February 12, 2009

MEMORANDUM TO: Charles E. Ader, Director
Division of Safety Systems and Risk Assessment
Office of New Reactors

FROM: Donald A. Dube, Senior Technical Advisor **/RA/**
Division of Safety Systems and Risk Assessment
Office of New Reactors

SUBJECT: WHITE PAPER AS BASIS FOR DISCUSSION AT THE FEBRUARY 18, 2009 PUBLIC MEETING ON IMPLEMENTATION OF RISK METRICS FOR NEW REACTORS

Enclosed is the white paper that provides a basis for discussion of implementation issues regarding risk metrics for new reactor risk-informed applications at the February 18, 2009 public meeting in T2-B3 of NRC Headquarters from 8:30 am to noon.

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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]

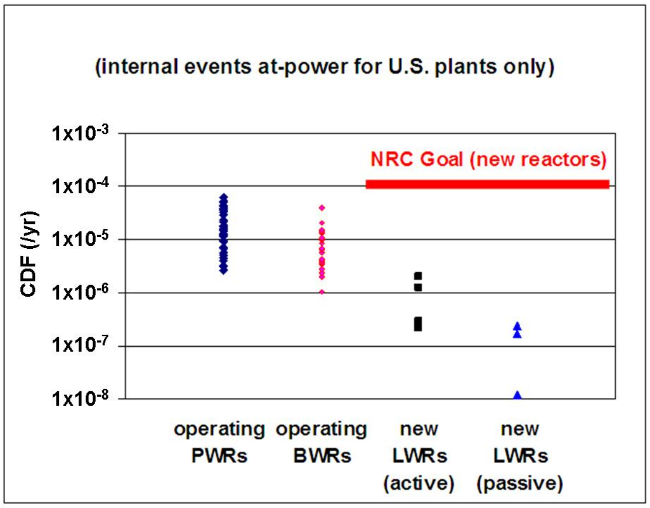

Figure 1 CDF estimates by plant type

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.

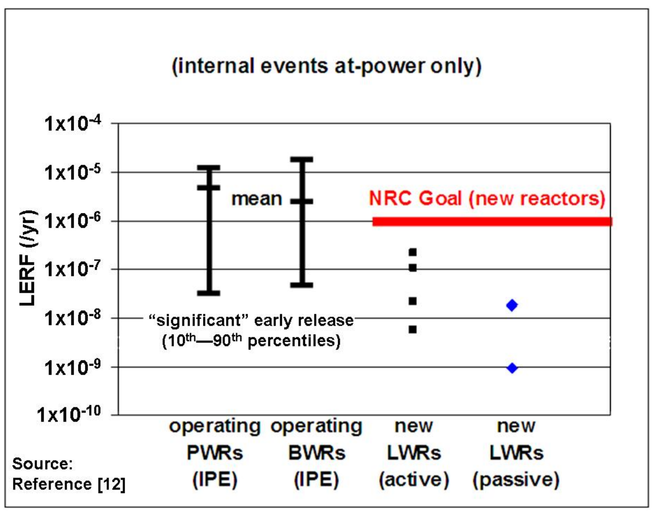

Figure 2 LRF estimates by plant type

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 $10^{-6} < \text{MSPI} \leq 10^{-5}$
- "Yellow": for $10^{-5} < \text{MSPI} \leq 10^{-4}$
- "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.)

<u>Management Directive 8.3, “NRC Incident Investigation Program”</u>

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.

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

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. [21,22,2] 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 2×10^{-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 4×10^{-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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[22] RG 1.178, “An Approach for Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping,” September 2003.

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