March 18, 2005

Mr. Alexander Marion
Director, Engineering Department
Nuclear Generation Division
Nuclear Energy Institute

SUBJECT: NUCLEAR REGULATORY COMMISSION (NRC) DISPOSITION OF COMMENTS ON NEI 04-02, " GUIDANCE FOR IMPLEMENTING A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM UNDER 10 CFR 50.48(c)," REVISION F

Dear Mr. Marion,

We are writing to provide additional clarifications we plan to include in the subject regulatory guide. These clarifications were developed after reviewing the public comments and two subsequent public meetings. We would prefer to have these critical issues written into your guidance so we could refer to Nuclear Energy Institute (NEI) 04-02 with minimal or no exceptions.

The following provides some clarifications on the industry comments relating to delegation of Authority Having Jurisdiction (AHJ), transition License Amendment Request (LAR) submittals, and radiation release requirements. The enclosure contains further clarifications on recovery actions, graded risk checks for deterministic changes, circuit analysis for safe shutdown, transition of existing engineering equivalency evaluations, comments on defense-in-depth, and the applicability of 10 CFR 50.69.

Industry raised the question regarding the delegation of limited AHJ approval authority to licensees with respect to the fundamental fire protection program design elements in Chapter 3 of NFPA 805. Based on additional investigations, we have concluded that the NRC cannot delegate this authority. However, to avoid numerous unnecessary LAR submittals for standard equivalences of little or no safety significance, the staff plans to develop guidelines during the Transition Pilot Program to clarify the types of changes that will require AHJ approval.

NRC's planned strategy on LAR reviews, as reflected by the draft regulatory guide relied heavily on the Reactor Oversight Process (ROP) rather than a review by the NRC licensing staff. Industry wanted a Safety Evaluation Report (SER) from the licensing staff covering all critical aspects of the new program to minimize licensing basis problems during future ROP inspections. We have agreed to provide additional flexibility to the licensee to submit additional information for staff review. Please revise NEI 04-02 to incorporate industry comments on this issue.

With respect to the radiation release performance criteria, in consideration of the very low risk associated with radioactivity releases from sources other than the reactor core, the NRC agrees with NEI's proposal as reflected in Revision F of NEI 04-02.

We are currently working with Revision F of NEI 04-02 and Revision 1 of NEI 00-01. It would be beneficial, both to the NRC and NEI, to reference the most current version and provide them to the ACRS for review. Since Revision 1 of NEI 00-01 is acceptable to us, we request the next revision to NEI 04-02 by March 31, 2005 to maintain our schedule with the ACRS.

Please contact Mr. Paul W. Lain of my staff at (301) 415-2346 if you have any questions or need additional information.

Sincerely,

//RA//

John N. Hannon, Chief
Plant Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: Suzanne Black, DSSA
Fred Emerson, NEI

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Transition of Manual Actions to Recovery Actions

Recovery Actions – Operator manual actions should be transitioned as "recovery actions" in the new NFPA 805 licensing bases. Repairs should also be transitioned as "recovery actions."

The information for operator manual actions that should be included in the summary for each fire area is: 1) whether the operator manual actions were previously reviewed and approved by the NRC's Office of Nuclear Reactor Regulations (NRR), and 2) reference to documentation that demonstrates prior review and approval by the NRC.

Repairs credited for cold shutdown equipment should also be transitioned on a fire area basis. Information that should be summarized includes reference to documentation that demonstrates the equipment necessary for the repair is staged, proceduralized, and achievable in the necessary time frame.

Operator manual actions that have been previously reviewed and approved by the NRC (as documented in an approved exemption request or deviation) can be transitioned without the need to use the change evaluation process. However, licensees may consider use of the change evaluation process for previously reviewed and approved operator manual actions so that the evaluation is consistent with operator manual actions not previously reviewed and approved by the NRC.

Operator manual actions (e.g., those credited for compliance with Appendix R sections III.G.2 and III.G.3) that have not been previously reviewed and approved by the NRC should be addressed for acceptability using the change evaluation process outlined in Chapter 5.3 of NEI 04-02.

NFPA 805 Section 4.2.3.1 does not allow recovery actions when using the deterministic approach to meet the nuclear safety performance. However, the use of recovery actions is allowed by NFPA 805 using a performance-based approach, provided that the additional risk presented by the recovery actions has been evaluated by the licensee in accordance with NFPA 805 Section 4.2.4. The two performance-based approaches allowed in NFPA 805 Section 4.2.4 are fire modeling (Section 4.2.4.1 of NFPA 805) and fire risk evaluation (Section 4.2.4.2 of NFPA 805).

Figure 5.1 provides the change evaluation process and flowchart that must be used for both the fire modeling and risk evaluation approach. Figure 5.1 requires a "risk check" for all recovery actions, independent of whether a licensee uses the fire modeling path or the risk evaluation path. This appears to impose a risk quantification burden on all recovery actions for both pathways. However, risk checks required under Figure 5.1 may be quantitative or qualitative. Characteristics of a reasonable qualitative or a quantitative risk check are provided in Section [***insert correct Section reference***] of this guide.

Regardless of which approach is used by a licensee, the recovery actions should demonstrate feasability. Furthermore, since NFPA 805 is a risk-informed, performance-based rule, detailed implementation of either approach must demonstrate that the recovery action has a reliability commensurate with its risk significance.

Criteria for Demonstrating Feasibility

One acceptable set of criteria that a licensee may use to evaluate the feasibility of recovery actions is outlined in NFPA 805 Appendix B.5.2(e). In accordance with general guidance used in the significance determination process and inspection procedures, the NRC staff also relies on a) written procedures, b) training, c) periodic drills that simulate the actual fire and environmental conditions to the extent practical, and d) consider availability of systems and indications essential to perform the recovery action as a minimum set of requirements to achieve feasibility of recovery actions.

If a licensee chooses to use the fire modeling path, Section 4.2.4.1.6 of NFPA 805 provides the requirement for feasibility. If a licensee chooses the risk evaluation path, a reasonable human reliability analysis should prompt the licensee to consider each of the feasibility criteria provided in Appendix B.5.2(e) of NFPA 805 and as supplemented above. A recovery action that does not meet the minimum feasibility criteria provided in Appendix B.5.2(e) and further supplemented above in this document will have a failure probability of approximately 1.0. Such recovery actions are not feasible and are therefore unacceptable.

Criteria for Demonstrating Reliability

The criteria in NFPA 805 Appendix B.5.2(e) and supplemented above are only to demonstrate feasibility of recovery actions. Recovery actions should be reliable as well. The reliability of the recovery action can be commensurate with its risk-significance. The reliability of the recovery actions should be considered by evaluating uncertainties associated with (i) human performance, (ii) the difference between field verification conditions and actual environmental and fire conditions, and (iii) design basis (e.g., thermal hydraulic analysis) versus actual time constraints. For example, under (i) there may be differences in the human performance from one plant person to another or between groups of personnel that may result in varying times to complete a recovery action; under (ii) there may be conditions (e.g., smoke, water, noise, etc.) that cannot be easily simulated when considering and evaluating recovery actions; and under (iii) the amount of time available to the licensee to complete the recovery action versus the time to actually complete the action should be considered and evaluated.

If a licensee uses the risk evaluation method, the risk significance associated with the recovery action is reflected in the core damage frequencies and the Risk Achievement Worths (RAWs) associated with that recovery action. Those quantitative values could be indicative of acceptable mean values of the recovery actions. For example, a failure probability of 0.5 may be acceptable for a recovery action which has a RAW of 1.01. However, that may not be acceptable for a recovery action with a RAW value of 2.0. The fundamental expectation is that any method used must assure that the level of reliability is commensurate with the risk significance.

If a licensee uses the fire modeling method, then the information assessed and documented under the “risk check” can be used to determine the level of reliability necessary to assure safety. In addition to the factors considered in the “risk check” a licensee must consider the frequency of the reliance of the recovery action. For example, a recovery action that is relied on to mitigate multiple failures has a higher risk-significance compared to a recovery action relied on infrequently.

A quantitative analysis results in conclusions that are relatively more certain then those based on qualitative assessments. Therefore, other examples of how a licensee may adjust reliability and accommodate uncertainties in accordance with a risk-informed approach are as follows.

If the recovery actions are required to transition to and maintain cold shutdown (where the plant is not required to go to cold shutdown to maintain the plant safe and stable), the licensee may use a qualitative risk evaluation for the risk evaluation approach (Section 4.2.4.2 of NFPA 805) of the change evaluation process.

If failure to perform the recovery action as required does not directly fail safe shutdown (e.g., through a single spurious or loss of component), the licensee may use a qualitative risk evaluation for the risk evaluation approach of the change evaluation process.

If successful completions of the recovery actions are the primary means to recover and reestablish the nuclear safety performance criteria (e.g., directly fails safe shutdown if not performed successfully), the licensee should use a quantitative risk evaluation approach of the change evaluation process.

Reference Basis for Acceptance Criteria of Recovery Actions

NFPA 805, Section 4.2.4.2 states that the risk evaluation should compare the risk associated with implementation of the deterministic requirements with the proposed alternative (in this case the recovery actions). If the manual action had always been credited (i.e., there was never any other compliance strategy, e.g., electrical raceway fire barrier system (ERFBS), 20 feet of separation) then the deterministic criteria chosen should be from Section 4.2.3.2. If the recovery action was credited in lieu of an ERFBS, then the deterministic criteria chosen should be from Section 4.2.3.3(a) or (c). The reference basis used to determine the acceptance criteria should be fixed. That is, the incremental increase in risk should be compared using the guideline provided above as the basis for all changes. This is necessary to prevent unacceptable risk increases resulting from a series of changes.

Aspects of fire risk that should be considered include fire ignition, fire growth, fire propagation, fire detection, fire suppression, fire-induced structure, system, and component (SSC) failures, and fire-affected human actions. For recovery actions, the licensee should evaluate each of the fire risk aspects and determine if the use of the recovery actions will require a risk check. For example, the licensee should perform a risk check when there is a change to the coverage of the fixed fire suppression system or where the change would result in consideration of a fixed fire suppression system where none currently exists. Using this example further, when a licensee uses a recovery action in lieu of fire barriers or fire separation required by the deterministic approach of NFPA 805 Section 4.2.3 and a fire suppression system is required by the deterministic approach but is not provided, then the licensee should evaluate the increase in risk at the beginning of the change evaluation process prior to performing the fire modeling approach or risk evaluation approach. This is necessary to ensure that the risk check appropriately considers all aspects of fire risk.

Additional Guidance in Selecting Qualitative Versus Quantitative Risk Assessment

NFPA 805 allows use of qualitative or quantitative evaluations for all recovery actions, including those necessary to perform a “risk-check.” However, a quantitative analysis results in conclusions that are relatively more certain than those based on qualitative assessments. Safety Margin (SM) and Defense-in-Depth (DID) are compensation for numerous types of uncertainties and licensees who chose the quantitative assessments would have more latitude in the manner on how they demonstrate availability of DID. An extreme example of this is as follows. If a recovery action is relied on with core damage frequency (CDF) sequences on the order of 1E-10/year and the 95% confidence limit that captures uncertainties creates an upper bound of 1E-08/year, the reliance on DID may be minimal.

Defense-in-Depth (DID)

High level expectation on DID are provided in Section 1.3 of NFPA 805. Further elaboration is provided in Section 5.3.2.2 of NEI 04-02. Regulatory guidance on DID is provided in RG 1.174. The following additional guidance specific to recovery actions is provided to a) ensure that a risk-informed approach does not result in unacceptable increases in risk to the public, and b) interpretation of DID does not unnecessarily inhibit the flexibility afforded to a licensee by this risk-informed rule:

- Where recovery actions are credited, eliminating the suppression system from an Appendix R, III.G.2 or III.G.3 compliant area or similar commitments (e.g., section 5.b of Standard Review Plan 9.5.1) is <u>unacceptable</u> with respect to DID, since that change would constitute an “over-reliance and increased length of time in performing programmatic activities to compensate for weakness” and “reasonable balance among prevention of fires, early detection and suppression of fires, and fire confinement.” Similarly, eliminating an ERFBS from an area with no suppression system and crediting a recovery action is unacceptable with respect to DID.

- Converting an “automatic” suppression system in a III.G.2 or III.G.3 compliant area from “automatic” to “manual” does not defeat DID as long as the recovery actions available are feasible and reliable with respect to early detection and suppression of fires, and fire confinement. Here early detection and suppression of fires, and fire confinement should be evaluated in terms of realistic fires and fire growth rates.

Additional guidance to supplement clarifications such as above would be developed during the Transition Pilot Program. Those clarifications would be included in future revisions of the Reg Guide and/or NEI 04-02 for NRC endorsement.

Safety Margins (SMs)

High-level expectations on SMs are provided in RG 1.174 with additional details in NUREG-0800, Standard Review Plan, Chapter 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance.” Further guidance is provided in Section 5.3.2.3 of NEI 04-02. The following additional guidance specific to recovery actions is provided to a) ensure that a risk-informed approach does not result in unacceptable increases in risk to the public, and b) interpretation of SM does not unnecessarily inhibit the flexibility afforded to a licensee by this risk-informed rule:

- The margin between maximum expected fire scenario (MEFS) and limiting fire scenario (LFS) should be considered and documented with a basis for acceptability. The magnitude of the margin may be case-specific depending on the consequences. For example, in some cases the margin between MEFS and LFS may need to be greater for sequences resulting in an interfacing systems loss of coolant accident (LOCA) versus sequences resulting in a non-interfacing systems LOCA.

- Safety margins in other factors (e.g., fire modeling, plant systems performance, probabilistic risk analysis (PRA) logic model) that provide significant (at least one order of magnitude) additional SM to compensate for uncertainties should be documented with a basis for acceptability.

Additional guidance to supplement clarifications such as above would be developed during the Transition Pilot Program. Those clarifications would be included in a future revision of the Reg Guide and NEI 04-02 for NRC endorsement.

Acceptable Risk Evaluations Methods for Deterministic Changes

NFPA 805 requires that a plant change evaluation be performed for a change to a previously approved fire protection program element. The evaluation process should include an assessment of the acceptability of the risk attributed to the change in accordance with Section 2.4.4 of NFPA 805. This requirement for a risk evaluation is part of the plant change evaluation, regardless of whether the plant change is evaluated using the deterministic approach or the performance-based approach (see Figure 2.2, Methodology, of NFPA 805). NFPA 805 Section 4.1, states that, “Deterministic requirements shall be “deemed to satisfy” the performance criteria and require no further engineering analysis.” However, Chapter 4 of NFPA 805 provides the requirements for the baseline evaluation of the fire protection program’s ability to achieve the performance criteria outlined in Section 1.5 of NFPA 805 and does not apply to the evaluation of plant changes.

While the risk impact of plant changes must be evaluated for all changes to a risk-informed fire protection program, the level of risk evaluation should be commensurate with the level of risk associated with the change. A graded approach can meet the requirements of NFPA 805, with a simple risk evaluation. Here a “simple” risk evaluation or the term “risk-check” should be interpreted as a qualitative check-off, a structured screening process, or an order-of-magnitude quantification for inconsequential increases. Potentially higher risk changes may require a quantitative evaluation spanning a spectrum from traditional approaches, such as FIVE, through more recent developments, such as the Fire Protection Significance Determination Process, up to a fire probabilistic safety assessment (PSA) itself.

This section provides guidelines for a simple risk evaluation (i.e., qualitative check-off, screening process, or order-of-magnitude quantification) that may be applied to deterministic changes to the fire protection program. Additional guidance to refine the clarifications provided above will be developed using the lessons learned from the pilot transition program. A “change” in this context can be a modification to the fire protection program, a modification to other aspects of the plant that impact the fire protection program, or a plant feature that does not comply with regulatory requirements for the fire protection program. The latter category of “change” must be addressed via the plant change process if the noncompliant feature is not corrected to bring it into compliance with regulatory requirements. This simple risk evaluation for deterministic changes that are insignificant with respect to meeting NFPA 805 performance criteria or of very low risk significance is referred to as a “risk-check.”

Characteristics of an Acceptable Risk-Check

Figure 5.1 of NEI 04-02 requires a risk-check for all change evaluations **[Figure 5.1 of Revision F of NEI 04-02 does not require the risk-check, but should be revised to conform to this writeup]**. The characteristics of an acceptable risk-check that meets the “assessment of the acceptability of risk” requirement of Section 2.4.4 of NFPA 805 are as follows:

- The quality of the check is sufficient to assure that non-negligible risk increases receive detailed risk assessments appropriate to the level of risk.

- The risk-check, whether qualitative or quantitative, must be documented and be available for inspection by the NRC.

- The risk-check does not defeat the intent of the Rule to provide success paths that do not require quantifications.

- The risk-check does not pose undue evaluation or maintenance burden.

An Acceptable Risk-Check Process and Examples

This section provides one acceptable risk-check method typical of a “qualitative check-off:”

- Identify and document the factors that contribute to the risk associated with the change. In general, these factors include changes in: a) frequency of all fire scenarios which are affected by the change, b) magnitude of expected fires, c) detection capability, d) suppression capability, e) post-fire capability of plant systems to prevent damage to the core, and f) post-fire capability of plant systems to prevent or minimize releases to the environment.
- The impact of the plant change on each of these factors can be qualitatively evaluated and categorized as: “no” impact, “negligible” impact or “high” impact. The nature of the change would enable a licensee to choose among the three categories. A licensee may refer to their IPEEE, the fire protection SDP, or other documents to determine whether the change could have “negligible” or “high” impact. The licensee should document the basis for the conclusion.

The following are examples of each of the impact levels for each of the factors listed:

- Frequency of fire scenarios: A “like-kind” replacement of fire protection equipment or systems has “no” impact on fire frequencies. Installing an electrical cabinet in a switchgear room could cause “negligible” impact on the fire frequency. Changing administrative procedures to allow welding in an area where it was previously prohibited could cause a “high” increase in the frequency of fire.

- Magnitude of expected fires : Replacing a cable with one of equivalent combustible loading and type has “no” impact on the magnitude of the expected fires. Routing a new cable through a switchgear room could cause “negligible” increase in the fire magnitude. Storing a drum of oil in the emergency diesel generator room could cause a “high” increase in the magnitude of expected of fire.

- Detection capability: Changes to safe shutdown equipment generally have “no” impact on the detection capability. A decrease in the normal area occupancy level where manual suppression and automatic detection are available could cause a small decrease in the fire detection capability. A decrease in normal area occupancy level where manual suppression is available but no automatic detection is provided could cause a “high” decrease in the fire detection capability.

- Suppression capability: Adding a few new cables to a cable tray without reducing the separation between redundant trains or adding an obstacle to a sprinkler spray-down path has “no” impact on the suppression capability. A decrease in the number of fire extinguishers available to fight fires or an equipment change that creates an minor obstruction to a suppression system spray pattern could cause “negligible” decrease in the fire suppression capability. Converting an automatic suppression system to a fixed manual fire suppression system could cause a “high” reduction in the fire suppression capability.

- Post-fire capability of plant systems to prevent damage to the core: Replacing a component with a similar component will typically have “no” impact on plant systems’ post-fire capability to prevent damage to the core, as long as the location of the component and cable routing remain unchanged. Rerouting one cable associated with a very low risk-significant system could cause “negligible” decrease in the plant systems’ capability to prevent damage to the core. Rerouting cables of a safety-related or a risk-significant system where separation is reduced or replacing a check valve with a motor-operated valve could cause a “high” decrease in the plant systems’ capability to prevent damage to the core. (Procedural changes should also be evaluated as part of this evaluation factor.)

- Post-fire capability of plant systems to prevent or minimize releases to the environment: Replacing a component with a similar component will typically have “no” impact on plant systems’ capability to prevent or minimize releases to the environment, as long as the location of the component and cable routing remain unchanged. Rerouting one cable associated with a very low risk-significant system that supports containment isolation of a penetration that is 2" or less, could cause “negligible” decrease in plant systems’ capability to prevent a large early release. Rerouting cables of a safety-related or a risk-significant system where separation is reduced or replacing a check valve with a motor-operated valve of a containment penetrating system whose pipe diameter is greater than 2" could cause a “high” decrease in plant systems’ capability to prevent a large early release of radioactivity.

- Other risk-checks: The licensee should also consider the potential for common cause effects of a given plant change on the above factors. For example, an increase in combustible loading in an area can impact all of the factors.

If a plant change could cause a “high” negative impact with respect to more than one of the above factors, or could result in a common cause impact on more than one of the above factors, licensees are encouraged to perform risk assessments of the more detailed, quantitative variety. Licensees are not required to track cumulative risk increases associated with changes which contribute negligibly, provided they evaluate “negligible” consistent with the guidelines of RG 1.174. Licensees are required to track all plant changes in accordance with their existing procedures in order to make sure that synergistic effects of cumulative changes are captured in the risk analysis.

Licensees are not required to perform the more detailed risk assessments of the quantitative variety to evaluate the risk impact of a deterministic change. Licensees are required to document the basis of their conclusion from any “risk-check” type of evaluation and make them available for inspection by the NRC. An inspector may chose to perform a more detailed, quantitative evaluation for a change for which the licensee has performed only a risk-check type of evaluation. If such an evaluation reveals that the licensee has performed changes that are unacceptable with respect to the requirements of NFPA 805, appropriate actions will be taken in accordance with the reactor oversight process.

Additional guidance to refine the clarifications provided above will be developed using the lessons learned from the Transition Pilot Program.

Circuit Analysis during Transition to NFPA 805

If the existing licensing basis is vague or silent on the methodologies identified, then a licensing basis should be clearly defined during the transition period. For example, if the existing licensing basis is vague or silent on the methodology for circuit analysis (selection and/or protection of circuits) or evaluation of the failures of circuits within a fire area (single failure, any-and-all, one-at-a-time, sequential/concurrent, cumulative effects) a licensing basis should be established against which changes can be assessed post-transition.

A licensee may choose to submit a summary of its licensing basis on circuits for NRC review and approval. At a minimum, the summary must contain sufficient information relevant to methods, tools, and acceptance criteria used to enable the staff to determine the acceptability of the licensee’s methodology. The NRC staff may request additional information necessary to adequately assess the licensee’s submittal. To assist those licensees who wish to submit a summary, the NRC plans to develop templates for licensee submittals during the Transition Pilot Program.

The NRC does not prescribe how a licensee should establish its licensing basis. The options to establish a licensing basis include 1) crediting a well documented design basis which meets minimum staff expectations, or 2) using other methods approved by the AHJ for selection of circuits and for using risk-insights to evaluate the consequences.

Minimum staff expectations include (however are not limited to) addressing single spurious and risk significant multiple spurious failures, DID and SM.

The NRC staff has reviewed Rev. 0 of NEI 00-01 and concluded that Chapter 3 provides an acceptable way to select circuits, and Chapter 4 provides an acceptable way to determine risk-significance of circuit findings. In addition, the staff has reviewed a methodology provided by Duke Energy [attach Duke Energy's proposed method] and agree that it provides an acceptable approach for screening out non-risk-significant issues.

NRC Inspection Manual Chapter 0609, Appendix F can also be used within the appropriate context. If Appendix F is used, the licensee must determine the applicability of its data to the licensee's facility.

Post Transition Circuit Analysis

Circuit analysis changes performed post-transition should be evaluated using the criteria in RG 1.174. Applying this criteria on a circuit-by-circuit basis or even an area-by-area basis may not adequately consider the cumulative effects. Therefore, in order to maintain reasonable assurance that cumulative effects of individual changes do not exceed the high-level acceptance criteria established in RG 1.174, a licensee may: 1) consider all circuit analysis changes, during the transition and post-transition, 2) perform plant or procedure changes that make the change risk neutral or decreases risk, or 3) apply an AHJ approved threshold for individual changes. The thresholds provided in the methodology provided in attachment [refer to Duke's proposal here] are considered an acceptable set of acceptance criteria.

Guidance for Existing Engineering Equivalency Evaluations

NFPA 805, Section 2.2.7 describes the application of Existing Engineering Equivalency Evaluations (EEEE's) when using a deterministic approach during the transition to an NFPA 805 licensing basis. One type of EEEE, commonly referred to as a "Generic Letter 86-10 evaluation (GL 86-10)," allows licensees who have adopted the standard fire protection license condition to make changes to the approved fire protection program without prior NRC approval if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. A GL 86-10 evaluation is one acceptable means of meeting the NFPA 805 EEEE acceptance criteria of "an equivalent level of fire protection compared to the deterministic requirements."

NEI 04-02, Section 4.1.1, Transition Process Overview, notes that the licensee will review EEEE's during the transition process to ensure the quality level and the basis for acceptability are still valid. Satisfactory results from this review will provide adequate basis to transition EEEE's for the deterministic requirements of Chapter 4 of NFPA 805.

As noted in the guidance for recovery actions, those credited for protection of redundant trains in Appendix R, III.G.2 areas do no meet the deterministic requirements of Chapter 4 of NFPA 805. Consequently, these recovery actions should be addressed as a plant change in accordance with Section 2.4.4 of NFPA 805. In accordance with 10 CFR 50.48(c) and NFPA 805, EEEE's which evaluate deviations from NFPA 805, Chapter 3 requirements must be submitted to the NRC for approval as a license amendment request.

Defense-In-Depth

NEI 04-02, Section 5.3.2.2, need to replace reference to "suppression" with "extinguishment" to align with the requirements in NFPA 805, Section 1.2 and NEI 00-01.

Applicability of 10 CFR 50.69

10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors" allows certain SSCs to be treated in a risk informed manner. 10 CFR 50.69 is not applicable to 10 CFR 50.48, "Fire protection" since 50.48 is not included in Section (b) of 50.69. 10 CFR 50.48(c) allows the voluntary application of a performance-based fire protection rule. The treatment of SSCs under 10 CFR 50.48(c) shall be in accordance with the rule, the referenced NFPA standard, and the applicable licensee's licensing basis.