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Anthony R. Pietrangelo
SENIOR DIRECTOR, RISK REGULATION
NUCLEAR GENERATION

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OFFICE OF SECRETARY
RULEMAKINGS AND
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March 6, 2006

Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTN: Rulemakings and Adjudications Staff

SUBJECT: Comments on Proposed Rule 10 CFR Part 50, RIN 3150-AH29, *Risk-Informed Changes to Loss-of-Coolant-Accident Technical Requirements* (70 *Federal Register* 67598, November 7, 2005)

The Nuclear Energy Institute[1] appreciates the opportunity to comment on the subject proposed rule. We commend the NRC for moving to the proposed rule stage of this important effort that will establish a realistically conservative design basis break size for analyzing the performance of emergency core cooling systems (ECCSs) during loss of coolant accidents (LOCAs). Even without a single design change to plants that adopt the final version of 10 CFR 50.46a, this rule will enhance safety by improving the focus of plant operators and the NRC on more safety significant matters. This change to the regulations is long overdue.

The impetus for this rulemaking is captured in one simple insight that has been gleaned through operating experience, engineering analyses and expert judgment. Large, robust steel pipes are highly unlikely to break catastrophically, and are much less likely to break than smaller steel pipes. While we believe it is prudent to maintain defense in depth through mitigation capability for the unlikely event of a large, catastrophic pipe break, it is not prudent to center a substantial portion of the regulatory framework on such an event. In fact, the near-sightedness imposed by the current rule leads to plant configurations that are less safe and processes and requirements that squander both licensee and NRC attention and resources.

While we fully support the concept of redefining the large break LOCA embodied in this rulemaking, we are seriously concerned that the proposed rule itself fails to

[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including regulatory aspects of generic operational and technical issues. NEI members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

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provide a practical, effective means of implementation that meets any of the objectives of risk-informed, performance-based regulation. In particular, the paragraphs of the proposed rule dealing with operational restrictions and the risk-informed safety performance (RISP) assessment for change control are unnecessarily and excessively burdensome. Rather than building on and taking credit for existing regulatory requirements and processes, these paragraphs confuse and complicate fundamental elements of the regulatory framework, including plant technical specifications and current licensing bases. In short, the proposed rule as drafted is not a viable option for licensees. As an alternative, we propose a performance-based approach that demonstrates sound risk management of the plant configuration after adoption of the rule.

Finally, we note that substantive interactions on the development of the proposed rule were curtailed by the NRC staff after two public meetings in the summer of 2003. We believe the current proposed rule suffers in many respects from a lack of stakeholder participation in the development process. We can only hope that in future rulemaking efforts, the NRC better recognizes the benefit of public interactions.

The enclosures include the following comments:

- Enclosure 1 – Responses to the questions posed in the *Federal Register* notice
- Enclosure 2 – Specific comments on the proposed rule language
- Enclosure 3 – Suggested revisions to the proposed rule
- Enclosure 4 – Seismic effects and assessments

We would be happy to discuss our comments with the NRC staff or Commission. Our intent is to achieve a final rule viable for industry-wide implementation that meets all of the objectives of risk-informed, performance-based regulation.

Sincerely,

Anthony R. Pietrangelo

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Enclosures

Specific Topics Identified for Public Comment on Proposed Changes to 10 CFR Part 50, ECCS LOCA Redefinition Rule

The NRC seeks specific public comments on numerous questions and issues. Specific topics for comment are identified below:

1. In proposed Sec. 50.46a(b), the Commission specifically precluded the application of the Sec. 50.46a alternative requirements to future reactors. However, future light water reactors might benefit from Sec. 50.46a. The Commission requests specific public comments regarding whether Sec. 50.46a should be made available to future light water reactors.

Comment: Sec. 50.46a should be applicable to future light water reactors. It does not make sense to ignore the insights that provide the basis for changing the existing 50.46 for future light water reactors. Regulation of advanced light water reactors should be done in light of all operating experience and analytical insights gained from the current fleet of light water reactors.

2. The TBS specified by the NRC in the proposed rule does not include an adjustment to address the effects of seismically-induced LOCAs. NRC is currently performing work to obtain better estimates of the likelihood of seismically-induced LOCAs larger than the TBS. By limiting the extent of degradation of reactor coolant system piping, the likelihood of seismically-induced LOCAs may not affect the basis for selecting the proposed TBS. However, if the results of the ongoing work indicate that seismic events could have a significant effect on overall LOCA frequencies, the NRC may need to develop a new TBS. To facilitate public comment on this issue, a report from this evaluation will be posted on the NRC rulemaking Web site at http://frwebgate.access.gpo.gov/cgi-bin/leaving.cgi?from=leavingFR.html&log=linklog&to=http://ruleforum.llnl.gov before the end of the comment period. In December 2005, stakeholders should periodically check the NRC rulemaking web site for this information. The NRC requests specific public comments on the effects of pipe degradation on seismically-induced LOCA frequencies and the potential for affecting the selection of the TBS. The NRC also requests public comments on the results of the NRC evaluation that will be made available during the comment period. (See Section III.B.3 of this supplementary information.)

Comment: The NRC issued the above mentioned seismic report in December 2005 titled "Seismic Considerations for the Transition Break Size". We concur with the NRC conclusions from the study that the recommended TBS is not adversely affected by the consideration of seismic risk. The basis for this concurrence includes agreement with the arguments presented within the NRC study and the following points:

- The median seismic capacities for both the primary piping system and the primary system components are higher than most other safety related power plant components within the nuclear power plant. At the very high accelerations (very low return period seismic hazard) associated with the point at which the primary piping or the primary system components would be calculated to fail, many other safety related structures, systems and components with lower capacities would already be postulated to have failed and thus control the seismic risk.
- The change in risk (delta risk defined in Regulatory Guide 1.174) due to seismic is estimated to be extremely low. The creation of the TBS by itself does not produce a physical change to the plant that would result in an appreciable change in seismic risk.

Enclosure 4 provides additional detailed comments.

3. Depending on the outcome of an ongoing NRC study (see Section III.B.3 of this supplementary information), the final rule could include requirements for licensees to perform plant-specific assessments of seismically-induced pipe breaks. These assessments would need to consider piping degradation that would not be prejudiced by implementation of the licensee's inspection and repair programs. The assessments would have to demonstrate that reactor coolant system piping will withstand earthquakes such that the seismic contribution to the overall frequency of pipe breaks larger than the TBS is insignificant. The NRC requests specific public comments on this and any other potential options and approaches to address this issue.

Comment: As stated in the response to Topic #2 above, the NRC study, *Seismic Considerations for the Transition Break Size,* concluded that the likelihood of seismically-induced LOCAs larger than the TBS was less than the 10^{-5} per year threshold of interest. As such, plant-specific assessments of seismically-induced pipe breaks should not be required. EPRI has conducted some limited studies into the indirect seismically-induced LOCA risk and confirmed for those samples that the risk is less than 10^{-5} per year. In addition, the change in risk described in RG 1.165 associated with the change in transition break size is judged to be negligible from the seismic perspective based on its associated negligible effects on either the seismic hazard or the seismic fragilities.

Enclosure 4 provides additional detailed comments.

4. The ACRS noted that "a better quantitative understanding of the possible benefits of a smaller break size is needed before finalizing the selection of the transition break size." The TBS to be included in the final rule should be selected to maximize the potential safety improvements. Thus, the NRC is

soliciting comments on the relationship between the size of the TBS and potential safety improvements that might be made possible by reducing the maximum design-basis accident break size.

Comment: The NSSS Owners Groups are submitting specific comments on this topic which we fully endorse.

5. The proposed Sec. 50.46a includes an integrated, risk-informed change process to allow for changes to the facility following reanalysis of beyond design basis LOCAs larger than the TBS. However, the current regulations in 10 CFR Part 50 already have requirements addressing changes to the facility (Sec. 50.59 and Sec. 50.90). It might be more efficient to include the integrated, risk-informed change (RISP) requirements, for plants that use Sec. 50.46a, under these existing change processes. The Commission solicits specific public comments on whether to revise existing Sec. Sec. 50.59 and 50.90 to accommodate the requirements for making plant changes under Sec. 50.46a.

Comment: The existing change control processes in the regulations are functioning properly in determining those changes to the facility, technical specifications or procedures that are within the licensing basis of the plant that require prior NRC review and approval. In addition, for license amendment requests that are risk-informed, the regulatory process (defined by RG 1.174 and other guidance) is well-established. Therefore, we would strongly discourage revisions of Sec. 50.59 or 50.90. However, this comment should not be construed as supportive of the RISP requirement in the proposed rule. The RISP requirement would unnecessarily move much of the existing regulatory guidance into the rule and would add excessive burden on both licensees and the NRC staff in processing non-risk significant changes and amendment requests. An alternative to the RISP requirement is proposed in Enclosure 3.

6. The proposed Sec. 50.46a rule would rely on risk information. The NRC has included specifically applicable PRA quality and scope requirements in the proposed rule. However, there are other NRC regulations that also rely on risk information (e.g. Sec. 50.65 maintenance rule and Sec. 50.69 alternative special treatment requirements). Consistent with the Commission policy on a phased approach to PRA quality, it might be more efficient and effective to describe PRA requirements (e.g., contents, scope, reporting, changes, etc.), in one location in the regulations so that the PRA requirements would be consistent among all regulations. The NRC is seeking specific public comments on whether it would be better to consolidate all PRA requirements into a single location in the regulations so that they were consistent for all applications or to locate them separately with the specific regulatory applications that they support.

Comment: The only other regulation that relies on risk information is § 50.69, where PRA is a primary element of the categorization process that leads to a safety focused scope of NRC special treatment requirements. The technical basis for the change to 50.46 (i.e., redefining the large break LOCA break size) does not rely on PRA. Rather, risk information is an input to subsequent plant changes enabled by the rule. We believe it is premature at this time to attempt to centralize PRA requirements into a single location based on an optional final rule and a proposed rule. In addition, different risk information will likely be needed, depending on the application. Centralization should be a prime consideration in the development of a true risk-informed performance-based regulatory framework.

7. The proposed Sec. 50.46a rule would include the requirement that all allowable at-power operating configurations be included in the analysis of LOCAs larger than the TBS and demonstrated to meet the ECCS acceptance criteria. Historically, operational restrictions have not been contained in Sec. 50.46 but were controlled through other requirements (e.g., technical specifications and maintenance rule requirements). It might be more practical to control the availability of equipment credited in the beyond design-basis LOCA analyses in a manner more consistent with other operational restrictions. As a result, the NRC is soliciting public comments on the most effective means for implementing appropriate operational restrictions and controlling equipment availability to ensure that ECCS acceptance criteria are continually met for beyond design-basis LOCAs.

Comment: The operational restrictions in the proposed rule would remove any potential benefit from the adoption of Sec. 50.46a. The proposed rule completely ignores the risk assessment and management requirements of 10 CFR 50.65(a)(4), which are more than adequate to address beyond design bases/severe accident considerations. The maintenance rule also requires performance monitoring (availability and reliability) of key equipment consistent with their PRA success criteria. In addition, the existing technical specifications continue to provide operational restrictions that are conservative. There is no need for additional operational restrictions in this rule.

8. Given the Commission's intent (See SRM for SECY-04-0037) that plant changes made possible by this rule should be constrained in areas where the current design requirements ``contribute significantly to the `built-in capability' of the plant to resist security threats,'' the Commission seeks examples on either side of this threshold (plant changes allowed vs. changes prohibited), and additionally any examples of changes made possible by Sec. 50.46a that could enhance plant security and defense against radiological sabotage or attack. (See Section III.G.2 of this supplementary information.)

The Commission also solicits comments on whether the Sec. 50.46a rule should explicitly include a requirement to maintain plant security when making changes under Sec. 50.46a or otherwise rely on a separate rulemaking now being considered by the NRC to more globally address safety and security requirements when making plant changes under Sec. Sec. 50.59 and 50.90. Any examples of plant changes that involve Safeguards Information should be marked and submitted using the appropriate procedures.

Comment: There is a common misperception associated with this proposed rule (and as reflected in this particular question) that it will somehow enable plant changes that will reduce plant safety margins as well as the capacity to deal with security threats. In reality, the opposite is true, because this rule can only increase the safety focus on risk significant events and mitigating equipment. It will also improve the reliability and availability of this equipment by removing excessive conservatism from the design basis. Finally, existing change control requirements in the regulations preclude significant reductions in safety or security and defense. Reduction of the large break LOCA through a revision to 50.46 does not allow licensees to not comply with any other regulations.

9. Given the potential impact to the licensee (since the backfit rule would not apply) of the NRC's periodic re-evaluation of estimated LOCA frequencies which could cause the NRC to increase the TBS, should the rule require licensees to maintain the capability to bring the plant into compliance with an increased transition break size (TBS), within a reasonable period of time?

Comment: First, we believe the backfit rule should apply to this rule because it ensures that an appropriate safety focus is maintained and does not dilute licensee and NRC attention and resources unnecessarily. Second, the rule requires maintaining a mitigating capability up to the largest LOCA, regardless of the size of the TBS. Thus, there is no need for the requirement discussed in this question.

10. Is the proposed rule sufficiently clear as to be ``inspectable?'' That is, does the rule language lend itself to timely and objective NRC conclusions regarding whether or not a licensee is in compliance with the rule, given all the facts? In particular, are the proposed requirements for PRA quality sufficient in this regard?

Comment: The proposed rule would be difficult to inspect because it overlaps with many existing regulatory requirements. The operational restrictions alone would cause havoc in compliance space relative to the existing technical specifications. The change control and PRA requirements are excessive and drag regulatory process guidance reserved for review of specific license amendment requests into programmatic inspection space. The only solution is to simplify this rule into a more objective, performance-based approach.

11. The proposed Sec. 50.46a rule would impose no limitations on ``bundling'' of different facility changes together in a single application. Changes which would increase plant risk substantially or create risk outliers could be grouped with other plant changes which would reduce risk so that the net change would meet the risk acceptance criteria. Are the net change in risk acceptance criteria in the proposed rule adequate or should some additional limitations be imposed to avoid allowing facility changes which are known to increase plant risk?

Comment: Regulatory Guide 1.174 already provides guidance on combining changes under single license amendment requests. There is no need to repeat this guidance in rule space.

12. Is there an alternative to tracking the cumulative risk increases associated with plant changes made after implementing Sec. 50.46a that is sufficient to provide reasonable assurance of protection to public health and safety and common defense and security? (See Section III.D.1 of this supplementary information.)

Comment: Yes. A periodic assessment, including update of the plant-specific PRA as necessary, would demonstrate the effectiveness of the licensee's risk management of the plant configuration. There is nothing wrong with a qualitative judgment based on the updated PRA that considered changes made in the preceding period to provide reasonable assurance that safety and security were maintained.

13. The Commission requests specific public comments on the acceptability of applying the change in risk acceptance guidelines in RG 1.174 to the total cumulative change in risk from all changes in the plant after adoption of Sec. 50.46a. Should other risk guidelines be used and, if so, what guidelines should be used? (See Section III.D.1.c of this supplementary information.)

Comment: The RG 1.174 guidance is appropriate. The emphasis should be on sound risk management of the plant configuration, not on the accounting of infinitesimally small changes.

14. After approval to implement Sec. 50.46a, the proposed rule would require tracking risk associated with all proposed plant changes but would not require a licensee to include risk increases caused by previous risk-informed changes that were implemented before Sec. 50.46a was adopted. Licensees who adopt Sec. 50.46a before implementing other risk-informed applications will have a

smaller risk increase ``available" compared to licensees who have already incorporated some risk-informed changes into their overall plant risk before adopting Sec. 50.46a. The Commission does not consider this a safety issue but requests specific public comments on whether this potential inconsistency should be addressed and, if so, how? (See Section III.D.1 of this supplementary information.)

Comment: The emphasis should be on sound risk management of the plant configuration, not on the accounting of infinitesimally small changes, either before or after 50.46a was implemented.

15. The proposed Sec. 50.46a would require licensees to report every 24 months all ``minimal" risk facility changes made under Sec. 50.46a(f)(1) without NRC review. Are there less burdensome or more effective ways of ensuring that the cumulative impact of an unbounded number of ``minimal" changes remains inconsequential? (See Section III.E.3 of this supplementary information.)

Comment: Yes. First, under the current 10 CFR 50.59, licensees are required to submit a report to the NRC at least every 24 months that summarizes the changes that were made that did not require a license amendment. This requirement, together with reporting on the new CDF and LERF values resulting from updating the plant PRA at least every 48 months would be a performance-based approach to demonstrating sound risk management of the plant configuration. NRC would have a summary report that briefly describes all of the plant changes and their impact on the plant CDF and LERF. This approach is both efficient and effective and does not dilute both NRC and licensee resources to meaningless, non-risk significant matters.

16. Should the Sec. 50.46a rule itself include high-level criteria and requirements for the risk evaluation process and acceptance criteria described in Reg Guide 1.174, as is currently proposed? If these criteria were included in the regulatory guide only, and not in the rule, how could the NRC take enforcement action for licensees who failed to meet the acceptance criteria?

Comment: The acceptance criteria in RG 1.174 are guidelines and were never intended to be hard numerical limits. They are not appropriate for inclusion in the rule. We believe the rule should include a requirement for risk management of the plant configuration going forward, to be accomplished through a periodic assessment of the changes that were made during the period. From that information, the NRC should be able to conclude that sound risk management of the plant configuration was practiced, and if it wasn't (i.e., substantial increases in risk were incurred from the changes made), then the NRC would have adequate basis for enforcement action.

Specific Comments on Proposed Rule 10CFR Part 50 Risk-Informed Changes to the Loss-of-Coolant Accident Technical Requirements

§ 50.34 paragraphs (a)(4) & (b)(4); § 50.46 paragraph (a); § 50.46a paragraph (b)(1)
§ 50.46a should be made available to future light water reactors so that the associated safety and operational benefits of a realistically conservative design basis break size can be realized. The language in the above paragraphs should be modified such that § 50.46a is also applicable to future light water reactors.

§ 50.46 paragraph (a) and § 50.46a paragraph (b)(1)
The rule language, as drafted, perpetuates the specific inclusion of only Zircaloy and Zirlo cladding. This would continue the need for M5 to be licensed by exemption. M5 is currently being used in 11 nuclear power reactors of varying designs across the US. Each of these plants continues to require the formality of an exemption for their license. It is obvious that M5 is an acceptable and desirable cladding material for use in nuclear power reactors. With a change to the regulations being made, it will serve efficiency to include M5 and eliminate the need for exemptions. The language in the above paragraphs should include M5 cladding.

§ 50.46a

Paragraph (a)(3) Operating configuration...
§ 50.46 has historically addressed the technical analysis requirements and acceptance criteria for ECCS performance during LOCAs. It has not been a rule that addresses the operating configuration of plants. That is addressed by § 50.36, Technical Specifications, as well as by § 50.65(a)(4), the maintenance rule configuration risk management requirement. We believe that § 50.46a should focus on the plant design configuration consistent with the historical precedent. We suggest the following definition as a replacement: "Plant design configuration means the physical plant design and equipment capability that affect plant response to a LOCA."

Paragraph (a)(4) Transition break size (TBS)...
The transition break size specified for boiling water reactors (BWRs) is overly conservative and may unnecessarily limit or preclude benefits for BWRs. We suggest the following language: "The specified piping for a BWR is equivalent to 16 inch schedule 80 piping in the shutdown cooling suction line inside containment." We endorse the comments submitted by the BWR Owners Group that provide the technical basis for this specification.

The transition break size specified for pressurized water reactors (PWRs) is overly conservative. For PWRs with large piping connected to both the hot and cold legs, the transition break size for the hot leg should be based on the largest connecting pipe on the hot leg, and the transition break size for the cold leg should be based on the largest connecting pipe on the cold leg. These are logical break sizes and avoid the arbitrary nature of the size of a connecting pipe on the hot leg also being applied to the cold leg. This should be defensible based on focusing on the high stress locations in the primary loop, and the ruggedness of the main loop piping. For those plants with no large piping connected to the cold legs, it should be acceptable to apply the same transition break size for the cold leg as is applied for the hot legs. We also endorse the comments submitted by the Westinghouse Owners Group on the TBS.

<u>Paragraph (d)(2) Operational restrictions for LOCAs larger than the TBS</u>
If a plant were to adopt § 50.46a, LOCAs larger than the TBS would now be considered beyond design basis events. As paragraph (e)(2) of the proposed rule notes, ECCS analyses for these events would not require the assumption of a single failure and may take credit for the availability of offsite power. In addition, the availability of non-safety related equipment may also be credited in this mitigating analysis. Paragraph (d)(2) would impose operating restrictions for these beyond design bases events that would preclude any at-power operating configuration not addressed by the mitigating analysis. This requirement is problematic for several reasons:

- Existing technical specifications control the initial conditions for equipment credited in the safety analyses of design bases events. The new TBS is still considered a large break LOCA, and all of the same equipment remains in the technical specifications, and the allowed outage times for that equipment do not change as a result of this rulemaking. This equipment will also continue to provide mitigation for breaks larger than the TBS. Thus, it follows that the existing operational restrictions on this equipment, which are based on the more conservative traditional ECCS analyses, are more than sufficient to provide reasonable assurance that the same equipment can mitigate breaks larger than the TBS, as analyzed in the realistically conservative mitigating analysis.

- It does not make sense to waive the single failure criterion and credit redundant trains of mitigation equipment in one part of the rule, and then not allow a train to be removed from service for the testing and maintenance that is allowed today in another part of the rule.

- The spectrum of breaks from the TBS up to the largest break are not risk significant. It is inconsistent with the intent of the rulemaking to add new requirements for non-risk significant matters.

- § 50.65(a)(4) already requires licensees to assess and manage operating configuration risk when equipment is removed from service. This requirement covers beyond design basis/severe accident risk for all modes of plant operation. Paragraph (d)(2) is redundant to this requirement.

For the above reasons, paragraph (d)(2) should be deleted.

RISP assessment process - paragraphs (c), (d) and (f)
The development and application of a RISP assessment process is detailed in several different requirements in the proposed rule. Our general concern with the RISP is that it is redundant to existing regulatory change control processes, is not safety-focused, and would be an extreme and unnecessary regulatory burden on both licensees and the NRC staff to implement. This concern is based on the following:

- § 50.90 governs amendments to a license including technical specifications. Any change that affects the license or technical specifications requires an amendment request by the licensee to the NRC for prior review and approval. As noted earlier, the mitigating equipment credited in safety analyses is already governed by technical specifications, and the adoption of § 50.46a itself does not change those specifications.

- § 50.59 governs changes to a licensed facility that may be conducted without obtaining a license amendment pursuant to § 50.90. Generally, § 50.59 provides a regulatory threshold for determining which changes a licensee may implement without prior NRC review and approval. While many of the criteria which establish this regulatory threshold are geared toward allowing only those changes which may minimally increase the probability or consequences of design bases events described in the Updated Final Safety Analysis Report, there are two criteria aimed at circumstances not previously evaluated in the UFSAR. These are § 50.59(c)(2)(v) and (vi):

    - Create a possibility for an accident of a different type than any previously evaluated in the UFSAR;
    - Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

- In addition, § 50.59(d)(2) requires the licensee to submit a report to the NRC that describes the changes and summarizes the evaluation of each at least every 24 months.

- § 50.71(e) requires periodic updates of the UFSAR to reflect the effects of any changes made to the plant during the reporting period.

- A licensee adopting § 50.46a will still be required to maintain the design bases accident analyses, including those for the spectrum of LOCAs up to the TBS. In addition, the licensee must maintain the mitigation analysis for the LOCAs up to the largest pipe in the reactor coolant system. Reporting requirements for deviations from these analyses are in § 50.46a(g).

- The adoption of § 50.46a does not allow non-compliance with any of the existing non-LOCA related regulations, e.g., fire protection, Part 100, etc.

- Any risk-informed amendment request that would be generated by one or more of the above regulatory controls follows the regulatory guidance for such amendments provided in RG 1.174. In addition, the NRC staff is empowered to request risk information, when warranted, on deterministically based amendment requests.

Our point in citing the above existing change controls and requirements in the regulatory framework is that it is virtually impossible for a licensee to make a significant adverse change to the risk profile of the plant that would not require an amendment request/prior NRC review and approval.

The RISP requirements in the proposed rule would impose risk assessment requirements for any change after adoption of the rule and would establish a new risk threshold for amendments in the 1E-7 range. This would be entirely inconsistent with the objective of risk-informed regulation and would drive both licensee and NRC attention and resources toward matters of residual risk significance. In addition, the RISP requirements would make the current licensing basis of a plant limitless, and would force the NRC to review and approve changes to areas that had never been reviewed and approved by the NRC. In short, the RISP provisions in the proposed rule would actually inhibit, not enable, both needed and optional changes to a plant and, as such, would not be a viable option for licensees.

While PRA itself was not a part of the expert elicitation that led to the development of the TBS in this proposed rule, we believe it is appropriate, as part of the overall effort to risk-inform the technical requirements in Part 50 (Option 3 from SECY 98-300), to provide reasonable assurance that the risk profile of the plant design configuration is maintained appropriately. The existing change control provisions in the regulatory framework noted above provide reasonable checks and balances to assure prior NRC review and approval of changes that could significantly impact the risk profile of the plant. What is missing from the current framework is a requirement for licensees to periodically assess the cumulative impact of changes on the risk profile. Such a requirement would provide additional assurance that post adoption of § 50.46a, the licensee is maintaining the risk profile of the plant appropriately.

Proposed § 50.46a(d)(5) is very similar to the intent of the type of requirement noted above that would complement the existing requirements in the regulatory framework.

Paragraph (e)(2) ECCS analysis for LOCAs involving breaks larger that the TBS
This paragraph discusses requirements for the calculations that demonstrate mitigation capability for beyond TBS breaks. The sentence requiring comparison to experimental data, "Comparisons to applicable experimental data must be made," is redundant and potentially problematic in regard to the previous sentence stating that there must be "sufficient justification." Certainly a successful comparison to experimental data is cause for justification. However, other approaches such as comparison of results to accepted analysis techniques or to text book approaches are also appropriate. The "sufficient justification" clause allows for a demonstration of the calculation approach that is appropriate to the importance of the phenomena without the specific requirement to benchmark data.

Paragraph (f)(4) Requirements for risk assessment – PRA
The wording of §50.46a(f)(4) "...to the extent that a PRA is used in the RISP risk assessment, it must..." is confusing. This language implies that only a PRA meeting the requirements of the next four paragraphs may be used. However, §50.46a(f)(5) allows other risk assessment methods to be used to address certain initiators, which would be a conflict with the current wording of §50.46a(f)(4).

The PRA scope requirements of §50.46a(f)(4)(i) appear excessive, and should invoke NRC policy regarding PRA scope requirements relative to application. NRC SECY-04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," states the following:

> "Furthermore, the PRA scope is such that all operational modes and initiating events that could change the regulatory decision *substantially* are included in the model quantitatively."

While the concept of "substantial to the decision" is captured in paragraph §50.46a(f)(4)(i), the rule itself should define what initiators and modes are substantial to the decision. As written, it appears that the licensee is required to justify anything other than an all modes all scope PRA would be sufficient. This is not regulatory clarity. In reality, PRAs currently do not exist for "all modes, all initiators" (for the purposes of discussion below, we interpret this to mean internal events, fire, seismic, and other external events at power, and internal events during shutdown). Since, as discussed below, we believe the internal events at power PRA is central to this rulemaking, we recommend rule language similar to that used in the final 10 CFR 50.69, which is the only other existing regulation invoking PRA requirements:

(i) *Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.*

(ii) *The licensee shall review changes to the plant, operational practices, applicable industry operational experience, and, as appropriate, update the PRA... The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.*

Relative to "substantial to the decision," LOCAs are internal event at power sequences, and are modeled in the internal events at power PRA. NRC believes that robustness of design for LBLOCA contributes to mitigation of other initiators, and this is true. However, adoption of § 50.46a by a licensee does not change other regulations that preserve safety and constrain the ability to make changes that could substantially reduce mitigation capability for any initiators. Thus, these other initiators are not "substantial to the decision" and the rule should constrain the PRA requirement to internal events at power. In addition, the internal events PRA in conjunction with other internal/external hazard risks (i.e., fire and seismic using up-to-date screening or margins studies) is adequate for assessing these changes because the hazard does not change, and mitigation capability will be maintained. Therefore, post adoption of § 50.46a, changes can be adequately assessed with less than a "full scope" PRA. Other regulatory controls, not affected by this rulemaking, include the following:

- Plant technical specifications, which provide specific performance requirements for all important mitigation systems and cannot be changed without NRC approval under 10 CFR 50.90. This process would be conducted under NRC Regulatory Guide 1.174, which would require NRC review and approval of the risk evaluation for all contributors.

- Fire protection regulations 10 CFR 50.48 and 10 CFR 50 Appendix B, which provide specific requirements for maintaining safe shutdown in the event of a fire.

- Maintenance rule (a)(4), which requires risk assessment and management of plant configuration, and includes specific and enforceable requirements for shutdown safety through a defense in depth approach.

- Seismic design requirements contained in 10 CFR Part 100

We believe that other initiators and modes of operation can be addressed by non-PRA analyses in considering plant changes enabled by § 50.46a, and could also be used in periodic assessments of the cumulative effect of changes. This could include the following basic elements:

Fire: Still must meet all existing fire protection requirements. Must preserve existing Appendix R/ 50.48 safe shutdown mitigation pathways. Proposed 50.46a modifications should be reviewed against existing screening analysis (FIVE) or existing fire PRA to determine that 1) screening remains valid, and 2) there is not a significant effect on the mitigation capability for equipment in the shutdown paths. The fire analysis should reflect the as built plant (should be updated from IPEEE as necessary)

Seismic: Still must meet seismic design bases. Modifications driven by 50.46a should be screened out unless they affect seismic fragilities or other specific seismic analyses (in which case compliance with other seismic requirements must be reviewed and maintained) Modifications should be reviewed against existing screening analysis (SMA) or existing seismic PRA to determine that 1) screening remains valid, and 2) that there is not a significant effect on the mitigation capability for equipment in the shutdown paths. The seismic analysis should reflect the as built plant (should be updated from IPEEE as necessary)

Shutdown: Still must meet existing requirements of 50.65(a)(4)

Paragraph (g)(1)(i) Reporting
The new reporting requirements include the addition of a 0.4 percent change in oxidation as a threshold for determining if a change, or the sum of changes, is significant. The rationale for selecting 0.4 percent is that it is the same, on a percentage basis, as the existing PCT change reporting requirement. This rationale is only true if one considers the range of interest of PCT as 0-2200oF [(50oF/2200oF)*(17 percent) = 0.4 percent]. If instead, one considers the range of interest of PCT as 1700-2200oF or 1800-2200oF, from the perspective of transient oxide build-up, this same rationale gives a significance threshold of 1.7 or 2.1 percent. On this basis, it is recommended that the significance threshold for changes in oxidation be revised to 2.0 percent.

Paragraph (g)(2) Reporting
We believe this requirement should address the result of the periodic assessment of the cumulative changes made during the period, including any PRA updates or other non-PRA analyses. (See recommended language in Enclosure C)

Paragraph (g)(3) Report summarizing changes
This requirement is redundant to § 50.59(d)(2) as well as § 50.71(e), which address both a summary of all changes and their evaluations and their effects on the information in the UFSAR. This requirement should be deleted.

Paragraph (h) Documentation
Same as comment on paragraph (g)(3) above. This requirement is redundant and should be deleted.

Paragraph (m) Changes to TBS
We believe that any subsequent changes to the TBS should be accomplished by rulemaking, and that § 50.109 should apply as it does today. The provision in proposed § 50.109(b)(2) should be deleted. We endorse the more specific comments on this paragraph submitted by Duke Power.

All of the comments made in this enclosure are reflected in Enclosure 3, *Suggested Revision to the Proposed Rules.*

Suggested Revisions to the Proposed Rules

§ 50.34(a)(4) *Preliminary Safety Analysis Report*

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 or § 50.46a, and § 50.46b.

§ 50.34(b)(4) *Final Safety Analysis Report*

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents LOCAs shall must be performed in accordance with the requirements of § 50.46 or 50.46a, and 50.46b.

§ 50.46 *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.*

(a) (1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy, ZIRLO, or M5 cladding must be provided with an emergency core cooling system (ECCS). Reactors must be designed in accordance with the requirements of either this section or § 50.46a.

§ 50.46a Alternative acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(a) *Definitions.* Definitions for the purposes of this section:

(1) *Evaluation model* means the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in

the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

(2) *Loss-of-coolant accidents (LOCAs)* means the hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(3) *Plant design configuration means* the physical plant design and equipment capability that affect plant response to a LOCA.

(4) *Transition break size (TBS)* is a break of area equal to the cross-sectional flow area of the inside diameter of specified piping for a specific reactor. For a pressurized water reactor with large piping connected to both the hot and cold legs, the specified piping for the transition break size for the hot leg is the largest connecting pipe on the hot leg, and the specified piping for the cold leg is the largest connecting pipe on the cold leg. For PWRs with no large piping connected to the cold legs, the transition break size for the cold leg is the same as for the hot leg. The specified piping for a boiling water reactor is equivalent to 16 inch schedule 80 piping in the shutdown cooling suction line inside containment.

(b) *Applicability and scope.*

(1) The requirements of this section apply to each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy, ZIRLO or M5 cladding.

(2) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part, with the exception of § 50.46. The criteria set forth in paragraph (e)(3) and (e)(4), with cooling performance calculated in accordance with an acceptable evaluation model or analysis method under paragraph (e)(1) and (e)(2) of this section, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A to this part.

(c) *Application.*

(1) A licensee voluntarily choosing to implement this section shall submit an application for a license amendment under Sec. 50.90 that contains the following information:

(i) A description of the method(s) for demonstrating compliance with the ECCS criteria in paragraph (e) of this section;

(ii) A description of the PRA or other risk analyses used to comply with paragraph (f) of this section.

(d) *Requirements for implementation.* A licensee whose application under paragraph (c) of this section is approved by the NRC shall comply with the following requirements until the licensee submits the certifications required by Sec. 50.82(a):

(1) The licensee shall maintain ECCS model(s) and/or analysis method(s) meeting the acceptance requirements in paragraphs (e)(1) and (e)(2) of this section;

(2) The licensee shall periodically assess the cumulative effect of changes to the plant design configuration, and update as necessary, the PRA and other risk analyses described under paragraph (f) of this section to address changes to the plant, operational practices, equipment performance, plant operational experience, and revisions in analysis methods, model scope, data, and modeling assumptions. The assessment must be completed every two refueling outages, not to exceed four years.

(3) The licensee shall use the results of the assessments to provide reasonable assurance that the plant design configuration risk is maintained appropriately.

(e) *ECCS Performance.* Each nuclear power reactor subject to this section must be provided with an ECCS that must be designed so that its ECCS calculated cooling performance following postulated LOCAs conforms to the criteria set forth in this section. The evaluation models for LOCAs involving breaks at or below the TBS must meet the criteria in this paragraph, and must be approved for use by the NRC. Appendix K, Part II, 10 CFR Part 50, sets forth the documentation requirements for evaluation models for LOCAs involving breaks at or below the TBS. The analysis methods for LOCAs involving breaks larger than the TBS must be maintained, available for inspection, and include the analytical approaches, equations, approximations, and assumptions.

(1) *ECCS evaluation for LOCAs involving breaks at or below the TBS.* ECCS cooling performance at or below the TBS must be calculated in accordance with an evaluation model that meets the requirements of either section I to Appendix K of this part, or the following requirements, and demonstrate that the acceptance criteria in paragraph (e)(3) of this section are satisfied. The evaluation model must be used for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs involving breaks at or below the TBS are analyzed. The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (e)(3) of this section, there is a high level of probability that the criteria would not be exceeded.

(2) *ECCS analyses for LOCAs involving breaks larger than the TBS.* ECCS cooling performance for LOCAs involving breaks larger than the TBS must be calculated and must demonstrate that the acceptance criteria in paragraph (e)(4) of this section are satisfied. The analysis method must address the most important phenomena in analyzing the course of the accident. Sufficient supporting justification, including the methodology used, must be available to show that the analytical technique reasonably describes the behavior of the reactor system during a LOCA from the TBS up to the double-ended rupture of the largest reactor coolant system pipe. The analysis must be performed for the double-ended rupture of the largest pipe in the reactor coolant system at the most severe location. The analysis may take credit for the availability of offsite power and does not require the assumption of a single failure. Realistic initial conditions and availability of equipment may be assumed if supported by plant-specific data or analysis.

(3) *Acceptance criteria for LOCAs involving breaks at or below the TBS.* The following acceptance criteria must be used in determining the acceptability of ECCS cooling performance:

(i) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature must not exceed 2200°F.

(ii) *Maximum cladding oxidation.* The calculated total oxidation of the cladding must not at any location exceed 0.17 times the total cladding thickness before oxidation. As used in this paragraph, total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding must be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness must be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(iii) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam must not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(iv) *Coolable geometry.* Calculated changes in core geometry must be such that the core remains amenable to cooling.

(v) *Long term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature must be maintained at an acceptably low value and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(4) *Acceptance criteria for LOCAs involving breaks larger than the TBS.* The following acceptance criteria must be used in determining the acceptability of ECCS cooling performance:

(i) *Coolable geometry.* Calculated changes in core geometry must be such that the core remains amenable to cooling.

(ii) *Long term cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature must be maintained at an acceptably low value and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(5) *Imposition of restrictions.* The Director of the Office of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraph (e) of this section.

(f) *Risk analysis requirements.* The risk analyses used in accordance with the update requirements in paragraph (d)(2) of this section shall meet the following:

(1) The PRA must, at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(2) To the extent that other risk analyses are used to support consideration of non internal events at power initiators and operating modes, they must be of sufficient quality and level of detail to support this application.

(3) Analyses, including PRAs, used to meet this paragraph must reasonably reflect the plant configuration and operating practices.

(g) *Reporting.*

(1) Each licensee shall estimate the effect of any change to or error in evaluation models or analysis methods or in the application of such models or methods to determine if the change or error is significant. For each change to or error discovered in an ECCS evaluation model or analysis method or in the application of such a model that affects the calculated results, the licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in Sec. 50.4. If the change or error is significant, the licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with Sec. 50.46a requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC-approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraphs (e)(3) or (e)(4) of this section is a reportable event as described in Sec. Sec. 50.55(e), 50.72 and 50.73. The licensee shall propose immediate steps to demonstrate compliance or bring plant design or

operation into compliance with Sec. 50.46a requirements. For the purpose of this paragraph, a significant change or error is:

(i) For LOCAs involving pipe breaks at or below the TBS, one which results either in a calculated peak fuel cladding temperature different by more than 50 [deg]F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 [deg]F; or a change in the calculated oxidation, or the sum of the absolute value of the changes in calculated oxidation, equals or exceeds 2.0 percent oxidation; or

(ii) For LOCAs involving pipe breaks larger than the TBS, one which results in a significant reduction in the capability to meet the requirements of paragraph (e)(4) of this section.

(2) Each licensee shall report the results of the assessment completed under paragraph (d)(2) of this section every two refueling outages, not to exceed four years.

Seismic Effects and Assessments

Section J of the statement of considerations for the new rule 10CFR 50.46a "Specific Topics Identified for Public Comment" identifies two seismic related topics.

NRC Specific Topic #2.

NRC Request - *"The TBS specified by the NRC in the proposed rule does not include an adjustment to address the effects of seismically-induced LOCAs. NRC is currently performing work to obtain better estimates of the likelihood of seismically-induced LOCAs larger than the TBS. By limiting the extent of degradation of reactor coolant system piping, the likelihood of seismically-induced LOCAs may not affect the basis for selecting the proposed TBS. However, if the results of the ongoing work indicate that seismic events could have a significant effect on overall LOCA frequencies, the NRC may need to develop a new TBS. To facilitate public comment on this issue, a report from this evaluation will be posted on the NRC rulemaking Web site before the end of the comment period. In December 2005, stakeholders should periodically check the NRC rulemaking web site for this information. The NRC requests specific public comments on the effects of pipe degradation on seismically-induced LOCA frequencies and the potential for affecting the selection of the TBS. The NRC also requests public comments on the results of the NRC evaluation that will be made available during the comment period."*

NEI Response - The NRC issued the above mentioned seismic report in December 2005 titled "*Seismic Considerations for the Transition Break Size*". The stated purpose of this NRC study was to obtain better estimates of the likelihood of seismically-induced LOCAs larger than the TBS. These likelihood of seismically-induced LOCA estimates were intended to be the criteria for determining whether seismic considerations affected the basis for selecting the proposed TBS. The NRC study of potential seismic effects on the TBS encompassed a review of:

1. Seismic experience data
2. Results of past seismic PRAs
3. Review of LLNL load combination program methods and results
4. Estimation of failure probability of unflawed piping based on results of piping test programs and current ASME code design procedures
5. Evaluation of flawed piping considering crack growth from normal and transient operating conditions with superimposed seismic events
6. Estimation of the probability of indirectly induced DEGB due to failure of primary coolant system component supports

The NRC report concluded that the likelihood of seismically induced LOCAs was small enough (smaller than 10^{-5} per year) that the NRC selected transition break size would not be affected. The report addressed both the direct seismic large LOCA failure likelihood (with and without piping flaws) and the indirectly induced large LOCA seismic risk, with the following conclusions taken from the report:

- Direct Seismic Large LOCA Failures

  - Unflawed Piping – "Analyses performed in this study show that seismic-induced failure probabilities of unflawed piping, as defined in Section 2, are significantly low compared to the frequency of 10^{-5} per year used as a basis to establish the TBS."
  - Flawed Piping – "In summary, this study has demonstrated that the critical flaws associated with the stresses induced by seismic events of 10^{-5} and 10^{-6} annual probability of exceedance are large, and coupled with other mitigative aspects, the probabilities of pipe breaks larger than the TBS are likely to be less than 10^{-5} per year."
- Indirect Seismic Large LOCA Failures –
  - "For the two cases considered, indirectly induced piping failure attributable to major component support failure has probabilities of occurrence of less than 10^{-5} per year – a threshold of interest. ...., the results indicate that indirectly induced piping failure is unlikely to govern the combined failure of piping."

We concur with the NRC conclusions from the study that the recommended TBS is not adversely affected by the consideration of seismic risk. This concurrence is based on agreement with the NRC study and the following:

  - The median seismic capacities for both the primary piping system and the primary system components are higher than most other safety related power plant components within the nuclear power plant. At the very high accelerations (very low return period seismic hazard) associated with the point at which the primary piping or the primary system components would be calculated to fail, many other safety related structures, systems and components with lower capacities would already be postulated to have failed and thus control the seismic risk.
  - The change in risk (delta risk) defined in Regulatory Guide 1.174 due to seismic is estimated to be extremely low. The creation of the TBS by itself does not produce a physical change to the plant that would result in an appreciable change in seismic risk.

<u>NRC Specific Topic #3.</u>

NRC Request - *"Depending on the outcome of an ongoing NRC study (see Section III.B.3 of this supplementary information), the final rule could include requirements for licensees to perform plant-specific assessments of seismically-induced pipe breaks. These assessments would need to consider piping degradation that would not be prejudiced by implementation of the licensee's inspection and repair programs. The assessments would have to demonstrate that reactor coolant system piping will withstand earthquakes such that the seismic contribution to the overall frequency of pipe breaks larger than the TBS is insignificant. The NRC requests specific public* *comments on this and any other potential options and approaches to address this issue."*

NEI Response - As stated in the response to Topic #2 above, the NRC study, *Seismic Considerations for the Transition Break Size,* concluded that the likelihood of seismically-induced LOCAs larger than the TBS was less than the 10^{-5} per year threshold of interest. As such, plant-specific assessments of seismically-induced pipe breaks should not be required. Supporting this conclusion are the following:

- Baseline seismic risk – the NRC study used the absolute (baseline) seismic risk of 10^{-5} per year as a metric to determine whether the proposed rule should be adjusted to reflect the effects of seismically-induced LOCAs. The NRC study utilized relatively dated seismic hazard data for their evaluation. EPRI has evaluated a limited number of example indirect LOCA seismic risks using updated seismic hazard data from the new plant program research studies and found these risks to also be less than 10^{-5}. The EPRI study found that while the latest seismic hazard has increased for some parts of the CEUS (central and eastern US) there are several mitigating phenomena that have been established within the new plant seismic program which tend to counter much of that increase, i.e.:
  - Ground motion incoherence
  - Truncation of lognormal distribution on the seismic hazard
  - Filtering of low magnitude earthquakes from the seismic hazard using the Cumulative Absolute Velocity (CAV) function
  - Negligible inelastic behavior to high frequency response
- Delta seismic risk - For a risk informed application, the change is risk should be the primary metric for decision making. The change in risk (delta risk) relative to seismic events is estimated to be negligible based on the fact that the TBS threshold does not directly impact either the seismic hazard (the seismic hazard is the same whether the 50.46a rule is established or not) or the plant SSC seismic fragilities (the fragilities are a function of the structures/equipment construction, their design, their load path, their anchorage, etc. Establishing a TBS does not alter these from a seismic fragility perspective).

From: "PIETRANGELO, Tony" <arp@nei.org>
To: <SECY@nrc.gov>
Date: Mon, Mar 6, 2006 9:17 AM
Subject: NEI Comments on Proposed Rule 10 CFR Part 50, RIN 3150-AH29

Attached are NEI comments on the proposed rule on Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements.

Mail Envelope Properties (440C4456.452 : 14 : 54354)

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