White Paper with Staff Recommendations for the Reactor Oversight Process for New Reactors

PURPOSE:

The purpose of this paper is to brief the ACRS on the staff recommended changes to the Reactor Oversight Process (ROP) for new reactor designs. 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), directed the staff to (1) submit a paper to the Commission 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 Performance Indicator would be applied to the passive safety-related components in Generation III+ reactors.

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. 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 (ESBWR)).

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:

Although the SRM to SECY-13-0137 specifically requested papers regarding proposed changes to the SDP and PI program, 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 new reactor designs. The review process demonstrated that the current ROP framework is flexible enough to ensure adequate oversight of new reactor designs with only modest modifications to the PI, SDP, and baseline inspection program areas. For the PI program, the staff analysis was specific to the AP1000 design. The staff would have to complete a similar analysis for other reactor designs to determine the viability of each of the PIs.

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 (NRO), the Office of Nuclear Regulatory Research (RES), the regional offices, and the ACRS. External stakeholder participants included representatives from the Nuclear Energy Institute (NEI), 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 during 13 ROP Working Group public meetings beginning in early 2015, shortly after the Commission issued SRM-SECY-2013-0137. Although the agency posted notices about these meetings and conducted them as public meetings, NRC staff and industry representatives were the primary participants in the discussions. During these public meetings, the NRC staff and industry exchanged white papers and comments on those papers, discussed the plan for oversight of the AP1000 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. Based on discussions and feedback from the public meetings conducted during the development of the draft paper, participants generally agreed with the evaluations, conclusions, and recommendations provided in this paper.

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, will 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 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.

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 be appropriate. 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 Mitigating Systems Performance Index (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. The staff intends to engage with industry in a public forum to revise that guidance in 2018.

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. Specifically, 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 nonsafety “front line” systems, including RTNSS, for potential PIs. However, their risk worth is so low that, in combination with the low baseline core damage frequency (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 RTNSS. The staff also evaluated a new risk-informed valve unreliability indicator that would monitor explosive squib, air-operated, motor-operated, and solenoid-operated valves relied upon by the passive safety systems for successful operation. The staff concluded that sufficient industry data on the active components within the passive safety systems does not currently exist. Because of the low numbers of expected demands for these components, along with their variable risk worth, a risk-informed PI focused on unreliability could change by several orders of magnitude by simply adding extra demands or changing the risk worth through plant modifications or PRA updates. The staff concluded that the limited size of the data set does not support the statistical analysis and conclusions needed to predictably and appropriately assess licensee performance. Additionally, for many components, given the low frequency of testing, the number of actuations in a 3-year period could be less than the number required to cross the threshold from Green to White. In other words, the valves could fail every demand, and the indicator would still be Green. The staff determined this would be inappropriate as a measure of licensee performance. Without the MSPI indicators, and no new PIs identified for the mitigating systems cornerstone, the staff will adjust the inspection procedures to ensure the mitigating cornerstone objectives are met, as discussed below.

The Commission also tasked the staff in the SRM to SECY-13-0137 to “further explore how the current Safety System Functional Failure (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 and Virgil C. Summer sites 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 pressurized-water reactor (PWR) designs should be adequate.

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).

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, “Regulatory Assessment Performance Indicator Guidelines, Revision 7,” dated August 31, 2013 (ADAMS Accession No. ML13261A116). For the mitigating systems cornerstone, the MSPI indicators would not be applicable, so the only PI would be SSFF. At this time, the staff is not considering any new PIs to monitor licensee performance in any cornerstone. The staff will assess the viability of new PIs to replace the current MSPI indicators once sufficient operating experience has been gained. Without the MSPI indicators, 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. 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.

Baseline Inspection Program

During initial ROP development, the Office of Nuclear Regulatory Research developed risk information matrices (RIMs) to identify the inspectable areas, frequency, sample sizes, and expected resource effort for the baseline inspection program. The agency developed risk information matrices for most PWRs and boiling-water reactors (BWRs) at the time based on the Individual Plant Examination (IPE), Individual Plant Examination External Events (IPEEE), and risk achievement worth (RAW).

Using the current ROP RIMs and the AP1000 safety performance verification matrix, the staff developed a draft risk importance RIM (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 structures, systems, and components, 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. The ROP baseline IPs have been reviewed for applicability to the AP1000 reactor design. The staff review determined that 20 baseline IPs could potentially 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 analyses 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. In addition, the staff is proposing to add a reference to inspecting systems subject to RTNSS in the IPs because of the importance of these systems. 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 (MCR) fans, instrumentation room fans, hydrogen igniters, onsite AC power, offsite AC power, ancillary diesel generators, non-Class 1E direct current (DC) and uninterruptible power supplies (UPSs) for the DAS anticipated transient without scram (ATWS) 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 from which to select as samples and the 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 staff is considering modifying inspection sample ranges listed in applicable IPs (e.g., Inspection Procedure 71111.04, “Equipment

Alignment") for AP1000 reactor designs. The risk importance of each system is defined in IMC 2519, "Construction Significance Determination Process," and is determined by the ΔCDF when the structure, system, or component (SSC) is assumed to be completely unavailable. The risk levels are defined as follows:

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

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 additional inspections are not needed to compensate for not using the MSPI indicators for the AP1000 design.

However, the staff is considering other changes to baseline inspections to account for the limited availably of appropriate risk-informed sample opportunities at-power and the design's reliance on passive safety systems. Many of the risk-significant systems reside 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, "Equipment Alignment," regarding the inspection of RTNSS and passive systems. After gaining inspection experience on the new AP1000 units, the staff will assess the effectiveness and availability of appropriate samples using Inspection Manual Chapter (IMC) 0307 Appendix B, "Reactor Oversight Process Baseline Inspection Procedure Reviews," and adjust or recommend changes to the AP1000 baseline inspections.

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 at least several 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 low level of risks 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 an additional resident inspector. This additional inspector would assist with conducting as-left equipment walkdowns, surveillance testing, post-maintenance testing, and containment closeout. Other team members would be determined by the region and might include engineering and health physics inspectors necessary to complete inservice inspection (ISI), inservice testing (IST), and radiation protection inspections. The staff is still evaluating the feasibility of this outage team concept.

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 Inspection Manual Chapter (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 probabilistic risk assessment (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 (ΔLERF) 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 very 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;” and 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 (I&C), and human performance issues uniquely associated with operational practices in Generation III+ reactor designs.

The staff will need 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, the staff will need to revise IMC 0609 Appendix H to reflect these different values.

The staff may modify IMC 0609 Appendix M to develop a structured qualitative assessment for conditions that are not evaluated in the 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. 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.

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 SRM-SECY-10-0121 and SRM-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 IMC 0308 Attachment 3, “Significance Determination Process Technical Basis Document” (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 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 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 NRC staff making the 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 construction SDP.

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 an effective 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 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 instrumentation and control, 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 instrumentation and control (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 core damage frequencies (CDFs) for existing reactors. The AP1000 has substantially different at-power and shutdown CDF profiles compared to existing 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 supporting plant risk models. The planned revision to IMC 0609 Appendix M would specify

[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.

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 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 one 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 Title 10 of the *Code of Federal Regulations* 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 radiological consequences, or potential 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 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 systems, structures, and components, 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 additional requirement. The additional requirement is that new reactors must assume full-room burnout in all of their fire areas. The AP1000 design is licensed to Branch Technical Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976," as is the current operating fleet. This additional requirement does not affect the use of Appendix F for inspection findings.
Appendix I, “Operator Requalification Human Performance Significance Determination Process [SDP]”	Although new reactors may be less susceptible to core damage from postulated events and operator errors, it is the staff’s position that licensed operator performance can affect plant risk for new 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 all potential more than minor inspection findings associated with the development and implementation of guidance and strategies as required by NRC Orders EA-02-026, EA-12-049, and EA-12-051. All three NRC orders 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 the orders. However, the appendices are generic with respect to plant type and technology, and are focused on deficiencies with respect to equipment and strategies associated with the Orders. Even though the requirements and methods of implementation for the AP1000 may differ slightly from legacy reactors, 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.