POLICY ISSUE
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FOR: The Commissioners

FROM: Margaret M. Doane
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SUBJECT: RECOMMENDATIONS FOR MODIFYING THE REACTOR OVERSIGHT PROCESS FOR NEW LARGE LIGHT WATER REACTORS WITH PASSIVE SAFETY SYSTEMS SUCH AS THE AP1000 (GENERATION III+ REACTOR DESIGNS)

PURPOSE:

The purpose of this paper is to request U.S. Nuclear Regulatory Commission (NRC) approval of the recommended changes to the Reactor Oversight Process (ROP) for new reactor designs. This paper responds to the Staff Requirements Memorandum (SRM) for SECY-13-0137, "Recommendations for Risk-Informing the Reactor Oversight Process for New Reactors," dated June 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14181B398). Specifically, this paper addresses the Commission's direction to (1) submit a paper with the staff's proposed approach for any revisions to the Significance Determination Process (SDP) for new reactors, (2) develop any necessary updates to the performance indicators (PIs) and submit them to the Commission for approval, and (3) further explore how the current Safety System Functional Failure (SSFF) PI would be applied to the passive safety-related components in Generation III+ reactors. Generation III+ reactors are advanced light-water reactor designs that rely more on passive safety systems, e.g., the AP1000 and the economic simplified boiling-water reactor. Because the first new Generation III+ reactor type that will be subject to the ROP will be the AP1000 reactors that are currently under construction, this paper will refer to the AP1000 design, but the information contained in this document is generally applicable to other Generation III+ reactor designs.

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SUMMARY:

The NRC staff has completed a comprehensive, integrated review of the ROP to determine what revisions to the current oversight program are necessary to provide reasonable assurance of the safe operations of new reactor designs. The staff focused its efforts on the AP1000 design, but concluded the process used in developing the recommended changes to the ROP would be identical for other Generation III+ designs. The staff did not include consideration of small modular reactors or reactors based on advanced non-light water technologies in this review. The applicability of the ROP to those types of operating reactors will be subject of future reviews, as warranted.

For the AP1000, the staff has concluded that the existing ROP is sufficiently flexible to accommodate new large light water reactor technologies through modest adjustments to the program areas of PIs, baseline inspection, and the SDP.

In SRM to SECY-13-0137, "Recommendations for Risk-Informing the Reactor Oversight Process for New Reactors," the Commission noted that the overall structure of the existing ROP should be preserved and directed the staff to enhance the SDP by developing a structured, qualitative assessment tool for events or conditions that are not evaluated in the supporting plant risk models. The Commission also directed that the SDP should continue to place emphasis on the use of the existing quantitative measures of the change in plant risk for both operating and new reactors. Further, the staff was directed to develop guidance to address circumstances that are unique to new reactors, for example due to uncertainty of the reliability of passive structures, systems, and components (SSCs) or other SSCs with limited operational experience.

The staff recognizes that Generation III+ reactors like the AP1000 have core damage frequencies that are lower than currently operating reactors due, in large part, to incorporating operating experience and risk insights at the design phase. This experience was gained during hundreds of plant-years of operations over the last several decades. In addition, the AP1000 design utilizes passive safety features as well as active defense-in-depth systems. Consequently the risk profiles of the AP1000 are lower, and the plants should be safer to operate. The staff's evaluation of the AP1000 standard design may be found in NUREG-1793, Revision 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," issued August 5, 2011.

In its integrated review in assessing oversight resources for the AP1000, the staff attempted to ensure that agency is able to obtain sufficient information regarding licensee performance in each of the ROP cornerstones while reducing effort to oversee performance in the reactor safety cornerstones to reflect the AP1000's inherently safer design. As described further in this document, the staff anticipates a reduction in the NRC's oversight efforts of the AP1000 operating reactors of between 25 to 36 percent as compared to currently operating reactors.

In carrying out the direction in the SRM and after a thorough review of the SDP, the staff concluded that the following four SDP documents should be modified to address the AP1000 design:

- Inspection Manual Chapter (IMC) 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," (ADAMS Accession No. ML101400574)
- IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," (ADAMS Accession No. ML13050A933)
- IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," (ADAMS Accession No. ML041340009)
- IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," (ADAMS Accession No. ML101550365)

The staff has concluded that all of the existing PIs will remain valid for the AP1000, with the exception of the Mitigating Systems Performance Index (MSPI) PIs. The staff is recommending that no new PIs be developed at this time to replace MSPI with the limited AP1000 performance data available. The staff has concluded that, with some changes, the baseline inspection program will ensure the mitigating systems cornerstone objectives are met without the MSPI for the AP1000 design.

The staff has also provided a discussion of planned changes to the baseline inspection program, since all of these modifications to the ROP are best considered in an integrated manner. To evaluate the baseline inspection program, the staff used risk information matrices (RIMs) for currently operating reactors, as well as a draft RIM to determine the risk importance of the AP1000 safety systems and systems subject to Regulatory Treatment of Non-Safety Systems (RTNSS) to identify inspectable areas, frequency, sample sizes, and expected resource effort. The staff reviewed all baseline inspection procedures (IPs) for possible revision and subsequently completed a gap analysis of those procedures. The gap analysis confirmed that current inspection procedures were written at a level of detail such that few changes are required to accommodate new reactors, specifically the AP1000. However, the staff concluded that adjustments to sample sizes and resource estimates would be warranted because there are fewer risk-significant components in designs with passive safety systems from which to sample. Changes to inspection frequencies and implementation will also be required, in some cases, to account for the significant portion of safety systems located inside containment and not accessible during power operations. The staff anticipates a reduction in resource requirements to implement the baseline inspection program. A scoping analysis projected a range of potential reductions for the AP1000 compared to a standard two-unit pressurized water reactor (PWR), as described in the "Baseline Inspection Program" section of this paper. In addition, the staff is proposing to add to the IPs a reference to inspecting systems subject to RTNSS, because of the importance of these systems for defense-in-depth.

BACKGROUND:

Baseline risk estimates for most new reactor designs, including estimates of the risk of both internally- and externally-initiated events, are expected to be lower than currently operating reactors, potentially by an order of magnitude or more. The expected lower risk values raised questions about how to modify the ROP to provide for an appropriate regulatory response to licensee performance. Over the past several years, the staff has interacted with the Advisory Committee on Reactor Safeguards (ACRS) and its Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) regarding proposals to modify the ROP as necessary to accommodate potential new light-water reactors. The staff has also sought approval of staff recommendations on this topic in Commission papers referenced throughout this document.

Most recently, in its SRM on SECY-13-0137, the Commission disapproved the staff's recommendation to develop an integrated risk-informed approach for evaluating the safety significance of inspection findings for new reactor designs using qualitative measures to supplement the risk evaluations. Rather, the Commission directed the staff to enhance the SDP by developing a structured, qualitative assessment tool for events or conditions that are not evaluated in the supporting plant risk models. The Commission stated that areas where such a qualitative assessment may prove useful include evaluation of performance deficiencies associated with passive safety systems, digital instrumentation and controls, and human performance issues. The Commission also directed that the SDP should continue to place emphasis on the use of the existing quantitative measures of the change in plant risk for both operating and new reactors. Further, the staff was directed to develop guidance to address circumstances that are unique to new reactors, for example due to uncertainty of the reliability of passive structures, systems, and components (SSCs) or other SSCs with limited operational experience.

In the same SRM, the Commission approved the staff's recommendation to develop appropriate PIs and thresholds for new reactors, specifically those PIs in the initiating events and mitigating systems cornerstones, or develop additional inspection guidance to address identified shortfalls to ensure that all cornerstone objectives are adequately met. Specifically, the staff was directed to develop, with appropriate stakeholder input, the necessary updates to the PIs, including any new PIs or changes to thresholds, and submit them to the Commission for approval prior to power operation for the first new reactor units. The Commission also directed the staff to further explore how it would apply the current Safety System Functional Failure PI to the passive safety-related components in Generation III+ reactors before deciding upon whether or how to apply this PI for new reactors.

The Commission also noted that the overall structure of the existing ROP should be preserved. Additionally, direction was given that the staff should notify the Commission through the annual report on the ROP self-assessment if the staff identifies any further changes that are necessary, once the staff has gained operating experience with the new Generation III+ plants.

DISCUSSION:

The SRM to SECY-13-0137 directed the staff to provide a proposed approach for revising the SDP for new reactors and to develop necessary updates to the PIs. The staff is also including a discussion of planned changes to the baseline inspection program, since all of these modifications to the ROP are best considered in an integrated manner.

While the subject of this paper is to describe ROP modifications for all new reactor designs, the staff focused its efforts on the AP1000 design initially because of the current new reactor construction schedule. The review process conducted for the AP1000 design would be identical for all Generation III+ designs. The staff would have to complete a similar analysis for other reactor designs, such as small modular reactors or non-light water reactors, to determine the viability of or necessary revisions to each PI, SDP, and inspection procedure.

The staff actively engaged with a variety of internal and external stakeholders with interest and expertise in ROP implementation, risk applications, and new reactor designs. NRC participants included staff from the Office of Nuclear Reactor Regulation (NRR), the Office of New Reactors, the Office of Nuclear Regulatory Research (RES), the regional offices, and the ACRS. External stakeholder participants included representatives from the Nuclear Energy Institute (NEI), current and new reactor licensees, industry consultants, and the public.

The staff conducted several public meetings with stakeholders to solicit input and comments on the ROP for new reactors. This topic was discussed as agenda items as part of 13 ROP Working Group public meetings beginning in early 2015, shortly after the Commission issued the SRM to SECY-13-0137. During these public meetings, the NRC staff and industry exchanged white papers and comments on those papers, discussed the plan for oversight of operating AP1000s as the units transition from construction to commercial operations, discussed PIs that would be valid for the AP1000 design, and discussed planned revisions to the baseline inspection program and SDP to support oversight of the AP1000. In the public meetings that were conducted during the development of the draft paper, industry participants provided information that evaluated each PI and concluded that with the exception of the five MSPI indicators, all of the current PIs should apply to the AP1000 design. Further, the industry concluded that most PIs could be applied with no additional guidance, although some changes to the guidance for the Unplanned Scrams with Complications PI will be necessary. The staff intends to engage with industry in a public forum to discuss necessary revisions to that guidance which is contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guidelines, Revision 7," dated August 31, 2013 (ADAMS Accession No. ML13261A116).

ROP Framework

In developing the ROP for new reactors, the staff used the same principles that guided the development of the original ROP. The principles include independence, openness, efficiency, clarity, and reliability. The agency designed the ROP to ensure that it meets its intended goals of being objective, risk informed, predictable, and understandable. The staff is preserving the existing overall ROP structure for new reactors consistent with Commission direction provided in the SRM to SECY-13-0137. The existing ROP is flexible enough to accommodate new reactor technologies through relatively modest adjustments to the program areas of baseline inspection, PIs, and the SDP. The ROP's risk-informed processes will continue to integrate risk insights with more traditional deterministic factors (such as defense-in-depth and safety margins) to guide regulatory decision-making. The proposed ROP changes described below, and future changes to the ROP, would increase the use of risk information in decision-making activities regarding the oversight of operating nuclear power plants, consistent with the guidance in SRM-M170511, "Briefing On Risk-Informed Regulation," dated June 26, 2017 (ADAMS Accession No. ML17177A397).

The regulatory framework for reactor oversight consists of three key strategic performance areas: reactor safety, radiation safety, and safeguards. Within these strategic performance areas are seven cornerstones that reflect the essential safety aspects of facility operation: initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and security. Satisfactory licensee performance in the cornerstones provides reasonable assurance that licensees are safely operating their facilities and that the NRC is accomplishing its safety and security mission. Each cornerstone contains inspection procedures and PIs to verify that its objectives are being met. The NRC staff evaluates both inspection findings and PIs and gives a color designation based on their safety or security significance. The NRC considers color designations for the inspection findings and

PIs in the ROP Action Matrix to determine a predictable regulatory response. The staff is proposing no changes to the operating reactor assessment program or the ROP Action Matrix for the oversight of new reactors.

Significance Determination Process

Within the ROP, the SDP is used to characterize the safety and security significance of inspection findings. SDP implementation guidance is contained in IMC 0609, "Significance Determination Process" (ADAMS Accession No. ML14153A633) and its appendices. The staff is anticipating no changes to the SDPs for the emergency preparedness, public radiation safety, occupational radiation safety, and security cornerstones. For those cornerstones that rely primarily on PRA (i.e., initiating events, mitigating systems, and barrier integrity), significance determination of inspection findings is based on increases in core damage frequency (ΔCDF) and large early release frequency from a plant's baseline risk. The staff is maintaining those thresholds consistent with the Commission affirmation in its SRM to SECY-10-0121, that the existing safety goals, safety performance expectations, subsidiary risk goals, and associated risk guidance are sufficient for new plants.

The staff performed a comprehensive gap analysis of the existing SDP with respect to the AP1000 design to identify process changes necessary to determine significance of inspection findings for new reactors. The gap analysis concluded that the staff would need to modify only a few SDP procedures. SDP documents that will require modification include:

- IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power"
- IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process"
- IMC 0609, Appendix H, "Containment Integrity Significance Determination Process"
- IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria"

The necessary modifications include new screening questions for the safety cornerstones of initiating events, mitigating systems, and barrier integrity, as well as addressing findings associated with the reliability of passive SSCs, digital instrumentation and control, and human performance issues uniquely associated with operational practices in Generation III+ reactor designs.

The staff plans to revise IMC 0609, Appendix G in order to appropriately credit passive SSCs for performance deficiencies involving traditional (i.e., active) SSCs. Because new reactors have substantially different at-power and shutdown CDFs compared to existing PWRs and boiling-water reactors (BWR), the staff will need to revise IMC 0609, Appendix H to reflect these different values.

The staff is currently developing a revision to IMC 0609, Appendix M. A future revision will consider the AP1000 design, in which the staff will develop a structured qualitative assessment for conditions that are not evaluated in the supporting plant risk models. It will also specify discreet entry conditions and provide a structured framework to both identify and assess the appropriate decision-making attributes and to integrate the results in an objective, reliable, and repeatable manner. The staff will engage both internal and external stakeholders on any proposed revisions to Appendix M. If the staff recommends a significant modification to Appendix M, as defined in the SRM to COMSECY 16-0022, "Proposed Criteria for Reactor Oversight Process Changes Requiring Commission Approval and Notification," dated

May 12, 2017 (ADAMS Accession No. ML17132A364), it will submit the revision to the Commission for approval in accordance with Commission direction in that SRM.

Enclosure 1 provides a detailed summary of the modifications to the SDP to support new reactor designs.

Performance Indicators

The staff completed an extensive review of the existing PIs to determine which PIs would still be valid with the new reactor designs and which PIs would not provide a meaningful indication of plant performance. The staff held discussions with internal and external stakeholders through the ROP Working Group to attempt to either develop new PIs or to modify the existing PIs to be able to monitor operating performance of new reactor designs.

The industry documented their assessment of the validity of PIs for the AP1000 design in two white papers. The first paper (ADAMS Accession No. ML16189A414) evaluated each PI and concluded that with the exception of the five MSPI indicators, all of the current PIs should apply to the AP1000 design. Further, the industry concluded that it could apply most PIs with no additional guidance, although some changes to the guidance for the Unplanned Scrams with Complications PI will be necessary. Guidance is located in NEI 99-02, "Regulatory Assessment Performance Indicator Guidelines, Revision 7," dated August 31, 2013 (ADAMS Accession No. ML13261A116). The staff intends to engage with industry in a public forum to revise that guidance.

In the second white paper (ADAMS Accession No. ML16189A418), the industry performed a focused evaluation of the MSPI for the AP1000 and enlisted the help of former NRC employees who assisted in the initial risk-informed PI development. The analysis confirmed that the MSPI PIs (emergency alternating current (AC) power, high pressure injection, heat removal, residual heat removal, and cooling water systems) could not be applied to the AP1000 reactors. These PIs measure the unavailability and unreliability of the active safety systems relied upon to mitigate the effects of an initiating event. The available performance data on passive systems and components is insufficient to develop meaningful industry-averaged performance baselines that are a key aspect of the MSPI formulation. When MSPI was developed for the current operating reactors, decades of performance data had been developed for the different designs. The paper did note that it could be possible to gather enough plant-specific data over a 3 year monitoring period; however, it may never be sufficient to provide meaningful and robust MSPI values.

The industry white paper also considered non-safety "front line" systems, including those subject to RTNSS, for potential PIs. However, RTNSS system risk worth is so low that, in combination with the low baseline CDF for the AP1000, risk-based PIs such as the MSPI would remain Green under virtually all circumstances for these systems. Green indicates cornerstone objectives are met and licensee performance does not warrant additional regulatory oversight.

The staff documented its review of PIs in a white paper (ADAMS Accession No. ML16251A018) concurring with the industry conclusions regarding the use of MSPI, for both passive safety systems and systems subject to RTNSS.

Options for PIs

The staff has identified the following options with regard to PIs for the AP1000 reactor design:

Option 1: Eliminate the use of the MSPI indicators with no new PIs

Under this option, the staff will eliminate the MSPI indicators without replacing them with new PIs. The SSFF PI coupled with the baseline inspection program would ensure adequate oversight of the mitigating systems cornerstone. The baseline inspection program is flexible enough to ensure the cornerstone objectives continue to be met with only minor adjustments to inspection guidance. This option would result in the least impact on the current ROP framework, and require the least staff resources to implement. The disadvantage to this option is that there would only be one PI monitoring licensee performance for the mitigating systems cornerstone.

Option 2: Eliminate MSPI and develop new PIs for systems classified as RTNSS

Under this option, the staff would attempt to develop PIs for availability or reliability of systems classified as RTNSS. Many of these systems are currently monitored under the MSPI indicators for operating reactors. In the AP1000 design, these systems are not safety-related systems, but they are still important for defense-in-depth considerations. Because the analysis shows that the unavailability and unreliability of these systems would likely not result in a risk-informed threshold being exceeded for the AP1000 design, the staff would have to develop a suitable performance-based threshold, i.e., a certain number of failures to start or failures to run would trip a significance threshold. The MSPI indicators have existing performance limits that act as backstops to indicate when the performance of monitored equipment in an MSPI system is significantly lower than expected industry performance. The performance-based limits are for components with low Birnbaum values where significant degradation in performance could occur before the risk significance crosses the Green-White threshold. The Birnbaum values for the components in the AP1000 will generally be low. Birnbaum importance measures the change in total risk as a result of changes to the probability of an individual basic event. In other words, it measures the sensitivity in the PRA model's output (typically core damage frequency) to changes in the failure probability of a particular structure, system, or component.

Staff case studies, described in SECY-12-0081, "Risk-Informed Regulatory Framework for New Reactors," dated June 6, 2012 (ADAMS Accession No. ML121170025), identified that it would take greater than 25 emergency diesel generator (EDG) start failures or greater than 25 EDG run failures for the U. S. EPR, for example, to exceed the Green-White risk threshold, and 12 failures to reach the performance limit. In another case, it would take 14 or more turbine-driven emergency feedwater pump failures or greater than 25 motor-driven pump failures for the US-APWR to exceed the Green-White threshold using the licensee's PRA model, and the performance limit would not be exceeded until six or more pump failures in a 3 year timeframe occurred. If directed to pursue this option, the staff would likely need to adjust performance limits because it is unlikely that a licensee would exceed a significance threshold with the current limits.

While there may be insights into licensee performance achieved with such a PI, this could be viewed as moving away from a risk-informed ROP. If pursuing this option, the staff would develop appropriate significance thresholds with stakeholder input. This option would require a moderate resource effort. The staff believes there would be little support of this initiative from the industry.

Option 3: Develop a new PI to monitor performance of active components in the passive safety systems

There are 45 valves in the AP1000 passive safety systems (primarily motor-operated, air-operated, and squib valves), 13 of which are considered passive because they are in their safety-related position during normal reactor operations, so if they fail, they would fail such that they could still perform their safety function. The remaining 32 valves are considered active. There are 12 squib valves, with 20 percent of the charges of the explosively actuated valves being tested every 2 years. Together with the continuity checks required for each valve every 2 years, there could be anywhere from 15 to 30 demands on the squib valves over a 3 year period, depending on whether there are one or two refueling outages during that period. This total is significantly lower than the demand frequency for other components, such as EDGs at operating plants that currently average approximately 50 demands per year, or 150 demands in the same 3 year period. The impact of this low demand frequency for squib valves would have to be assessed to ensure that monitoring of the squib valves would produce a viable performance indicator. The remaining 20 motor-operated and air-operated valves are subject to American Society of Mechanical Engineers testing requirements, and therefore will have more demands during a 3 year operating period.

In the current MSPI indicators, the key components being monitored are pumps and EDGs, which have a moderate failure probability, and there is significant industry data from which to develop meaningful risk-informed performance indicators. The AP1000 passive safety systems have active valves that have significantly lower failure probabilities and require significantly fewer demands than those components monitored by MSPI. In addition, these active valves have very low Birnbaum values and are expected to be insensitive to the formulation used by MSPI, i.e., there could be many failures, and the indicator would not cross a significance threshold. The staff concluded that all potential indicators for these components proved to be insensitive when applying the data analysis technique currently used in MSPI. However, several indicators did not appear to be insensitive when employing a different data analysis technique, maximum likelihood estimation, which makes no use of historical information, and derives an estimate from current failure and demand information. A more comprehensive assessment of this method needs to be performed if implemented.

While the staff is not currently capable of developing a statistically meaningful PI for these valves, the staff could initiate a research effort to develop such a PI. The disadvantage of this option is the expected large resource effort, which would include engaging with a national laboratory on the research effort to develop a statistical approach, outreach efforts with the public, Commission interaction, and implementing the new PI, in order to monitor components that have a low failure probability. The purpose of PIs is to call licensee and inspector attention to potentially degrading performance in a monitored cornerstone of safety. These components will be monitored under the maintenance rule and in-service testing, and failures of these components will likely be evaluated through the reactive inspection process because of their safety significance, so a PI to monitor their performance may be considered unnecessary.

The staff recommends that the Commission approve Option 1. This option ensures the mitigating cornerstone objectives are met, requires the least staff resource effort, and ensures the ROP remains risk-informed to the extent possible. The staff will evaluate the viability of new PIs after sufficient operating experience is gained.

The staff also considered whether or not the NRC should adjust PI thresholds based on the new reactor designs. In the SRM to SECY 10-0121, "Modifying the Risk-Informed Regulatory Guidance for New Reactors," dated March 2, 2010 (ADAMS Accession No. ML110610166), the Commission reaffirmed "that the existing safety goals, safety performance expectations, subsidiary risk goals and associated risk guidance..., key principles and quantitative metrics for implementing risk-informed decision making, are sufficient for new plants." The staff completed an analysis and concluded that the existing PI thresholds should remain unchanged until sufficient operating experience is gained to determine if thresholds should be adjusted. This analysis was documented in SECY-13-0137 (ADAMS Accession No. ML13263A351). For example, the thresholds for the initiating events cornerstone PIs were established incorporating both performance and risk data to be commensurate with a generally achievable level of performance that takes into account the statistical variability across the current plant designs. While the lower risk profile associated with the AP1000 may ultimately result in a change to the existing thresholds, the staff intends to maintain those thresholds until sufficient operating experience exists to inform the threshold changes.

Application of the Current SSFF PI

The Commission also tasked the staff in the SRM to SECY-13-0137 to "further explore how the current SSFF PI would be applied to the passive safety-related components in Generation III+ reactors before deciding upon whether or how to apply this PI for new reactors." The staff reviewed the technical specifications for the AP1000 units under construction at the Vogtle site and determined that the SSFF PI could be adequately applied to these new designs with no changes. Given that the passive systems all have multiple trains of actuation valves, the Green-White threshold of five failures applicable to current PWR designs is expected to be adequate.

In summary, the staff concluded that 12 of the 17 PIs monitoring the performance of the current reactor fleet are applicable to new reactor designs with minimal revision to NEI 99-02. For the mitigating systems cornerstone, the MSPI indicators would not be applicable, so the only PI would be SSFF. If the Commission approves Option 1 above, the staff will adjust the baseline inspection program to ensure the mitigating systems cornerstone objectives are adequately met. Those adjustments are described in the next section. The staff will maintain the existing PI thresholds until enough operating experience exists to perform additional analysis. The staff anticipates that 3 years of operating experience data will be needed before assessing changes to the PI thresholds. The staff will assess the viability of new PIs to replace the current MSPI indicators once sufficient operating experience has been gained. As with all PIs, the staff will continuously evaluate the need to adjust PIs or thresholds as a part of the annual ROP self-assessment process.

<u>Baseline Inspection Program</u>

During initial ROP development, RES developed RIMs to identify the inspectable areas, frequency, sample sizes, and expected resource effort for the baseline inspection program. The agency developed RIMs for most PWRs and BWRs based on the Individual Plant Examinations, Individual Plant Examination External Events, and risk achievement worth values for various components.

Using the RIMs for the currently operating reactor types and the AP1000 safety performance verification matrix, the staff developed a draft RIM to determine the risk importance of the AP1000 RTNSS and safety systems (ADAMS Accession No. ML16244A160) and a draft inspection procedure RIM (ADAMS Accession No. ML16244A148) for the AP1000. The AP1000 safety performance verification matrix was developed to present the key attributes of the AP1000 SSCs, and how they will be evaluated and assessed in the ROP. The staff shared the RIMs with the industry on September 21, 2016, during an ROP public meeting (ADAMS Accession No. ML16288A215), as well as with the NRC regional construction staff. All ROP baseline IPs were reviewed for applicability to the AP1000 reactor design. The staff review determined that several baseline IPs in the initiating events, mitigating systems, and barrier integrity cornerstones could be revised because of the reduced risk and reduced number of components associated with the passive safety system design of the AP1000. The staff subsequently completed a gap analysis of the existing procedures to determine what changes, if any, might be required to ensure adequate inspection coverage of the new reactor design. The gap analysis confirmed that the NRC had written inspection procedures at a level of detail such that few changes were required to accommodate new reactor designs. However, the staff concluded that adjustments to sample sizes and resource estimates are warranted.

Proposed Revisions to Baseline IPs

The staff is proposing to add a reference to inspecting systems subject to RTNSS in the IPs because of the importance of these systems to defense-in-depth. These systems include: the diverse actuation system (DAS), normal residual heat removal system, component cooling water system, service water system, post-72-hour makeup water sources, main control room fans, instrumentation room fans, hydrogen igniters, onsite AC power, offsite AC power, ancillary diesel generators, non-Class 1E direct current and uninterruptible power supplies for the DAS anticipated transient without scram mitigation function and reactor vessel insulation.

The staff concluded that sample sizes for several IPs could be reduced because there are fewer components in the AP1000 design to select as samples and the components have lower baseline risk estimates. The staff expects to establish a sample range based on the risk importance (high and intermediate) of a system and whether the system is classified as RTNSS. The staff used the list of systems for the AP1000 from the safety performance verification matrix, the Virgil C. Summer combined license (ADAMS Accession No. ML14100A092), and the AP1000 technology manual in conducting the review. The risk importance of each system is defined in IMC 2519, "Construction Significance Determination Process," and is determined by the ΔCDF when the SSC is assumed to be completely unavailable. The risk levels are defined as follows:

- High risk is defined as a ΔCDF greater than 1E-4.
- Intermediate risk is a ΔCDF less than 1E-4 but greater than 1E-5.
- Low risk is a ΔCDF less than 1E-5 but greater than 1E-6.
- Very low risk is a ΔCDF less than 1E-6.

The staff plans to adjust sample sizes for several baseline inspections based on the limited availability of appropriate risked-informed sample opportunities. In addition, changes to inspection frequencies and implementation will also be required in some cases to account for the significant portion of safety systems located inside containment and not accessible during power operations.

Based on an initial review, the staff concluded that the following IPs will likely have fewer samples required because there are fewer components associated with the AP1000 design:

- 71111.12, "Maintenance Effectiveness"
- 71111.13, "Maintenance Risk Assessments and Emergent Work Control"
- 71111.15, "Operability Determinations and Functionality Assessments"
- 71111.19, "Post Maintenance Testing"
- 71111.21M, "Design Bases Assurance Inspection (Teams)"

- 71111.22, "Surveillance Testing"

Sample requirements for several other IPs may be reduced because of the lower overall risk of the new reactor design.

Under IP 71111.04, "Equipment Alignment," the staff anticipates conducting full system alignments for passive safety systems inside containment prior to containment closeout following refueling outages, and partial or selected system walkdowns following emergent outages, to ensure safety systems are available through the following operating cycle.

The staff concluded that the IPs associated with radiation protection, emergency preparedness, and security, would remain unchanged initially because they are not dependent on system design or numbers of components. The staff will assess necessary changes to all baseline IPs as part of the annual ROP self-assessment process and biennial ROP realignment effort.

The staff completed an analysis of all baseline inspection procedures to determine appropriate sample sizes and resource estimates. The cumulative impact of these proposed changes to the inspection program could potentially result in a reduction of 25 percent (about 570 hours) in direct inspection effort required to complete baseline inspection activities for a two-unit AP1000 compared to a standard two-unit PWR as a standalone facility.

The staff completed an analysis of resource implications when co-locating a two-unit AP1000 plant with an existing operating reactor plant, e.g., Vogtle. The staff intends to treat the facility as a single four-unit site for security and emergency preparedness inspections. The staff also plans to conduct one Force-on-Force inspection for the entire facility, instead of two separate exercises. Efficiencies can also be gained by performing a consolidated emergency preparedness exercise, as well as a biennial problem identification and resolution inspection. The staff concluded that these economies of scale will result in a 36 percent reduction (840 hours) in direct inspection for the AP1000 compared to a standard two-unit PWR. Additionally, the staff determined that there will also be a 12 percent reduction (270 hours) in direct inspection effort attributed to the existing operating units because of the shared inspection effort.

It should be noted that the staff's estimates of inspection efforts for Vogtle are subject to change. The degree to which operations, engineering, maintenance, and corrective action programs are common to the four Vogtle units will affect the estimates. In addition, operating experience from Sanmen and Haiyang could influence estimates. Finally, the estimates are for a nominally-operating AP1000. Inspection resources during testing and initial operations will be higher.

The staff is also considering other changes to baseline inspections to account for the limited availability of appropriate risk-informed sample opportunities at-power and the design's reliance on passive safety systems. Many of the risk-significant systems are located in containment and can only be inspected during outages. For example, the staff is considering adding guidance to the IP 71111.20, "Refueling and Other Outages," and IP 71111.04, regarding the inspection of RTNSS and passive systems. In addition, the AP1000 containment is an integral part of the passive containment cooling system, acting as a heat exchanger. Therefore, the staff is considering adding guidance to IP 71111.07, "Heat Sink Performance," to ensure any degradation that could affect heat transfer performance of the containment is considered. These proposed changes to the inspection program will result in a greater inspection resource effort during refueling outages, with less inspection effort expected during the operating cycle.

After gaining inspection experience on the new AP1000 units, the staff will assess the effectiveness and availability of appropriate samples using IMC 0307 Appendix B, "Reactor Oversight Process Baseline Inspection Procedure Reviews," and adjust or recommend changes to the AP1000 baseline inspections.

Addressing Baseline Inspection of Passive Systems Without MSPI

In the ROP, the PI and baseline inspection programs are complementary, i.e., baseline inspections are conducted in areas not adequately covered by PIs, or where a PI does not fully address the objectives of the cornerstone. The objective of the mitigating systems cornerstone is to monitor the availability, reliability, and capability of systems that mitigate the effects of initiating events to prevent core damage. Licensees reduce the likelihood of reactor accidents by maintaining the availability and reliability of mitigating systems. The purpose of the MSPI performance indicators is to monitor availability and reliability of safety systems necessary to mitigate accidents. Because the MSPI performance indicators would be ineffective for monitoring licensee performance for the AP1000, the staff conducted a review of inspection procedures to determine if any changes to the baseline inspection program were warranted in order to ensure the mitigating systems cornerstone objectives are met. Therefore, the staff focused its review on inspection procedures that could be used to monitor availability and reliability of the unique passive safety systems associated with the AP1000 reactor design.

Considering the breadth of baseline inspections that assess the availability, reliability, and capability of mitigating systems and the purpose of the MSPI, which is to monitor the readiness of important safety systems to perform their safety functions in response to off-normal events or accidents, the staff has determined that no new inspections are needed to compensate for elimination of the MSPI indicators for the AP1000 design. The staff found that the current suite of baseline inspection procedures is sufficient to monitor the performance of licensees operating new reactor designs with regard to availability and reliability of the passive safety systems. However, the staff intends to provide additional inspection guidance in some existing IPs to ensure inspectors focus efforts on those unique passive safety systems. Revisions to inspection guidance are still being developed. Because the passive nature of these safety systems make them inherently reliable, the staff intends to focus inspection efforts on ensuring availability. For example, the staff is considering adding guidance to IP 71111.04 to verify proper alignment of the passive safety systems prior to containment closeout following an outage to ensure availability of those systems during operation. Also, the staff is considering adding guidance to IP 71111.22 to ensure availability and reliability of the active components (motor-operated valves, air-operated valves, and squib valves) associated with the passive safety systems. The guidance would ensure inspectors verify that the licensee is conducting appropriate surveillance testing, especially for the explosive squib valves, in accordance with the approved technical specifications, and the design certification as described in the safety evaluation report for the AP1000.

Staffing

Upon transition to the ROP, the staff plans to conduct additional baseline inspection to monitor the initiating events and mitigating systems cornerstones until the PIs become valid.

Based on experience with the transition of Watts Bar Unit 2 from construction to operation, and due to the lack of experience with the new reactor plant designs, the staff is planning a larger-than-normal complement of inspectors onsite for a few months after startup. The long-term post-commercial operations resident inspector staffing will be established with consideration of the sites' existing operating units and the AP1000 ROP implementation requirements, in collaboration with NRR. The level of resident inspector staffing will reflect consideration of the initial start-up phase of the plants and then, for the longer term, the enhanced safety and lower level of risk inherent in the AP1000 design.

During outages, safety-related systems, high and intermediate risk important systems, and systems identified as RTNSS become more accessible for inspection. The staff is considering formation of an outage inspection team to support the inspections of those systems. Because so many of the baseline inspections conducted during the outage fall under the operations discipline, the team might be augmented by additional inspection staff. This additional staff would assist with conducting as-left equipment walkdowns, surveillance testing, post-maintenance testing, and containment closeout. Other team members determined by the region might include engineering and health physics inspectors necessary to complete in-service inspection, in-service testing, and radiation protection inspections. This will result in a greater inspection resource effort during outages that offset the reduced inspection effort during the operating cycle. The net result will be a reduction of 25 percent in inspection resource expenditures for a standalone AP1000 facility, and 36 percent for an AP1000 co-located with an existing operating reactor plant.

Conclusions

The ROP is a constantly evolving program and is flexible enough to accommodate Generation III+ reactor designs, such as the AP1000, with relatively modest changes. The current framework ensures that the NRC will continue to meet its mission of protecting the public health and safety and the environment. The majority of PIs continue to apply to the AP1000 technology, and inspection efforts can compensate for the elimination of the MSPI indicators to ensure the mitigating systems cornerstone objectives continue to be met. The baseline inspection procedures currently in place provide for sufficient oversight of all seven cornerstones of the ROP. Because of the increased reliance on passive safety systems and fewer components associated with the AP1000 design, several baseline inspection procedures will have reduced inspection sample requirements. The staff is recommending very few changes to the SDPs for evaluating inspection findings for the AP1000, and no changes to the assessment program. Additionally, the staff anticipates a 25 – 36 percent reduction in the resources needed in carrying out the ROP activities for the AP1000 operating reactors.

RECOMMENDATIONS:

The staff recommends that the Commission approve Option 1, to eliminate the MSPI performance indicators for the AP1000, with no new PIs being developed during initial operation, with limited modifications being made to the baseline inspection program. For any other changes being made to the ROP that meet the definition of "significant" as defined in the SRM to COMSECY-16-0022, the staff will seek Commission approval prior to implementing those changes. For example, if the staff determines that significant modifications to the SDP are needed, then the staff will submit a separate paper requesting Commission approval of those modifications.

RESOURCES:

The staff expects that implementation of the described revisions to the SDP and PI programs of the ROP will have a minimal impact on resources. The staff anticipates resource savings of 25 to 36 percent implementing the baseline inspection program for new reactor designs because of the proposed reduction in sample sizes in several inspection procedures. The staff will use a 25 to 36 percent reduction as a planning estimate, and will adjust as further experience is gained. The resource impact is anticipated to be for fiscal year 2020 and beyond, and the staff will address this impact through the planning, budget, 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.

Margaret M. Doane

Margaret M. Doane
Executive Director
for Operations

Enclosure:
Significance Determination Process

Commissioners' completed vote sheets/comments should be provided directly to the Office of the Secretary by COB Thursday, September 27, 2018.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT COB Thursday, September 20, 2018, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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Significance Determination Process

The staff performed a comprehensive review of the existing Significance Determination Process (SDP) to identify gaps in the process to account for unique elements associated with new reactor designs, specifically the AP1000. The gap analysis took into account the following from the Staff Requirements Memoranda (SRM) to SECY-10-0121 and SECY-13-0137.

- There would be no change in significance determination thresholds for Green, White, Yellow, and Red inspection findings.

- The use of an integrated risk-informed approach for evaluating safety significance of all inspection findings for new reactor designs using qualitative measures to supplement the risk evaluations was disapproved by the Commission.

- Staff should enhance the SDP by developing a structured qualitative assessment for events or conditions that are not evaluated in the supporting plant risk models. Examples include inspection findings associated with passive safety systems, digital instrumentation and control (I&C), and human performance issues.

- The revised SDP should continue to place emphasis on the use of the existing quantitative measures of the change in plant risk for both operating and new reactors.

- Staff should develop guidance to address circumstances that are unique to new reactors, for example due to uncertainty of the reliability of passive systems, structures, systems and components (SSCs) or other SSCs with limited operational experience.

The gap analysis concluded that the staff would need to modify very few SDP procedures. From a higher tier program perspective, the staff will need to revise Inspection Manual Chapter (IMC) 0308 Attachment 3, "Significance Determination Process Technical Basis Document" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15268A268), to accommodate the passive nature of new reactor designs and the corresponding lower core damage frequencies with the significance thresholds being unaffected. The staff will modify the main SDP program document (IMC 0609, "Significance Determination Process") to provide guidance that inspection findings related to implementation of operational programs identified prior to the U.S. Nuclear Regulatory Commission (NRC) staff making the Title 10 of the *Code of Federal Regulations* Paragraph 52.103(g) (10 CFR 52.103(g)) finding will be dispositioned using IMC 0609, the operational SDP. Additionally, any findings related to the development of operational programs identified after the 10 CFR 52.103(g) finding will be dispositioned using IMC 2519, "Construction Significance Determination Process."

The staff expects that the other higher tier SDP program documents listed below will not require modification due to their design neutral nature.

- IMC 0609 Attachment 1, "Significance and Enforcement Review Panel (SERP) Process"
- IMC 0609 Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)"
- IMC 0609 Attachment 3, "Senior Reactor Analyst Support Expectations"
- IMC 0609 Attachment 4," Initial Characterization of Findings"

The review of other lower tier SDP program documents (i.e., SSC-specific SDP appendices A through M and their associated technical basis documents) concluded that only four appendices would require modification. These include the following:

- IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power"
- IMC 0609 Appendix G, "Shutdown Operations Significance Determination Process"
- IMC 0609 Appendix H, "Containment Integrity Significance Determination Process"
- IMC 0609 Appendix M, "Significance Determination Process Using Qualitative Criteria"

Regarding IMC 0609, Appendix A, the staff will revise the screening questions to address the unique design and operational practices of advanced reactor plants. The staff will develop a set of screening questions for the safety cornerstones of initiating events, mitigating systems, and barrier integrity to screen out very low risk-significant findings because the internal events baseline risk of Generation III+ reactor plant designs are typically very low (e.g., baseline core damage frequency (CDF) of 1E-7 per year or less). However, since the baseline risk for external events (e.g., fire, flood, seismic events, etc.) may vary from CDF values of 1E-6 to 1E-5 per year, these events will be considered in this appendix. In addition to the unique differences in baseline risk profile, the staff will modify the IMC 0609, Appendix A screening questions for Generation III+ reactor power plants to address findings associated with the reliability of passive SSCs, digital I&C, and human performance issues uniquely associated with operational practices in these designs.

Regarding IMC 0609, Appendix G, the general approach of the IMC will work for the AP1000. However, the procedure's detailed analysis will be revised to reflect the AP1000's passive design features. Specifically, the staff needs to modify IMC 0609, Appendix G in two areas. First, the evaluation process must be modified to address inspection findings associated with the reliability of passive SSCs, digital I&C, and human performance issues uniquely associated with operational practices. Second, the evaluation process must be modified so that those same passive SSCs are appropriately and sufficiently credited in performance deficiencies involving traditional (i.e., active) SSCs.

Regarding IMC 0609, Appendix H, the general approach of the IMC will work for the AP1000. However, the staff needs to review and revise the procedure's detailed analysis to reflect the AP1000's passive and other unique design features. Specifically, IMC 0609, Appendix H builds upon calculated generic baseline at-power and shutdown CDFs for existing reactors. The AP1000 has substantially different at-power and shutdown CDF profiles compared to existing pressurized water reactors (PWRs). Therefore, the staff will need to revise this appendix to reflect these different values. In addition, the existing appendix does not take into consideration passive containment cooling systems. Therefore, the staff needs to revise the appendix first to credit these passive systems in its risk analysis, and second to evaluate any inspection findings identified in those passive systems.

The NRC staff may modify IMC 0609 Appendix M, to address Commission direction to develop a structured qualitative assessment for events or conditions[1] that are not evaluated in the

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[1] The SDP assesses licensee performance deficiencies and associated degraded conditions that are determined to be of more than minor significance. Plant events are not assessed by the SDP. Rather Management Directive 8.3, "NRC Incident Investigation Program" is used to assess the significance of plant events.

supporting plant risk models. The planned revision to IMC 0609, Appendix M would specify discreet entry conditions and provide a structured framework to both identify and assess the appropriate decision-making attributes and to integrate the results in an objective, reliable, and repeatable manner. These revisions to Appendix M to account for the AP1000 design would be in addition to other revisions being developed to provide clarity of existing entry conditions and decision-making attributes. The staff would engage both internal and external stakeholders and the Commission, if needed, to solicit all points of view. Use of IMC 0609, Appendix M for these situations is consistent with the existing SDP program.

The staff will develop the necessary changes mentioned above working with industry representatives to ensure the SDP procedures will be ready for use by at least 1 year prior to the 10 CFR 52.103(g) finding for the first unit to complete construction.

Other SSC-specific SDP appendices and the technical basis for not needing modification are presented below in Table 1.

Table 1

Significance Determination Process Appendix	Basis for No Modification
Appendix B, "Emergency Preparedness Significance Determination Process"	The significance determination logic in Appendix B is largely based on the 16 planning standards, which were broadly written in a design-neutral manner and have been applied to the current fleet of plants licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and the AP1000 reactor sites. The staff identified no gaps.
Appendix C, "Occupational Radiation Safety Significance Determination Process" and Appendix D, "Public Radiation Safety Significance Determination Process"	The staff confirmed that Appendix C and D will continue to fulfill the objectives of IMC 0609. The staff designed Appendix C and D to evaluate inspection findings with regards to their actual or potential radiological consequences; this approach is independent of reactor design. The staff identified no gaps.
Appendix E, "Security Significance Determination Process" Parts I–IV	The security baseline inspection program is based on the verification of licensee performance and compliance with the applicable physical protection requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials," and implements a risk-informed approach for the conduct and scope of inspection activities, as well as the application of significance determination for enforcement under the Reactor Oversight Process (ROP). Following a review of the security baseline inspection program and associated SDPs, staff determined that the

	program addressed all areas with no gaps identified. The cyber security SDP has provided accurate, predictable, and repeatable significance assessments. As a part of the ROP self-assessment and realignment process, the staff and industry identified enhancements relative to cyber security guidance. Because the cyber security SDP takes into account plant-specific SSCs, there are no anticipated changes needed relative to new reactors.
Appendix F and associated Attachments 1-8, "Fire Protection Significance Determination Process"	New reactors are licensed to the same fire protection standards as existing reactors, with one significant additional requirement. The additional requirement is that new reactors must assume full-room burnout in all of their fire areas. This additional requirement does not affect the use of Appendix F for inspection findings. Although new reactors may be less susceptible to core damage from postulated events and operator errors, licensed operator performance can affect plant risk for new
Appendix I, "Operator Requalification Human Performance Significance Determination Process"	reactor technologies. As such, licensed operator performance on requalification examinations, and the ability of facility licensees to properly develop and administer these examinations are valid assessment areas for new reactors. The staff has determined that the methodology contained in IMC 0609 Appendix I is equally valid for assessing licensed operator requalification inspection findings for both operating and new reactors. Additionally, the significance thresholds for inspection findings addressed by IMC 0609 Appendix I are appropriate for new reactors without modification, and the staff determined that no new inspection areas or finding thresholds are necessary to accommodate new reactors.

Appendix J, "Steam Generator Tube Integrity Findings Significance Determination Process"	The licensing basis for AP1000 steam generators is similar to the current licensing basis for operating PWRs. Therefore, the methodology that would be used for AP1000 steam generator tube integrity issues is consistent with the current methodology in Appendix J. In addition, the AP1000 steam generators are similar to the steam generators found in Combustion Engineering plants. Therefore, no revisions to Appendix J will be necessary.
Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process"	The staff determined that the current Appendix K is suitable for use with AP1000 nuclear power plants. This is because (a) plants licensed under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," are required to follow 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," in the same manner as plants licensed under 10 CFR Part 50, and (b) the differences in design and licensing under 10 CFR Part 52, do not require changes to be made to this appendix for new reactor designs.
Appendix L, "B.5.b Significance Determination Process" Appendix O, "Significance Determination Process for Mitigating Strategies and Spent Fuel Pool Instrumentation"	IMC's 0609 Appendices L and O are screening tools used in the ROP for inspection findings associated with the development and implementation of guidance and strategies as required by 10 CFR 50.54(hh), NRC Order EA-12-049, and NRC Order EA-12-051. These requirements are applicable to the AP1000 series reactors. Therefore, IMC 0609 Appendices L and O would be used to assess the safety significance of inspection findings at AP1000 reactors associated with these requirements. The appendices are generic with respect to plant type and technology, and are focused on deficiencies with respect to equipment and strategies associated with these requirements. Even though the requirements and methods of implementation for the AP1000 may differ slightly from previously licensed and operating reactors, the process for assessing the safety significance of related inspection findings would remain the same. As a result, the staff should not need to specifically update IMC 0609 Appendices L and O to accommodate technical differences associated with the AP1000 series reactors.

SUBJECT: RECOMMENDATIONS FOR MODIFYING THE REACTOR OVERSIGHT PROCESS FOR NEW REACTORS DATED:

ADAMS Accession Number.: ML17166A238 **SRM-S13-0137-1, 3, and 4** *concurred via email

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