CEOG COMBUSTION ENGINEERING OWNERS GROUP

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CEOG Comments Related to Draft NUREG-1606

Final Report

Task 1000

Prepared for the
C-E OWNERS GROUP
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ABB Combustion Engineering Nuclear Operations

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Compiled By

Ian C. Rickard, Project Manager

Contributions from:

ABB
Arizona Public Service Co.
Baltimore Gas & Electric Co.
Consumers Energy Co.
Entergy Operations
Florida Power & Light
Maine Yankee Atomic Power Co.
Northeast Utilities Service Co.
Omaha Public Power District
Southern California Edison Co.

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Executive Summary

The purpose of this report is to provide the comments of the Combustion Engineering Owners Group (CEOG)[1] on the NUREG-1606 guidance for implementation of 10 CFR 50.59[2]. Section 50.59 has been successfully implemented by the US domestic nuclear power industry since 1962. Since 1989 the industry has used NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations" to provide guidance to licensees on the performance of 10 CFR 50.59 evaluations. Although the NRC has never endorsed the use of NSAC-125, NRC has praised the quality of evaluations performed under the guidance of NSAC-125[3,4]. A recent evaluation by NEI[5] confirms that the industry guidance has a strong legal basis and warns against the unintended adverse consequences of modifying a long standing regulatory scheme.

The CEOG believes that the new guidance proposed in NUREG-1606 will result in a large increase in the number of changes that are determined to be Unreviewed Safety Questions. The result will be a large increase in paperwork and reduced attention by the NRC staff for safety-significant items. In addition, because of the increased burden of the regulatory process, licensees will be less likely to introduce changes that will have a positive impact on the safety of the plant. The CEOG believes that the overall impact of NUREG-1606 will be detrimental to the public health and safety and will result in increased costs for both licensee staffs and the NRC.

The NRC staff has stated that the re-evaluation of the implementation of Section 50.59 results from the findings of the Millstone Lessons Learned Task Group[6]. In fact, a review of that report shows that NUREG-1606 would have had little or no impact on the events at Millstone. Rather than modifying the current guidance, more uniform and strict adherence to the guidance of NSAC-125 would achieve the full intent of 10 CFR 50.59. CEOG recommends that the NRC abandon the guidance of NUREG-1606 and endorse the continued use of the industry guidance contained in NSAC-125 or the recently issued NEI 96-07.

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[1] Comments were supplied on behalf of ABB (as a non-voting member of CEOG), and other members of the CEOG and were compiled on behalf of CEOG by ABB.

[2] NUREG-1606, Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests, or Experiments), Draft Report for Comment, U.S. Nuclear Regulatory Commission, April 1997.

[3] Brian K. Grimes (NRC) to Thomas E. Tipton (NEI), NSAC-125 Guidelines for 10 CFR 50.59 Safety Evaluations, June 25, 1993: letter states that "....because some of the guidelines in NSAC-125 go beyond the requirements of 10 CFR 50.59, we do not believe the guidelines are appropriate for endorsement as regulatory guidance." : in other words, NSAC-125 exceeds the minimum requirements of 10 CFR 50.59.

[4] NEI has re-issued NSAC-125 as NEI 96-07. NEI 96-07 (Draft Revision C), June 1997. Revision C contains wording changes to clarify and strengthen the guidance of NSAC-125. In general in this report, NEI 96-07 and NSAC-125 can be used interchangeably.

[5] Analysis of Industry Guidance for Implementing 10 CFR 50.59, Draft NEI Report, May 1997.

[6] Millstone Lessons Learned Task Group Report, Part 1: Review and Findings, U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation, September, 1966.

Overall Comments

Scope of the Impact

Almost any engineering or design analysis is subject to a review under Section 50.59 if it results in a change to a parameter or description contained in the SAR. ABB CE provides NSSS design engineering services, nuclear fuel, reload fuel engineering services and other general engineering services which may be subject to review under Section 50.59. Since ABB is a vendor rather than a licensee, ABB does not maintain its own 50.59 implementing procedures. However, ABB engineers must be familiar with the requirements of Section 50.59 and ABB engineers frequently provide input to a licensee's Section 50.59 review. Typical areas where ABB might be called upon to provide input to a Section 50.59 review, using utility and NSAC-125 guidelines, are listed below:

- Cycle specific reload application (as a whole, the reload being considered the change)
- Acceptability of various cycle specific changes as input to safety related evaluations for the reload (e.g. changes to peaking factors, fluences etc.)
- Transients and accidents impacted by changed core inputs as a result of the cycle specific reload design
- Changes to reload analysis groundrules
- Changes to the COLR
- Differences between the as-operated core and that assumed in the reload analysis (the reload analysis is performed assuming a projected operational history; a 50.59 evaluation may be necessary to evaluate the effect of differences between the actual operating history and that assumed in the reload analysis)
- Lead test assemblies
- Changes to mechanical design of the fuel (minor changes to fuel rod length, pellet design, welds, grids etc.)
- Degraded equipment, e.g., reduction in number of operable In Core Instruments, changes in RTD responses
- Effect of changes in response of various equipment as a result of equipment aging and/or surveillance (e.g., changes in valve tolerances, time response or lift pressure)
- Acceptability of replacement equipment
- Changes in approved analysis methodology including changes to design input assumptions
- Changes to analytical biases and uncertainties

All of the above areas will be impacted by NUREG-1606.

The first step is to determine a) that the change is safe and b) that the change does not involve a change in Technical Specifications. If not, a Section 50.59 review is performed.

This review is performed on behalf of the licensee using NSAC-125 and the licensee's procedure for such review. Although there are differences amongst the procedures used by each utility as well as differences in application/scope, it is our experience that these reviews are performed in a professional and conscientious manner consistent with the intent of Section 50.59. It is our experience that in the majority of cases, the licensee reaches the conclusion that no Unreviewed Safety Question exists. **In the opinion of the CEOG, the new interpretation of Section 50.59 presented in NUREG-1606 will result in a large increase in the number of such changes being classified as Unreviewed Safety Questions and a very large increase in the volume of paperwork and review requested from the NRC staff**. We do not believe that there is any corresponding increase in plant safety or regulatory oversight improvement which would justify such a change. In fact, the increase in paper work may detract from a utility's desire to implement beneficial plant changes and would certainly detract from the NRC's ability to monitor real safety issues.

Relationship to the Millstone Lessons Learned Report[7]

NUREG-1606 refers to the Millstone Lessons Learned as the reason why additional Section 50.59 implementation guidance is needed. However, there is no direct correlation shown between NUREG-1606 (i.e., change in interpretation of 50.59 as it has been applied since 1962) and Millstone problems. More appropriate guidance might result if the NRC staff clearly identified the Millstone problems and the changes proposed to fix those problems. Without such a correlation, one must conclude that much of NUREG-1606 is designed to fix something that isn't broken.

In October 1993, Northeast Utilities (NU) submitted a licensee event report[8] indicating that the plant had operated outside of the plant's design basis during refueling outages. Although these practices were of little actual safety significance, they raised questions of regulatory significance. The report of the Millstone Lessons Learned Task Group discusses the regulatory issues raised by the Millstone event. The review uncovered considerable confusion and divergence of opinion on the role of the updated FSAR, the proper definition of its contents and its relationship to both the plant design basis and licensing basis. As a result, the report recommends that the interpretation and implementation of Section 50.71 (e) should be reevaluated. With regard to 10 CFR 50.59, the report does not implicate current industry guidance using NSAC-125. Instead the report finds that, as a policy issue, NRC should examine the relationship between the implementation of Section 50.59 and the contents and reporting requirements for the FSAR. In addition Part 1 concludes that additional guidance is required with respect to existing and as-found conditions and to clearly establish actions the agency expects licensees to take in resolving degraded or non-conforming conditions, including the role of Section 50.59.

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[7] Millstone Lessons Learned Task Group Report, Part 1: Review and Findings, U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation, September, 1966.
[8] LER 93-11

The Part 2 report[9] observes that "Implementation of Section 50.59 also is affected by the variability in FSARs." The proposed action to resolve this is to ensure licensees are updating their FSARs with the appropriate information; determine if it is necessary to establish a standard level of detail for FSAR updates; determine if additional information should be added to updated FSARs. Again, this does not point to any problems with the current industry guidance with respect to Section 50.59.

Parts 1 and 2 of the Millstone Lessons Learned Task Group report do not contain any conclusions that justify the need to modify current industry practice and interpretation of Section 50.59. The lessons learned suggest that the desired improvements could be achieved by NRC endorsement of NSAC-125 (or NEI 96-07[10]) together with clarification on the requirements for handling degraded or non-conforming conditions. We concur with NRC's belief that additional guidance is required with respect to the role of the FSAR, FSAR updates and contents.

Format of the Report

The format of NUREG-1606 is awkward and confusing. The report identifies 22 interrelated "issues or concerns" together with 2 related policy issues. Although the report does a good job of identifying possible issues and laying out the industry position as well as the proposed NRC position, this type of word by word analysis does not produce good regulatory guidance. The overall intent and impact of the rule depends on the entirety of the wording, not on a precise literal interpretation of each word. Any final guidance should be focused on regulatory clarity rather than legal justification. NRC should take into account that industry has successfully implemented 50.59 since 1962 and that a reinterpretation of the rule will result in confusion and unexpected adverse consequences. The CEOG recommends that NRC endorse the continued use of the guidance contained in NSAC-125 or its successor document, NEI 96-07.

Detailed comments on the contents of NUREG-1606

I. Introduction

The objectives of the guide are listed as follows:

- Reaffirm existing regulatory practice
- Clarify NRC staff's expectations and regulatory positions in areas where industry practice or position differs from the NRC staff's expectations for implementation of 10 CFR 50.59.

[9] Millstone Lessons Learned Report, Part 2: Policy Issues, Nuclear Regulatory Commission, February 1997.

[10] NEI 96-07 (Draft Revision C), Guidelines for 10 CFR 50.59 Safety Evaluations, Nuclear Energy Institute, June, 1997.

- Establish guidance in areas where previous guidance did not exist

These objectives are fine in so far as they go. However, ABB believes that the NRC staff should clearly identify those areas where actual industry practice has failed to live up to the legal intent of 50.59, and then show how each proposed change would rectify the resulting safety concerns. There have been 30 years of experience with the implementation of Section 50.59, yet only a handful of abuses have actually been identified. Adequate guidance exists currently and industry practice is overwhelmingly in compliance with the intent of 10 CFR 50.59.

II. Relationship of Review of Changes for Effects on Safety and for 10 CFR 50.59 Evaluation Purposes

The CEOG unequivocally endorses the fact that in evaluating a proposed new activity or change, a licensee is responsible for determining that the change, test or experiment is safe. We also agree that this is the first step in the change process that must be completed. We believe that this position is consistent with industry practice (as stated in NSAC-125) and we also believe that the industry appreciates the distinction between determining that a change is safe and determining that no Unreviewed Safety Questions exist.

III. Discussion of Regulatory Issues or Concerns

NUREG-1606 identifies 22 interrelated "issues or concerns" with respect to the industry's implementation of Section 50.59. In general, the CEOG endorses the detailed comments generated by NEI[11]. The following are the specific comments on the NRC "issues and concerns" which have been generated by the CEOG.

A. Definition of Change

NRC proposes to interpret "change" to mean any modification or replacement with something that is not identical to the original design requirements. Presumably, "no change" would be involved and no 10 CFR 50.59 evaluation would be required if a component is replaced with an identical component. The control of the facility configuration during an identical component replacement is maintained by compliance with the Technical Specification and not the SAR. For example, when the unit is shutdown typically only one train of safety-related equipment is required to be operable per the Technical Specifications. This allows extensive work to be accomplished on the inoperable train with limited risk concerns. NRC should recognize a clear distinction between maintenance and modification activities, and permit functionally identical replacements as maintenance activities. One utility estimates that this definition change alone would require about five hundred full 50.59 evaluations for routine plant work that is currently screened out as not needing full 50.59 evaluation because it is not a change.

[11] Industry Comments on NUREG-1606 (Draft C), Nuclear Energy Institute, June 4, 1997. These were, in part, based on: Draft Comments on NRC Letter SECY-97-035, John H. Hesser (APS) to Ralph Beedle (NEI), April 22 1997.

In summary, the CEOG recommends that replacement with functionally equivalent components should be exempted from a 10 CFR 50.59 evaluation since no change is being proposed by the licensee and compliance with the Technical Specifications provides configuration control. It is noted that this does not obviate the need or the responsibility of the licensee to determine the safety of such intermediate configurations: rather, it avoids the paperwork associated with obtaining NRC approval of such intermediate configurations.

In addition, the CEOG believes that "changes" that would normally be considered routine maintenance and be subject to the Maintenance Rule should not be considered a change under Section 50.59. (See example, below).

Example[12]

Definition of What Constitutes a Change

A plant has three non-Safety Related condensate pumps installed, with only two pumps required for full power operations. The third pump has a standby feature which allows it to automatically start in the event of failure of the installed pumps. The third pump and the standby feature are described in the UFSAR, which notes that "normally, two pumps are running with the third in standby." The plant's accident analysis does not, however, credit the response of the standby pump in evaluating any accident. The licensee desires to conduct preventive maintenance on the one standby pump while the reactor is on-line.

Because the pump is not subject to any Tech Spec action statements and does not affect the operability of any Tech Spec components, under current practice, the licensee has considered routine maintenance, whether corrective or preventive, consistent with the design basis purpose of having a third pump. Therefore, the third pump's removal from service has not been considered a change, test, or experiment subject to 50.59. If its removal had been considered under 50.59, the potential for a plant trip should either of the other pumps fail is increased, although the increase in probability of an accident could reasonably be judged negligible. Maintenance performed on the pump, including system unavailability, would be subject to the Maintenance Rule.

The guidance of NUREG-1606 would apply 50.59 to maintenance and corrective action configurations, and the unavailability of the standby pump would reasonably be considered to increase the probability of a loss of feedwater event and a plant trip. This would require determination that

[12] Example recorded by Region 1 Engineering Managers and reported by John Osborne (BGE), May 1997.

it constitutes a USQ. The likely result would be that the licensee would seek explicit recognition of the maintenance configuration in the UFSAR via license amendment. This would be effectively the equivalent of seeking a Tech Spec Limiting Condition of Operation for this non safety-related component as a means of obtaining NRC concurrence with its maintenance intentions. Alternatively, the licensee could forego any repairs except when one pump was clearly non-functional and then seek NRC approval for repair.

B. Definition of Facility

The NRC proposes in NUREG-1606 to define the term "facility" to include all SSCs, the requirements for their design, construction and operation and the design bases and safety analysis information associated with those SSCs that are described in the SAR. This is potentially very confusing and the use of the term "design bases" adds to the possible interpretations of this proposed guidance.

ABB recommends that NRC clarify the intended use of the term "facility" and carefully distinguish between changes in Systems, Structures and Components (SSCs) and changes in analyses, models and "design bases".

Examples

See examples under Section U, below, Use of New Methods.

C. Definition of Procedures

In Section III.C.4 of NUREG-1606, when discussing certain procedures and programs described or referenced in the SAR, the NRC staff states that specific licensee programs, such as emergency preparedness plans, security plans, and quality assurance plans have change control processes explicitly established by regulation (viz., 10 CFR 50.54). These change control processes have different criteria than 10 CFR 50.59 and are generally more restrictive than the 10 CFR 50.59 criterion. The CEOG believes that for consistency and clarity, the NRC staff guidance should list all such cases when a 10 CFR 50.59 evaluation is not required due to deferment to another controlling regulation.

D. Definition of Test or Experiment

Simple data collection from a system that is in its normal operating alignment should not be considered a test or experiment in the context of 10 CFR 50.59. The act of collecting data cannot cause the system to be incapable of performing or accomplishing its normal function(s).

E. Definition of "as described"

NUREG-1606 concludes that since an SSC that is not described in the SAR can affect others that are - if a change to one part of the facility results in some other change to "the facility as described in the SAR", the first change is within the scope of 10 CFR 50.59. The change could also be at a level of detail that is not explicitly described in the SAR, but could affect a function or SSC that is described. The intent of this guidance appears to be consistent with NSAC-125.

However, the NRC appears to go beyond the industry guidance in the discussion of levels of detail not found in the SAR. To apply the rule to changes that are at a level of detail below that described in the SAR is illogical. Presumably these details were never explicitly reviewed by the NRC for the original configuration and it should not be necessary to review changes at this level of detail. The CEOG recommends that NRC adopt the industry guidance on this issue. In addition, we recommend that NRC clarify the intent of the guidance by providing examples of changes that do not fall within the scope of 10 CFR 50.59.

F. Definition of Final Safety Analysis Report

The CEOG has no comments on this issue.

G. Industry Use of a Screening Process

The CEOG has no comments on this issue.

H. Definition of Accident Previously Evaluated

Problems can arise in the evaluation of changes when the SAR does not include a complete record of all previously evaluated transients. A SAR typically contains descriptions of limiting events (relative to some criterion) in a number of different categories. It is possible that an event scenario can become more limiting after the change.

> Because of the potential for problems in this area the CEOG recommends that the NRC clarify the guidance in this area and emphasize the relationship to the guidance under "as described".

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

The Design Basis event sequences presented in the FSAR and the SRP are selected conservatively, as are other input and methods assumptions. As a result, the Design Basis analyses presented in the FSAR are conservative relative to what would be expected to occur should a plant experience one of the initiating events. The development and

analysis of steam generator tube rupture events is a good example. We believe that the underlying intent and benefit of this conservatism is that the results reported in the FSAR envelope the many sequences not specifically analyzed. Therefore, we believe that a significantly different mode of failure or sequence of events not previously analyzed could represent a USQ, but a different mode of failure or sequence that is similar to what has been analyzed would not be a USQ. Again, judgment is required; it cannot be legislated without undue burdens on both the industry and NRC staff.

In general, the mechanistic form of the failure was not considered in the SAR analysis. The SAR analysis is specifically concerned only with the outcomes - the performance of the safety systems - not with what caused a piece of equipment to be unavailable. Since the original initiator is usually not described explicitly in the SAR (and has not been reviewed in detail by the NRC staff), changes to the form of initiators are usually irrelevant to the sequence of events presented in the SAR.

J. Licensee Implementation of Modifications Associated with Technical Specifications

The CEOG has no comments on this issue.

K. Need for Plant Specific 50.59 Evaluations When Implementing Generic Modifications

The CEOG has no comments on this issue.

L. Licensee Identification of Technical Specifications That are not Adequate to Assure Compliance with the Design Bases

The CEOG has no comments on this issue.

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

The CEOG agrees that PRA does have a role to play in some Section 50.59 evaluations. In particular, PRA is uniquely suitable for providing insights into initiating event frequencies, equipment reliability and system unavailability. This information can play an important role in answering the questions related to an increase in probability of occurrence. CEOG expects that PRA techniques will increasingly be used to provide risk insights and that risk informed regulation will increasingly replace deterministic safety evaluations. In particular, changes could be evaluated by comparing the relative risk of the new design to that previously reviewed and licensed by the NRC staff.

The CEOG recommends that future rulemaking on Section 50.59, if required, include changes to allow the use of probabilistic risk analysis . This might include the use of a

"risk significance"[13] test to differentiate those changes that would require prior NRC approval from those changes that might be allowed without prior approval, but subject, perhaps, to a more timely reporting requirement.

N. Licensee Practice of Deleting Information from Safety Analysis Reports

In Section III.N.4 of NUREG-1606, when discussing the deletion of information in the SAR, The NRC staff states that "there is no established policy or guidance with respect to removal of information from the FSAR not associated with changes to the facility or procedures." Regulatory Guide 1.70 Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," however, provides the general expectations regarding the format and content of the FSAR. General expectations include that the information in the SAR be (1) timely, (2) accurate, (3) complete, and (4) organized in a format that provides easy access. Timeliness is addressed in 10 CFR 50.71(e) and accuracy is addressed through 10 CFR 50.9. Although completeness and ease of access are not sufficiently specific requirements, newer plants can address these two issues by using the format delineated in Regulatory Guide 1.70 and the level of detail for the various sections in NUREG-0800, "Standard Review Plan for the review of Safety Analysis reports for Nuclear Power Plants."

The CEOG believes that "deleting" selected information is appropriate and advisable to enhance the ease of access and the accuracy of information in the SAR.

Example:

(a) *Information may be deleted from the SAR to avoid confusing or ambiguous statements, or unnecessarily verbose descriptions that do not contribute to expeditious technical review and understanding. In other cases, it is convenient and more appropriate to provide the technical bases in a separate report that is referenced in the SAR. This is particularly true if proprietary information is involved. Referenced proprietary reports should be listed at the end of the SAR section in which they are referenced.*

(b) *Another reason to "delete" information in the SAR is to avoid duplication of information. While similar or identical information is relevant to more than one section, it is acceptable and is often advisable that information be presented only once (in the principal section) in the SAR and then referenced by other sections. This helps to ensure that the SAR remains internally consistent.*

[13] NUREG-1606, Section IV.B.3, this option is discussed by the NRC staff as a possible policy issue for rulemaking.

O. Application of 10 CFR 50.59 to the Resolution of Degraded and Nonconforming Conditions

Section III.O.4 of NUREG-1606 states that a 10 CFR 50.59 evaluation is required when the licensee decides to accept a nonconforming condition rather than restore the plant to its SAR-described condition. The plant is always in a constant state of "change." This change is circumstantial. That is, several Age-Related Degradation Mechanisms (ARDMs) (e.g., erosion, corrosion, fatigue, etc.) are continuously affecting the facility. Significant ARDMs are monitored, controlled, and mitigated by programs either established voluntarily by the licensee or required by other regulatory requirements. The results of these programs feedback into a determination of system operability, as necessary. Some operability determinations derive directly through Technical Specifications surveillance results and others emerge indirectly through the licensees corrective action program, which is required by 10 CFR 50 Appendix B.

Circumstantial changes should not require a 10 CFR 50.59 evaluation because they do not satisfy the language of the rule which reads that "a holder of a license...may make changes in the facility as described in the SAR...". Changes due to ARDMs are circumstantial and are not pre-meditated changes proposed by the licensee.

<u>The CEOG recommends that degraded and nonconforming conditions must be evaluated based on an "operability" test and should not be subject to the "regulatory" test of 10 CFR 50.59 for which it is not intended. It should be recognized, however, that any compensatory actions taken by the licensee to assure operability may be subject to 10 CFR 50.59 if they constitute a "change to the facility as described in the SAR".</u>

Example[14]:

Use of 50.59 for Degraded Conditions and Compensatory Measures for Degraded Conditions

> *The level transmitter for one Reactor Coolant Pump oil reservoir failed while at power. The transmitter provides an alarm and indication function, but not an automatic protective action function. The alarm is described in the UFSAR, but no Tech Spec applies. Loss of the transmitter does not result in the loss of OPERABILITY for any Tech Spec equipment. The transmitter fails in a direction resulting in a hanging alarm in the Control Room. Alarm circuitry provides one common alarm for the upper and lower reservoir circuits, so the failure causes loss of alarm indication for one functional transmitter in addition to the failed circuit.*

[14] Example recorded by Region 1 Engineering Managers and reported by John Osborne (BGE), May 1977

The necessary action to restore the alarm from the functional channel is to lift a lead from the failed transmitter. Current practice justifies this on the basis that:

a. *it restores the functional channel to the condition described in the UFSAR and,*

by substituting periodic (monthly) boroscope inspections to verify oil level of the affected channel and monitoring of bearing temperature indication, the loss of continuous monitoring results in a negligible increase in the probability of a Reactor Coolant Pump seized rotor. Time required to lift the lead and restore the functional alarm circuit was typically a few hours.

Under NUREG-1606, because the RCP Oil Level Transmitter ***as described in the UFSAR*** *is fully functional, lifting the leads to its alarm would be analyzed as removing a described alarm function. Substitution of periodic (monthly) monitoring for continuous monitoring would result in a slightly increased probability of an accident previously analyzed (RCP rotor seizure) and an increased probability of malfunction of equipment important to safety (the failed reservoir transmitter). The changes in probability would be negligible but the staff's proposal considers that any increase in probability, regardless of extent, is an USQ. This would result in treating the lifted lead as a USQ requiring NRC approval before clearing the hanging alarm. While NRC concurrence for such minor matters has not heretofore been required, past observation indicates that a substantial delay would occur because of the need for NRC staff approval. Safety would be degraded by this delay.*

P. Definition of Increase in the Probability of Occurrence

ABB-CE believes that the NRC staff's proposal to require that any increase in the probability of an event must be considered a USQ runs counter to the NRC's recent efforts to encourage risk-informed management. Users of the 50.59 process should be given clear guidance that allows for use of risk assessments or, in some cases, judgment. We are not aware that 50.59 users have significantly misused the privilege of judgment in the past and, therefore, we encourage continued use of judgment and risk assessments in the future. Users should be encouraged to use risk assessment methods when changes cannot be clearly assessed with deterministic analysis or judgment. Use of risk assessment methods may be feasible and useful in quantifying the risk associated with an increased probability, but the feasibility of using such methods may be case-dependent and should not be required. If NRC's normal inspection process identifies a user that is not making proper judgments, then corrective action may be appropriate for that user; it does not appear necessary to regulate the whole industry to the lowest performer.

Example:

> *Consider a change to a plant parameter, say the MTC. The plant transient response will change as a result. How does this, for example, affect the probability of a pipe break? The FSAR establishes the pipe design (or the plant transient response) such that the pipe is not expected to break. But the pipe break is evaluated as an accident, not knowing mechanistically how or why the pipe breaks. While the transient response changes due to the MTC change, the transient response remains acceptable (both because the MTC limits are Tech. Specs. and the response is assumed in this example to be always within the acceptance criteria). Currently, we would conclude that there is no increase in the probability of an accident. But if the effect of the MTC change is to make the transient more severe (or increase the time during the cycle that the transient is more severe) it is difficult to argue that although no pipe failure is expected (because the transient remains less severe than that expected to fail the pipe) that the probability of a pipe break "may" not increase however insignificantly. After reading the NRC guidance, it is unclear whether this scenario would be judged a USQ. We do not believe that this is the intended effect but the NUREG-1606 wording lacks regulatory clarity.*

Q. Increase in Probability Still Within Design Basis

NUREG-1606 would define any instance where the probability of an accident may increase as a USQ. This normally will not have a significant impact on the ABB scope of evaluations since most of what is addressed by ABB are analytical changes that affect consequences, rather than probability (however, see example quoted under Section III.P). But this could have a larger impact on utilities especially as they replace obsolete equipment with similar, but not identical equipment.

R. Definition of Increase in Consequences

While increases in analysis results that were originally close to an SRP limit may very well represent a USQ, small increases in consequences that are far from limits should not be classified as a USQ. Analysts understand well the difference between an analysis result that is close to a limit and one that is far from a limit; and that the level of detail in FSAR and SER documentation reflect the "closeness" to a limit. Based on experience at ABB-CE and CEOG licensee experience, analysts are conscientious and conservative with respect to their judgments as to whether the analysis results are close or far from a limit. The normal NRC inspection process should be able to identify any abuse of this process. On the other hand, requiring that any increase in consequences be classified as a

USQ would create an undue burden on both licensee and NRC staff resources that could be better spent on other issues more closely related to safety.

NUREG-1606 would define as a USQ any increase in consequence beyond that quoted in the FSAR. NUREG-1606 results in a requirement for prior NRC approval of insignificant changes. A typical reload will probably result in such insignificant changes to one or more event consequences. The main impact of this would be to force the reload analysis to be performed much earlier to allow for the possible necessity of NRC review (would this take 6 months?, 9 months?). Most utilities routinely conclude that a reload does not involve a USQ. NUREG-1606 would invalidate that conclusion and require lengthy NRC review of the insignificant changes associated with a core reload. <u>The bottom line is that if the NRC staff believes they have to review insignificant changes, then the NRC staff should develop a process for rapidly scheduling and reviewing such changes</u>.

When discussing the issue of whether a change could increase the consequences of an accident previously evaluated in the SAR, the NUREG-1606 states that "the staff does not generally approve the methods or results of SAR analysis, but finds the consequences of the accident acceptable if the staff-calculated results meet the applicable acceptance guidelines." The NRC staff notes that this fact would make it difficult for a licensee to consider the NRC acceptance value (i.e., 10 CFR 100 limits) rather than the NRC staff results as the benchmark for determining whether an increase is within the bounds of what the NRC staff has previously reviewed. Based on this logic, the NRC staff-calculation becomes the Design Basis Document of record for the facility with respect to radiological doses. If the NRC staff-calculations are Design Basis Documents, the licensee needs to understand the assumptions, methods, and results of the NRC staff calculations in order to determine if there is any increase in consequences. Rather than DBDs, NRC staff-calculations are independent assessments[15] to determine the reasonability of licensee results which are the Design Basis Documents. Without this conclusion, the licensee would require full access to the NRC staff-calculations in order to be able to properly perform a 10 CFR 50.59 evaluation.

<u>The CEOG believes that a judgment of increase in consequences should be based on the NRC acceptance values and not on the previously reported calculation results reported in the SAR.</u>

Examples:

Use of UFSAR as sole basis to define acceptable level of consequences resulting in inconsistent application of guidance:

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[15] These calculations, in effect, constitute an alternate analysis for purposes of verification, similar to those prescribed in 10 CFR 50 Appendix B.

a) *At least one utility interprets 50.71(e) as requiring the FSAR to be updated to reflect any analyses re-performed as part of the reload analysis (others change the FSAR only if the results of an analysis are materially different or results are more adverse). As a result, the FSAR may be updated to show event results (consequences per the NRC guidance) less severe than acceptance criteria and less severe than that shown previously in the FSAR and previously reviewed by the NRC. In subsequent cycles, results may become more severe than that now shown in the UFSAR, but still within those results previously approved. If the UFSAR is the sole basis for judging if event consequences have increased, this would be a USQ, even though the NRC has previously reviewed and accepted this event with more severe consequences.*

Specifically, in this example the original FSAR analysis showed a limited amount of fuel failure for an event and the associated radionuclide release. No fuel failure was predicted in subsequent reload cycles, and the FSAR and associated releases were updated. In a subsequent cycle, a small amount of fuel failure was predicted. As a result, releases increased above that in the FSAR, but were well within those approved in the original FSAR analysis, and well within NRC acceptance criteria.

It appears that this would be a USQ under the NUREG-1606 since the consequences were greater than that in the (current) FSAR. Further, since the predicted consequences of events will always change somewhat from cycle-to-cycle[16]*, sometimes becoming slightly more adverse, sometime less, efforts to diligently update the FSAR will always lead to USQs under NUREG-1606.*

b) *The analysis demonstrates a very small amount of fuel failure and consequently releases are predicted to be much less than criteria. These releases are reported in the FSAR. The SER concludes that the releases are acceptable because they are less than that allowed by the SRP. In a subsequent cycle, the amount of fuel failure increases slightly, resulting in releases that exceed those reported in the FSAR but that are still consistent with those of the SER (i.e., less than that allowed by the SRP). If the FSAR is the sole basis for judging acceptable consequences, this becomes a USQ.*

[16] The original transient analyses in many FSARs were based on the concept of an "equilibrium cycle" where the physical parameters would remain essentially constant from cycle to cycle. In practice, operational and economic issues have resulted in very few plants actually reaching the ideal of an equilibrium cycle.

Further examples of the use of the FSAR as the limit of consequences against which USQs are determined:

a) *The safety analysis demonstrates a very small amount of fuel failure and consequently the calculated radiological releases are much less than the criteria. These releases are reported in the FSAR. The SER concludes that the releases are acceptable because they are less than that allowed by the SRP. In a subsequent cycle, the amount of fuel failure increases slightly, resulting in releases that are still consistent with those of the SER (i.e., less than that allowed by the SRP). For two different plants, one has reported actual values from the analysis, another has reported results as less than criteria. Any increase results in a USQ for the first but not for the second.*

b) *A Section 50.59 evaluation was performed for one unit where, as a result of the fuel reload, fuel failure for an event increased from 1.1% to 1.2%. This resulted in an insignificant increase in dose (well within SRP acceptance criteria), both because the increase in fuel failure was small and because the calculated dose is dominated by the initial coolant activity allowed by the Technical Specifications, rather than the dose resulting from fuel that is failed as a result of the transient. This result was judged not to be a USQ on the basis of no significant increase in consequences, where the consequences are well within the acceptance criteria. Under NUREG-1606, this result would be judged to be a USQ.*

c) *A Section 50.59 review was performed of the reload analysis compared to the as-built and as-operated cycle. The review evaluated all differences in analysis assumptions compared to the as-built/operated inputs and concluded that the small changes between the two would not result in a significant impact on the consequences of events evaluated. If any increase in consequence, independent of significance, are considered to be a USQ, this type of evaluation will routinely result in Unreviewed Safety Questions.*

d) *Any reload evaluation has the potential for some increase in consequences compared to those reported in the FSAR. For a standard reload, where no changes other than the reload itself are being introduced, the small cycle-by-cycle variation in core parameters would not be expected to, and do not, result in significant changes to event consequences. In such cases, the reload analysis is typically performed on a time schedule as late as possible prior to refueling to assure that it best reflects current cycle operation and next cycle plans. If changes in consequence, independent of significance, is the basis for judging a USQ, then it becomes much more probable that*

the reload analysis will result in an USQ for some event. This would necessitate performing the reload analysis 6-9 months earlier than is current practice to allow for the increased possibility that NRC review of insignificant changes in consequences would be required. Earlier analyses increase the likelihood that analysis assumptions will not meet actual conditions of the fuel at the time of refueling. Under NUREG-1606, there would be last minute re-analyses with significant time and cost impacts and the possible need for rushed reviews of USQs by the Staff. Use of the SRP acceptance criteria as the basis for determining whether the margin of safety has been reduced, rather that the analysis result reported in the FSAR, would ensure consistent application of 50.59.

S. Definition of Reduction in Margin of Safety

The NRC staff should develop a more tractable approach for addressing insignificant changes. The NRC staff's position that, while it may not be necessary from a technical standpoint to address the impact of insignificant changes, their hands are tied because 50.59 says "may increase", is not compelling. As several of the Commissioners pointed out[17], the NRC staff's position does not appear consistent with the intent or practice when 50.59 was developed, and it certainly isn't consistent with practice over the past 10-15 years. The CEOG does not believe it is necessary to define "may"; rather, it is only necessary to interpret "increase in consequence" or "increase in probability" as that which would cause established acceptance criteria to be exceeded.

The NRC has, through rulemaking and the promulgation of the SRPs, clearly established the acceptance criteria for the probability of occurrence and consequences of various malfunctions and postulated accidents. The acceptance criteria for older plants, which may not fully, or at all, conform to the SRPs, were established as a result of NRC FSAR review. The methodology used to evaluate the consequences of such events has also been subject to NRC review, either on a plant specific basis or generically. Consequently, any change when evaluated using methods that have been reviewed and approved and which employ conservative assumptions that have been reviewed and approved in the FSAR should not be considered a USQ if the consequences or the probability of the event remain within established acceptance criteria. Industry practice and guidance in this area remains adequate and acceptable.

Examples:

See Section R, above, for examples of related problems.

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[17] Briefing on 10 CFR 50.59 Regulatory Process Improvements - Public Meeting, US Nuclear Regulatory Commission, March 10 1997.

T. Information That Establishes the Basis for any Technical Specification

The CEOG has no comments on this issue.

U. Determination of Unreviewed Safety Questions When Licensees Use New Methods (Analysis Methods, Assumptions) to Evaluate Plant Changes or Conditions

NUREG-1606 would require a change to be evaluated with the same methodology as used in the FSAR (or alternately, for the new methodology to be used before and after the change) so as to determine if consequences or probability of accidents increases. ABB thinks that this is incorrect when the new methodology has been generically approved by the NRC or specifically approved for a particular plant. The FSAR evaluation was approved on the basis that the event under discussion met some acceptance criteria as evaluated with the then current methodology. NRC approval of that methodology is based on the conclusion that the new methodology still yields acceptable conservative results. If such an approved methodology is used to evaluate a change and the consequences (or probability) of the event still meet the original acceptance criteria, then the change should not be considered an USQ. The definition of a methodology change is unacceptably broad and imprecise. Under NUREG-1606, any "sharpening of the pencil" as is routinely employed in engineering would likely result in an USQ.

Examples: Use of Different Methods

a) *There are often various choices of approved methods that can be employed to evaluate the consequences of events: simple (typically highly conservative) methods and more complex (typically less conservative) methods. Much of the licensing effort to obtain approval of the simple methods was to demonstrate that they were sufficiently conservative compared to the more phenomenologically correct complex methods to allow their use. The FSAR analysis typically uses the simpler (more conservative) methods when acceptable consequences can be shown, using the more complex methods only when necessary; this is particularly true of older analyses when computer time was costly.*

b) *If as a result of a change, the consequences using a simplified method become more adverse such that they can no longer be shown to be acceptable, we currently would use the more complex model to demonstrate acceptable consequences. We would not consider this a USQ since the more complex model was reviewed and approved and could have been used in the FSAR analysis. Under NUREG-1606 this would be a USQ since the consequences have increased, although they remain within acceptance criteria as evaluated using an approved methodology.*

c) A variation on this scenario is the case where a new method has been *approved subsequent to the FSAR analysis. We currently would feel free to employ that method, provided that it had prior NRC review and approval.*

The key difference between ABB (and I believe the industry) approach and the NRC guidance is that we attempt to evaluate the acceptability of a change by using approved methodology and FSAR established assumptions to compare to acceptance criteria. The NRC proposed guidance focuses on whether the change alters the consequences, independent of the acceptability of the altered consequences.

d) *The impact of NUREG-1606 would be to require NRC review of changes that clearly are acceptable when evaluated using approved methodology. Also, the methods used in the FSAR may not be still available, or may no longer be judged suitable today. The NUREG-1606 guidance is to evaluate the change by applying the new method before and after the change. In many cases, the engineering effort is doubled with no increased safety benefit.*

Impact of changes in analysis assumptions:

e) *Within an approved methodology, analysts have always had the option, within the limits of the approved methodology (as stated in the SER), to modify design input. Such changes are controlled by 10 CFR 50 Appendix B. In many cases, it is necessary (and desirable) to modify analysis inputs based on actual plant experience. For example, as more operating experience is gathered, the biases and uncertainties associated with analytical methods should be updated based on that experience. The NUREG-1606 guidance will likely discourage that process because of the likelihood of resulting in an USQ.*

V. Consideration of Compensating Effects When Making an Evaluation of Whether an Unreviewed Safety Question Exists

When discussing the issue of whether the use of new methods or compensatory actions could result in a USQ, the NRC staff states that "if new methods or assumptions are necessary to demonstrate that consequences have not increased or that the margin of safety is not reduced..., it is likely that a USQ is involved." In other words, the NRC staff position is that a USQ exists if "the change by itself would adversely affect consequences or margin of safety and it is only consideration of other factors that makes the net effect on the analysis be no increase (or no reduction)."

This NRC staff position is not supported by the language of the current rule. The licensee is free to determine the extent of the change and is not constrained by the rule to consider each element of the change individually. For example, if the NRC staff has generically approved a new analytical methodology and that methodology is used by a licensee to ensure that a change does not increase consequences or reduce margin, then there should be no USQ even if the change, absent the new methodology, would have increased consequences or reduced margins.

Even when the new methodology has been previously reviewed and approved by the NRC, the NRC staff notes that the "acceptance of the facility may have been based on certain conservatisms in the (original) analysis methods and assumptions." This conclusion would mean that the licensee does not have a clear and full understanding of the design basis of its facility. The Design Basis must, therefore, include undocumented reliance by the NRC staff on certain attributes of analytical methodology present at the time of the relevant licensing action that have not been subjected to Quality Assurance and technical reviews. If this conclusion is left to stand, it could undermine a licensee's ability to perform any changes under 10 CFR 50.59 since it would not be possible to have reasonable assurance regarding what undocumented basis the NRC staff relied on to approve the existing configuration. In our experience, assumptions and methods specifically relied upon by the NRC staff are sufficiently delineated in NRC SERs when those assumptions and methods played a significant role in a safety decision process. We find it difficult to believe that individual staff members would ignore basic nuclear principles of quality assurance and technical review. We also find it difficult to believe that NRC management would tolerate such practices.

Examples:

a) *Every reload analysis involves many changes. Some of these changes are small (a few days change in cycle length, for example) and some are major (changes in fuel design, for example). Each of these changes can have an impact on "consequences", either positive or negative. It is simply not possible (or meaningful) to distinguish between the effects of each individual change. Many of the changes are designed to produce a compensating effect for known adverse changes (increased numbers of burnable poisons to compensate for increased reactivity in the fuel, for example). A literal following of this guidance would require that all reload analyses be declared USQs.*

See Section U, above, for additional examples.

Policy Issues

The CEOG believes that adequate guidance for the performance of Section 50.59 exists using the existing NSAC-125 guidance (and as reissued as NEI 96-07) and that the

original intent of 50.59 remains valid and achievable under the current rule. However, if the NRC proposed guidance in NUREG-1606 is adopted, we believe that this will result in an untenable situation in which an overwhelmingly large number of trivial changes will require license amendments and the attention of the NRC staff. As a result, rulemaking will be required to put in place a rule that can achieve the original intent of 50.59 i.e., that of allowing safe changes to be implemented by licensees without the licensees and the NRC becoming buried in paperwork. In the event that rulemaking is required, the CEOG believes that the NRC staff should take advantage of the opportunity to simplify the wording and clarify the intent to more readily achieve the original regulatory goals.

Other Considerations:

a) In the future, when the NRC staff imposes significant additions to the facility licensing basis (e.g., Station Blackout or Anticipated Transient Without Scram), clear guidance should be provided with respect to the NRC staff expectations regarding updating the FSAR.

b) In future rulemaking activities, the NRC staff should consider deleting the question regarding "malfunction of a different type" from the 10 CFR 50.59 evaluation. The important aspect to safety is not the cause of a malfunction but whether the probability of failure increases or the consequence of failure is changed. For example, it is irrelevant what causes an electrical circuit (whether analog or digital) to fail "open or closed". The important question is whether the change makes it more likely to fail, or whether the end state is different from that previously evaluated.

c) In future rulemaking activities, the NRC staff should consider deleting the word "safety" from USQ since the intent of 10 CFR 50.59 is to perform a "regulatory test" and not a "safety test", as the NRC itself has stated. Therefore, a "positive" 10 CFR 50.59 evaluation would result in an Unreviewed Question or Unreviewed Regulatory Question.

A. Scope of Section 50.59

The NRC staff offers four possible approaches to defining better the scope of Section 50.59. CEOG would have to see the specifics of any NRC proposal before developing firm comments.

B. Unreviewed Safety Question Threshold

As the NRC acknowledges, the question of the USQ threshold is important because of the different actions required depending on whether a USQ is involved. If a change involves a USQ, a license amendment must be submitted and approved. This may be appropriate

for a significant change (one leading to a condition clearly outside of the plant's current licensing basis) but is very inappropriate or unreasonable for changes that have very little true significance.

The industry has consistently and successfully defined the USQ threshold for 30 years. The CEOG believes that any change in the rule should, at a minimum, conform to the guidance currently promulgated by NSAC-125 and NEI 96-07. This could then be a launching point for further improvements and refinement including the addition of risk-significance to the test for an USQ. The CEOG would have to see the specifics of any NRC proposal before developing firm comments.