April 30, 2007

MEMORANDUM TO: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons

FROM: Luis A. Reyes ***/RA/***
Executive Director for Operations

SUBJECT: ADDITIONAL INFORMATION ON THE APPROACH (DETERMINISTIC OR RISK-INFORMED) EXPECTED TO BE TAKEN BY NEW REACTOR APPLICANTS (M070228)

On February 28, 2007, the U.S. Nuclear Regulatory Commission (NRC) Office of New Reactors (NRO) provided to the Commission a *Periodic Briefing on New Reactor Issues.* Following the NRO briefing, the Commission, in its related Staff Requirements Memorandum (SRM) M070228, dated March 12, 2007, directed the staff to "provide the Commission more information on the approach (deterministic or risk-informed) expected to be taken by new reactor applicants, including fire protection and any other relevant areas." This direction followed a discussion at the briefing regarding the approach likely to be used by new reactors in the area of fire protection, in light of the voluntary risk-informed fire protection regulation. This memorandum responds to the Commission's SRM by providing additional information on the requirements for fire protection for new reactors and on other regulations that provide a voluntary risk-informed and/or performance-based option for implementation.

Risk insights have been used by the NRC as a valuable complement to the NRC's deterministic approach for a number of years. Their use has increased in recent years following the issuance of the Commission's probabilistic risk assessment (PRA) policy statement in 1995 and the SRM dated April 15, 1997, "Staff Requirements - COMSECY-96-061 – Risk-Informed Performance-Based Regulation (DSI 12)," which stated, in part, that "the use of PRA technology should be increased in all regulatory matters...in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy...." Furthermore, in response to the staff's recommendations in SECY-98-300, "Options for Risk-informed Revisions to 10 CFR Part 50- 'Domestic Licensing of Production and Utilization Facilities,'" the Commission, in its SRM dated June 8, 1999, approved the staff's recommendation that risk-informed implementation of Part 50 should be voluntary for current licensees.

Following that guidance, the staff has developed revisions to 10 CFR Part 50 requirements that provide licensees the flexibility to use risk insights and adopt cost-effective methods for implementing the safety objectives underlying the requirements of regulations, when applicable.

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In response to the recent SRM, the staff performed a review to identify the more significant regulations that provide alternate approaches for implementation, and their applicability to new reactors. In addition, the staff has had some discussion with the Design-Centered Working Groups to determine the approach (deterministic, risk-informed, and/or performance-based) that is expected to be taken by the new reactor applicants. As a result, the staff identified six regulations and one current proposed rule, as specified below, that provide either a deterministic or a risk-informed and/or performance-based approach:

- 10 CFR 50.48, "Fire Protection"
- 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"
- 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"
- Proposed 10 CFR 50.46a, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements"
- 10 CFR 100.23, "Geologic and Seismic Siting Criteria"
- 10 CFR 50.55a(g) / 10 CFR 50.55a(a)(3), "Alternative Approach to Inservice Inspection Requirements"
- 10 CFR 50.55a(f) / 10 CFR 50.55a(a)(3), "Alternative Approach to Inservice Testing Requirements"

Although potential combined license applicants have not yet committed to a risk-informed or performance-based approach, the Enclosure to this memorandum provides additional information on the voluntary risk-informed and/or performance-based options for new reactor applicants associated with the above regulations.

In all of these activities, the staff has followed the Commission's direction to allow the use of risk-informed and performance-based approaches on a voluntary basis. Although the staff, the Advisory Committee on Reactor Safeguards, and the Commission recognize the added complexity of alternatives, the judgment was made that the added opportunity for effectiveness and efficiency more than offset the disadvantage.

Enclosure:
Risk-Informed and/or Performance-Based
Options for New Reactor Applicants

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- 10 CFR 50.55a(g) / 10 CFR 50.55a(a)(3), "Alternative Approach to Inservice Inspection Requirements"
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Risk-Informed and/or Performance-Based Options for New Reactor Applicants

The following discussion provides additional information concerning regulations that were identified by the staff as providing the option of using either a deterministic or a risk-informed and/or performance-based approach:

<u>10 CFR 50.48, "Fire Protection"</u>

The fire protection programs for new reactor applicants that submit applications in accordance with 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," are subject to 10 CFR 50.48(a). The criteria for enhanced fire protection are described in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements;" SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs;" and SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs." SECY-90-016 proposed enhanced fire protection criteria for evolutionary light-water reactors. SECY-93-087 recommended that the enhanced criteria be extended to include passive reactor designs. The Commission approved the staff recommendations in SECY-90-016 and SECY-93-087.

The criteria for enhanced fire protection as discussed in these SECY papers are based on the fire protection risk insights acquired from regulatory experience with existing operating reactors. Current certified designs and license applications also reference the industry consensus standard National Fire Protection Association (NFPA) 804, "Fire Protection for Advanced Light Water Reactor Electric Generating Plants," which provides deterministic requirements for advanced light-water reactor fire protection programs that are consistent with the NRC's enhanced fire protection requirements. NFPA 805 entitled "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," which the NRC endorsed in 10 CFR 50.48(c), is only applicable to existing plants.

The NFPA is developing a consensus standard for a risk-informed and/or performance-based approach to the fire protection program for new reactors in NFPA 806, "Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants." Committee participants include representatives from the NFPA, industry, and the NRC staff. This standard would allow both a deterministic and a risk-informed and/or performance-based approach to fire protection. The issuance of NFPA 806 is currently scheduled for 2010, therefore, new reactor design certifications (DCs) and combined licenses (COLs) that are granted before issuance of the standard will not be able to reference this standard. In the meantime, one DC applicant has indicated that it plans on using a basic methodology consistent with NFPA 805 for an internal fire assessment.

A new reactor license applicant that wants to adopt a risk-informed and/or performance-based approach to fire protection prior to issuance of NFPA 806 may submit its proposed program for staff review and approval as part of the license application. However, the staff is not aware of industry interest in following that approach. The staff anticipates that if licensees later elect to adopt the risk-informed approach of NFPA 806, it will be for the purpose of evaluating plant changes that impact the fire protection program subsequent to receiving a license.

10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors”

In 2004, the NRC amended its regulations to provide an alternative approach for establishing the requirements for treatment of structures, systems, and components (SSCs) for nuclear power reactors using a risk-informed method. Section 50.69 revised requirements with respect to ‘‘special treatment,’’ that is, those requirements that provide increased assurance (beyond normal industrial practices) that SSCs perform their design-basis functions. This amendment permitted licensees (and applicants for licenses) to use risk insights to remove SSCs of low safety significance from the scope of certain identified special treatment requirements and revised requirements for SSCs of greater safety significance.

This rule provides a voluntary risk-informed alternative to the traditional deterministic approach under which the regulatory requirements for nuclear reactors are based largely on a deterministic approach for protection against a defined set of design-basis events (DBEs). The risk-informed alternative allows licensees to establish requirements for treatment of SSCs using a risk-informed method of categorizing SSCs according to their safety significance. The categorization process uses a blend of deterministic and risk insights to develop an integrated assessment of the safety significance of particular SSCs, and then specifies requirements commensurate with the significance.

At the time the rule was promulgated, applicants for standard DCs were excluded from taking advantage of the rule. However, a new reactor applicant for a COL is allowed to request approval of implementation of 10 CFR 50.69 as part of its application and could then implement this risk-informed alternative after its license is granted. To date, potential COL applicants have not committed to this risk-informed and/or performance-based approach.

10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors”

On September 26, 1995, the Commission revised 10 CFR Part 50, Appendix J, to provide an Option B, which is a performance-based approach to leakage testing requirements. Option A of the rule specifies the leakage testing frequency for Type A tests, often referred to as integrated leak rate tests (ILRTs), based on a deterministic approach that requires a set of three ILRTs to be performed at approximately equal intervals during each 10-year service period. The rule change was based in large part on the risk insights developed from studies such as that described in NUREG-1493, “Performance-Based Containment Leak-Test Program,” issued September 1995, which demonstrated that the risk to the public would not significantly increase due to the proposed relief from the requirements of Appendix J to 10 CFR Part 50. Option B provides licensees the option of reducing the frequency of these leakage-rate tests based on how well the overall containment and its penetrations have performed in tests since the plant began operation. The provisions allow extending the ILRT surveillance interval from three in 10 years to one in 10 years. All licensees currently use Option B for their testing programs. Since 2001, most licensees have requested (and the NRC has granted) one-time ILRT interval extensions of 15 years using industry performance data and risk-informed guidance based on NUREG-1493. The staff is currently reviewing an industry topical report that may support a permanent 15-year ILRT interval.

The AP1000 DC and Economic Simplified Boiling Water Reactor DC application include both Options A and B of Appendix J, either of which can be chosen by the COL applicants for meeting the requirements of this appendix. However, based on the current industry approach, the staff expects that Option B will be requested by new reactor COL applicants.

Proposed 10 CFR 50.46a, Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements

On November 7, 2005, the staff published for public comment a proposed rule adding a new Section 50.46a, “Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” to provide an alternative, risk-informed set of requirements for emergency core cooling systems (ECCSs). The proposed rule would allow current light-water reactor licensees to voluntarily adopt these requirements, which are intended to give licensees additional flexibility to change the designs of reactor ECCSs.

The proposed rule divides the current spectrum of loss-of-coolant accident (LOCA) break sizes into two regions. The division between the two regions is determined by a “transition” break size (TBS). The first region includes small breaks up to and including the TBS. The second region includes breaks larger than the TBS, up to and including the double-ended guillotine break of the largest reactor coolant system pipe. LOCAs in the smaller break size region would continue to be considered “design-basis accidents” and would be analyzed by current design-basis accident methods, assumptions, and acceptance criteria. LOCAs in the larger break size region would also need to be mitigated, but the proposed rule would allow licensees to analyze them using more realistic analytical methods and initial conditions without postulating the loss of offsite power or the worst case single failure.

The public comment period on the proposed rule ended on March 8, 2006. Of the 13 sets of comments received, 11 came from the nuclear industry. While the staff was evaluating the comments, it posted revised draft rule language on the NRC Web site to facilitate stakeholder involvement as the issues underwent resolution. The staff received comments stating that the rule should apply to new light-water reactors that are similar in design to existing plants. As a result, the staff modified the draft rule to apply to new light-water reactor designs that are determined by the NRC to be similar to existing light-water reactors. In light of the public comments, the staff expects that when the final rule is published, it would allow new reactor applicants to use the new 10 CFR 50.46a rule.

10 CFR 100.23, “Geologic and Seismic Siting Criteria”

Section 100.23, paragraph (d)(1), “Determination of the Safe Shutdown Earthquake Ground Motion,” requires that uncertainty inherent in estimates of the safe shutdown earthquake (SSE) must be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA) or a suitable sensitivity analysis.

Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” to 10 CFR Part 100, “Reactor Site Criteria,” provides the deterministic seismic design requirements, including the definition of the SSE, for currently operating nuclear power reactors. Past licensing experience in applying Appendix A has demonstrated the need to formulate procedures that quantitatively incorporate uncertainty in the evaluation of seismic hazards.

A single deterministic representation of seismic sources and ground motions at a site may not explicitly provide a quantitative representation of the uncertainties in geological, seismological, and geophysical data and alternative scientific interpretations.

In the 1980s, PSHA was developed to complement the deterministic approach that was included in Appendix A. As a result, 10 CFR 100.23, which specifies PSHA as a suitable method for determining the seismic hazard, was developed in the mid 1990s and applies to early site permits (ESPs) and COLs issued on or after January 10, 1997. In Regulatory Guide (RG) 1.165, “Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion,” issued in March 1997, the agency provides general guidance on procedures acceptable to the NRC for conducting a PSHA. The RG specifies a risk-informed probabilistic method for determining the site SSE. In March 2007, the agency issued RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” to provide an alternative to RG 1.165 for defining the site-specific earthquake ground motion that satisfies the requirements of 10 CFR 100.23 for establishing of the SSE. RG 1.208 specifies more detailed performance-based criteria for determining the site-specific SSE than RG 1.165; however, both approaches satisfy the requirements of 10 CFR 100.23.

New reactor applicants will be required to comply with the probabilistic approach specified in 10 CFR 100.23 using either of the risk-informed approaches described in RG 1.165 or RG 1.208. Two of the four ESP applicants have used the performance-based approach of RG 1.208 and it is expected that all of the COL applicants will also use this approach.

<u>10 CFR 50.55a(g) /10 CFR 50.55a(a)(3), Alternative Approach to Inservice Inspection Requirements</u>

The current inservice inspection (ISI) requirements for piping components are found in 10 CFR 50.55a and the General Design Criteria listed in Appendix A to 10 CFR 50. Section 50.55a (g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Code entitled "Rules for Inservice Inspection of Nuclear Power Plant Components," except where specific written relief has been granted by the Commission pursuant to Paragraphs (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a.

The scope for ASME Code, Section XI, ISI programs is based largely on the deterministic approach contained in design stress reports. As an alternative, the industry proposed the application of a risk-informed approach in order to determine the scope of ISI programs.

As a result, Westinghouse Topical Report WCAP-14572, Revision 1-NP-A, “Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection," and Electric Power Research Institute Topical Report TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection [RI-ISI] Procedure," have been developed and were accepted by the NRC as alternative approaches to the current ISI requirements. Subsequently, the NRC issued RG 1.178, “An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping,” to provide guidance on acceptable alternative approaches to meeting the requirements for the scope and frequency of inspection of ISI programs.

The risk-informed ISI programs yield significant benefits to the existing power plants in terms of monitoring the integrity of safety significant piping components. Hence, a new reactor applicant may pursue this risk-informed alternative after its license is granted. To date, potential COL applicants have not committed to the ISI risk-informed approach.

10 CFR 50.55a(f) / 10 CFR 50.55a(a)(3), Alternative Approach to Inservice Testing Requirements

Section 50.55a requires that, for new plants, Inservice Testing (IST) shall be performed in accordance with the specified ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to Paragraphs (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In support of the alternative concept of the IST program, RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," was issued in August 1998 to describe an acceptable alternative approach applying risk insights from probabilistic risk assessment to make changes to an IST program.

Additionally, 10 CFR 50.55a incorporated by reference RG 1.192, which endorses (with conditions) ASME Code Case OMN-3 that provides a risk-informed process that may be used when developing IST programs. This process may be used without prior staff review and approval and establishes the component safety methodology for classifying certain SSCs into high safety significant and low safety significant categories.

New reactor applicants may pursue this risk-informed alternative after their licenses are granted. To date, potential COL applicants have not committed to the IST risk-informed approach.