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Reference: Draft Report NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10CFR50.59 (Changes, Tests, or Experiments)", dated April 1997

This letter submits comments from South Texas Project regarding the referenced report published by the Nuclear Regulatory Commission on proposed regulatory guidance for Section 50.59 of Title 10 of the Code of Federal Regulations (10CFR50.59), "Changes, Tests and Experiments". South Texas Project generally endorses the Industry position communicated through both the Nuclear Energy Institute response letter planned for transmittal on July 7, 1997 and the Region IV Engineering Managers Working Group response letter dated June 26, 1997. Attachment 1 to this letter provides specific South Texas Project comments on rule interpretations proposed by the regulatory guidance. We believe that a number of Nuclear Regulatory Commission positions expressed in Draft Report NUREG-1606 are inconsistent or represent new requirements without supportive regulatory analysis which could have potential significant impact on South Texas Project operations and implementation practices.

We believe that the proposed guidance of NUREG-1606 will significantly lower the threshold for changes which require Nuclear Regulatory Commission review prior to implementation. It is not clear why this proposed guidance is considered necessary, particularly in view of Nuclear Regulatory Commission comments in SECY-97-035 which concludes the present Industry guidance is generally sound and has given a reasonable foundation to establish a process that will produce effective evaluations related to changes in plant design or procedures. Albeit that the Nuclear Regulatory Commission considers the issuance of regulatory guidance necessary, we believe the end result will be that more changes will be identified as unreviewed safety questions because a new regulatory threshold is not met and not because an issue of safety importance is involved. Neither the Nuclear Regulatory Commission nor the Industry have resources to handle this increased workload. We believe the impact will actually be detrimental to safety in at least two aspects. First, this new threshold will divert resources to tasks of

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minimal, if any, safety benefit. Second, it appears likely that changes which actually improve safety can be designated as unreviewed safety questions requiring Nuclear Regulatory Commission review. As a result, these changes would either be cancelled or experience long delays awaiting Nuclear Regulatory Commission approval. Attachment 2 to this letter presents examples where the use of the proposed NUREG-1606 guidance would potentially delay making changes for little or no safety benefit.

Conservative implementation of the 10CFR50.59 process at South Texas Project has successfully preserved our design bases and safety margins. Although guidance can help ensure that 10CFR50.59 safety evaluations are performed in a consistent manner, we believe the proposed regulatory guidance is a change in regulatory requirements and a potential major impact on the operation of our facility with little benefit in improving safety. We believe that Nuclear Energy Institute's Draft NEI 96-07, "Guidelines for 10CFR50.59 Safety Evaluations", is the best approach for providing appropriate guidance and represents a well established and proven method of implementing 10CFR50.59. NEI 96-07 was promulgated as an improvement to the Industry's previous guidance NSAC-125. NSAC-125 has given the Industry a reasonable foundation and established a sound evaluation process that produces effective evaluations related to changes to plant design and procedures. Inspection results have confirmed that the quality of the evaluation of changes has improved since licensees began implementing the NSAC-125 guidance. We believe a consistent application of NEI 96-07 guidance will result in conservative compliance with the requirements of 10CFR50.59 even beyond that previously realized through the application of NSAC-125. We strongly recommend that the Nuclear Regulatory Commission work with the Nuclear Energy Institute to reach mutually satisfactory changes to NEI 96-07 guidance rather than generating a separate guidance document.

The present rule has preserved safety margins and has served well in maintaining the health and safety of the general public. Presently, rulemaking regarding 10CFR50.59 is not needed to preserve the design bases and safety margins of operating plants. However, we welcome the proposal of rulemaking to endorse the use of current Probabilistic Safety Assessment techniques in the determination for what constitutes an unreviewed safety question. We believe this has the potential to substantially reduce the resources applied to tasks with little or no safety benefit.

W T Cottle

W. T. Cottle
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KJT/

Attachments: 1. South Texas Project Comments Regarding Draft Report NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10CFR50.59 (Changes, Tests, or Experiments)"

2. Examples of NUREG-1606 Potential Impact

South Texas Project Comments Regarding Draft Report NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10CFR50.59 (Changes, Tests, or Experiments)"

Note: Subject paragraph numbers reference the numbers found in NUREG-1606

III.A Definition of Change

NUREG-1606 indicates the Nuclear Regulatory Commission staff has interpreted "change" to include any modifications or replacement of something, whether temporary or permanent, with something that is not identical to the original in design requirements. South Texas Project believes a replacement of plant equipment, components or parts with "equivalent" equipment, components or parts should not be considered a "change" governed by the requirements of 10CFR50.59. We consider an "equivalent" change to be defined as a hardware change that results in the installation of an item, not identical to the original item, which meets the design bases of both the item and applicable interface. This equivalency is determined by the engineering staff at South Texas Project.

A second comment regarding this section of NUREG-1606 concerns whether an activity being contemplated is a change rather than maintenance or an activity already reviewed. We believe that the Nuclear Regulatory Commission staff guidance needs clarification. One interpretation of the proposed guidance is that every work order activity not specifically described in the Safety Analysis Report or a Technical Specification Limiting Condition for Operation would require a safety evaluation. This is considered an overly burdensome expectation with little safety benefit. South Texas Project maintenance activities have sufficient controls to assess operability and the impact on plant safety. We believe a safety evaluation is required if the maintenance activity requires a deviation from a procedure described in the Safety Analysis Report or the results of the maintenance puts the plant in a condition where it functions differently than described in the Safety Analysis Report.

III.D Definition of Test or Experiment

The Nuclear Regulatory Commission staff's definition of a test or experiment in NUREG-1606 is not considered explicit enough. The decision whether tests or experiments not described in the Safety Analysis Report receive a 10CFR50.59 evaluation should not necessarily focus on the purpose of the test or experiment. Rather, the focus should be on the impact of the test or experiment on plant design bases and safety margins. Tests or experiments that have no impact on the design, function, or method of performing the function of a structure, system or component described in the Safety Analysis Report should not require 10CFR50.59 evaluation.

III.E Definition of "as described"

NUREG-1606 indicated the Nuclear Regulatory Commission staff concludes that a broad interpretation of the phrase "as described" is appropriate when evaluating proposed changes under 10CFR50.59. Although we do not categorically disagree with this conclusion, we believe the following are instances where changes can be made to information as described in the Safety Analysis Report without a 10CFR50.59 evaluation.

- Editorial changes, clarifications, and changes to the Safety Analysis Report which have no impact on safety-related systems, structures, or components should be possible without a 10CFR50.59 evaluation.

- All information provided in the Safety Analysis Report is not design basis information. Many areas of the Safety Analysis Report contain only descriptive information that has no impact on systems, structures or components. Changes should be able to be made to text or figures in the Safety Analysis Report without a 10CFR50.59 evaluation based on a determination that the design, function, or method of performing the function of systems, structures, or components is not impacted by the change.

III.F Definition of Final Safety Analysis Report

NUREG-1606 indicates the Safety Analysis Report description includes the text, tables, figures, and drawings. Most drawings in South Texas Project's Safety Analysis Report are provided for clarification and are not necessarily considered the description of system operation to meet the design bases. South Texas Project's believes that drawings in the Safety Analysis Report are provided for clarification and revisions to drawings should not be considered a change pursuant to 10CFR50.59 and can be made without a 10CFR50.59 evaluation as long as the change does not affect the design, function, or method of performing the function of a structure, system, or component described in the Safety Analysis Report.

III.I Malfunction of Equipment Important to Safety - of a Different Type

NUREG-1606 indicated that in determining whether a malfunction of a different type, the licensee needs to consider not only the "effect" of the malfunction on equipment or plant response but also what "causes" the malfunction. South Texas Project believes that the effect of the malfunction of equipment should be evaluated and not the potential cause of a malfunction.

III.M Role of Probabilistic Risk Analysis (PRA) in Section 50.59 Evaluations

South Texas Project does not agree with the explanation in NUREG-1606 why PRA is not suitable as a decision making tool. The definition of an unreviewed safety question inherently goes beyond the single failure criterion. The concept of "probability of ... malfunction of equipment important to safety" is not compatible with the assumption that the worst single failure is guaranteed while everything else works perfectly. The typical severe accident PRA analysis may not be applicable, but with the appropriate boundary conditions, success criteria, and failure data, PRA techniques can be successfully applied to evaluate the probability of accidents and equipment malfunctions within the scope of the design basis.

III.M Role of Probabilistic Risk Analysis (PRA) in Section 50.59 Evaluations (Continued)

Regarding the Nuclear Regulatory Commission staff position or guidance in the fourth paragraph of this section of NUREG-1606 which states: "Where PRAs were used as part of the basis for a previous licensing decision (as documented in the safety analysis report), facility changes that increase the related initiating event frequencies or equipment unreliability or accident consequences would constitute unreviewed safety questions"; South Texas Project believes that the proposed guidance is burdensome, inappropriate, and has little safety value. PRAs, as currently used, are best estimate rather than limiting analyses. At South Texas Project, we maintain a "living" PRA that evolves continuously, based not only on facility changes, but on plant historical data and methodology refinements and extensions. The parameters mentioned in NUREG-1606, initiating event frequencies, equipment unreliability and accident consequences, are not the same as Safety Analysis Report assumptions, but can and often do change with each PRA update. Therefore, 10CFR50.59 rules should not be applied to them.

South Texas Project has used PRA methods as part of the bases of Technical Specification changes, which had to do with allowed outage times to perform maintenance. The effect of these maintenance outages on core damage frequency is continuously monitored through the Comprehensive Risk Management Program as required by Technical Specifications. Although the effect of every facility change on the PRA is not evaluated, the risk due to changes that result in equipment outages, as allowed by Technical Specifications for equipment within the PRA scope, is continuously monitored.

The use of PRA has the potential to enhance plant safety and direct resources to focus on issues of safety importance. A risk informed 10CFR50.59 process is achievable and should be pursued. Risk informed criteria which could be considered for inclusion in such a process are:

- No new vulnerabilities are introduced,
- Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) does not change significantly (i.e., less than that recommended in the PSA Applications Guide or Regulatory Guide, DG-1061),
- Sequence of dominant contributors to CDF or LERF does not change,
- The current licensing bases conclusions relative to the reliability of risk significant components are not changed.

Applying inappropriate and restrictive rules on the use of PRA will discourage its use and ultimately have a negative impact on safety.

III.N Licensee Practice of Deleting Information from Safety Analysis Reports

NUREG-1606 indicated the Nuclear Regulatory Commission staff's position is that licensees may not remove material from safety analysis reports unless the material is changed as a direct result of a change to the facility. South Texas Project believes that obsolete and less meaningful information which would have no impact on the design, function, or method of performing the function of systems, structures, or components described in the Safety Analysis Report or would have no impact on procedures described in the Safety Analysis Report can be removed pursuant to 10CFR50.59. In fact, failure to remove such obsolete information detracts from the vitality of the Safety Analysis Report.

III.O Application of 10CFR50.59 to the Resolution of Degraded and Nonconforming Conditions

The Nuclear Regulatory Commission staff's position is that a 10CFR50.59 evaluation is required when a discovered nonconforming or degraded condition is not permanently resolved at the first available opportunity. Once the nonconforming or degraded condition is determined not to affect operability, South Texas Project believes the more appropriate response to these conditions is to take prompt corrective action required by Criterion XVI of Appendix B to 10CFR50.

The staff position regarding the situation of a plant currently operating with a condition involving an unreviewed safety question should recognize there are other resolutions in addition to submitting a license amendment such as restoring the condition to the licensing bases in a prompt manner commensurate with safety. In either case, the Nuclear Regulatory Commission should be expeditiously informed of the existence of any unreviewed safety question.

A plant should not be restricted from starting up if an unreviewed safety question exists provided Technical Specification requirements, design bases and plant safety margins are met. The Nuclear Regulatory Commission staff position that allow a plant operating with an unreviewed safety question to continue operation but would not allow a shutdown plant with an unreviewed safety question to restart appears inconsistent and may not adversely affect safety.

III.P Definition of Increase in the Probability of Occurrence

The increase in the probability of occurrence should be measurable for an unreviewed safety question to exist. The Nuclear Regulatory Commission staff's position that any uncertainty or doubt about whether an increase, even a neglible one, has occurred should lead to the conclusion that an unreviewed safety question is involved and will most likely result in an increased number of staff reviews of licensee submittals involving conditions where no clear indication of increase exists. In fact, a diversion of staff resources from matters of safety importance may be the result. Another result may be needlessly delaying a change that improves plant safety or reliability. In addition, the staff position appears to discourage the use of improved statistical tools to demonstrate that an increase is neglible. These tools are useful in facilitating engineering judgment. The staff's position presented in Section III.Q of NUREG-1606 appears to be a better approach to the increase in probability issue.

III.R Definition of Increase in Consequences

In the past, the Nuclear Regulatory Commission has indicated that the acceptance basis in their Safety Evaluation Report is the proper licensing basis for the plant. Any value for dose consequences which remains less than the acceptance basis should not be considered an unreviewed safety question.

If the dose calculated in the Safety Analysis Report is considered as the threshold for when an increase in consequences results, licensees should not be required to submit licensee amendments for past occasions when their Safety Analysis Report was revised pursuant to 10CFR50.59 and reported pursuant to 10CFR50.71? We believe the staff's position as stated in NUREG-1606 is a new requirement; and if approved through appropriate processes, should not be applied retroactively.

In NUREG-1606, the Nuclear Regulatory Commission staff noted that the staff Safety Evaluation Report conclusions are generally based upon independent calculations performed by the staff. It should be noted that calculations performed by the Nuclear Regulatory Commission staff are not normally subjected to the criterion of 10CFR50, Appendix B; whereas, the calculations performed by the licensee are subject to this regulatory requirement.

III.S Definition of Reduction in Margin of Safety

In NUREG-1606, the Nuclear Regulatory Commission staff indicated that if the staff's acceptance limit for determining margin of safety is not explicit in their Safety Evaluation Report, then the acceptance limit is the value as reported in the Safety Analysis Report. It should be recognized that licensees have reported detailed system or component performance results in their safety analysis reports that do not necessarily represent acceptance limits. South Texas Project believes that acceptance limits are the values per regulatory guidance or in the Nuclear Regulatory Commission's Safety Evaluation Report. Changes to detailed system or component performance analysis that result in values within a regulatory acceptance limit should not be considered an unreviewed safety question.

III.U Determination of Unreviewed Safety Questions When Licensees Use New Methods to Evaluate Plant Changes or Conditions

In NUREG-1606, the Nuclear Regulatory Commission staff indicated that in order to judge the effect of a change, test, or experiment when a new analysis methodology is used, the analysis must be done for the cases of before and after the change and both analyses must be performed with the same methodology. South Texas Project believes that new methodologies that have become accepted practice and are not inconsistent with the criteria of any similar methodology approved by the Nuclear Regulatory Commission should not have to be applied to the old design as well as the new design if the new design continues to meet the licensing and design basis of the system, structure, or component being changed. The focus should be a determination of whether the acceptance limits are met.

Examples of NUREG-1606 Potential Impact

EXAMPLE 1

MALFUNCTION OF EQUIPMENT OF DIFFERENT TYPE

Proposed Change

The existing pair of inline solenoid operated valves located in each of the bulk sampling lines from the Steam Generators will be replaced with one air operated valve. The existing solenoid operated valves have experienced numerous failures, such as inability to close, isolation failures, seat leakage, and other substantial problems affecting valve reliability. The air operated valves have no known failures as evidenced by a Nuclear Plant Reliability Data System search which produced no findings other than normal wear and aging of internal valve components. The Steam Generator Blowdown System performs no function associated with safe shutdown of the plant or to mitigate the consequences of an accident. Containment isolation and Steam Generator isolation are the only safety functions of these valves. The replacement valves are General Design Criterion 57 "(Closed System) Containment Isolation Valves" of 10CFR50, Appendix A. The air operated valves are normally closed and fail closed and receive a redundant Engineered Safety Features Actuation System signal to close on the same basis as the valves being replaced.

Evaluation of Change Using NEI 96-07 Guidance

Separate from a 10CFR50.59 evaluation, the replacement of the valves would first be evaluated to ensure that the change met all of the applicable design standards for the application of the system and would not result in a failure to meet design or safety requirements.

The proposed change would then be reviewed for a potential unreviewed safety question. Under the existing and previous guidance of NSAC-125/NEI 96-07 regarding a malfunction of a different type, it would be concluded that both the replaced valves and replacement valve in each Steam Generator Blowdown System fail closed in the safe direction. The replacement valve has no impact on the piping and supports, and meets the appropriate seismic criteria and environmental requirements for the room. Therfore, the effect of the malfunction of the valve on the supported system is the same as the replaced valves. The change would not result in an unreviewed safety question.

Evaluation of Change Using NUREG-1606 Guidance

In the guidance of Section III.I.4 of NUREG-1606, the Nuclear Regulatory Commission indicates that the licensee in determining a malfunction of a different type needs to also evaluate the "cause" of the malfunction in addition to the effect of the malfunction. In this case, the cause of the malfunction now includes a loss of instrument air. This would be considered to involve an unreviewed safety question.

Impact of Using NUREG-1606 Guidance

A change to improve system safety and reliability for ensuring Containment isolation and Steam Generator isolation when required would be delayed while waiting for Nuclear Regulatory Commission review and approval of a license amendment. The new "cause" of the malfunction still results in the more reliable valve failing in the safe condition.

EXAMPLE 2

INCREASE IN CONSEQUENCES/NONCONFORMING CONDITION

Proposed Change

Nuclear Regulatory Commission Information Notice 91-56 was issued regarding potential radioactive leakage to tanks vented to atmosphere. Licensees were alerted to potential problems resulting from the leakage of isolation valves in emergency core cooling system recirculation lines to the safety injection water storage tank, which may be vented to atmosphere. The Information Notice stated at least one licensee concluded that the dose rates resulting from the estimated valve leakage, together with other assumed radiological sources during a maximum hypothetical accident, could cause "regulatory limits" to be exceeded. Thus, an unreviewed safety question was determined to exist. As a result of this information, a second licensee with a different plant design adds an assumption of the new potential valve leakage paths in evaluating the radiological consequences of a loss of coolant accident described in their Safety Analysis Report. The second licensee's evaluation concludes that the radiological consequences do increase but remain well below (i.e., "a small fraction of") the limits of General Design Criterion 19, "Control Room", of 10CFR50, Appendix A and 10CFR100.

Evaluating the Change Using NEI 96-07 Guidance

Upon receipt of the Nuclear Regulatory Commission Information Notice, a Condition Report would be written and the potential condition would be evaluated for operability/reportability.

The next action would be to evaluate the condition for an unreviewed safety question using the NEI 96-07 Guidance. To determine whether an increase in consequences is involved, the action would be to determine whether the Nuclear Regulatory Commission reported the radiological results in the Safety Analysis Report were the acceptance limit, the regulatory limits of General Design Criterion 19 of 10CFR50, Appendix A and 10CFR100 were the acceptance limits, or whether the Nuclear Regulatory Commission staff was silent. As long as the staff had not established the acceptance limit of the previously reported radiological consequences in the licensee's Safety Analysis Report, the licensee could go up to the General Design Criterion 19 of 10CFR50, Appendix A and 10CFR100 limits without it being an unreviewed safety question. Therefore, no unreviewed safety question existed as the result of the second licensee's evaluation.

Evaluating the Change Using NUREG-1606 Guidance

The approach taken in Section III.R.4 of NUREG-1606 indicates that any value above that reported in the Safety Analysis Report, no matter whether the Nuclear Regulatory Commission accepted a higher value, would be considered an unreviewed safety question. Therefore, a small change in radiological consequences with the new result remaining a small fraction of regulatory limits becomes an unreviewed safety question because it is an increase in the dose previously reported in the Safety Analysis Report.

Impact of Using NUREG-1606 Guidance

The approach taken in Section III.O.4 of NUREG-1606 indicates the staff would not allow plant startup with the existence of an unreviewed safety question until staff approval is received. If ready to restart following an outage when the evaluation results regarding Nuclear Regulatory Commission Information Notice 91-56 were completed, the licensee would be required to remain shutdown for reasons of no safety significance.