for long-term overpressure protection, but which do not contain any SSCs with CDF or LERF based importance measures above the guideline values, should be identified and the safety significance explicitly assigned. For example, the containment spray systems for PWRs may not contribute to the prevention or mitigation of core damage or large early release.

An important factor to ensure that defense-in-depth and safety margin considerations are not degraded during the implementation of GQA is control of potential common mode failures. As discussed in Regulatory Position 2.1.2.1, groups of nominally identical SSCs, utilized in multiple systems throughout the plant, can as an aggregate have high safety significance.

Principle 4 in Regulatory Guide 1.174 (Ref. 3) states that any proposed increase in CDF and risk are small and are consistent with the intent of the Commission's Policy Statement (Ref. 1). Although the risk impact of GQA changes on individual components is expected to be minimal, reduced QA oversight may be applied to a large number of SSCs. It is recognized that limited data are available to define the impact of QA programs on SSC reliability. Accordingly, the licensee should perform a bounding analysis in which the failure rates or probabilities for basic events representing SSCs that may be subjected to reduced QA controls are set at some increased level (chosen and justified by the licensee). Alternatively, the licensee may choose to address the bounding analyses by modifying the uncertainty distributions in some manner (also chosen and justified by the licensee).

The bounding analysis should include all SSCs modeled in the PRA on which QA controls may be

LICENSEE HANDOUT

REVISIONS TO SELECTED PAGES OF PROPOSED FSAR SECTION

Enclosure 5

ENGINEERING EVALUATIONS FOR EQ - - INSERT FOR UFSAR SECTION 13.7.3.3.4

Technical requirements (including applicable environmental conditions) are specified for items to be procured, which are based on the original design inputs and assumptions for the item. One or more of the following methods are used to determine that the procured item can perform its function under design basis conditions, including applicable environmental conditions:

- Vendor Documentation - - The performance characteristics for the item, as specified in vendor documentation (e.g., catalog information, certificate of conformance), satisfy STP's technical requirements.

- Equivalency Evaluation - - An equivalency evaluation determines that the procured item has a form, fit, and function under design basis conditions that is equivalent to the item being replaced.

- Engineering Evaluation - - An engineering evaluation compares the differences between the procured item and original item and determines that the differences are not sufficient to preclude the procured item from performing its function under design basis conditions.

- Engineering Analysis - - In cases involving design changes or substantial differences between the procured item and replacement item, an engineering analysis is performed to determine that the procured item can perform its function under design basis conditions. The engineering analysis may be based upon a computer calculation, evaluations by multiple disciplines, test data, or operating experience related to the procured item.

Documentation of the implementation of these methods is maintained. Additionally, documentation is maintained to identify the preventive maintenance needed to preserve the capability of the procured item to perform its function under applicable environmental conditions.

INSTALLATION TESTING - - INSERT FOR UFSAR SECTION 13.7.3.3.5

A test is performed if the installation could affect an SSC's design function. The test verifies that the SSC is operating within expected parameters and functionality is verified prior to return to service.

PERIODIC TESTING - - INSERT FOR UFSAR SECTION 13.7.3.3.7

ASME pumps and valves are subject to routine operation or periodic tests. This includes one or more of the following:

- Running of the pump or actuation of the valve during normal operation, system alignment changes, or mode changes.

- Testing of the pump or valve using the inservice test (IST) methodology specified in 10 CFR 50.55a(f), but at a reduced frequency and without the other special treatment required by that section.

- Periodic testing of the pump or valve using a method that is less rigorous than the IST methodology specified in 10 CFR 50.55a(f) but still sufficient to provide confidence that the component has not failed.

NATIONAL CONSENSUS STANDARDS - - INSERTS FOR UFSAR SECTION 13.7.3.3

STP uses the following national consensus standards in the _________ process, as necessary to ensure the functionality of components:

- Standards required by the State of Texas to be used for the process.

- Existing standards at the time of the granting of the exemption, where STP determines that it is reasonable to apply those standards to the process.

- Future standards, at STP's discretion or in lieu of a standard in use at the time of the granting of the exemption.

STP is not required to itemize the standards in use at STP or to perform an evaluation of all national consensus standards.

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 experienced 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.**

The Expert Panel and Working Group have expertise in the areas of risk assessment, quality assurance, licensing, engineering, and operations and maintenance. The combined membership of the Expert Panel and Working Group includes at least three individuals with a minimum of five years experience at STP or a similar nuclear plant, and at least one individual who has worked on the modeling and updating of the PRA for STP or a similar plant for a minimum of three years. *[Note to STP: NRC's 7/19/00 Draft Review Guidelines do not allow for such experience to be on "a similar plant."]*

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.

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.**

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. 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 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**

~~**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.**~~
Procedures provide for the qualification, training, and certification of personnel, commensurate with the functions they perform. Experienced personnel may be exempted from prerequisite training. STP considers vendor recommendations and commercially accepted national accepted consensus standards in the training, qualification, and certification of personnel. STP may deviate from these recommendations and standards based on specific circumstances and sound business practices. Such deviations are not required to be documented.

Documentation, review, and retention requirements of completed work activities are governed by administrative procedures and work instructions.

Procedures identify the types of inspection, test, and surveillance equipment requiring control and calibration, and the interval of calibration. Equipment that is in error or defective is removed from service or properly tagged to indicate the error or defect, and a determination will be made of the functionality of the SSC that was checked using that equipment.

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.**~~