POLICY ISSUE
(Notation Vote)

September 14, 2010 SECY-10-0121

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: MODIFYING THE RISK-INFORMED REGULATORY GUIDANCE FOR NEW REACTORS

PURPOSE:

Since the U.S. Nuclear Regulatory Commission (NRC) published its probabilistic risk assessment (PRA) policy statement in 1995, the NRC staff has developed or endorsed many guidance documents to support risk-informed changes to the licensing basis and the Reactor Oversight Process (ROP). The purpose of this paper is to request Commission approval of the staff's recommendation to modify the risk-informed regulatory guidance to (1) recognize the lower risk profiles of new reactors[1] and (2) prevent a significant decrease in the enhanced levels of safety provided by new reactors.

SUMMARY:

In early 2009, the staff provided the Commission with a memorandum and white paper that identified potential issues with applying the current guidance for risk-informed changes to the licensing basis (including operational programs such as risk-managed technical specifications) and the ROP to new reactors with lower risk estimates. In the memorandum, the staff informed the Commission about the staff's intent to engage external stakeholders in the development of potential options to modify risk-informed regulatory guidance for new reactors.

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[1] For the purpose of this paper, the term "new reactor" refers to large light-water reactors (LWRs) that have been certified or are under review as standard designs by the NRC.

This paper defines several options for consideration by the Commission. The NRC staff recommends an option in which the NRC staff, together with stakeholders, identifies appropriate changes to the existing risk-informed guidance for changes to the licensing basis, including operational programs, and to the ROP.

BACKGROUND:

For both operating and new reactors, regulatory interactions in the areas of licensing and oversight rely upon a number of regulatory processes and guidance, some of which are risk-informed. The current framework that supports risk-informed regulation for reactors consists of guidance that can be grouped into four major categories:

(1) *Guidance for changes to a licensee's approved licensing basis without prior NRC approval.* In this category, the NRC's endorsement of Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," in Regulatory Guide (RG) 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," supports implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, Tests and Experiments."

(2) *Risk-informed guidance to support changes to a licensee's approved licensing basis, including operational programs, with prior NRC approval.* In this category, RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and associated guidance (e.g., RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications") provide a risk-informed integrated decisionmaking framework.

(3) *Guidance to support implementation of risk-informed regulations.* In this category, NRC endorsement of Nuclear Management and Resources Council 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," supports implementation of the Maintenance Rule (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants").

(4) *Guidance to support implementation of the ROP.* Management Directive (MD) 8.13, "Reactor Oversight Process," dated June 19, 2002, documents the staff's oversight process under the ROP. The NRC Inspection Manual describes the implementation of specific aspects of the ROP.

Given Commission guidance on its expectations about enhanced safety for new reactor designs, implementing the above guidance when reviewing applications under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," presents several challenges. In its policy statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985 (50 *Federal Register* (FR) 32138), the Commission stated that it "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe-accident safety performance than their prior designs." The policy statement, "Regulation of Advanced Nuclear Power Plants," dated July 8, 1986 (and restated July 12, 1994; 59 FR 35461), further states that "the Commission expects that advanced

reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions." This policy is effectively implemented by design certifications, which codify in rules the severe-accident enhancements in the new reactor designs, and by environmental reviews, which consider severe-accident mitigation design alternatives based on the lower risk profile estimates of the new reactor design.

In the staff requirements memorandum dated February 15, 1991, for SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," dated November 8, 1990, the Commission approved a process similar to that in 10 CFR 50.59 for making changes to Tier 2 information between issuance of a combined license (COL) and authorization for operation. The Commission stated that "the staff should ensure that this process requires preservation of the severe accident, human factors, and operating experience insights that are part of the certified design." Under 10 CFR Part 52, the process for changes to and departures from each certified reactor design appears in Section VIII of the appendix that contains its design certification rule.

Further, the Statement of Considerations of the standard design certification for the Advanced Boiling Water Reactor (ABWR) design (62 FR 25800, 25810; May 12, 1997) highlights the Commission's position on the change process as it relates to the PRA and severe accidents:

> The Commission recognizes that the ABWR design not only meets the Commission's safety goals for internal events, but also offers a substantial overall enhancement in safety as compared, generally, with current generation of operating power reactors. The Commission recognizes that the safety enhancement is the result of many elements of the design, and that much but not all of it is reflected in the results of the probabilistic risk assessment (PRA) performed and documented for them. In adopting a rule that the safety enhancement should not be eroded significantly by exemption requests, the Commission recognizes and expects that this will require both careful analysis and sound judgment, especially considering uncertainties in the PRA and the lack of a precise, quantified definition of the enhancement which would be used as the standard.

The Statement of Considerations also includes the following Commission statement:

> The Commission on its part also has a reasonable expectation that vendors and utilities will cooperate with the Commission in assuring that the level of enhanced safety believed to be achieved with this design will be reasonably maintained for the period of the certification (including renewal). This expectation that industry will cooperate with NRC in maintaining the safety level of the certified designs applies to design changes suggested by new information, to renewals, and to changes under section VIII.B.5 of the final rule. If this reasonable expectation is not realized, the Commission would carefully review the underlying reasons and, if the circumstances were sufficiently persuasive, consider the need to reexamine the backfitting and renewal standards in Part 52 and the criteria for Tier 2 changes under section VIII.B.5.

On February 12, 2009, the staff provided the Commission a memorandum with an enclosed white paper, "White Paper on Options for Risk Metrics for New Reactors" (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML090150636 and ML090160004, respectively) (Enclosure 1) that identified potential issues with applying the current guidance for risk-informed changes to the licensing basis (including operational programs such as risk-managed technical specifications) and the ROP to new reactors with lower risk estimates. In the memorandum, the staff informed the Commission about the staff's intent to engage external stakeholders in the development of potential options. As discussed below, the staff held many dialogues with external stakeholders on the issues raised in the 2009 white paper.

DISCUSSION:

With the implementation of an enhanced level of severe-accident prevention and mitigation design capability being confirmed through the review of applications for design certification for new LWRs, the staff is identifying potential issues that may arise with the transition to operations and the use of the existing risk-informed framework. Although RG 1.174 and the current ROP have no specific provisions precluding their application to new reactor designs, the NRC experience with implementing both RG 1.174 and the ROP has only involved currently operating plants. As discussed in the 2009 white paper, the staff identified a number of potential issues posed by the lower risk estimates of new reactors using the current risk-informed guidance that could potentially allow for a significant erosion of the enhanced safety of new reactors as originally licensed. As a result, the staff is considering whether changes to RG 1.174 and the ROP are needed in light of the differing risk profiles and the 10 CFR Part 52 process (e.g., design certification rulemaking on enhanced severe-accident features per Section VIII.B.5 discussed above). The staff is currently reviewing one application for risk-informed technical specifications initiatives 4b and 5b (on completion times and surveillance test intervals, respectively) as part of the U.S. Advanced Pressurized-Water Reactor design certification. In addition, other industry representatives have expressed interest in pursuing risk-informed inservice inspection of piping for new reactors, and the staff expects additional risk-informed applications for new reactors in the future.

Risk-Informed Changes to the Licensing Basis and Operational Programs

RG 1.174 provides an approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis for current reactors. This guide provides the basis for many other risk-informed programs (e.g., risk-informed inservice testing, risk-informed inservice inspection of piping, and risk-managed technical specifications).

RG 1.174 describes five principles for making risk-informed decisions. Specifically, the proposed change should be shown to do the following:

- Meet current regulations, unless the change is explicitly related to a requested exemption.
- Be consistent with the defense-in-depth philosophy.
- Maintain sufficient safety margins.

- Result in an increase in core damage frequency (CDF) or risk that is small and consistent with the intent of the Commission's safety goal policy statement.

- Include monitoring that uses performance measurement strategies.

Figures 3 and 4 of RG 1.174 provide acceptance guidelines for what constitutes "small changes" in both CDF (ΔCDF) and large early release frequency (ΔLERF). In RG 1.174, the acceptance guidelines for "small" and "very small" are defined relative to the Commission's safety goal policy statement and not to the specific plant's risk profile. For most new LWRs, which have baseline CDF estimates at or substantially below 10^{-6} per year, a ΔCDF of 10^{-6} or even 10^{-7} would not constitute a "small change" on a relative basis to the plant's risk profile. A change that is considered a "small increase" for current reactors under RG 1.174 may not have the same ramifications when applied to new reactors. Furthermore, RG 1.174 does not explicitly consider the impact of changes on the enhanced severe-accident safety features included in new reactor designs, which could result in the increased levels of safety achieved by these enhanced features being significantly reduced during operations unless specific guidance is developed to maintain these enhanced levels. RG 1.174 also does not address whether changes in large release frequency, which is used in new reactor licensing, should be considered when evaluating "small changes."

In addition, a number of important operational programs also have close ties to the current risk-informed regulatory framework. The extent to which these operational programs rely on quantitative risk metric guidelines varies. In risk-informed technical specifications initiative 4b, the derived completion times have a relationship to the PRA results, although they contain deterministic backstops consistent with the PRA policy statement that the PRA should complement the traditional deterministic approach and not replace it. In other cases, the analysis may be less quantitative and more qualitative in nature. For example, under 10 CFR 50.65(a)(4), the licensee "shall assess and manage the increase in risk that may result from the proposed maintenance activities" before performing the maintenance. The maintenance risk can be assessed using risk insights that are qualitative or quantitative in nature. Here again, the question of what constitute "small changes" in CDF and risk when applied to new reactors for these and other operational programs needs to be addressed. Without changes to the guidance documents for risk-informed changes to the licensing basis and operational programs, the Commission's expectations for new plants may not be met.

Reactor Oversight Process

The regulatory framework for reactor oversight is a risk-informed, tiered approach to overseeing plant safety. The framework has three key strategic performance areas: reactor safety, radiation safety, and security. Each strategic performance area has cornerstones that reflect the essential safety aspects of facility operation. Satisfactory licensee performance of the cornerstones provides reasonable assurance of safe facility operation and that the NRC's safety mission is being accomplished. Within this framework, the ROP provides a means of collecting information about licensee performance, assessing the information for its safety significance, responding to degraded licensee performance, and ensuring that licensees take appropriate corrective actions. Because there are many aspects of facility operation and maintenance, the NRC inspects licensee programs and processes on a risk-informed sampling basis to obtain representative information.

With regard to setting numerical thresholds, SECY-99-007, "Recommendations for Reactor Oversight Process Improvements," dated January 8, 1999, discusses a close link to RG 1.174:

> The concept for setting performance thresholds includes consideration of risk and regulatory response to different levels of licensee performance. The approach is intended to be consistent with other NRC risk-informed regulatory applications and policies as well as consistent with regulatory requirements and limits…(2) the thresholds should be risk informed to the extent practical, but should accommodate defense in depth and indications based on existing regulatory requirements and safety analyses; (3) the risk implications and regulatory actions associated with each performance band and associated threshold should be consistent with other NRC risk applications, and based on existing criteria where possible (e.g., Regulatory Guide 1.174).

Additionally, consistent with the principles of RG 1.174, SECY-99-007 provides the framework for meeting cornerstone objectives with minimal reduction in safety margin.

The ROP is designed to respond to declining performance, utilizing risk insights and other factors to focus inspections and regulatory response. Because the ROP is *risk-informed,* thresholds for regulatory engagement are largely based on quantification of ΔCDF and ΔLERF. And since a new reactor generally has a lower risk profile than currently operating reactors, applying the same thresholds used for the current reactors to licensee safety performance at a new reactor site could allow more significant *relative* degradation in performance before NRC engagement would be invoked by the ROP.

One of the staff's concerns is that the existing ROP may not provide for meaningful regulatory oversight for new reactors that can support the NRC's regulatory actions and inspection as performance declines. The current risk-informed baseline inspection program and risk-informed thresholds for performance indicators may not trigger a regulatory response before significant erosion occurs to the enhanced defense in depth and safety margins of the plant.

Interactions with Stakeholders

The staff developed an initial set of possible options for risk metrics for new reactors in early 2009. Through subsequent public meetings, the staff engaged stakeholders, including the Advisory Committee on Reactor Safeguards (ACRS), to further assess these options. Industry representatives expressed the opinion that new and currently operating reactors should be treated the same with respect to risk-informed changes to the licensing basis and the ROP (i.e., status quo). NEI issued its own white paper (Enclosure 2) describing why it believes that the current metrics are technically justified and appropriate for all plants, based on reasonable assurance of public health and safety, including operation at a prudent margin above adequate protection. A Union of Concerned Scientists representative expressed the opinion that it was premature to consider any options so far in advance of reactor construction and operation. The representative further stated that, although new reactors appear to be safer than the currently operating fleet, the public should get the benefit of this safety through the implementation of more stringent acceptance guidelines for licensing and thresholds in the ROP. Finally, the staff discussed the options presented in this paper with ACRS at a June 10, 2010, full committee meeting. In a letter to the Commission dated July 27, 2010 (ADAMS Accession

No. ML102000422) (Enclosure 3), ACRS agreed with the staff's position on the proposed framework as described in Option 2. The staff reviewed the ACRS letter and responded on August 25, 2010 (ADAMS Accession No. ML102210553) (Enclosure 4).

Options for Modifying the Risk-Informed Regulatory Guidance

The staff is requesting Commission direction before the staff implements new guidance, if any, for the review of a number of industry-proposed risk-informed initiatives for new reactors. The staff believes that potential policy issues associated with the ROP for new reactors should be addressed at this time because of the link to RG 1.174 and the goal to maintain consistency with other risk applications. Specifically, the staff is requesting Commission direction on its expectations for enhanced severe-accident safety performance for new reactors. This direction will determine the staff's approach to risk-informed changes to the licensing basis that could be viewed as voluntary changes to the design or operational programs (e.g., risk-managed technical specifications and risk-informed inservice inspection of piping), as well as to the risk-informed elements of the ROP for new reactors.

Option 1: No changes to the existing risk-informed guidance for the ROP and for changes to the licensing basis, or status quo.

Under this option, the staff would continue to use the existing risk-informed framework for licensing changes and the ROP. This option could provide incentives to build reactors with enhanced severe-accident safety features; applicants and licensees who invest in and maintain additional safety features would have more flexibility to operate the plants with a reduction in regulatory interactions. However, Option 1 may not meet Commission expectations because it may not prevent significant decrease in enhanced safety through changes to the licensing basis and plant operations over plant life. In addition, Option 1 may not provide for meaningful regulatory oversight that supports the NRC's regulatory actions and inspection.

Option 2: Identify and implement appropriate changes to the existing risk-informed guidance.

Under this option, the staff would continue to work with stakeholders to (1) identify specific changes to the guidance for risk-informed licensing-basis changes that would prevent a significant decrease in the new reactor's level of safety over its life and (2) identify specific changes to the risk-informed guidance for the ROP to provide for meaningful regulatory oversight. This option would support the Commission's expectations for new plants. The implementation details would differ for changes to the guidance for risk-informing the licensing basis and changes to the ROP because of the differences in the scope of NRC and industry documents that would be affected and the general time frames for implementation of each process, as discussed below.

For changes to the licensing basis and operational programs, the staff would modify the risk-informed guidance to prevent a significant decrease in the level of safety provided by certified designs. Implementation of this option will support the Commission's expectation about the maintenance of the level of severe-accident safety performance of new designs. The staff would supplement the CDF and LERF acceptance guidelines to recognize the lower risk profiles

of new reactors, including revisiting the definition of “small” change when implementing RG 1.174. Specifically, the staff would do the following:

- Use stakeholder involvement in the evaluation and development of detailed changes to risk-informed regulatory guidance.

- Evaluate the merits of developing additional criteria (e.g., deterministic, defense in depth) to support the change process.

- Evaluate proposed changes to guidance to ensure that the changes do not create unintended consequences, such as creating disincentives for safer designs or allowing degradation of passive safety system performance. This would include developing guidance to implement Section VIII.B.5.c of the design certification rules.

For oversight, the staff would identify appropriate changes to the risk-informed elements of the ROP. These changes would reflect the enhanced level of severe-accident safety performance of new reactors while providing for meaningful regulatory oversight that supports the NRC’s regulatory actions and inspection, recognizing that the staff will continue to independently assess licensee performance in the area of safety culture, which addresses common underlying factors that affect plant safety. Specifically, the staff would do the following:

- Use stakeholder involvement in the evaluation and development of changes to the guidance.

- Evaluate the criteria for plant placement in the action matrix to assess whether or not the current process would ensure that operational performance resulting in significant reductions in the level of safety provided by the certified design is fully understood by the licensee and the NRC and is effectively corrected.

- Evaluate the merits of developing additional criteria (e.g., deterministic, change in risk) to support the NRC’s response to findings and performance trends.

- Evaluate any potential ROP changes to avoid unintended consequences, such as creating disincentives for safer designs, allowing degradation of passive safety system performance, or diverting the attention of NRC inspectors from issues of higher safety significance in currently operating reactors.

- Consider the need to risk-weigh or otherwise weigh findings associated with passive systems to reflect the difficulty of recognizing the degradation of passive systems.

- Evaluate maintaining or changing the current thresholds for green, white, yellow, and red risk-significant findings and performance indicators, given that low-risk designs may rarely, if ever, cross the current white threshold.

- Consider the advantages and disadvantages of applying any potential changes to the ROP to currently operating reactors.

A key advantage of Option 2 is that it would reaffirm the Commission's expectation of enhanced severe-accident safety performance for new reactors and the expectation that this level of enhanced safety will be reasonably maintained throughout plant life. The option addresses both plant design and operations, including licensing basis changes, operational programs, and oversight. Furthermore, Option 2 acknowledges that there are safety-margin and defense-in-depth considerations beyond the quantitative risk-informed thresholds. However, a disadvantage of Option 2 is the short time available to revise the guidance needed to support the staff's review of a number of risk-informed initiatives expected to be proposed by design certification and COL applicants, including risk-informed technical specifications initiatives 4b and 5b. Further, some stakeholders may view <u>any</u> change to thresholds that might be considered under Option 2 to be inconsistent with the underlying technical basis for the current thresholds that are derived from the Commission's safety goals and implemented in RG 1.174.

In addition to revising RG 1.174, Option 2 would necessitate changes to associated guidance for specific risk-informed applications. Changes to the ROP, including MD 8.13 and some Inspection Manual Chapters, would be necessary. Several industry documents endorsed by the staff may also be affected.

<u>Option 3</u>: Modify the risk-informed guidance to include a new risk metric for the ROP and changes to the licensing basis.

Under this option, acceptance guidelines for risk-informed changes to the licensing basis and/or numerical thresholds in the ROP would be lowered. Like Option 2, this option would reaffirm the Commission's expectation of enhanced severe-accident safety performance for new reactors and the expectation that this level of enhanced safety will be maintained throughout plant life. However, some internal and external stakeholders have indicated that this option goes beyond the Commission's expectation by essentially requiring that new reactors be measured against more stringent risk guidelines. Thus, they believe this option may be inconsistent with the NRC response to the comment by the North Carolina Waste Awareness and Reduction Network on the Commission's "Policy Statement on the Regulation of Advanced Reactors," dated October 14, 2008 (73 FR 60612) that advanced reactors need to be made safer, more robust and effective. The NRC's response says that the "policy statement does not state that advanced reactor designs must be safer than the current generation of reactors."

Option 3 would thus create a risk-informed framework that is, in effect, inconsistent with the underlying technical basis for the current thresholds that are derived from the Commission's safety goals and implemented in RG 1.174. This option may also have unintended consequences in that new reactors with enhanced safety features would have less operational flexibility than the current fleet of reactors; applicants who invest in additional safety features expect more flexibility to operate the plants with a reduction in regulatory interactions.

Option 3 would require major revision to RG 1.174 and associated guidance for specific risk-informed applications. Significant changes to ROP-related documents also would be necessary. Many industry documents endorsed by the staff would be affected.

RECOMMENDATION:

The staff recommends Option 2. The staff believes that Option 2 meets the Commission's expectation that there is "no significant decrease in the level of safety" over the life of the new reactor design. This option also creates a regulatory environment that encourages the design of new reactors with higher levels of severe-accident safety performance, including greater redundancy of safety systems, which may allow for greater operational flexibility. Stakeholder involvement in the development of the new guidance for changes to the licensing basis and in the identification of potential changes to the risk-informed elements of the ROP is a key feature of this option. If directed by the Commission to implement Option 2, the staff would keep the Commission apprised of progress in the development of such guidance.

RESOURCES:

Option 1 (no changes) would require no additional resources to carry out.

Option 2 (identify and implement appropriate changes to existing guidance) would require staff resources to engage stakeholders, evaluate proposed changes, and draft any needed updates to guidance documents. Based on recent experience with the development of risk-informed regulatory guidance, this effort is estimated to require no more than 1.0 full-time equivalent during fiscal year 2011. Although this activity is not specifically included in the fiscal year 2011 budget, resources for licensing support are available within the new reactors business line to complete this work.

Option 3 (modify guidance to include new risk metric) would require additional staff resources beyond those currently budgeted, depending on the extent of the changes needed to the guidance documents. Because the scope of these revisions is not yet known, resource estimates would be developed following stakeholder interaction, and appropriate adjustments would be made through the planning, budgeting, and performance management process.

COORDINATION:

This paper has been coordinated with the Office of the General Counsel, which has no legal objection. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. A copy of this paper in draft form was provided to ACRS on May 12, 2010. The staff discussed the options presented in this paper with ACRS at a June 10, 2010, full committee meeting.

/RA/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Memorandum to Commission and White Paper
2. NEI Letter - Transmission of Industry White Paper
3. ACRS Letter - Risk-Informed Regulatory Guidance for New Reactors
4. Letter to Dr. Said Abdel-Khalik, ACRS

February 12, 2009

MEMORANDUM TO: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
Commissioner Svinicki

FROM: R. W. Borchardt ***/RA Bruce S. Mallett for/***
Executive Director for Operations

SUBJECT: ALTERNATIVE RISK METRICS FOR NEW LIGHT-WATER REACTOR RISK-INFORMED APPLICATIONS

The purpose of this memorandum is to inform the Commission of the staff's intent to engage stakeholders in the consideration of alternative risk metrics for new light-water reactor (LWR) risk-informed applications. This memorandum also provides additional information on the discussions regarding risk-informed and performance-based regulation for new reactors provided during the February 4, 2009, Commission Briefing on Risk-Informed Performance Based.

Reactor risk metrics refer to the quantitative measures of risk from reactor operations up to and including severe core damage accidents. The two most common metrics used for operating reactors are core damage frequency (CDF) and large early release frequency (LERF).

The Commission has provided guidance regarding risk metrics and safety margins in the past. For example, the Commission stated in its policy statement of 1985 regarding severe reactor accidents for future designs and existing plants that it "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." Moreover, in its policy statement of 1994 regarding the regulation of advanced nuclear power plants, the Commission stated its expectation "that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions." As noted in the staff requirements memorandum dated June 26, 1990, on SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the Commission supported a goal of 10^{-4} per year of reactor operation for CDF and 10^{-6} per year for large release frequency (LRF) for use by the staff in its assessment of evolutionary LWRs. Additionally, the Commission approved a conditional containment failure probability (CCFP) objective of 0.1 as a basis for establishing regulatory guidance for evolutionary designs.

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CDF estimates for new reactors[1] are observed to be typically 10 to 1,000 times lower than those for currently operating reactors when internally initiated events and those externally initiated events that have been quantified are included.

Correspondingly, LRF (or LERF) estimates are 10 to 10,000 times lower for new reactors, while CCFP estimates are typically 3 to 10 times lower. The lower risk estimates for new reactors raise several issues regarding how to apply acceptance guidelines for changes to the licensing basis and thresholds in the Reactor Oversight Process (ROP) that were developed for currently operating reactors.

The enclosed white paper presents a full discussion of the issues and options for applying or modifying the current set of reactor risk metrics to new reactors. The paper discusses the issues posed by the lower risk estimates of new reactors in risk-informed applications, including changes to the licensing basis and the ROP, and describes the advantages and disadvantages of each option.

The staff is currently reviewing one application for risk-informed technical specifications initiatives 4b and 5b (concerning completion times and surveillance test intervals, respectively) as part of the US-APWR design certification and the Comanche Peak combined license application. In addition, other industry representatives have expressed interest in pursuing risk-informed inservice inspection of piping for new reactors, and the staff expects additional risk-informed applications for new reactors in the future.

During the staff's consideration of these risk-informed initiatives, the question arose of whether the current numerical risk metric goals for CDF and LERF should be applicable to new LWRs, or whether alternate metrics for CDF and LRF should be developed consistent with the Commission's safety expectations and approved goals for new reactors. If alternate metrics are developed, a related question is whether the new LWR risk metrics should be considered for both licensing and operations reviews.

The staff has developed an initial set of possible options for risk metrics for new reactors. The staff intends to engage stakeholders regarding the issues surrounding the application of the current set of reactor risk metrics to new reactors and to further assess and evaluate these options. Through these interactions, the staff will finalize a set of options for Commission consideration, including possible alternate risk metrics that could be applied to new reactors.

[1] For the purpose of this memo and the attached white paper, the term "new reactor" refers to evolutionary and advanced LWRs, including the plants using multi-train, mostly active engineered safeguards (Advanced Boiling Water Reactor (ABWR), System 80+, U.S. Advanced Pressurized-Water Reactor (US-APWR), U.S. Evolutionary Power Reactor (U.S. EPR)), as well as those plants with mainly passive safeguards systems (Advanced Passive 600 (AP600), Advanced Passive 1000 (AP1000), Economic Simplified Boiling-Water Reactor (ESBWR)).

The attached white paper provides a basis for discussion with stakeholders. These interactions will include one or more public meetings, one or more briefings with the Advisory Committee on Reactor Safeguards, and solicitation of written comments from interested stakeholders.

Enclosure:
As stated

cc: SECY
OGC
OCA
OPA
CFO

White Paper on
Options for Risk Metrics for New Reactors

I. PURPOSE

The purpose of this paper is to identify the issues posed by the lower risk estimates of new reactors in risk-informed applications, including the Reactor Oversight Process (ROP) and changes to the licensing basis. "New reactor" in this context refers to near-term, evolutionary, and advanced light-water reactors (LWRs). This paper, intended to provide a basis for further discussion, identifies several options for addressing these issues and describes the advantages and disadvantages of each.

II. SUMMARY

Reactor risk metrics refer to the quantitative measures of risk to the public from reactor operations up to and including severe core damage accidents. The two most common metrics are core damage frequency (CDF) and large early release frequency (LERF). These two measures are typically used as surrogates for latent and early fatality risks, respectively, from the Commission's quantitative health objectives (QHO) in the Safety Goal Policy Statement. [1] Additional metrics in the form of risk importance measures have been commonly applied in Title 10, Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (known as the maintenance rule), of the *Code of Federal Regulations* (10 CFR 50.65), and 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components [SSC] for Nuclear Power Reactors." [2,3] The current set of reactor metrics has met the agency's needs to date.

The Commission has provided guidance regarding risk metrics and safety margins in the past. The Commission stated that it "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." [4] Moreover, "the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions." [5] However, the Commission disapproved the use of 10^{-5} per year (/yr) CDF for advanced designs. [6] As noted in the staff requirements memorandum (SRM) dated June 26, 1990, on SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the Commission supports goals of 10^{-4}/yr of reactor operation for CDF and 10^{-6}/yr for large release frequency (LRF). [6] Additionally, the Commission has approved a conditional containment failure probability (CCFP) objective of 0.1 as a basis for establishing regulatory guidance for evolutionary designs. [6, 7]

CDF estimates for new reactors are typically 1 to 3 orders of magnitude lower than those for current designs when the contributions from external events that have been quantified, such as fire, are included. Correspondingly, LRF (or LERF) estimates are 1 to 4 orders of magnitude lower for new reactors. The lower risk estimates for new reactors raise several issues regarding how to apply acceptance guidelines for changes to the licensing basis and thresholds in the ROP that were developed with current reactors in mind.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis," provides an approach for using probabilistic risk assessment (PRA) in risk-informed decisions on plant-specific changes to the licensing basis for current reactors. [8] This guide is the foundation on which many other risk-informed programs (e.g., risk-informed inservice testing, risk-informed inservice inspectionof piping, and risk-managed technical specifications (TS)) are based at the agency. Figures 3 and 4 of RG 1.174 provide acceptance guidelines for changes in CDF (ΔCDF) and LERF (ΔLERF), respectively. For most new LWRs with baseline CDF estimates substantially below 10^{-6}/yr, a 10^{-6} ΔCDF, or even an order of magnitude lower 10^{-7} ΔCDF, would no longer constitute a "small change" *on a relative basis*. "Small increase" for current reactors may not have the same ramifications when applied to new reactors. Furthermore, RG 1.174 is based, in part, on LERF, while new reactors are reviewed against the LRF goal of 10^{-6}/yr.

As discussed below, most programs and processes relying on an *absolute* measure of risk such as Birnbaum importance, CDF, and LERF (and their associated increments) could raise several issues when applied to new reactors. Those programs and regulations relying to a greater extent on *relative* measures of risk including Fussell-Vesely (FV) and risk achievement worth (RAW) importance would appear to pose less of an issue for new reactors.

This paper suggests six possible options for addressing the issues related to the application of risk metrics for new reactors and discusses the advantages and disadvantages of each option. The options range from the status quo (i.e., applying current ROP thresholds and acceptance guidelines to new reactors), to the use of altogether new guidelines and thresholds for new reactors, and finally to postponing any significant change to the process and evaluating new reactors on a case-by-case basis for an indeterminate period. The purpose of these options is to provide a basis for further discussion of the issues surrounding the use of risk metrics for new reactors.

III. BACKGROUND

III.a Current Definitions Related to Risk Metrics

In general terms, "reactor risk metrics" refers to the quantitative measures of risk to the public from reactor operations up to and including severe core damage accidents. The two most common metrics are CDF and LERF. These two measures are typically used as surrogates for latent and early fatality risks, respectively, from the Commission's QHO in the Safety Goal Policy Statement. [1] Various risk-informed applications use several derivatives of these measures, including ΔCDF, ΔLERF, incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP). Additional metrics in the form of risk importance measures have been commonly applied in 10 CFR 50.65 and 10 CFR 50.69. LRF pertains primarily to new reactor design certification and combined license applications.

The current risk metrics have evolved to meet the needs of various program objectives. Some metrics such as CDF and LERF have definitions that vary slightly depending on the source, but these differences are usually minor. For example, most definitions of LERF (often calculated in limited Level 2 PRAs) contain some statement of "significant" or "rapid, unmitigated" release. They also contain some elements of a Level 3 PRA (e.g., "prior to effective evacuation" or "before the effective implementation of off-site emergency response").

Regarding LRF, the Commission requested the staff to provide a definition of LRF, but in SECY-93-138, "Recommendation on Large Release Definition," dated May 19, 1993 [9], the staff recommended to the Commission that work on a definition be terminated. As a result, the definitions of LRF in the design certification documents of the standard designs referenced in combined license applications all differ to varying extents.

Consistent with Commission direction, the staff has been flexible regarding the application of the definition of CCFP, which has a 0.1 objective for evolutionary designs. [7] In new reactor design certifications, the staff has determined the CCFP by first calculating the probability of the containment remaining intact (no release beyond design-basis leakage) for core damage events, and then taking the mathematical complement.

For the purpose of this paper, the term "new reactor" refers to evolutionary and advanced LWRs, including the plants using multi-train, mostly active engineered safeguards (Advanced Boiling Water Reactor (ABWR), System 80+, U.S. Advanced Pressurized-Water Reactor (US-APWR), U.S. Evolutionary Power Reactor (U.S. EPR)), as well as those plants with mainly passive safeguards systems (Advanced Passive 600 (AP600), Advanced Passive 1000 (AP1000), Economic Simplified Boiling-Water Reactor (ESBWR)). For this set of new designs, the existing risk metrics including CDF and LERF have applicability, as does perhaps LRF. This paper addresses the issues that could arise from using these and other risk metrics in risk-informed applications and the ROP.

For advanced reactors (non-LWRs), CDF may no longer be a useful metric. The framework for a risk-informed, performance-based regulatory structure for advanced reactor licensing has assessed alternate approaches to meet the QHO. [10] This paper does not discuss issues involving the application of risk metrics to advanced reactors.

III.b New Reactor Risk Goals and Objectives

The Commission issued its policy statement entitled, "Safety Goals for the Operation of Nuclear Power Plants," on August 4, 1986. [1] The policy statement established two qualitative safety goals and two QHOs. In subsequent implementation of the Commission policy statement, the staff demonstrated that CDF of 10^{-4}/yr and LERF of 10^{-5}/yr are acceptable surrogates for the latent and early QHO. [10]

The Commission stated that it "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." [4] Moreover, "the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions." [5] The Utility Requirements Document (URD) for advanced (evolutionary) LWRs has a more restrictive goal for CDF of 10^{-5}/yr and a goal of 10^{-6}/yr (the same as the Commission's) for LRF. [11] There is no design objective for CCFP. However, the Commission disapproved the use of 10^{-5}/yr CDF for advanced designs, noting that "although the Commission strongly supports the use of the information and experience gained from the current generation of reactors as a basis for improving the safety performance of new designs, the NRC should not adopt industry objectives as a basis for establishing new requirements." [6] As noted in the SRM on SECY-90-016, the Commission supports the use of 10^{-4}/yr of reactor operation as a CDF goal and 10^{-6}/yr for LRF. [6] (If defined consistently in terms of "large" release, one would expect LERF to be less than LRF for a particular plant design.)

Additionally, the Commission approved a CCFP objective of 0.1 as a basis for establishing regulatory guidance for evolutionary designs. [6,7] Thus, Commission policy has established the following design goals/objectives for new reactors:

- CDF $<10^{-4}$/yr
- CCFP <0.1
- LRF $<10^{-6}$/yr

Assuming for the purposes of discussion that containment failure necessarily results in a large release, the following relationship can be shown:

$$LRF = CDF * CCFP$$

Note that meeting the CDF goal of less than 10^{-4}/yr and CCFP of less than 0.1 does not ensure that the LRF goal of less than 10^{-6}/yr can be met. A CCFP of approximately 0.1 would require a CDF of less than 10^{-5}/yr to meet the LRF goal. Likewise, given a CDF of 10^{-4}/yr, the CCFP would have to be less than 10^{-2}. Even with engineered severe accident mitigation capabilities, the current technology would make the 10^{-2} CCFP value difficult to achieve for most new LWR designs. In effect, given the currently achievable CCFP of approximately 0.1 for new LWR designs, the Commission's goal for CDF basically defaults to the URD value of less than 10^{-5}/yr.

However, the URD goals, with no constraint on CCFP, essentially allow the LRF goal of 10^{-6}/yr to be met entirely by CDF $<10^{-6}$/yr, or by a combination of 10^{-5}/yr>CDF>10^{-6}/yr and the corresponding CCFP between 0.1 and 1 (such that CDF * CCFP $<10^{-6}$/yr).

The net effect of the two sets of goals and objectives (both Commission and utility requirements) is that the safety goals for new reactors essentially default to the following:

- CDF $<10^{-5}$/yr
- CCFP <0.1
- LRF $<10^{-6}$yr

III.c Estimates of Risk-Related Values for New Reactor Designs

Figure 1 shows the CDF estimates for the seven new LWR designs. These estimates include only the conventional internal events contribution (no internal flooding or fire). The designs with passive safety features appear to have somewhat lower CDFs than those with active safety systems, although there is some overlap. Also provided are relatively recent industry values for operating plants tabulated from the plants' Mitigating Systems Performance Index (MSPI) basis documents of April 2006. The average CDF for boiling-water reactors (BWRs) is $8x10^{-6}$/yr, while for pressurized-water reactors (PWRs) it is $2x10^{-5}$/yr. (These values from the MSPI are about a factor of 3 lower than the Individual Plant Examination (IPE) results [12], reflecting enhancements in design and operation, as well as improvements in PRA modeling.) Overall, the internal events CDF estimates from the new LWRs are about 1 to 3 orders of magnitude lower than the mean of operating reactors. Similar observations can be made when comparing the contribution of externally initiated events for new reactors to Individual Plant Examination of External Events (IPEEE) results. [13]

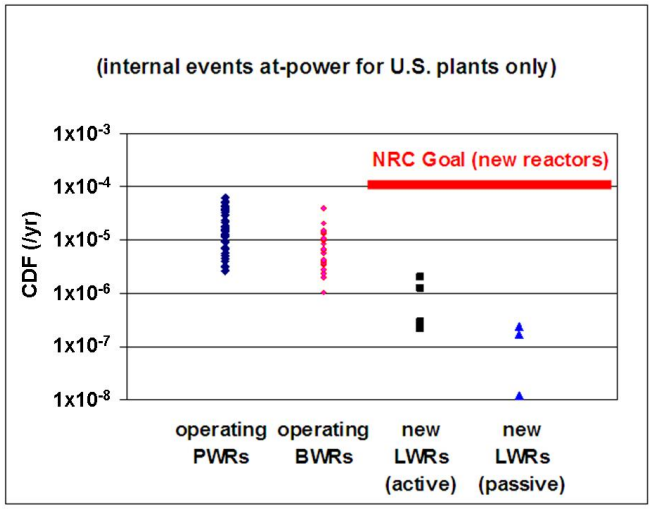

Figure 1 CDF estimates by plant type

Figure 2 shows the internal-events LRF estimates for new LWR designs. A comparison of LRF estimates to operating plants is problematic in that there is no consistent definition of "large release," and operating plants have used LERF as the primary Level 2 PRA metric. Moreover, unlike CDF values from the MSPI, there is no single comprehensive source of updated LERF values across the industry. Thus, *significant* early release frequency distributions from the IPE results are provided for reference. Given these qualifiers, it is generally observed that the estimated frequencies of large radiological release from new LWR designs are approximately 1 to 4 orders of magnitude lower than the mean of operating plants.

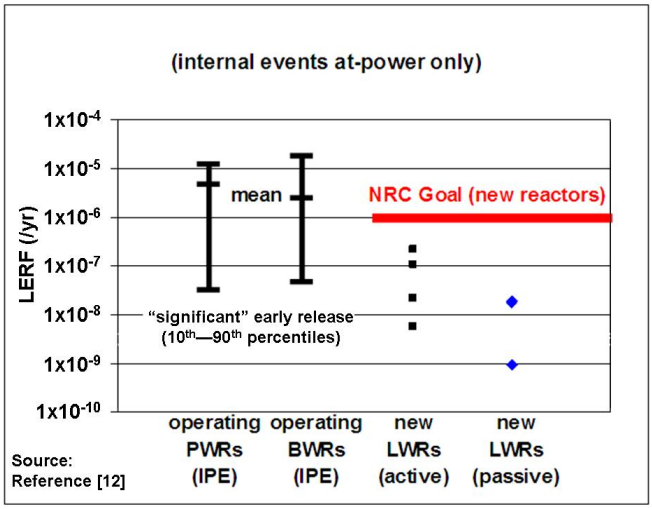

Figure 2 LRF estimates by plant type

The orders of magnitude differences in CDF and LERF (LRF) estimates between current reactors and new designs demonstrate great strides from an overall safety perspective. However, the lower risk estimates for new reactors also create issues regarding how to apply acceptance guidelines for changes to the licensing basis and thresholds in the ROP that were developed for current reactors. The following section discusses these issues in greater detail.

III.d Issues with the Application of Current Risk Metrics to New Reactors

Risk metrics are used in many risk-informed applications throughout the agency, including changes to the licensing basis and the ROP. It is assumed for the purposes of this paper that the reader is generally familiar with current programs and processes such as risk-informed changes to TS and the significance determination process (SDP) for characterizing the safety significance of performance deficiencies. Therefore, the discussion is limited to how risk metric values for new reactors might affect the decisions and outcomes of the various programs and processes.

III.d.1 Reactor Oversight Process

Mitigating Systems Performance Index

The MSPI is formulated as a simplified approximation of the change in CDF attributable to changes in reliability and availability of risk-significant elements of the system during internal events with the reactor operating at power. [14] Changes in train unavailability and monitored component unreliability from baseline values are weighted by Birnbaum importance measures of the basic events for the corresponding train and monitored components in the system. The Birnbaum importance can be represented in several ways, but its algebraic representation for unreliability is given in the MSPI by

$$CDF * FV/UR$$

where

CDF is the at-power plant CDF for internal events,
FV is the FV importance measure for the unreliability of the monitored component, and
UR is the unreliability of the component.

Therefore, to a first approximation, the MSPI value for a system is proportional to the baseline CDF.

The bands of performance for the MSPI are consistent with other elements of the ROP:

- "Green": for MSPI $\leq 10^{-6}$
- "White": for $10^{-6} < MSPI \leq 10^{-5}$
- "Yellow": for $10^{-5} < MSPI \leq 10^{-4}$
- "Red": for MSPI $> 10^{-4}$

It can be demonstrated that the number of failures of a monitored component necessary to cross a performance threshold (e.g., "Green/White"), is inversely related to the Birnbaum importance for the basic event of the failed component. Components with low basic event Birnbaum importance measures will require a greater number of failures during the monitored period (36 months) to cross a performance threshold than will those components with high Birnbaum importances. Correspondingly, plants with lower CDF will generally require more failures to cross a performance threshold than plants with higher CDF, all other things being equal. This is the risk-informed, plant-specific nature of the MSPI.

By way of an example, the nominal number of failures of emergency diesel generators necessary to cross the "Green/White" threshold in a plant with average CDF may be two to four

over a 3-year period. For some plants with a significantly lower CDF, say a factor of 3 lower, the number of failures might be 4 to 11. To address the concern that under some circumstances the MSPI formulation may result in needing an unacceptably large number of failures within a system to cross the "Green/White" threshold, a "backstop" or component performance limit was added. The backstop is based on a statistical correlation of actual versus expected number of failures over a 3-year monitoring period. It was designed to be invoked on rare occasion (a few percent of all the "White" performance indicators).

The nominal plant-to-plant variation in CDF for existing reactors is about a factor of 3 to 5. However, as indicated in Figure 1, CDF estimates from internal events for new reactors are expected to be 1 to 3 orders of magnitude lower than those of current reactors. Given the discussion above regarding the sensitivity of the MSPI to baseline CDF, it is fair to conclude that the MSPI values for new reactors would be very low. In essence, the MSPI would be largely insensitive to system performance in terms of reliability and availability. It could be demonstrated that under many circumstances train unavailability could be essentially 1.0 (always unavailable), and the MSPI would still be $<10^{-6}$.

Significance Determination Process (SDP)

Because NRC Inspection Manual Chapter (IMC) 0609 addresses the SDP, this paper will not discuss the process at any length. [15] Phase 3 of the SDP uses appropriate PRA techniques to evaluate the significance of the performance deficiency. For a degraded condition assessment, the change in CDF, along with the exposure time involved, provides the figure-of-merit for the evaluation (i.e., change in core damage probability (ΔCDP)). Conditional core damage probability (CCDP) is the figure-of-merit for events. The bands for significance are similar to those discussed for the MSPI (e.g., $>10^{-6}$ but $\leq 10^{-5}$ for "White"), although the SDP uses probability and the MSPI uses frequency. (The SDP also assesses changes in LERF and corresponding probabilities, with thresholds 1 order of magnitude lower than ΔCDP and CCDP.)

To illustrate the situation that may arise for new reactors with lower baseline CDF estimates, a hypothetical but realistic example is provided.

First, assume that a very important valve has failed as a result of a performance deficiency in a current reactor. The RAW is given as 5 for all events including internal and external initiators. Assume that the baseline CDF is $2x10^{-5}$/yr (using nominal unavailabilities for all equipment) and that the condition existed for 72 days (i.e., 0.2 yr). To a good approximation, ΔCDP is given by the following:

[CDF(with failed valve, using nominal unavailabilities for all other equipment) - CDF(baseline, using nominal unavailabilities for all equipment)] * 0.2 yr

or

$[(5 * 2x10^{-5}) - (2x10^{-5})] * 0.2 = 1.6x10^{-5}$, which would be "Yellow"

In a new reactor, a different valve with comparable risk worth has been in a failed state for the same assumed time. Assuming that the baseline CDF is $5x10^{-7}$/yr including internal and external events, a calculation similar to the one above would yield the following:

$[(5 * 5x10^{-7}) - (5x10^{-7})] * 0.2 = 4x10^{-7}$, which would be "Green"

In this hypothetical example, the valve would have to be unavailable for about half a year before the “White” significance level is approached.

In essence, for new reactors with baseline CDF estimates 1 to 3 orders of magnitude lower than the norm for current reactors, it would be very rare to reach the “White” significance level, with “Yellow” and “Red” almost impossible to reach unless multiple system failures occurred. (In the above example, a 100-fold increase in baseline CDF of a new reactor for about 2 months would be necessary to cross into the “Yellow” band, and a 1,000-fold increase in CDF for the same 2 months to reach “Red” significance.)

Management Directive 8.3, “NRC Incident Investigation Program”

Issues similar to those discussed above for the SDP could result from the implementation of the quantitative risk considerations of Management Directive (MD) 8.3 for new reactors. [16] MD 8.3 details deterministic criteria for identifying what constitutes a significant operational event. One such criterion is “operations that exceeded, or were not included in, the design bases of the facility.”

Such operational events could occur at new reactors, just as they have occurred at current plants. However, MD 8.3 also provides quantitative guidelines to assist in determining the appropriate reactor operational event response options. The figure in MD 8.3 provides the following risk-informed guidelines:

- CCDP between 10^{-6} and 10^{-4} special inspection
- CCDP between 10^{-5} and mid-10^{-3} augmented inspection
- CCDP above mid-10^{-4} incident investigation

The CCDP varies from plant to plant and from initiator to initiator. Assume, by way of example, a nominal loss of offsite power (LOOP) frequency of $4x10^{-2}$/yr and a LOOP-induced CDF of $4x10^{-6}$/yr for a current reactor. [17] These values would correspond to a CCDP of $1.0x10^{-4}$ with no other equipment failures. A LOOP alone would call for a special inspection or augmented inspection based on the quantitative guidelines. An additional failure of an emergency diesel generator during the event would place this event in the incident investigation regime.

For one new reactor design, the estimated LOOP-induced CDF is approximately 10^{-9}/yr. This is a factor of about 4000 lower than the average for current reactors. The values also correspond to a CCDP of $2.5x10^{-8}$, which is outside the lowest threshold for response follow-up. Even an additional diesel generator failure would place the CCDP below or just at the 10^{-6} threshold for a special inspection. While it is understood that some new reactor designs are insensitive to LOOP events by design, the insight is not unique to LOOP events. With a steam generator tube rupture (SGTR) initiator frequency of $4x10^{-3}$/yr and SGTR-induced CDF estimate for one new reactor of $7x10^{-9}$/yr, the CCDP of $1.8x10^{-6}$ would barely place the event response in the special inspection category. Suffice to say, with CDF estimates of 1 to 3 orders of magnitude below those of current reactors, many “significant” operational events at new reactors would not reach the quantitative risk guideline thresholds.

III.d.2 Changes to the Licensing Basis

Regulatory Guide 1.174

RG 1.174 provides an approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis. As such, this document is intended to improve consistency in regulatory decisions in areas where the results of risk analyses are used to help justify regulatory action. The guide is the foundation for many other risk-informed programs (e.g., inservice testing, inservice inspection of piping) at the agency.

RG 1.174 describes five key principles of the risk-informed, integrated decisionmaking process. In Principle 4, it links small changes in CDF and risk to the QHOs of the Commission's Safety Goal Policy Statement. In addition to the stated principles, it presents quantitative acceptance guidelines for CDF and LERF, as depicted in Figures 3 and 4 of the guide. Regions representing different acceptance guidelines are established in a stepwise manner depending on the baseline risk metric (CDF, LERF, or both) and the change in the metrics (ΔCDF, ΔLERF, or both) resulting from the licensing basis change. The largest changes in CDF and LERF that fall into Region III are 10^{-6} and 10^{-7}, respectively.

As emphasized, one of the overriding principles of the approach is that increases in estimated CDF and LERF resulting from proposed licensing basis changes will be limited to small increments, and the cumulative effect of such changes should be tracked and considered in the decision process. For a baseline CDF of 10^{-5}/yr (assuming all contributions from internal events, external events, and low power/shutdown), a 10^{-6}/yr ΔCDF would constitute a 10-percent increase.

In the development of RG 1.174, there was a basis for using *absolute* change in CDF and LERF rather than *relative* change. The 10^{-5}/yr threshold was set on the basis of the regulatory analysis guidelines related to when a backfit can be considered, irrespective of the absolute value of CDF. The 10^{-6}/yr ΔCDF limit was set based on a value that the staff, at that time, believed was close to the limit of resolution of PRA models in the sense that one could always find a "valid" means to change the PRA model that would negate a 10^{-6}/yr CDF increase.

The CDF estimates for new reactors are 1 to 3 orders of magnitude below those for current reactors. As shown in Figure 1 above, these estimates range from the low 10^{-6}/yr for one design with active safety systems to the mid-10^{-8}/yr (estimate for all events) for a new design with passive safety systems. Certainly, for most of the new LWR designs, the baseline CDF and LERF estimates are "off-the-chart" at the low end of the x-axes of Figures 3 and 4 of RG 1.174. A 10^{-6} ΔCDF, or even 2 orders of magnitude lower 10^{-8} ΔCDF, would no longer constitute a "small change" *on a relative basis* in comparison to a baseline CDF of mid-10^{-8} as implied by the principle. Should the principle of "small increase" be interpreted to mean on a *relative* or an *absolute* basis for ΔCDF and ΔLERF? Should RG 1.174 present an additional ΔLRF acceptance guideline strictly for new reactors? Section IV of this paper discusses some of the ramifications of this issue, including the possible range of options.

Regulatory Guide 1.177

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," describes methods acceptable to the NRC staff for assessing the nature and impact of proposed changes to TS by considering engineering issues and applying risk insights. [18] The five key principles from RG 1.174 are applied to TS changes as well. Quantitative acceptance guidelines from RG 1.174 are directly applicable to changes in allowed outage times (AOT) and surveillance test intervals. The numerical guidelines used to decide an acceptable

TS change are taken into account along with other traditional considerations, operating experience, lessons learned from previous changes, and practical considerations associated with test and maintenance practices. The final acceptability of the proposed change should be based on all these considerations and not solely on the comparison of PRA-informed results to numerical acceptance guidelines.

An additional risk metric for changes to the AOT is the ICCDP. As defined in RG 1.177, it is given mathematically as the following:

> ICCDP = [(conditional CDF with the subject equipment out of service) - (baseline CDF with nominal expected equipment unavailabilities)] * (duration of single AOT under consideration)

An ICCDP of less than $5x10^{-7}$ is considered small for a single TS AOT change. For ICLERP, a value less than $5x10^{-8}$ is considered small. Additional conditions and considerations apply. For example, notwithstanding the risk-derived value for AOT, the staff would not allow large values of AOT beyond certain deterministic limits based on practice. However, because of the lower baseline CDF and LERF estimates for new reactors, all other things being equal, the requested TS change to AOT could be large and yet not even approach the ICCDP and ICLERP guidelines of $5x10^{-7}$ and $5x10^{-8}$, respectively.

While not discussed in any detail, the same new reactor risk metric issues would apply to the derivation of completion times in risk management TS initiative 4b. Given the risk estimates for some new LWRs, many of the SSCs could meet the criteria to justify up to the 30-day backstop for single equipment outages.

<u>10 CFR 50.69 and RG 1.201 Risk-Informed Categorization and Treatment of Structures, Systems and Components</u>

The NRC has promulgated regulations to permit power reactor licensees and license applicants to implement an alternative regulatory framework with respect to “special treatment,” where special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that SSCs will perform their design-basis function. Figure 1 of RG 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” provides a conceptual understanding of the new risk-informed SSC categorization scheme. [19] The four risk-informed safety class (RISC) categories are based on deterministic and risk-informed considerations. The reader is referred to 10 CFR 50.69 and the corresponding RG 1.201 (for trial use), as well as Nuclear Energy Institute guidance document NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” for detailed discussion. [20]

NEI 00-04 presents many deterministic and qualitative considerations, including the five key principles of RG 1.174. The guidance describes, for example, the system engineering assessment and integrated decisionmaking panel review of the process. It also addresses defense-in-depth considerations and the performance of risk sensitivity studies. Of particular interest to this paper are the quantitative risk considerations and the impact of the lower risk estimates posed by new reactors on the implementation of 10 CFR 50.69. The two specific aspects considered here are (1) risk-informed SSC categorization using importance measures and (2) risk sensitivity studies.

The following are the importance measure criteria used to identify candidate safety significance:

- sum of FV for all basic events modeling the SSC of interest, including common-cause events >0.005
- maximum of component basic event RAW values >2
- maximum of applicable common cause basic events RAW values >20

If any of these criteria are exceeded, then the SSC is considered a safety-significant candidate.

FV and RAW are measures of *relative* risk importance unlike CDF, LERF, ΔCDF, CCDP, ICCDP, Birnbaum importance, and so on, which are more *absolute* measures of risk or change in risk. The fundamental definition of FV implies a fractional contribution of the component basic event to CDF. Likewise, the RAW is a normalized measure (or ratio) that considers the CDF with the component basic event that is presumed failed divided by the nominal CDF. The extent to which a new reactor might have a CDF estimate several orders of magnitude lower than the current generation of reactors is in effect offset in large measure by this normalization.

New reactors, especially those with passive safety systems, have significantly different systems and designs from the current reactors. They also have different risk profiles than the current generation of plants in terms of dominant sequences, cut sets, and component basic event importance. However, the same could be said regarding the differences among the reactors in the current generation of reactors, yet one set of quantitative criteria in the 10 CFR 50.69 characterization has been established for the wide range of more than 100 plant designs.

A cursory review of importance measures in the PRA for one new reactor design was performed. Indeed, few SSC basic events have FV and risk reduction worth (RRW) importance measures that are very high (i.e., RRW >1.1). Only one basic event has an RRW above 1.1. For RAW, the new reactor design does have a moderate number of component basic events with high value. However, the distribution of RAW values does not appear to be measurably different than that for current reactors. For comparative purposes, Table 1 provides the RAW and FV values that correspond approximately to the above safety-significant criteria for the new reactor design and the standardized plant analysis risk (SPAR) model for a currently operating PWR.

Table 1 Comparison of Number of Basic Events with Significant Importance Measures

Importance Measure*	New Reactor	SPAR Model for Currently Operating Reactor
Modeled SSC basic events with FV >0.005	About 60	About 40
Modeled SSC basic events with RAW >2	About 60	About 70
Modeled SSC common cause basic events with RAW >20	About 25	About 30

* Because of difficulties identifying the various failure modes for the same component and interpreting the basic event definitions, no attempt has been made to sum FVs for the same component or to use the maximum RAW rule.

As discussed in the footnote to Table 1, this is only an approximate comparison because FVs were not combined and the RAWs may have been double-counted. Notwithstanding this limitation, the comparison shows generally good agreement of the number of basic events approximating the criteria for risk-significance. While the new reactor design has only one SSC with an RRW above 1.1, the SPAR model for the currently operating reactor has about a half-dozen. These are mainly auxiliary feedwater (AFW) pumps and direct-current batteries,

illustrating the important support system dependencies of AFW for some electrically initiated transients. While it is difficult to generalize the comparison of one new reactor design and one current generation plant to the entire set of possible new reactors, the results tend to support the contention that risk metrics for new reactors that are *relative* in nature should not pose as much concern as those relying on the *absolute* magnitude of CDF and LERF.

The second aspect of 10 CFR 50.69 implementation for new reactors concerns the risk sensitivity studies. The regulation in 10 CFR 50.69(c)(1)(iv) states the following:

> Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

Applying the acceptance guidelines per RG 1.174 to the definition of "small" will result in the same issues discussed above that stem from the lower baseline CDF and LERF estimates of new reactors. In effect, virtually all risk changes measured on an *absolute* scale will be demonstrated as "small," even though on a *relative* scale, the CDF and/or LERF change might be significant.

Other Programs, Processes, and Regulations

The application of new reactor risk metrics to the programs and processes described above is not exhaustive. Many other programs and regulations, such as risk-informed inservice testing (RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing"), risk-informed inservice inspection of piping (RG 1.178, "An Approach for Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping"), and some elements of the maintenance rule (10 CFR 50.65), rely to some extent on the risk metrics discussed above. [22,23,2] It would appear, based on the limited discussion and comparisons, that most programs and processes relying on an *absolute* measure of risk such as Birnbaum importance, CDF, and LERF (and their associated increments) could raise issues when applied to new reactors. Those programs and regulations relying to a greater extent on *relative* measures of risk, including FV and RAW importance, would appear to pose less of an issue for new reactors.

IV. OPTIONS

This section presents six possible options for addressing the issues related to the application of risk metrics for new reactors and discusses the advantages and disadvantages of each option. The options range from the status quo (i.e., applying current ROP thresholds and acceptance guidelines to new reactors), to the use of altogether new guidelines and thresholds for new reactors, and finally to postponing any significant change to the process and evaluating new reactors on a case-by-case basis for an indeterminate period. These options are intended to provide a basis for further discussion of the issues. During these discussions, other options may be identified.

Option 1: Status Quo

Under this option, the current ROP thresholds and acceptance guidelines in RG 1.174 (and associated regulatory guides) would also be applied to new reactors. In effect, there would be no distinction between existing reactors and new reactors with regard to treatment and staff reviews. Risk increases resulting from changes to the licensing basis would be considered on

an *absolute* scale consistent with the considerations of Figures 3 and 4 of RG 1.174 and the five key principles. ROP thresholds for MSPI, SDP, and MD 8.3 would retain the same numerical values.

Advantages

- This option provides a consistent set of acceptance guidelines and ROP thresholds for both existing and new reactors.
- It acknowledges and gives credit to new reactors for lower risk estimates.

Disadvantages

- This option may not be consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."
- It would use a less restrictive change process than the Commission established for the review of new reactors.
- It could raise concerns regarding the insensitivity of the MSPI and of SDP findings under the ROP.
- Compared to the baseline CDF and LERF estimates for new reactor designs, this option could allow large *relative* increases in CDF and LERF.

Option 2: Convert to *Relative* Risk Changes

Under this option, RG 1.174 and associated regulatory guides would be converted so that risk changes would be assessed on a *relative* basis for current and new reactors. The five key principles in RG 1.174 would continue to apply, but the acceptance guidelines in Figures 3 and 4 would be on a relative scale (i.e., percent change). Similarly, ROP thresholds would be converted to relative values (i.e., percent change) rather than the current 10^{-6}, 10^{-5}, and 10^{-4}. Programs, processes, and regulations that largely use relative risk metrics such as FV and RAW importance measures would not need to change.

Advantages

- This option recognizes that "small increase" is a relative measure, and precludes large percent change in CDF and/or LERF for new reactors.
- It precludes the situation noted above for the ROP whereby new reactor performance indicators would be insensitive for safety systems.
- It precludes a situation in which inspection findings for performance deficiencies in new reactor systems would be relatively insensitive to the deficiency.

Disadvantages

- This option would be inconsistent with the underlying technical basis for the current *absolute* thresholds in RG 1.174 as discussed above in Section III.d.2.

- There could be substantial disagreement between industry and staff regarding what constitutes the "baseline" for CDF and LERF changes.

- Major changes to current regulatory guides and other processes would be required.

- The MSPI formulation would require substantial revision.

- This option would impact currently operating reactors.

- This option would result in inconsistency between existing and new reactors. A licensing basis change to the surveillance test interval of an instrument loop with $2x10^{-7}$ ΔCDF for a current reactor may be deemed "acceptable," but the same change for a new reactor would not.

- The transition from the existing *absolute* acceptance guidelines and ROP thresholds to *relative* (percent changes) could be difficult.

- Depending on the chosen limits for acceptance guidelines and ROP thresholds, past conditions that were deemed acceptable might not be found acceptable under the new formulation and vice versa. Past decisions would need to be "grandfathered." Possibly, a past inspection finding that was deemed of "White" significance under the SDP could be "Green" in the new formulation. Such situations could affect the agency's response to plant performance; IMC 0308, Attachment 4, documents the plan for this response in an "action matrix." [23]

Option 3: Reduce Acceptance Guidelines and ROP Thresholds for New Reactors

Under this option, the acceptance guidelines in RG 1.174 (and associated regulatory guides) and ROP thresholds, including MSPI, SDP, and MD 8.3 event response options, would be lowered by 1 or more orders of magnitude solely for new reactors. For example, if the current threshold for "White" performance in the MSPI is greater than 10^{-6} (but less than or equal to 10^{-5}) and 1 order of magnitude were chosen as the reduction factor, then for new reactors, the threshold for "White" would be set at greater than 10^{-7}. Likewise, the "y scales" in Figures 3 and 4 of RG 1.174 would be established as 1 order of magnitude lower for this option (for new reactors). Programs, processes, and regulations that largely use relative risk metrics such as FV and RAW importance measures would not need to change.

Advantages

- This option acknowledges that new reactor CDF/LERF estimates are significantly lower than existing reactors and adjusts acceptance guidelines and ROP thresholds accordingly.

- It would preclude the expected insensitivity if the current ROP thresholds were applied to new reactors (e.g., MSPI would be insensitive to system performance in terms of reliability and availability, and the SDP "White" threshold would rarely be reached).

- It is consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."

Disadvantages

- This option would be inconsistent with the underlying technical basis for the current *absolute* thresholds in RG 1.174 as discussed above in Section III.d.2.
- It penalizes new reactors for having lower risk estimates.
- It results in different treatment at new and current reactors of a proposed licensing basis change resulting in a ΔLERF of $4x10^{-8}$/yr, even though these changes would have essentially the same public health impact.
- It may be inconsistent with the Commission's Safety Goal Policy Statement on acceptable level of risk.

Option 4: Use a Combination of Existing and New Acceptance Guidelines and Reactor Oversight Process Thresholds

This option considers a combination of the status quo (Option 1) and Options 2 and 3. The option has several variations. For example, one sub-option would be to apply the status quo (existing acceptance guidelines and ROP thresholds) to both existing and new reactors. Additionally, new reactors would have to meet a second set of acceptance guidelines and ROP thresholds based on *relative* risk changes. For new reactors, the acceptance guidelines and ROP thresholds would be limited by the more stringent (conservative) acceptance guidelines or ROP thresholds, whether *absolute* or *relative.* (Based on the examples provided above, this would inevitably be the *relative* set of conditions.) A second sub-option would be not to distinguish between existing and new reactors and to apply both an *absolute* and *relative* set of acceptance guidelines and ROP thresholds to all reactors. Yet a third sub-option would be to modify Figures 3 and 4 of RG 1.174. Rather than the regions being stepwise, the region thresholds at the lower end of the CDF and LERF baseline scales would be sloped to, in effect, apply *relative* risk change to new reactors that have risk estimates several orders of magnitude lower than those of current reactors. ROP thresholds would be similarly adjusted.

Advantages

- This option addresses some of the concerns discussed above regarding large *relative* changes to risk with new reactors.
- It would preclude the expected insensitivity if the current ROP thresholds were applied to new reactors (e.g., MSPI would be insensitive to system performance in terms of reliability and availability, and the SDP "White" threshold would rarely be reached).
- It is consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."

Disadvantages

- This option would be inconsistent with the underlying technical basis for the current *absolute* thresholds in RG 1.174, as discussed above in Section III.d.2.
- It penalizes new reactors for having lower risk estimates.

- It results in different treatment at new and current reactors of a proposed licensing basis change resulting in a ΔLERF of $4x10^{-8}$/yr, even though these changes would have essentially the same public health impact.

- Major changes to current regulatory guides and other processes would be required.

- The MSPI formulation would require substantial revision.

- Implementation would be complex, since the "baseline" CDF would have to be established to determine how to apply the varying thresholds in RG 1.174 and the ROP.

Option 5: Use Existing Acceptance Guidelines and ROP Thresholds for Current and New Reactors (*Status Quo*), but Establish an LRF-Based Acceptance Guideline for New Reactors

Under this option, the existing acceptance guidelines for ΔCDF and ΔLERF in RG 1.174 (and associated regulatory guides) would be the same for current and new reactors. A new acceptance guideline for ΔLRF would be added strictly for new reactors, with scales on the abscissa and ordinate 1 order of magnitude lower than those for LERF and ΔLERF shown in Figure 4 of RG 1.174. (A variation to this option would be to replace altogether ΔLERF as shown in RG 1.174 with the ΔLRF acceptance guideline for new reactors only.) However, the ROP thresholds would remain the same for currently operating and new reactors.

Advantages

- This option is consistent with the goals that the Commission established for the review of new reactors.

- It provides a consistent set of acceptance guidelines and ROP thresholds for both existing and new reactors with regard to ΔCDF, consistent with Commission goals.

- It is consistent with the underlying technical basis for the current *absolute* thresholds for ΔCDF and ΔLERF in RG 1.174, as discussed above in Section III.d.2 and as modified to reflect Commission policy regarding ΔLRF for new reactors.

- This option acknowledges and gives credit in the ROP to the lower risk estimates of new reactors.

- Because it establishes an LRF-based acceptance guideline, this option is consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."

- This option would not require the significant effort involved in revising the ROP thresholds (MSPI, SDP, MD 8.3) for new reactors.

- It allows a number of anticipated risk-informed initiatives, such as inservice inspection of piping and TS initiative 4b and 5b, to move forward before or after issuance of a combined license.

Disadvantages

- This option does not address the issues discussed above regarding the insensitivity of the MSPI and of SDP findings under the ROP.

- It could allow large *relative* increases in CDF and LERF compared to the baseline CDF and LERF estimates for new reactor designs, although in most cases the stricter acceptance guideline for ΔLRF would probably be limiting.

- It would require significant changes to regulatory guides.

Option 6: Assess New Reactors on a Case-by-Case Basis

Under this option, risk-informed regulation and processes for existing reactors would not change. The decision as to how to treat new reactors would be postponed until operational experience has been obtained and assessed. In effect, new reactor performance measurement and oversight would default to the process used before the current ROP. Performance monitoring would rely less on numerical thresholds and more on subjective factors. Likewise, the staff would evaluate changes to the licensing basis by using a combination of deterministic and probabilistic considerations. The five key principles of RG 1.174 would still apply, but existing numerical guidelines would not be applied universally.

Advantages

- No changes would be needed to the ROP, regulatory guides, and related documents for current reactors. (Some guidance would be required regarding the processes for new reactors.)

- The staff could await the accumulation of sufficient new reactor operating experience before making a decision on the treatment of new reactors. The number of new reactors in operation will grow slowly, and sufficient operating experience will not be accumulated until 2018 or later.

- Treating a small number of new reactors on a case-by-case basis for the first few years for ROP and various risk-informed applications should not place an undue burden on the staff.

Disadvantages

- Current reactors and new reactors would be treated inconsistently.

- Some stakeholders may object to the subjective treatment of performance monitoring for new reactors in the ROP. The lack of objective, numerical criteria is regressive.

- New reactor licensees would not know what the acceptance guidelines are for risk-informed changes to the licensing basis.

- Some combined license applicants for new reactors may be proposing risk-informed initiatives or special treatment under 10 CFR 50.69 in the near future. Other possible applications include risk-informed AOTs under RG 1.177 and TS initiatives 4b and 5b. The absence of detailed guidance could be problematic.

- This option simply defers any decision on the treatment of new reactors. At some point, more definitive guidance will be necessary.

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NUCLEAR ENERGY INSTITUTE

Designated Original

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March 27, 2009

Mr. Girija S. Shukla
Cognizant Staff
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Transmission of Industry White Paper, "Risk Metrics for Operating New Reactors," for ACRS Review.

Project Number: 689

Dear Mr. Shukla:

Since the issuance of the Commission's PRA Policy Statement in 1995, the ACRS has provided the NRC and the industry with valuable input and insights regarding risk-informed regulation. The industry understands that the ACRS is scheduled to discuss risk metrics for new light-water reactor risk-informed applications during their April meeting and is providing the enclosed white paper, "Risk Metrics for Operating New Reactors," for ACRS review. This paper discusses the evolution of risk metrics, the role of risk metrics in design and licensing, and quantitative thresholds for operating new reactors, and further offers an analysis of the anticipated difficulties associated with the establishment of alternative risk metrics or thresholds for operating new reactors. The industry is prepared to discuss this white paper during the April ACRS meeting.

If you have any other questions on this topic, please feel free to contact me at 202.739.8083; reb@nei.org or Victoria Anderson at 202.739.8101; vka@nei.org.

Sincerely,

Biff Bradley

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Risk Metrics for Operating New Reactors

Introduction

As new reactors progress through the 10 CFR 52 licensing process (i.e., Part 52), the focus is shifting from evaluation of the basic standardized design toward operational risk considerations. This transition has become somewhat confounded due to the differences in the metrics used in the Part 52 licensing process versus those used in the regulatory programs that apply to operating reactors.

The purpose of this paper is to consider the value and applicability of various risk metrics for the two distinct phases of plant licensing: design and operation. Furthermore, the paper discusses the quantitative thresholds that may apply to those risk metrics.

Evolution of Risk Metrics

In order to frame the situation, a brief summary is provided on the evolution of risk metrics that has occurred over the past 20 years.

The Part 52 licensing process and associated industry guidance for consideration of severe accident risks was promulgated in the early 1990s. At that time, the Safety Goal Policy Statement [Ref. 1] had only recently been issued and the industry had relatively little experience with using risk information in regulatory decision-making.

The Part 52 design certification process [Ref. 2] includes a requirement that the licensing of advanced reactor design would include a probabilistic risk assessment (PRA) for internal hazards (i.e., internal events, internal flooding, and internal fires) for at-power and selected shutdown operating states. External hazards that are site-specific (e.g., seismic events, high winds, external floods, etc.) are primarily addressed using design features and capabilities intended to minimize risks. For seismic hazards, a PRA-based seismic margin analysis (SMA) is performed to demonstrate that there is a high confidence of a low probability of core damage scenarios occurring for seismic events even more severe than the Safe Shutdown Earthquake (SSE).

Through a series of SECYs and associated Staff Requirements Memorandums (SRMs) issued during 1990 [Ref. 3, 4], the Commission elected to include three risk metrics and associated quantitative goals in the design certification process:

- **Core Damage Frequency (CDF) < 1 x 10^{-4}/year** – a measure of overall safety performance in the prevention of severe accidents
- **Large Release Frequency (LRF) < 1 x 10^{-6}/year** – a measure of prevention of significant offsite consequences
- **Conditional Containment Failure Probability (CCFP) < 0.1** – a measure of the capability of the design to mitigate a severe accident

By the early-1990s, the Individual Plant Examinations (IPEs) for the operating fleet of reactors were drawing to a close and the Nuclear Regulatory Commission (NRC) began to grapple with how to best implement the Safety Goal Policy Statement and severe accident considerations in regulatory decision-making. In 1993, the Commission unanimously voted to abandon efforts to define "large release" after the staff found it impossible to provide a simple definition that was

not "several orders of magnitude more conservative than the QHOs" (Quantitative Health Objectives) [Ref. 5].

In 1995, the Nuclear Regulatory Commission (NRC) issued the PRA Policy Statement [Ref. 6] that encouraged the use of PRA in all regulatory matters. In 1998, the NRC issued Regulatory Guide (RG) 1.174 [Ref. 7], providing a risk-informed integrated decision-making framework. The development of RG 1.174 relied upon a body of work that demonstrated that CDF and large, early release frequency (LERF) were the appropriate surrogates for the Quantitative Health Objectives (QHOs) of the Safety Goal Policy Statement for operating reactors (and not CCFP and LRF). Of equal significance, the framework defined by RG 1.174 established the concept of "risk-informed" decision-making in which risk results are one input to a decision, along with other factors such as maintaining defense-in-depth, sufficient safety margins, and performance monitoring.

Today, the risk-informed process, metrics, and guidelines defined in RG 1.174 have been incorporated into numerous licensee and regulatory programs for operating reactors including the Maintenance Rule, Reactor Oversight Process, Technical Specifications, etc. Since the issuance of RG 1.174 and its companion application-specific RGs various derivative metrics have been included in licensee and regulatory programs, e.g., ΔCDF, ΔLERF, incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP). While RG 1.174 allows small risk increases, in practice the majority of regulatory applications by RG 1.174 have been developed and implemented to minimize any risk increase and this has resulted in improved overall safety. The improved safety focus gained through these applications has contributed to an overall reduction in CDFs industry-wide [Ref. 8].

In summary, while the operating reactors have adopted the RG 1.174 approach that uses CDF and LERF, the regulatory documents related to PRAs for Combined Operating Licenses (COLs) [Ref. 9] have continued to rely upon the original risk metrics defined in the original Part 52 design certification process, i.e., CDF, LRF, and CCFP.

However, as the advanced reactor designs are progressing through the licensing process and appear to be heading toward construction and operation, NRC has identified potential issues involving the risk metrics being applied to an operating reactor and those used in the design and licensing process under Part 52.

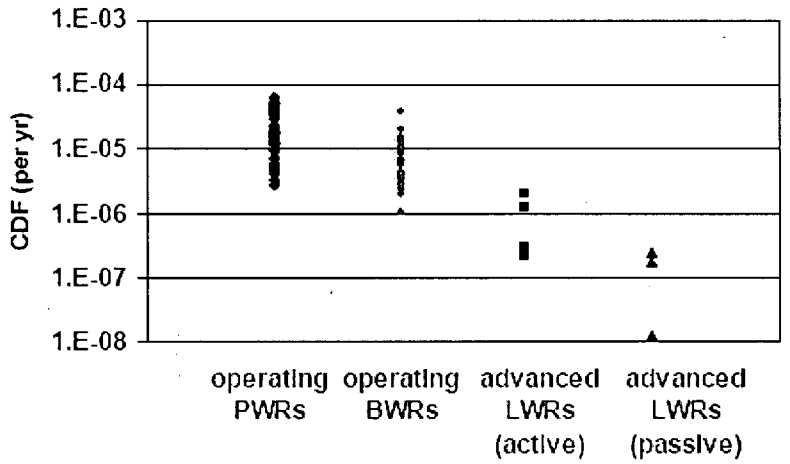
Role of PRA and Risk Metrics in Design and Licensing

PRA fundamentally addresses the risks associated with design and operation. The Part 52 design certification and COL process both use extensive PRA insights for all new reactors. PRA has been shown to be an effective tool to inform the design and minimize the potential for beyond design basis vulnerabilities. All of the certified designs have used PRA in this manner.

As the design certification PRAs have demonstrated, the computed risks for the new reactors are lower than comparable operating designs (i.e. PWR versus PWR) when only internal events are considered (Figure 1). Generally, the design certification PRAs are within about an order of magnitude of the lower end of the internal events CDF range for current reactors.

Figure 1
Internal Events Core Damage Frequency Perspectives [Ref. 10]

(internal events at-power for U.S. plants only)



The design certification PRA scope has generally been limited to internal events, internal flooding, and internal fires for power operations and selected shutdown operational states. Seismic risks have not been quantified, but have been addressed using PRA-based SMA.

Another important part of the severe accident capabilities of plants licensed under Part 52 involves the deterministic requirements for each design to include severe accident mitigation features such as:

- Hydrogen generation and control systems,
- Reactor primary system depressurization systems,
- Ex-vessel core debris cooling capability, and
- Robust containment design to prevent releases within 24 hours.

From a design perspective, the requirements and risk goals of the Part 52 process have put appropriate focus on the following:

- A low and balanced computed CDF,
- A low computed CCFP (<0.1) for the corresponding computed CDF, and
- A low LRF for the corresponding computed CDF.

The use of SMA ensures a pragmatically minimal seismic risk, but excludes the seismic hazard risk from the computed CDF.

Part 52 requirements for severe accident mitigation and the associated risk metrics have been effective in establishing robust designs with low computed risk levels, a computed CDF not dominated by any particular contributors (i.e., balanced risk profile without vulnerabilities), and strong severe accident containment performance. As noted, the computed CDF does not include seismic risk.

Large Release Frequency (LRF)

Each of the design certification submittals and COLs (through reference to the DCD) has been expected to demonstrate that the design supports a LRF goal of < 1 x 10^{-6}/year for the quantified risk contributors. However, given difficulties in defining "large release" that led to the NRC's decision to abandon the development of a specific definition, each reactor vendor has been left to provide their own definitions for LRF. In all cases, the criteria used would be considered much less than "large" (see Attachment 1 for further discussion of large release definitions). In some cases, the vendors simply stated that they did not define large release, but elected to use a criterion that is "much less than large" [Ref. 11]. In all cases, robust new plant designs, including severe accident mitigation features, have met the LRF goal, despite the use of conservative definitions.

Conditional Containment Failure Probability (CCFP)

The Commission also approved the use of CCFP as a risk metric in the design certification and COL process. Once again, a standard definition has not been established, so each vendor adopted their own approach to computing CCFP. In general, CCFP has been defined as the fraction of the computed CDF that results in LRF (i.e., CCFP = LRF/CDF). All designs have been able to show that CCFP < ~0.1 for the scope of quantified hazards, albeit to different CCFP definitions. Given the conservative definitions of LRF used and the balanced CDF profiles, this has served to confirm the robustness of the plant design and the effectiveness of the severe accident mitigation features required under Part 52.

General Conclusions on the Severe Accident Capability of New Reactors

Consistent with Commission policy, new reactor designs have an enhanced level of severe accident prevention and mitigation capability resulting from a number of common attributes [Ref. 12]:

- High level of redundancy
- Physical separation of safety systems
- Very low contribution from interfacing systems loss of coolant accident (ISLOCA)
- Low contribution to CDF from anticipated transient without scram (ATWS)
- Rapid reactor primary system depressurization capability
- Core melt mitigation capability
- Containment combustible gas control capability

As a result, the certified designs have achieved a higher standard of severe accident safety performance than prior designs and they provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety functions in preventing core damage, containment failure, and large release.

The Role of Risk Metrics in New Reactors Operational Phase

As plants licensed under Part 52 transition into operation, they will fall under the requirements and implementation practices of Part 50 for operating reactors. The two primary risk metrics used in evaluating operating reactors are CDF and LERF. During the 1990s, these metrics were demonstrated to be acceptable surrogates for the QHOs, and were elevated to subsidiary safety objectives as part of the implementation of RG 1.1.74. CDF is generally regarded as a surrogate for the individual latent cancer fatality risk QHO and LERF has been shown to be an adequate surrogate for the individual early fatality risk QHO [Ref. 13].

The origin of LERF as a risk metric was the PSA Applications Guide [Ref. 14] developed by EPRI in the mid-1990s. LERF was proposed by industry as a surrogate for the early fatality QHO, based on the insights from NUREG-1150 [Ref. 15] and other Level 3 PRAs. Such studies have shown that LERF is actually more than an order of magnitude greater than the frequency of a "large release," when a large release is defined as one that can result in one or more early fatalities (as was done in NUREG-1150). It is worth noting that the staff's original proposal to the Commission in SECY 89-102 was similar: "a large release is a release that has the potential for causing an offsite early fatality" [Ref 16]. The reason that a LERF of 10^{-5}/year is roughly comparable to a LRF of 10^{-6}/year is that not all large early releases occur at a time when conditions (e.g., wind direction and speed, stability class, evacuation progress, etc.) would cause an early fatality (See Attachment 1). So, in a sense, LERF is a surrogate for both LRF and the early fatality QHO. Furthermore, the factor of ten (10) difference in the quantitative guidelines for CDF and LERF aligns well with the concept of a CCFP < 0.1. Attachment 1 provides a more comprehensive discussion of the relation between LRF and LERF.

Quantitative Risk Thresholds for Operating New Reactors

The Commission has been consistent in maintaining that new reactors should not be measured against a lower quantitative CDF threshold than operating reactors. When the staff first proposed that new reactors be evaluated using a mean CDF target less than 1.0×10^{-5} event per reactor-year [Ref. 17], the Commission rejected the recommendation [Ref. 18] and reiterated their position from the previous year [Ref. 3], supporting the use of 10^{-4} per year of reactor operation as a core damage frequency goal.

Concerns that tighter risk metrics are needed to prevent the erosion of new plant safety and severe accident performance echo concerns that led to NRC staff proposals in the mid-1990s for a suite of new "applicable regulations" to codify requirements for severe accident features in each design certification rule. The basis for the Commission's rejection of the staff proposal [Ref. 19] is applicable to the current consideration of tighter risk metrics for new plants.

In 2008, in the release of the revised Advanced Reactor Policy Statement, the Commission again made it clear that:

> *... the policy statement does not state that advanced reactor designs must be safer than the current generation of reactors, but rather that they must provide the same degree of protection of the environment and public health and safety and the common defense and security that is required for current-generation light-water reactors. The goal of the policy statement update is to encourage advanced reactor designers to consider safety and security in the early stages of design in order to identify potential design features and/or mitigative measures*

> *that provide a more robust and effective security posture with less reliance on operational programs.* [Ref. 20]

In fact, as discussed above, the Part 52 design certification and COL process ensures that the designs do incorporate risk insights and design features that provide a robust safety capability.

Furthermore, in developing Regulatory Guide 1.174, NRC developed CDF and LERF acceptance guidelines that were derived from the QHOs of the safety goal. Regulatory Guide 1.174 states the following:

> *In theory, one could construct a more generous regulatory framework for consideration of those risk-informed changes that may have the effect of increasing risk to the public. Such a framework would include, of course, assurance of continued adequate protection (that level of protection of the public health and safety that must be reasonably assured regardless of economic cost). But it could also include provision for possible elimination of all measures not needed for adequate protection, which either do not effect a substantial reduction in overall risk or result in continuing costs that are not justified by the safety benefits. Instead, in this regulatory guide, the NRC has chosen a more restrictive policy that would permit only small increases in risk, and then only when it is reasonably assured, among other things, that sufficient defense in depth and sufficient margins are maintained. This policy is adopted because of uncertainties and to account for the fact that safety issues continue to emerge regarding design, construction, and operational matters notwithstanding the maturity of the nuclear power industry. These factors suggest that nuclear power reactors should operate routinely only at a prudent margin above adequate protection. The safety goal subsidiary objectives are used as an example of such a prudent margin.*

Given the Commission's consistent position on expectations for new reactors and the consistency between a LERF guideline of 10^{-5}/reactor year and a LRF of 10^{-6}/reactor year described above, it is clear that the quantitative acceptance guidelines of RG 1.174 could be considered not only appropriate, but consistent with previous Commission policy for new reactors.

Problems with Establishing Alternate Risk Metrics and/or Thresholds

The NRC has identified six risk metric options for consideration [Ref. 10]. Based upon the above discussion, the industry believes the current metrics (Option 1 of the NRC paper) are technically justified and appropriate for all plants, based on reasonable assurance of public health and safety, including operation at a prudent margin above adequate protection. The introduction of new risk metrics or thresholds could create a number of issues:

1. <u>Inconsistency with Commission Policy</u>: Most importantly, as discussed above, the existing risk metrics are consistent with and derived from the NRC Safety Goal Policy Statement which has been long accepted and reiterated in Commission statements such as the cited NRC SRM on SECY 90-16, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," June 26, 1990. In the intervening years there has been no Commission direction to provide alternate requirements or goals. Certain of the proposed NRC options would establish de facto new safety goals or subsidiary objectives without proper revision of the

underlying Policy Statement. The Commission has consistently stated that new reactors should not be measured by more restrictive quantitative risk metrics, including as recently as last year. Many of the issues raised in the staff paper have been previously considered and resolved. Thus, introduction of new or different risk metrics would be contrary to long-standing Commission positions.

2. New Risk Metrics Would Penalize Safer Plants: As described above, new plants have been designed using risk insights. Different risk metrics would create a double standard that would penalize plants for being safer (e.g., limit their operational flexibility and subject them to enforcement at low thresholds), and could lead to allocation of NRC inspection resources at thresholds where there is essentially no impact on public health and safety.

3. New Risk Metrics Would Create Public Perception Problems: Many of the new reactors in the COL process are planned to be on a site with an existing reactor. Having different metrics creates challenges for the NRC inspectors as well as the public in understanding how similar findings at two co-located sites could be considered to have different levels of significance.

4. New Risk Metrics Values Could be Associated with High Uncertainties: For some of the NRC proposed options, the risk metrics could be so low that they would challenge the resolution capability of PRA technology. Decisions would be made on extremely small risk values that are almost unnoticeable within uncertainty bands. Existing metrics already suffer from this problem to a degree, and this problem would be exacerbated with some of the proposed options, especially for the reactor oversight process (ROP) and the significance determination process (SDP) determinations. The imposition of de minimus risk thresholds could also have the unintended consequence of truncating risk-informed activities in new plants that would undermine the observed benefit of a risk-informed focus.

5. Current Risk Metrics are also Supported with Additional Requirements: Risk -informed regulation through RG 1.174 has not led to increased CDF/LERF values for operating plants – in fact, the opposite has been demonstrated. Risk-informed regulation has led to an increased focus on risk significant items and safety performance has improved. This reality contradicts a fundamental premise of the NRC paper: an apparent presumption that the entire risk margin available through RG 1.174 could be consumed. All RG 1.174 applications require advance NRC review and approval, and there is a significant body of practical experience from risk applications at operating plants. RG 1.174 is not solely risk-based and it requires four other regulatory considerations to be addressed, including safety margins and defense-in-depth. In reality, these other considerations are routinely employed by NRC staff to limit or reject the proposed changes even when risk thresholds are met. Additionally, even for current plants, the NRC has rarely granted changes outside of "very small" region of RG 1.174 Figures 3 and 4. In practice this has limited changes by an order of magnitude compared to the "allowable" acceptance guideline. This further decreases the margin between new reactor risk metrics and the quantitative thresholds.

6. New plants s are Subject to a Comprehensive Change Control Process with explicit consideration of severe accidents: New reactors licensed under Part 52 already have a comprehensive change control process with respect to severe accident capabilities. Changes to fundamental plant design or plant Technical Specifications are subject to

prior NRC review and approval. Changes in design or implementation details (so-called Tier 2 information) are subject to an enhanced 50.59-like process that explicitly addresses the potential to increase the likelihood or consequences of severe accidents, and if triggered, would require prior NRC review and approval.

7. Current Risk Metrics Contain Deterministic Backstops: Risk-informed applications generally contain deterministic backstops that protect against very small risk impacts leading to non-conservative operational decisions. For instance, Technical Specification Initiative 4B limits all risk-informed completion times (RICTs) to 30 days maximum, regardless of how small the magnitude of the computed increase in core damage probability is.

8. Risk Profile for New Reactors is Not Yet Complete: The staff's proposed options appear to be based on an incomplete risk picture (i.e., the DCD/COL PRAs). It can be reasonably expected that the DCD risk results will increase as new plants are required through 10 CFR 50.71(h)) to develop PRAs for NRC endorsed consensus standards, which currently would include internal events, fire, and external events including seismic (RG 1.200, Revision 2 was issued in March 2009). The NRC options paper is silent on this aspect and appears to presume the DCD values will carry forward as the required scope of the PRA increases. Given the above, the calculated risk metrics for new reactors are likely to increase and therefore be closer to current plants than being portrayed today. That is, the one to four orders of magnitude difference cited by the staff will decrease as other site-specific risk contributors, such as seismic, are more fully quantified. There already exists a variation in baseline risk values (CDF/LERF) for operating plants, and this was explicitly considered in the development of RG 1.174. Given this, the current approach should remain valid for new plants.

Conclusions

The Part 52 licensing process and Commission policy, quite appropriately, puts an increased emphasis on the severe accident capability of new reactors and the resulting computed risk metrics of CDF, CCFP, and LRF. Consistent with Commission policy, these risk metrics, in combination with other deterministic requirements, have resulted in new reactors having enhanced severe accident safety performance and enhanced margins of safety, as compared to prior designs.

As these new reactor designs transition to the operational phase, it is appropriate to transition the evaluation of severe accident safety to the risk-informed process that was developed by the NRC after the Part 52 process was implemented. Reliance on CDF and LERF provides risk metrics that are philosophically consistent with the Part 52 risk metrics and appropriately aligns the safety metrics of new reactors with the rest of the operating fleet.

The existing RG 1.174 risk metrics, quantitative acceptance guidelines, and integrated decision-making process fits well with the Commission objectives for new reactors. That is, the risk metrics of RG 1.174 and the derivative metrics used in other applications are consistent with the risk metrics used in Part 52 (CDF and LERF as a surrogate for LRF) and the defense-in-depth and safety margin principles will ensure that the robust severe accident design features provide enhanced safety and severe accident protection throughout the operational phase.

References

1) USNRC, "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986.

2) USNRC, 10 CFR 52.47(a)(27).

3) USNRC, SRM on SECY 89-102, "Implementation of Safety Goals," June 15, 1990.

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5) USNRC, SECY 93-138, "Recommendation on Large Release Definition," May 19, 1993 and associated SRM on June 10, 1993.

6) USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

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10) Memo from R.W. Borchart to NRC Commissioners, "Alternative Risk Metrics For New Light-Water Reactor Risk-Informed Applications," USNRC, February 12, 2009.

11) "ABWR Design Control Document/Tier 2," Rev. 0.

12) Donald Dube, "Comparison of New Light-Water Reactor Risk Profiles," ANS PSA 2008 Topical Meeting, Knoxville, Tennessee, September, 2008.

13) USNRC, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, Appendix D - Derivation of Risk Surrogates for LWRs," December, 2007.

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15) USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

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18) USNRC, SRM on SECY 90-16, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," June 26, 1990.

19) USNRC, SRM on SECY 96-077, "Certification of Two Evolutionary Designs," December 6, 1996.

20) USNRC, "Policy Statement on the Regulation of Advanced Reactors," 73 FR 60612, October 14, 2008.

Attachment 1

The Relationship of Large Early Release Frequency and Large Release Frequency

-- A Historical Perspective --

Recently, there has been renewed discussion regarding the definition of large release frequency (LRF), spawned by the need for operational decision-making metrics for new reactors licensed under 10 CFR 52. The purpose of this paper is to provide a historical perspective on the development of the risk metrics large early release frequency (LERF) and Large Release Frequency (LRF), and their relationship to each other.

History of "Large Release"

The notion of a risk metric related to "large release" dates back to the issuance in 1986 of the Nuclear Regulatory Commission's (NRC's) Safety Goal Policy Statement (SGPS) [Ref. A1-1]. The Safety Goal Policy Statement (SGPS) provides qualitative and quantitative definitions of "how safe is safe enough" for nuclear power plants. In the SGPS, the Commission directed the staff to develop guidance related to "large releases":

> *"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."*

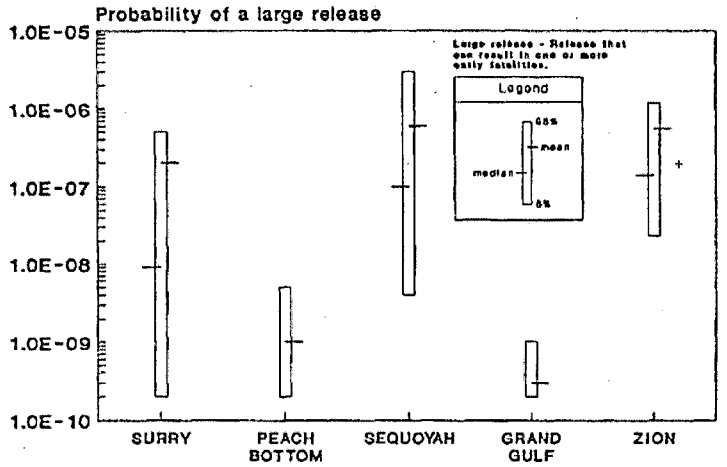
In response to the SGPS direction, the staff proposed a qualitative definition for "large release" in SECY 89-102 [Ref. A1-2]:

> *"A large release is a release that has a potential for causing an offsite early fatality."*

In the resulting staff requirements memorandum (SRM) on SECY 89-102 [Ref. A1-3], the Commission directed the staff to develop a more specific definition and supporting rationale, consistent with criteria provided by the Advisory Committee on Reactor Safeguards (ACRS).

In 1990, the NRC issued NUREG-1150 [Ref. A1-4], a risk reference document containing Level 3 PRAs for five U.S. nuclear power plants. In the figure below, the staff computed the large release frequencies (LRF) for each plant consistent with the staff's original definition, i.e., a release that can result in one or more early fatalities:

Figure 1
NUREG-1150 Large Release Frequencies for Internal Events
(Fig. 12-5 of Ref. A1-6)



In SECY 90-405 [Ref. A1-5], the staff provided two more specific definitions of large release for further investigation that met the criteria provided by the ACRS and Commission. In the SRM on SECY 90-405 [Ref. A1-6], the Commission expressed a preference for a quantitative definition of LRF in terms of an absolute value or equivalent curies released.

In 1993, the staff concluded research on a quantitative definition and sent the Commission SECY 93-138 [Ref. A1-7], which recommended termination of work on a quantitative definition of large release. The staff's conclusion was that "development of a large release definition and magnitude, beyond the simple qualitative statement released to the 10^{-6} per year large release frequency (such as is currently contained in the Commission's Safety Goal Policy Statement), is not practical or required for design or regulatory purposes." In the SRM on SECY 93-138 [Ref. A1-8], the Commission concurred with the Staff's recommendation.

History of "Large Early Release"

The origin of large early release frequency (LERF) ties back to the EPRI PSA Applications Guide [Ref. A1-9], published in 1995. The PSA Applications Guide was the industry's initial effort to define risk metrics and quantitative thresholds for decision making. The industry effort was aware of the NRC's decision to terminate the effort to define large release. It was recognized that CDF alone was not sufficient in demonstrating that the Quantitative Health Objectives (QHOs) of the SGPS were met. Work done on plant-specific PRAs, as well as NUREG-1150, had shown that CDF was a reasonable surrogate for the latent health effect QHO, but another metric was required for the early fatality QHO. The concept of LERF tied to a mechanistic definition of a Level 2 PRA endstate was created by the EPRI team. Rather than explicitly tying the metric to a quantitative definition of the release, the characteristics of the release were used. Specifically, the release had to be both "large" (i.e., a rapid unscrubbed release of fission products) *and* "early" (i.e., before emergency protective actions had been completed).

A few years later when the NRC developed the integrated risk-informed decision making process described in Regulatory Guide 1.174, the NRC also adopted LERF as a risk metric. Work done by the ACRS staff [Ref. A1-10] confirmed the alignment of LERF to the early fatality QHO. More recently, NUREG-1860 [Ref. A1-11] reiterated the validity of LERF as a surrogate for the QHO.

The Relationship of LERF and LRF

In the original development of LERF as a risk metric, EPRI evaluated the relationship between LERF and LRF. Although the quantitative thresholds for LERF are an order of magnitude greater than the original LRF frequency expressed in the Safety Goal Policy Statement, the two metrics are actually consistent. This is best explained via an example.

As shown above in Figure 1, when a Level 3 PRA is available, it is possible to quantify the original qualitative definition of LRF provided by the staff in SECY 89-102. In NUREG-1150, the staff computed LRF as the frequency of a release that can result in one or more early fatalities. The results are compiled below for the internal events based on Figure 1 above:

Table 1
NUREG-1150 Computed Large Release Frequency for Internal Events

Plant	Large Release Frequency (/year)
Peach Bottom	1.00E-09
Surry	2.00E-07
Grand Gulf	3.00E-10
Sequoyah	6.00E-07
Zion	6.00E-07

NUREG-1150 was published before the concept of LERF had been adopted for risk-informed decision-making. However, based on the frequency of the accident progression bins used to define containment status in the Level 2 portion of the NUREG-1150 analysis, it is possible to estimate LERF for each of the plants. Table 2 provides a summary of LERF based on the accident progression bins for each plant based on the applicable figure in NUREG-1150.

Table 2
NUREG-1150 Estimated LERF Results

Table 2.a
Peach Bottom Internal Events (Fig. 4.5 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB >200 psi, Early WWF	No	n/a
VB <200 psi, Early WWF	No	n/a
VB >200 psi, Early DWF	Yes	1.48E-06
VB <200 psi, Early DWF	Yes	7.94E-07
VB, Late WWF	No	n/a
VB, Late DWF	No	n/a
VB, CV	No	n/a
No CF	No	n/a
No VB	No	n/a
No Core Damage	No	n/a
	Total	2.27E-06

Table 2.b
Surry Internal Events (Fig. 3.5 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Alpha, Early CF	Yes	1.23E-07
VB > 200 psi, Early CF	Yes	1.64E-07
VB < 200 psi, Early CF	Yes	0.00E+00
VB, BMT, or Late CL	No	n/a
Bypass	Yes	5.00E-06
VB, No CF	No	n/a
No VB	No	n/a
	Total	5.29E-06

Table 2.c
Grand Gulf Internal Events (Fig. 6.4 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Early CF, Early SPB, No CS	Yes	6.46E-07
VB, Early CF, Early SPB, CS	No	n/a
VB, Early CF, Late SPB	No	n/a
VB, Early CF, No SPB	No	n/a
VB, Late CF	No	n/a
VB, Venting	No	n/a
VB, No CF	No	n/a
No VB	No	n/a
	Total	6.46E-07

Table 2.d
Sequoyah Internal Events (Fig. 5.4 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
VB, Early CF (During CD)	Yes	2.79E-07
VB, Alpha, Early CF (at VB)	Yes	1.12E-07
VB > 200 psi, Early CF (at VB)	Yes	1.95E-06
VB < 200 psi, Early CF (at VB)	Yes	1.28E-06
VB, Late CF	No	n/a
VB, BMT, Very Late CF	No	n/a
Bypass	Yes	3.12E-06
VB, No CF	No	n/a
No VB, Early CF (During CD)	No	n/a
No VB	No	n/a
	Total	6.75E-06

Table 2.e
Zion Internal Events (Fig. 7.3 of Ref. A1-4)

Accident Progression Bin	LERF?	Mean LERF
Early CF	Yes	4.73E-06
Late CF	No	n/a
Bypass	Yes	2.37E-06
No CF	No	n/a

These estimated LERF results can then be compared relative to the LRF results (Table 3)

Table 3
Comparison of NUREG-1150 LERF and LRF Results

Plant	Estimated LERF (/yr)	Large Release Frequency (/yr)	Conditional Probability of LRF Given LERF
Peach Bottom	2.27E-06	1.0E-09	0.04%
Surry	5.29E-06	2.0E-07	3.8%
Grand Gulf	6.46E-07	3.0E-10	0.05%
Sequoyah	6.75E-06	6.0E-07	8.9%
Zion	7.10E-06	6.0E-07	8.5%

In all cases, the LRF is more than an order of magnitude below the estimated LERF. The reason for this is that not every large early release results in an early fatality due to weather and/or partially implemented emergency protective actions. Thus, the LERF quantitative acceptance guideline of RG 1.174 of 10^{-5}/year is consistent with, or potentially conservative with respect to, a LRF acceptance guideline of 10^{-6}/year.

References

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A1-2) USNRC, SECY 89-102, "Implementation of Safety Goals," March 30, 1989.
A1-3) USNRC, SRM on SECY 89-102, "Implementation of Safety Goals," June 15, 1990.
A1-4) USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.
A1-5) USNRC, SECY 90-405, "Formulation of a Large Release Definition and Supporting Rationale," December 14, 1990.
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A1-9) D.E. True, et al, "PSA Applications Guide," EPRI TR-105396, August, 1995.
A1-10) Rick Sherry Memorandum to ACRS Members, "Large Early Release Frequency (LERF) Definition and Acceptance Criteria," February 18, 1997.
A1-11) USNRC, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, Appendix D - Derivation of Risk Surrogates for LWRs," December, 2007.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 27, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RISK-INFORMED REGULATORY GUIDANCE FOR NEW REACTORS

Dear Chairman Jaczko:

During the 574th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 2010, we completed our review of a draft Commission Paper on Modifying the Risk-Informed Regulatory Guidance for New Reactors. During our 573rd meeting, June 9-11, 2010, we met with representatives of the NRC staff to discuss this matter. Our Subcommittee on Reliability and Probabilistic Risk Assessment also met with the staff and representatives of the Nuclear Energy Institute and the Union of Concerned Scientists on June 2, 2009, to discuss preliminary options and proposals for the use of quantitative metrics in risk-informed applications for new reactors. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. We agree with the staff's position in the draft Commission Paper that the proposed framework should prevent significant decreases in the level of safety of the new reactor designs. The staff should continue to interact with internal and external stakeholders to develop guidance for an integrated risk-informed decision-making framework that consistently applies the principles of Regulatory Guide 1.174 for currently operating plants and new reactors. Pilot applications should be used to assess and refine details of the proposed framework.

2. The bases for risk significance determinations in the Reactor Oversight Process (ROP) should be consistent with the guidance for changes to the licensing basis.

3. The guidance should anticipate and account for plant-specific risk profiles that may be influenced significantly by external events such as severe earthquakes.

4. The staff should expedite the development of interim guidance for the use of numerical risk significance measures for selection of candidate structures, systems, and components (SSCs) in design certification and Combined License (COL) reliability assurance programs.

BACKGROUND

Quantitative metrics are used extensively to guide regulatory decisions regarding the risk significance of design, operations, testing, maintenance, and inspection issues that affect the

ROP and changes to the licensing basis for currently operating nuclear power plants. The specific metrics and their applications have evolved during the 15 years since the 1995 Commission Policy Statement on the use of Probabilistic Risk Assessment (PRA) methods in nuclear regulatory activities. Regulatory Guide 1.174 describes the key principles for implementing risk-informed decision making, and it contains guidance for the use of quantitative metrics to assess the measurable risk impacts from a proposed change.

Current guidance relies primarily on Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) as surrogate measures for the risk from nuclear power plant accidents. A modified metric, Large Release Frequency (LRF), is applied for new reactor designs to clarify the risk implications that are subsumed by the LERF metric. The CDF and LRF metrics are approximate pragmatic surrogates for a more complete assessment of risk that would be provided by a fully integrated Level 3 PRA. Our referenced reports contain extensive discussions about the complex relationships among the Commission Safety Goals, PRA scope and quality, efficacy of CDF and LERF as risk surrogates, and applied methods for measuring risk significance within the constraints of these metrics.

Regulatory pragmatism and the limitations of applied PRA technology have influenced the selection of these risk metrics and the guidance for their use. Current PRAs focus primarily on CDF (Level 1 PRA) with the most emphasis placed on internal initiating events that occur during full power operation. The scope of these PRAs typically includes limited assessments of the risk from internal plant fires, seismic events, other external events, or initiating events that may occur during low power and shutdown modes of operation. Most PRAs include only a limited evaluation of containment performance (Level 2 PRA) that is deemed sufficient to estimate the conditional probability of large (early) releases.

New plant designs have benefited substantially from the use of passive safety features and insights gained from PRAs for operating plants. Regulatory Guide 1.206 states that PRAs which are performed to support the certification of new reactor designs and COL applications "should be a Level 1 and Level 2 PRA that includes internal and external events and addresses all plant operating modes." The scope and level of detail in the design certification and COL PRAs are typically more limited than many PRAs that are performed for operating plants. The resulting estimates for CDF and LRF are typically 10 to 1,000 times lower than CDF and LERF estimates for the current operating fleet. These substantial differences require reexamination of the types of metrics that are used to assess risk, the criteria that are used to measure risk significance on an absolute level, and the criteria that are used to measure the significance of departures from a particular plant's baseline risk profile.

DISCUSSION

The following three issues are most pertinent to the draft Commission Paper.

(1) What principles should apply in near-term risk-informed guidance for new reactors?

Pragmatic considerations must continue to guide the regulatory decision-making process. Those considerations favor retention of the basic principles that are currently applied for risk-informed licensing decisions as well as inspection and enforcement actions under the ROP. The foundations for those principles are elaborated in Regulatory Guide 1.174.

The draft Commission Paper summarizes three conceptual approaches for the assessment of acceptable risk margins for new reactors. One approach considered by the staff (denoted as

Option 1) involves no changes to the metrics in the current regulatory guidance. That approach uses the absolute risk measures for currently operating reactors as a basis for determining the significance of departures from a new reactor's baseline risk profile. In effect, under this proposal, risk increases would not be deemed “significant” until a new reactor's cumulative risk approaches the range of CDF and LRF estimates that apply for currently operating plants. If a new reactor had a very low overall baseline risk at the time of its COL issuance, successive plant changes under the regulatory framework of Option 1 could permit substantial erosion of its original risk margin. This is contrary to Commission expectations that a new reactor should maintain an enhanced level of safety throughout its operating life. We concur with the staff's conclusion that the principles of Option 1 are not appropriate for new reactor regulatory guidance.

Another approach considered by the staff (denoted as Option 3) would modify regulatory guidance to apply a “new risk metric” for new reactors. The staff explained that this approach would effectively use current numerical measures of relative risk significance as a basis for determining the acceptability of departures from a new reactor's baseline risk profile. The regulatory framework of Option 3 would permit only small relative increases in risk, regardless of a plant's original baseline risk profile. This approach could inappropriately constrain practical risk-informed applications for new reactors whose original baseline risk and updated risk profile remain substantially below the estimates for currently operating plants. These restrictions are also inconsistent with the principles and regulatory decision-making framework that are applied in Regulatory Guide 1.174.

The stated goals of the staff's recommended approach (denoted as Option 2) are to develop guidance for risk-informed licensing-basis changes that would prevent a significant decrease in the level of safety of a new reactor over its life and to develop risk-informed guidance for the ROP to provide meaningful regulatory oversight for all operating plants.

We concur with the general principle that the regulatory process should seek to prevent a significant decrease in the level of safety over the life of a plant. However, plants that have lower overall risk should be allowed more regulatory flexibility, compared with plants that have higher overall risk. For example, a conceivable regulatory framework under Option 2 could apply a variable determination of the acceptable level of cumulative risk increase as a function of a plant's baseline risk profile. New reactors with very low overall risk would be allowed a larger relative increase in that risk, provided that their total risk remains substantially below the estimates for currently operating plants. These concepts are consistent with Commission expectations for new reactor safety margins, our past recommendations, and the basic principles in Regulatory Guide 1.174. The staff has not yet developed substantive recommendations for implementation of Option 2. We encourage the staff to continue their interactions with internal and external stakeholders to develop more detailed guidance for that conceptual framework.

The staff has noted that different interpretations and applications of the risk metrics may be needed for risk-informed changes to the licensing basis and for evaluations of plant performance during the ROP. We reiterate our past recommendations on the need for a consistent basis for the determination of risk significance in all regulatory activities.

The staff noted that development of the guidance under Option 2 may benefit from pilot evaluations of changes to plant hardware, testing and maintenance programs, Technical Specifications, new reactor reliability assurance programs, and inspection findings within the context of a few actual reactor designs and their supporting PRAs. These evaluations would

provide valuable practical insights, experience, and mutual understanding that benefit the development of consistent regulatory guidance. We recommend that these types of pilot applications should be incorporated into the staff's deliberations and stakeholder discussions.

(2) How should applied regulatory guidance account for the plant-specific risk profile and its contributors?

The licensing process in 10 CFR Part 52 should ensure a high degree of design consistency among individual plants that reference the same design certification. However, there may be substantial differences in plant-specific risk profiles. For example, the PRAs that have been prepared for current design certification and COL applications typically do not quantify the risk from site-specific external events such as large earthquakes, high winds, tornadoes, and external floods. At many sites, it is likely that a complete assessment of these contributors may produce risk results that are substantially larger than the quantified CDF and LRF values that consider only internal initiating events. The current PRAs also include only limited assessments of the risk during plant low power and shutdown modes of operation. Experience has shown that the contributors to shutdown risk often depend strongly on plant-specific outage management plans and practices. Thus, it is likely that full-scope risk assessments of two nominally "identical" plant designs at two different sites with different operating and maintenance practices may display significantly different risk profiles. The risk-informed regulatory framework should not presume that all new plants referencing the same certified design will have similar risk profiles, and the guidance should account for expected site-specific plant-to-plant variability.

The potential risk contributions from external hazards introduce another layer of conceptual complexity in the regulatory framework for new reactors. For example, a full-scope risk assessment may conclude that seismic events account for 80 percent of the plant-specific baseline risk, and all other contributors account collectively for the remaining 20 percent. The risk-informed regulatory framework should explicitly anticipate and account for this condition. Thus, it is important to decide whether the applied regulatory risk acceptance guidelines pertain to the total plant risk (i.e., including the seismic contribution) or to only a fraction of that risk (e.g., without the seismic contribution). Different guidance for acceptable risk increases may be appropriate for proposed changes to the licensing basis, which is concerned primarily with absolute risk, and regulatory actions under the ROP, which primarily address operational departures that involve equipment malfunctions, maintenance activities, and human performance.

(3) How should risk significance be determined for current design certifications and COL applications?

Quantitative risk metrics are being used to support licensing decisions for current design certifications and COL applications. In particular, numerical measures of risk significance are used to identify candidate non safety-related SSCs for inclusion in the Design Reliability Assurance Program (DRAP). The DRAP is developed during design certification, is adopted by the COL applicant, and is revised, as necessary, to include additional risk-significant site-specific SSCs.

Most applicants currently apply the numerical risk significance guidance that is endorsed in Regulatory Guide 1.200 and Regulatory Guide 1.201 (i.e., Fussell-Vesely Importance $\geq$ 0.005 or Risk Achievement Worth $\geq$ 2). Some applicants apply less restrictive selection criteria (e.g., Fussell-Vesely Importance $\geq$ 0.010 or Risk Achievement Worth $\geq$ 5), citing their low overall risk

values as justification. The former applicants effectively apply the principles of Option 3 discussed above, pending more explicit guidance on the applicable metrics. The latter applicants effectively apply a particular interpretation of Option 2.

While we endorse the conceptual principles of Option 2, we are concerned that the staff's approval of design-specific departures from Option 3 during current certification and COL licensing decisions could establish a precedent for “acceptable” measures of absolute and relative risk that may not be fully consistent with the eventual resolution of those quantitative metrics. Clear interim guidance should be developed expeditiously for the use of numerical risk significance measures to select candidate SSCs for design certification and COL reliability assurance programs.

We look forward to our continuing interactions with the staff to resolve these significant technical issues.

Additional comments by ACRS member John Stetkar are presented below.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

Additional Comments from ACRS Member John W. Stetkar

I concur fully with the content and recommendations in this letter. Consideration of these additional comments should not interrupt the staff's proposed approach in the draft Commission Paper. It is important to maintain consistency in the regulatory guidance for operating reactors and near-term new reactor applications. However, the investigation and development of risk metrics that will be applied for new reactors provide a valuable opportunity to judiciously consider a key issue that is not mentioned in the letter.

Are CDF and LRF the most appropriate metrics for risk-informed regulatory decision making?

Despite the general acceptance of these metrics, they are only surrogates that do not accurately assess the actual risk to a member of the public who resides near a particular nuclear power plant. An assessment of that risk can be made only by the performance of a full-scope Level 3 PRA that evaluates internal events and external hazards during all operating modes. Risk is measured by the integrated evaluation of the potential hazards, their likelihoods, the effectiveness of plant features to prevent core damage, the effectiveness of containment features and accident management plans to mitigate offsite releases, the local site meteorology and demography, the effectiveness of offsite emergency planning, and a consistent assessment of the uncertainties that are associated with each of these elements. Partial evaluations of CDF and LRF do not provide this integrated understanding of the risk and its contributors. Regulatory processes that focus too strongly on only these surrogate measures may not provide balanced decisions that are fully informed by an understanding of the actual risks.

Consideration of this issue is very timely. Insights from preliminary analyses of new reactor designs indicate that the most important contributors to their risk may involve hazards such as fires, floods, and severe seismic events that affect all elements of an integrated risk assessment. Thus, perceived distinctions among the contributors to core damage, containment failure, and offsite releases for new plants may be much less clear than those identified historically for operating plants with risks that are influenced strongly by internal events such as equipment failures.

It is conceivable that insights from plant-specific Level 3 PRAs may identify metrics other than CDF and LRF that are more appropriate representations of integrated risk. Those metrics would provide an improved rationale for regulatory decisions that are concerned primarily with potential effects on public health and safety.

With these observations in mind, I believe that the continuing evolution of risk-informed regulatory guidance and its applied metrics should encourage the extension of PRA technology to include full-scope Level 3 assessments that provide more complete estimates of actual public health risk and its contributors.

References:

1. Memorandum to Edwin M. Hackett, “Transmittal of Draft Commission Paper on Risk-Informed Regulatory Guidance for New Reactors,” 05/12/2010 (ML101090355)

2. Memorandum to the Commissioners, “Alternative Risk Metrics for New Light-Water Reactor Risk-Informed Applications,” and enclosed “White Paper on Options for Risk Metrics for New Reactors,” 02/12/2009, (ML090150636, ML090160004)

3. Letter to Girija S. Shukla, Transmission of Industry White Paper, “Risk Metrics for Operating New Reactors,” 03/27/2009 (ML090900674)

4. U.S. Nuclear Regulatory Commission, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” *Federal Register,* Vol. 60, (60 FR 42622), 08/16/1995

5. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 1, 11/2002 (ML023240437)

6. U.S. Nuclear Regulatory Commission, “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” *Federal Register*, Vol. 51, (51 FR 30028), 08/21/1986

7. Letter to Greta Joy Dicus, “Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities,” 10/12/1999 (ML992880016)

8. Letter to Richard A. Meserve, “Importance Measures Derived from Probabilistic Risk Assessments,” 02/11/2000 (ML003684519)

9. Letter to Dr. William D. Travers, “NUREG-CR-6813, Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making,” 04/29/2003 (ML031220311)

10. Letter to Nils J. Diaz, “Draft Final 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” 06/15/2004 (ML041690039)

11. Letter to Nils J. Diaz, “Report on Two Policy Issues Related to New Plant Licensing,” 09/21/2005 (ML052640580)

12. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” June 2007 (ML070720184)

13. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 1, January 2007 (ML070240001)

14. Regulatory Guide 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” Revision 1, May 2006 (ML061090627)

August 25, 2010

Dr. Said Abdel-Khalik, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RISK-INFORMED REGULATORY GUIDANCE FOR NEW REACTORS

Dear Dr. Abdel-Khalik:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter dated July 27, 2010, regarding the recommendations of the Advisory Committee on Reactor Safeguards (ACRS) on the draft Commission paper, "Modifying the Risk-Informed Regulatory Guidance for New Reactors." The ACRS received the draft Commission paper, sent by memorandum dated May 12, 2010, (ML101090355). The NRC staff discussed the draft paper with the ACRS on June 10, 2010. The NRC staff's response to the recommendations in your letter is discussed below.

ACRS Recommendation 1

We agree with the staff's position in the draft Commission Paper that the proposed framework should prevent significant decreases in the level of safety of the new reactor designs. The staff should continue to interact with internal and external stakeholders to develop guidance for an integrated risk-informed decision-making framework that consistently applies the principles of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," for currently operating plants and new reactors. Pilot applications should be used to assess and refine details of the proposed framework.

NRC Response

The NRC staff agrees with the ACRS recommendation. If the Commission directs the staff to implement Option 2, the staff will continue to interact with internal and external stakeholders to develop guidance for an integrated risk-informed decision-making framework that consistently applies the principles of Regulatory Guide 1.174 for currently operating plants and new reactors. The staff will also consider conducting one or more public workshops to evaluate the Part 52 change processes for a wide spectrum of potential plant changes, as well as some aspects of the current Reactor Oversight Process (ROP), and identify potential "gaps" in the existing risk-informed framework.

ACRS Recommendation 2

The bases for risk significance determinations in the ROP should be consistent with the guidance for changes to the licensing basis.

NRC Response

The NRC staff agrees with the ACRS recommendation. The staff recognizes that the bases for risk significance determinations in the ROP would follow from, and would need to be consistent with, the guidance for changes to the licensing basis for new reactors. As such, if the Commission directs the staff to implement Option 2, the staff would evaluate potential changes to the ROP to ensure that the ROP remains consistent with any changes to the risk-informed licensing guidance, just as the current ROP is consistent with the risk-informed licensing guidance currently contained in Regulatory Guide 1.174. The staff would continue to interact with internal and external stakeholders to identify appropriate changes to the risk-informed elements of the ROP to reflect the enhanced level of severe-accident safety performance of new reactors while providing for meaningful regulatory oversight.

ACRS Recommendation 3

The guidance should anticipate and account for plant-specific risk profiles that may be influenced significantly by external events such as severe earthquakes.

NRC Response

The NRC staff agrees with the ACRS recommendation. The staff recognizes that, for some new reactors, site-specific hazards such as severe earthquakes could be significant contributors to the overall risk profile. In modifying any risk-informed guidance, the staff will be mindful of the ACRS recommendation regarding site-specific plant-to-plant variability and consideration of external hazards.

ACRS Recommendation 4

The staff should expedite the development of interim guidance for the use of numerical risk significance measures for selection of candidate structures, systems, and components (SSCs) in design certification and Combined License (COL) reliability assurance programs.

NRC Response

The NRC staff understands the ACRS concerns and believes that it would be prudent to await Commission direction before proceeding with the development of guidance for the use of numerical risk significance measures for the selection of candidate SSCs. The NRC staff will take the ACRS concerns into consideration when the staff modifies the risk-informed regulatory guidance as necessary.

We appreciate the comments and recommendations provided by ACRS and look forward to continuing to work with the Committee as we begin to draft detailed guidance as directed by the Commission.

Sincerely,

/RA by Martin J. Virgilio for/

R. W. Borchardt
Executive Director
for Operations

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY