May 18, 2000

MEMORANDUM TO: Robert A. Gramm, Chief, Section 1
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: John A. Nakoski, Senior Project Manager, Section 1
Project Directorate IV and Decommissioning /***RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - DRAFT INFORMATION PROVIDED BY LICENSEE BETWEEN APRIL 26 AND MAY 15, 2000, FOR RESOLUTION OF THE REQUEST FOR ADDITIONAL INFORMATION FOR THE MULTIPART EXEMPTION REQUEST (TAC NOS. MA6057 AND MA6058)

The U.S. Nuclear Regulatory Commission (NRC) staff is in the process of reviewing the risk-informed exemption requests that the STP Nuclear Operating Company (STPNOC) submitted on July 13, 1999. As part of that process, the NRC staff issued a request for additional information (RAI) on January 18, 2000. Currently, the staff is working with STPNOC to ensure that STPNOC clearly understands the extent of the questions raised and for the NRC staff to gain a better understanding of the scope of the expected response by STPNOC. The NRC staff has agreed to participate in periodic teleconferences to discuss specific questions raised in the RAI. In preparation for these teleconferences, the licensee will frequently provide the NRC staff with information either using email or by fax. Likewise, the NRC staff will frequently provide information to the licensee using similar methods. All of the information exchanged by email or fax between the licensee and the NRC during this process will be made available to the public.

Enclosure 1, received from the licensee on April 26, 2000, responds to questions 20, 21, 25, 31, 33, 35, and 36. Enclosure 2, received from the licensee on May 2, 2000, responds to questions 1, 26, 27, and 28. Enclosure 3, received from the licensee on May 3, 2000, responds to question 22. Enclosure 4, received from the licensee on May 9, 2000, responds to questions 10 and 23. Enclosure 5, received from the licensee on May 10, 2000, responds to questions 4, 8, 19, and 32. Enclosure 6, received from the licensee on May 11, 2000, responds to questions 3 and 7. Enclosures 7, 8, and 9, received from the licensee on May 15, 2000, respond to questions 5, 14, 38, 44, and 46; 42; and 11, 15, and 34, respectively.

The draft information provided by the licensee was reformatted into WordPerfect format and distributed to the responsible technical reviewers. No changes were made to the text of the information provided by the licensee.

Enclosures:
1. Draft Response to RAI Questions 20, 21, 25, 31, 33, 35, and 36
2. Draft Response to RAI Questions 1, 26, 27, and 28
3. Draft Response to RAI Question 22
4. Draft Response to RAI Questions 10 and 23
5. Draft Response to RAI Questions 4, 8, 19, and 32
6. Draft Response to RAI Questions 3 and 7
7. Draft Response to RAI Questions 5, 14, 38, 44, and 46
8. Draft Response to RAI Question 42
9. Draft Response to RAI Questions 11, 15, and 34

Docket Nos. 50-498 and 50-499

The draft information provided by the licensee was reformatted into WordPerfect format and distributed to the responsible technical reviewers. No changes were made to the text of the information provided by the licensee.

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1. Draft Response to RAI Questions 20, 21, 25, 31, 33, 35, and 36
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Docket Nos. 50-498 and 50-499

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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: (A. Moldenhauer)

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:

$$FV_{Comp\ A} = FV_A + 1/2*FV_{[AB]} + 1/2*FV_{[AC]} + 1/3*FV_{[ABC]}$$

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:

$$FV_{Comp\ x} = FV_x + 1/2*FV_{Doublet} + 1/3*FV_{Triplet} + 1/4*FV_{Quadruplet} + ...$$

Where,

$FV_{Comp\ x}$ represents the total FV importance of component x.

Mathematically, the Risk Achievement Worth for component A is represented by:

$$RAW_{comp\ A} = 1 + RAW_A - 1 + 1/2*RAW_{[AB]} - 1 + 1/2*RAW_{[AC]} - 1 + 1/3*RAW_{[ABC]} - 1$$

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:

$$RAW_{Comp\ x} = 1 + RAW_x + 1/2*RAW_{Doublet} + 1/3*RAW_{Triplet} + 1/4*RAW_{Quadruplet} + ... - n$$

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 iiwh

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	No. 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: (A. Moldenhauer)

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, Qualitative Categorization Methodology. 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."

STP identifies all attributes for HSS safety related components, which are considered critical attributes. 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: (D. W. Stillwell)

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 Graded QA process is fully implemented, changes in equipment failure rates, if they occur, will be identified by the Maintenance Rule Program 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 components 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. Notwithstanding this conclusion, use of the special treatment exemption, when granted, will occur only as components are replaced. 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: (D. W. Stillwell)

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. Thus demonstrating the robustness of the PRA risk ranking process.

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 component categorized as LSS and which are modeled in the PRA. The results are as follows:

	Current Average (events/reactor year)	Sensitivity Study l_{LSS}*10 (events/reactor year)	Increase	% Increase
CDF	9.0781E-6	9.3232E-6	2.4510E-7	2.7%
LERF	1.3742E-7	1.5136E-7	1.3940E-8	10%

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: (D. W. Stillwell)

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: (D. W. Stillwell)

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: (D. W. Stillwell)

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 effect 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 rate will be 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 will be 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.

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).

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): (R. Chackal)

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

Weighted Score of 25 on any one question (ACC or EOP).....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 MEDIUM risk for this hypothetical component.

RESPONSE (part b): (R. Chackal)

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): (R. Chackal)

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. All of 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.

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): (A. Moldenhauer)

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): (A. Moldenhauer)

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 its importance during certain scheduled maintenance activities. The PDP had 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.

It is expected that with the PDP's recent improved reliability and availability, the PRA importance categorization will result in a lower classification. Consideration for the low reliability and availability of this component demonstrates the robustness of the GQA risk categorization process.

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: (R. Chackal)

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|CHARGING 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.
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).

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
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
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

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
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.
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

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
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
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

SY	TYPE	ID	COMPONENT DESCRIPTION	PRA	RISK	COMMENTS
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
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 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: (A.Moldenhauer)

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.

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: (R. Chackal)

The risk significance of any component in the plant can be determined using the established 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 of the component's failure on the system functions supported.

5. Identification of component critical attributes for safety related Low/Medium and non-safety related Medium/High components.

6. Approval by the Expert Panel.

7. Periodic performance feedback to ensure the appropriateness of the risk categorization.

Until a component is categorized in this manner, it remains conservatively under the Full QA program (if it is safety related) and is not in the scope of this exemption request.

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. 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. For LSS/NRS non-MED components, the allowances provided in this exemption request apply to these components also.

It should be emphasized that regulatory requirements not in the scope of this exemption request would continue to be applied for LOW and NRS non-MED components. For example, cabling would continue to meet separation requirements regardless of its risk significance. In addition, as with MED components, design requirements would still apply and could not be changed without being first evaluated under the design change process.

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: (R. Chackal)

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 ensure 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 ensure 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 ³ 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.

Type	Sys	Component	PRA Risk	Determ Risk	Final Risk	Basis
	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 is re-assessing the use of the 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 will occur to fully assess the overall documentation adequacy. The results of these corrective actions will be forwarded to the NRC within six weeks of the final submittal of the RAI responses.

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: (R. Chackal)

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 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)	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	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?	Could the failure result in loss or substantial degradation of another system's risk significant functions?

MAINT. RULE SCOPE	EQUIVALENT SCOPE IN RISK SIGNIFICANCE DETERMINATION PROCESS	COMMENTS
(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
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

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
	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
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
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

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
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

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): (R. Chackal)

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 control, use, and revision of General Notes in the risk categorization process. The results of the overall review of the General Notes and the revised procedural guidance will be incorporated into the final 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): (R. Chackal)

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): (R. Chackal)

As stated, STP has reviewed all evaluated systems that utilized the old-format notes to ensure consistency with the approved General Notes. Specific for the categorization of limit switches, 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 switches involved in the actuation of critical components. STP concludes that these switches should receive the same risk rank as their associated component, if their failure could prevent the actuation of that component. We have initiated a condition report to effect this change, to review all previously evaluated systems for this occurrence, and to revise the generic notes to specifically refer to this determination. The results of the overall review of limit switches will be incorporated into the final response.

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 generic 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.

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): (R. Chackal)

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 will request 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): (R. Chackal)

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 generic notes, which have been approved by the Expert Panel. For control room handswitches, the generic notes provide the following justifications:

1. Most time-sensitive operations are automatic and do not require handswitch manipulation.
2. Reliability of handswitches has been very good.
3. Redundant handswitches are available.

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.

Results Of Reviews To Compare Reliability Of Safety Related Versus Non-Safety Related Handswitches

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 then merged to provide a large-scope analysis to support responses for the STPEGS 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 said 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, STPEGS 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.

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.

For a component whose failure could cause the failure of electrically interconnected components, the classification process involves an evaluation of the potential failure modes, their probability, their impact 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 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 impacted highest risk electrically connected component. Under such a scenario, the risk of the electrical component could be higher than that of the attached mechanical component. In that case, the Working Group would re-evaluate the appropriateness of the mechanical component's risk categorization in light of the potential for a failure of the mechanical component to result in failure of 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 and 24.

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: (R. Grantom)

To clarify our position, STP requests a permanent and global exemption from the 50.59 evaluation process only for special treatment requirement issues for all 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 special treatment requirements (i.e., functional changes, design changes, etc) as applicable.

When changes are necessary for LSS or NRS components that only affect the special treatment requirements, these changes will be controlled through existing commercial treatment programs that provide reasonable assurance that the 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.

For example, if STP were to replace a failed safety-related LSS/NRS component with a functionally equivalent, commercial-grade, non-safety-related component, then a 50.59 process would not need to be performed. The 50.59 evaluation process would not be required in this situation since the change is solely associated 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 process 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.

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: (R. Grantom)

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:

- Changes in Technical Specifications, License Conditions, and Orders – 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.

- Changes in the Updated Final Safety Analysis Report (UFSAR) – 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 USFAR without performing a 50.59 evaluation or seeking prior NRC approval. Therefore, following grant of the exemption, STP will simply 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).

- Changes in the Quality Assurance Program (QAP) Description – 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 simply 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).

- Changes in Other Commitments – 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.

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, Risk-Based Snubber Inspection and Testing Guidelines. 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.

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 a 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.

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: (Allen Moldenhauer)

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 categorized as HSS or MSS. 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.

Equipment Survivability Analysis

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:

<u>Description</u>

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.

<u>Previous Reviews</u>

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 general 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.

32. (con't)

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 overall small importance external events have on the PRA.

32. (con't)

Have any SSC's been identified that are important only for external events?

RESPONSE: No

32. (con't)

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 activity 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. An 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 necessary to support operation of the various systems is determined based on discrete acceleration values. All components of a similar type (e.g., batteries or diesel generators) are assumed to fail at the same time based on these values. 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 <u>possible</u> changes. The risk associated with <u>possible</u> changes in equipment performance are addressed by increasing equipment failure rates by a factor of 10 and 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 <u>possible</u> 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 guarantee 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 <u>possible</u> 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 <u>possible</u> performance changes to LSS components have a negligible impact on CDF or LERF.

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) (K. Cope)

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 will be 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. However, if this LSS transmitter were to be relocated to a different area inside of Containment, a design change package would be generated 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.

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: (R. Grantom)

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 to establish 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 scope or testing strategy changes. Under the current ASME O&M code cases, only test frequency 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 requires monitoring of system and/or component functions that provide 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 would 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 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, 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.

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: (R. Chackal)

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. 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 – a) Use of planned and fully documented maintenance work packages. b) Quality Control hold points. c) Additional post-maintenance testing.

3. Maintenance Rule - Inclusion in the Maintenance Rule scope, if not already included.

4. Change Control – Use of the 10CFR50.59 process, if not already required, to evaluate proposed changes.

5. Preventive Maintenance –Inspections and preventive/predictive maintenance activities that are targeted toward the critical attributes.

6. 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. 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, such as 10CFR50.59, are considered to be adequate to preserve these risk significant critical attributes.

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): (J. Winters)

To clarify the STP position, STP is requesting an exemption from the requirements of 10CFR50.65 for all LSS and NRS components. The exemption request will be modified to clarify that the exemption applies to both safety-related and nonsafety-related components that are classified as LSS or NRS.

RESPONSE (part b): (J. Winters)

The exemption request will be 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): (J. Winters)

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.

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: (M. McGehearty)

(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 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 continue to be used. 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.

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 <u>plant/system/train level functions</u> 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): (R. Chackal)

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 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 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 LOW 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): (B. Stillwell)

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. Information for the station performance indicators is developed from the information collected to support the Maintenance Rule Program . The PRA is exercised to support periodic evaluations of SSC reliability under the Maintenance Rule Program 50.65(a)(3). Insights gained from this data are used to quantitatively assess SSC reliability under both normal and off-normal conditions.

RESPONSE (part c): (R. Chackal)

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 subjected to the special treatment requirements. 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): (J. Winters)

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 GQA reviews. There are several methods by which degrading performance of LSS and NRS components affecting the Maintenance Rule is fed back into the GQA risk ranking process. First, the Maintenance Rule Coordinator participates in the GQA 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 reccurrence. 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 GQA Working Group.

RESPONSE (part e): (J. Winters)

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 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.

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): (J. Winters)

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 question 14b 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): (J. Winters)

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 GQA Working Group periodic reviews. See the responses to questions 14b and 44d for more discussion concerning the monitoring of the performance of LSS and NRS SSCs.

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: (R. Grantom)

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 ANLAYSIS 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, gauges	1,165,607,472	1,572	467,257,680	452	1.35E-06	9.67E-07	NO	NO	NO
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

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
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

11. *The licensee's exemption request states that "LSS components generally include piping, locked open valves, hand switches, and <u>outside containment isolation valves sized 3" and under</u> [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: (R. Chackal/A. Moldenhauer)

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 when any of the following criteria are met:

- The subject system is a closed water-to-water system and failure to isolate the line would not lead to a radiation release to the outside environment.
- The piping systems have a much higher pressure rating than the containment building.
- Redundancy exists with both an inboard and an outboard isolation valve.

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 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

UNIT 1 TAG/TPNS	SERVICE_DESC	PRA Rank	GQA Rank
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 <u>all</u> 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 good performance for all equipment 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. 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.

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: (G. Schinzel)

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, 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 the appropriate remedial action 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 six-month reviews to ensure that the applied controls are effective. It should be noted that the component's impact on current operating conditions is done 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.

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: (R. Chackal/R. Grantom)

(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 when any of the following criteria are met:

- The subject system is a closed water-to-water system and failure to isolate the line would not lead to a radiation release to the outside environment.
- The piping systems have a much higher pressure rating than the containment building.
- Redundancy exists with both an inboard and an outboard isolation valve.

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.

(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 adequate 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 GQA 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 GQA feedback process (component level monitoring). 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 GQA 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 GQA 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 ensure 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 isolation function will still be monitored by the Maintenance Rule at the component level for HSS and MSS components and at the system/train level for LSS and NRS components. 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	Monitored via Maintenance Rule at the system/train level. Not normally within the surveillance program.	Monitored at the component level of the Maintenance Rule. 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	Not in the Maintenance Rule scope at a component level. Monitoring is done at the system/train/plant level.	In the Maintenance Rule scope. Monitoring is at the component level.
Containment Leakage Performance Indicator	Not in the scope.	In scope at a component level.