Options for Proceeding with Future
Level 3 Probabilistic Risk Assessment Activities

PURPOSE

This document provides more detailed information on (1) the basis for originally proposing a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 probabilistic risk assessment (PRA) for a nuclear power plant (NPP)[1], (2) potential future uses for Level 3 PRAs, (3) three primary options for proceeding with future Level 3 PRA activities[2], and (4) the activities that supported development of items 2 and 3.

A separate document included as the second enclosure to the notation vote SECY paper provides more detailed information on the structure and evolution of PRA and risk-informed regulation that led to the staff's original proposal for a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

[1] As used in this document and the SECY paper to which it is enclosed, a full-scope comprehensive site Level 3 PRA is a PRA that includes a quantitative assessment of the public risk from accidents involving all site reactor cores and spent nuclear fuel that can occur during any plant operating state, and that are caused by all initiating event hazards (internal events, fires, flooding, seismic events, and other site-specific external hazards).

[2] This document and the SECY paper to which it is enclosed distinguish between "Level 3 PRA activities" and "Level 3 PRAs." The latter refers to a PRA that includes specific technical elements or analyses to assess the public risk from a NPP, while the former refers to activities (e.g., research and development) specifically related to or in support of Level 3 PRAs.

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BASIS FOR PROPOSING NEW LEVEL 3 PRA ACTIVITIES

In 1995, the Commission established the current framework for risk-informed regulation by issuing a PRA Policy Statement[3] that stated the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art and in a manner that complements the U.S. Nuclear Regulatory Commission's (NRC's) deterministic approach and traditional defense-in-depth philosophy. In its approval, the Commission articulated its expectation that implementation of this policy would improve the regulatory process in three areas (1) through safety decisionmaking enhanced by the use of PRA insights, (2) through more efficient use of agency resources, and (3) through a reduction in unnecessary burdens on licensees.

Traditionally focused on accidents involving single-unit reactor cores, PRAs for NPPs can estimate risk metrics at three sequential levels or end states. A Level 1 PRA models system (plant and operator) response to various initiating events that challenge system operation to estimate reactor core damage frequency (CDF). A Level 2 PRA includes Level 1 PRA analyses and, in addition, models system and containment response to severe core damage accidents to estimate conditional containment failure probabilities, radioactive material release frequencies (e.g., large early release frequency [LERF]), and various source term characteristics. Finally, a Level 3 PRA includes Level 2 PRA analyses and, in addition, models the transport and dispersion of released radioactive materials to estimate various offsite radiological health and economic consequence measures. By combining radioactive material release frequencies from a Level 2 PRA with the offsite radiological consequences associated with each release, a Level 3 PRA estimates the public risk from all analyzed risk contributors[4]. Level 3 PRAs can provide valuable insights into the relative importance of various risk contributors to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

Using information from the last set of NRC-sponsored Level 3 PRAs conducted as part of the NUREG-1150 study[5], the staff determined that the reactor-specific risk metrics CDF and LERF can be used respectively as surrogates for the latent cancer risk and prompt fatality risk quantitative health objectives (QHOs) defined in the Commission's Safety Goal Policy Statement[6]. Therefore, instead of using Level 3 PRAs, the staff compares the results from Level 1 and limited-scope Level 2 PRAs to subsidiary numerical objectives based on CDF and LERF for regulatory decisionmaking involving plant-specific applications. Although Level 3 PRAs have since been performed to some extent within both the United States and international nuclear industries, the NRC has not sponsored development of a Level 3 PRA since NUREG-1150.

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[3] 60 FR 42622, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (August 16, 1995).

[4] As used in this SECY paper and its enclosures, risk contributors include: radiological sources (e.g., reactor core, spent nuclear fuel); initiating event hazards (e.g. internal events, fires, flooding, seismic events, other site-specific external hazards); plant operating states; accident sequences; failure of structures, systems, and components; and operator actions.

[5] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990).

[6] 51 FR 30028, "Safety Goals for the Operations of Nuclear Power Plants" (August 21, 1986).

The staff has identified several compelling reasons for proceeding with a new and more comprehensive site Level 3 PRA that can be organized into two broad categories (1) technical advances since NUREG-1150 and (2) additional scope considerations.

Technical Advances Since NUREG-1150

In the more than two decades since the last NRC-sponsored Level 3 PRAs were conducted as part of the NUREG-1150 study, numerous technical advances have been made that were not reflected in the NUREG-1150 PRA models. These technical advances can be organized into three categories (1) modifications to enhance NPP operational performance, safety, and security; (2) significantly improved understanding and modeling of severe accident phenomena; and (3) advances in PRA technology. Given the substantial role the NUREG-1150 results and risk insights have played in shaping the development and implementation of the current risk-informed regulatory framework, the potential impact of these advances warrants further investigation.

Modifications to Enhance NPP Operational Performance, Safety, and Security

PRA models should strive to be as realistic as practicable, representing the as-designed, as-built, and as-operated plant. Over the past two decades, the increased use of PRA results and insights by both the nuclear industry and the NRC has helped to improve NPP safety and operational flexibility and performance. These improvements have been realized in terms of observed reductions in the frequencies of the following types of events typically modeled in NPP PRAs: component unreliability (e.g., a pump failing to start or failing to run), component or train unavailability resulting from test or maintenance outages, special events covering operational issues (e.g., pump restarts injection valve reopening during unplanned demands), and initiating events.[7] In addition to the implementation of multiple risk-informed regulations, there have also been a number of modifications to plant design, operating and emergency procedures, and training, inspection, and maintenance practices. Finally, following the accident at Three Mile Island (TMI) and the terrorist attacks on September 11, 2011, the nuclear industry developed and implemented severe accident management and extensive damage mitigation strategies, respectively, to enhance both the safety and security of NPPs.

Notable examples of risk-informed regulations and guidance that have resulted in modifications to enhance NPP operational performance, safety, and security include:

- 10 CFR 50.44, Combustible gas control for nuclear power reactors
- 10 CFR 50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled NPPs
- 10 CFR 50.63, Loss of all alternating current power (Station blackout rule)
- 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at NPPs (Maintenance rule)
- 10 CFR 50.54(hh)(2), Conditions of licenses – potential aircraft threat. These are also known as extensive damage mitigation guidelines (EDMGs) or B.5.b mitigation strategies[8].

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[7] NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants" (February 2007).

[8] EA-02-026, "Order for Interim Safeguards and Security Compensatory Measures" (February 25, 2002). Section B.5.b requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment, and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond-design-basis aircraft impacts. These requirements were formalized through rulemaking in 10 CFR 50.54(hh)(2).

- Severe Accident Management Guidelines (SAMGs)

Significantly Improved Understanding and Modeling of Severe Accident Phenomena
Insights gained from substantial research programs implemented after the TMI accident have significantly improved both our understanding of severe accident phenomena and the modeling of these phenomena using computer codes to support severe accident progression and containment response analyses as part of Level 2 and Level 3 PRAs.

Advances in PRA Technology
Similarly, insights gained from PRA-related research, advances in information and computer technology, and the acquisition of over 20 additional years of operating experience, have led to advances in PRA methods, models, tools, and data—collectively referred to as "PRA technology." Examples of important advances in PRA technology include improved modeling of severe accident phenomena; development of improved methods for common cause failure (CCF) analysis and human reliability analysis (HRA); improved analytical tools such as those used in the development and demonstration of state-of-the-art integrated modeling and analysis of severe accident progression and offsite radiological consequences in the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project; and improved quality and quantity of data for initiating events, component failures, and operator errors. These advances in our knowledge and PRA technology through research and acquired operating experience should result in improved methods, models, tools, and data when compared to the NUREG-1150s; this in turn should lead to an associated reduction in the epistemic or "state-of-knowledge" uncertainties that can significantly impact the interpretation and use of PRA results and risk insights.

<u>Updated Seismic Hazard Data</u>
Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," recently investigated the safety and risk implications of updated earthquake-related data and models. These data and models suggested that the probability of earthquake ground shaking above the seismic design basis for some NPPs in the Central and Eastern United States (CEUS) is still low, but larger than previous estimates.

Key messages from the GI-199 Safety/Risk Assessment (ML100270582) included:

(1) **Operating NPPs are safe.** Plants have adequate safety margin for seismic issues. The NRC's Safety/Risk Assessment confirms that overall seismic risk estimates remain small and that adequate protection is maintained.

(2) **Though still small, some seismic hazard estimates have increased.** Updates to seismic data and models indicate increased seismic hazard estimates for some operating NPP sites in the CEUS.

(3) **Assessment of GI-199 will continue.** Plants are safe, but the NRC has separate criteria for evaluating whether plant improvements may be imposed. The NRC's Safety/Risk Assessment used readily available information and found that for about one-quarter of the currently operating plants, the estimated CDF change is large enough to warrant further attention. Action may include obtaining additional, updated information and developing methods to determine if plant improvements to reduce seismic risk are warranted.

These insights gained from the GI-199 Safety/Risk Assessment suggest a need for further evaluating the relative contribution of the seismic hazard to the public risk from all analyzed risk contributors associated with NPPs.

Additional Scope Considerations

In addition to these technical advances since NUREG-1150, the staff has identified additional scope considerations not previously considered that could be addressed by performing a new and more comprehensive site Level 3 PRA. Examples of these additional scope considerations include (1) consideration of multi-unit[9] site effects and (2) consideration of other site radiological sources (e.g., spent fuel pools [SFPs], dry storage casks, and multiple units). Each of these areas is explored in more detail below. Before doing so, however, some of the important limitations of previous Level 3 PRAs and current PRAs used to support regulatory applications that could be addressed by expanding the scope to include additional considerations are reviewed.

Scope Limitations of the NUREG-1150 PRAs

The NUREG-1150 PRAs were limited in scope to the assessment of single-unit reactor accidents initiated primarily by internal events occurring during at-power operations, with only a partial treatment of fires and seismic events for two of the five analyzed plants (Surry and Peach Bottom). Although a later study evaluated the risk associated with accident sequences occurring during low-power/shutdown operations for two of the five analyzed plants (Grand Gulf[10] and Surry[11]), this study examined only one plant operating state (POS) in detail and was unable to provide an integrated perspective with insights into the relative importance of multiple POSs to risk. In addition, the NUREG-1150 PRAs did not include an assessment of accidents involving other site radiological sources such as spent fuel pools, dry storage casks, and other units on site (including additional reactor cores and spent fuel pools). These considerations are discussed in more detail below.

Current Use of Limited-Scope PRAs for Regulatory Applications

Although Regulatory Guide (RG) 1.174[12] states that the CDF and LERF acceptance guidelines are intended for comparison with the results of a full-scope PRA that includes all risk contributors, it does allow for the use of limited-scope PRAs. When a limited-scope PRA is used, the contribution of out-of-scope items to risk must be assessed based on the margin between the PRA results and the acceptance guidelines.

However, with qualitative analyses of varying degrees of rigor being submitted to support scope limitations, guidance is needed for the staff to assess the impact of these scope limitations on conclusions that are made. Among others, a study sponsored by the Advisory Committee on

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[9] As used in this document and the SECY paper to which it is enclosed, a unit refers to a reactor core and, if applicable, an associated spent fuel pool.

[10] NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1" (March 1995).

[11] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1" (July 1994).

[12] Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (May 2011).

Reactor Safeguards (ACRS) to assess the agency's need for improved PRA technology to risk inform its regulations[13] identified the following point as needing further investigation:

> "While there are valid technical arguments that can be made to justify the exclusion of some portions of a full-scope PRA model for risk-informed regulation, there are resources that must be continually applied by the licensee and the NRC to check the validity of the risk-informed decisions in light of the use of an incomplete PRA model. At some point, it is reasonable to ask whether these additional resources are small or large in relation to the use of a full-scope PRA to start with."

Consideration of Multi-Unit Site Effects

Because the Commission's safety goals, QHOs, and subsidiary numerical objectives are applied on a per reactor basis, most PRAs developed to date do not explicitly consider multi-unit accidents in which initiating events lead to reactor core damage in multiple units at the same site. Current PRA models therefore do not generally identify and address dependencies between systems at multi-unit sites, particularly those with highly interdependent support systems involving systems and subsystems that are shared by multiple units.

To understand the contribution of these multi-unit effects to the risk associated with a NPP, PRA models need to be enhanced to include both initiating events that might simultaneously impact multiple units and equipment and human action dependencies in responding to multi-unit accidents.

Consideration of Other Site Radiological Sources

To be complete, estimation of total site accident risk should also include an assessment of the risk from accidents involving other site radiological sources, to include spent nuclear fuel.

In summary, the NRC has never sponsored a site Level 3 PRA that includes an assessment of not only accidents involving the reactor core of a single unit, but also accidents involving other site radiological sources such as spent fuel pools, dry storage casks, and other units on site (including additional reactor cores and spent fuel pools). The incorporation of these technical advances and additional scope considerations into a new full-scope comprehensive site Level 3 PRA could yield new and improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

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[13] NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology In Risk-Informed Decision Making" (April 2003).

COMMISSION TASKING

During the Annual Commission Briefing on Research Programs, Performance, and Future Plans on February 18, 2010, the staff proposed a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

In SRM M100218 (ML100780578) dated March 19, 2010, the Commission expressed conditional support for Level 3 PRA related activities and directed the staff to (1) continue internal coordination efforts and engage external stakeholders in formulating a plan and scope for future actions and (2) provide the Commission with various options for proceeding that include costs and perspectives on future uses for Level 3 PRAs.

This document and the notation vote SECY paper to which it is enclosed were developed in response to the SRM. The remainder of this document discusses potential future uses for Level 3 PRAs and three primary options for proceeding with future Level 3 PRA activities that were developed by the staff through multiple interactions with stakeholders.

APPROACH TO NEW LEVEL 3 PRA ACTIVITIES

In response to SRM M100218 and to optimize cost-benefit considerations by focusing NRC resources, the staff developed a three-phased approach to planning and conducting future Level 3 PRA activities.

The first phase consisted of a scoping study that began in April 2010 and ended upon submission of this SECY paper to the Commission. This scoping study was conducted by staff from the Office of Nuclear Regulatory Research (RES) with support from representatives from the following NRC program offices: Office of Nuclear Material Safety and Safeguards (NMSS), Office of New Reactors (NRO), Office of Nuclear Reactor Regulation (NRR), and Office of Nuclear Security and Incident Response (NSIR). The objectives and activities associated with this scoping study are discussed in more detail below.

The second phase would consist of proceeding with either one of the options developed by the staff as part of the scoping study or any other option directed by the Commission following submission of the notation vote SECY paper.

Based on the results and insights from the second phase, the staff would then assess the need for follow-on Level 3 PRA activities and then provide the Commission with additional options and recommendations for proceeding.

LEVEL 3 PRA SCOPING STUDY

Objectives
Based on Commission tasking in SRM M100218, the staff identified the following main objectives for the Level 3 PRA scoping study:

(1) To identify potential future uses for Level 3 PRAs;

(2) To develop various options for proceeding with future Level 3 PRA activities that include objectives, scope, PRA technology to be used, site selection considerations[14], and resource estimates;

(3) To determine the feasibility of proceeding with each of the developed options;

(4) To continue internal coordination efforts to identify the staff's recommendation for proceeding; and

(5) To engage external stakeholders to obtain their views on the staff's approach, potential future uses for Level 3 PRAs, options for proceeding with future Level 3 PRA activities, and recommendation for proceeding.

Internal Coordination Activities
Throughout the scoping study, the staff participated in numerous internal coordination activities to develop its approach, identify potential future uses for Level 3 PRAs, develop options for proceeding with future Level 3 PRA activities, and identify its recommendation for proceeding. These activities included workshops, coordination and alignment meetings, and internal stakeholder briefings.

Brainstorming Workshop
The scoping study began with a brainstorming workshop on April 28, 2010 that was attended by RES staff and managers, as well as select staff from NMSS, NRO, NRR, and NSIR. The primary objectives of this workshop were to (1) provide participants with the background and vision for new Level 3 PRA activities, (2) identify scoping issues associated with various technical elements of new Level 3 PRAs, (3) identify potential uses for future Level 3 PRAs, and (4) identify technical working groups for the scoping study and next steps for moving forward.

As a result of the workshop, the following six technical working groups comprised of staff from RES, NMSS, NRO, NRR, and NSIR were established to accomplish the scoping study objectives for specific Level 3 PRA technical elements that were viewed as particularly complex and challenging (1) Level 1 PRA and Interface to Level 2 PRA, (2) Level 2 PRA and Interface to Level 3 PRA, (3) Other (than internal events) Hazard Groups PRA, (4) Spent Fuel and Non-Reactor PRA, (5) Human Reliability Analysis, and (6) 21st Century PRA Documentation.

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[14] Because the Commission expressed only conditional support and directed the staff to provide various options for proceeding, the staff did not include selection of a site to participate in a future Level 3 PRA as one of the objectives of the scoping study. Instead, the staff identified various site selection considerations related to the quality and availability of relevant information that could impact the cost of a future Level 3 PRA. These considerations, which are provided later in this enclosure, can inform future site selection activities if the Commission directs the staff to proceed with a Level 3 PRA.

An Integration and Oversight working group was also established to oversee technical working group activities and to integrate the options developed by each technical working group to develop high-level options and a specific recommendation for proceeding. This group created a working group charter to provide the technical working groups with objectives, deliverables, working group roles and responsibilities, and guidance for developing various scoping options.

Alignment Workshop
On July 27, 2010, working group leaders and select staff and managers from RES, NMSS, NRO, and NRR participated in an alignment workshop to (1) obtain an overview of the scoping options being considered and developed by the technical working groups; (2) identify ongoing or planned research that supports and has cost implications for these scoping options; (3) identify and discuss working group interface issues; (4) ensure initial alignment on key messages among working group leaders, senior managers, and other representatives from offices who will be involved in the Commission paper concurrence process; and (5) consider site selection issues and options for engaging external stakeholders.

Coordination and Alignment Meetings
Throughout the scoping study, the staff held numerous coordination meetings with senior managers and technical staff in RES, NMSS, NRO, NRR, and NSIR to further develop and refine the various options for proceeding with future Level 3 PRA activities, including costs and potential future uses for Level 3 PRAs, and to identify the staff's recommendation for proceeding.

In accordance with guidance provided by the Office of the Executive Director for Operations (OEDO) on the process for developing SECY papers[15], the staff also participated in multiple alignment meetings to ensure senior management and technical staff were in agreement on expectations, scope, and key messages to be communicated in the notation vote SECY paper to which this document is attached.

Internal Stakeholder Briefings
Throughout the scoping study, the staff provided multiple briefings to various internal stakeholders. The purposes of these briefings varied depending on the audience, but in general included stimulating interest in the initiative, sharing information about the staff's current thinking, answering stakeholder questions, and seeking stakeholder feedback. Example briefings include NMSS/NSIR staff briefing on April 12, 2010; Senior Reactor Analyst (SRA) monthly call on August 16, 2011; SRA Counterpart Meeting on May 26, 2011; and briefing of the Chairman's task force for developing options for a more holistic risk-informed, performance-based regulatory approach[16] on May 31, 2011.

External Stakeholder Engagement Activities
In addition to internal coordination activities, the staff participated in numerous external stakeholder engagement activities during the scoping study. These activities included: ACRS interactions, Regulatory Information Conference (RIC) presentations, and a Category 2 public meeting with representatives from nuclear industry, vendor, research, interest group, and public media organizations.

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[15] OEDO Notice 2010-0380-01, "SECY Paper Development Process" (March 17, 2010).

[16] Memorandum from G.B. Jaczko to R.W. Borchardt, "Assessment of Options for More Holistic Risk-Informed, Performance-Based Regulatory Approach" (February 11, 2011).

ACRS Interactions
The staff interacted with the ACRS on three separate occasions. The first briefing occurred during a November 17, 2010 meeting of the ACRS Subcommittee on Reliability and PRA in which the staff presented its approach to planning for future Level 3 PRA activities. The second briefing occurred during a May 11, 2011 meeting of the ACRS Subcommittee on Reliability and PRA in which the staff presented its identified potential uses for future Level 3 PRAs, developed options for proceeding with future Level 3 PRA activities, and its recommendation for proceeding. The third and final briefing occurred during a June 8, 2011 meeting of the ACRS Full Committee in which the staff once again presented its identified potential uses for future Level 3 PRAs, developed options for proceeding with future Level 3 PRA activities, and its recommendation for proceeding. The goal of this final briefing was to obtain ACRS support for the staff's recommendation.

Following the briefing of the Full Committee, the ACRS recommended the staff (1) develop a phased approach and schedule that would enable the staff to complete a new full-scope comprehensive site Level 3 PRA while minimizing the near-term resource impact; (2) take maximum advantage of existing PRA technology; and (3) actively engage the participation of industry. These recommendations are consistent with the staff's plans for proceeding, if the Commission directs the staff to proceed with such a study.[17]

RIC Presentations
The NRC took advantage of unique opportunities to engage with both internal and external stakeholders at the 2010 and 2011 RIC. At the 2010 RIC, the staff introduced stakeholders to this new Level 3 PRA initiative by presenting the envisioned approach, objectives, and scope in the "Current Topics in Probabilistic Risk Analysis" technical session. At the 2011 RIC, the staff provided stakeholders with updated information and encouraged their engagement and participation in the subsequent ACRS interactions and the Category 2 public meeting discussed below by developing a poster presentation and by presenting in the "Current Topics in Probabilistic Risk Analysis" technical session. During and after the 2011 RIC, the staff received a number of stakeholder questions related to this new Level 3 PRA initiative. The staff developed responses to each of these questions and posted them to the external RIC website for stakeholders to review.

Category 2 Public Meeting
A Category 2 public meeting was held on April 11, 2011 at NRC Headquarters to obtain stakeholder views on options for proceeding with future Level 3 PRA activities. Key external stakeholders who were specifically invited to attend and who participated in the public meeting included representatives from the following organizations: the Nuclear Energy Institute (NEI), the Electric Power Research Institute, Inc. (EPRI), and the Union of Concerned Scientists (UCS). Representatives from other nuclear industry, vendor, research, interest group, and public media organizations also participated.

In general, meeting participants supported the options developed by the staff for proceeding with future Level 3 PRA activities. In particular, meeting participants supported a recommendation to proceed with a new full-scope comprehensive site Level 3 PRA, but expressed some concern about the potential scope, cost, and schedule of such a study. Some participants offered comments related to specific aspects of technical elements of a Level 3 PRA that the NRC should consider if a new full-scope comprehensive site Level 3 PRA is

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[17] Letter from Said Abdel-Khalik to The Honorable Gregory B. Jaczko, "Draft SECY Paper, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities"" (June 22, 2011).

planned. A public meeting summary, including a list of meeting participants and a copy of the meeting presentation slides handout is publicly available in the Agencywide Documents Access and Management System (ADAMS) at ML111400179.

POTENTIAL FUTURE USES FOR LEVEL 3 PRAs

In identifying potential future uses for Level 3 PRAs, a logical first step was to identify how the results and risk insights from the NUREG-1150 PRAs were used. In addition, the staff considered potential enhancements that could be made to the use of PRA in the existing risk-informed regulatory framework. In this way, the staff developed the following set of seven potential uses. These potential uses are meant to apply to future Level 3 PRAs in general, and not specifically to the full-scope comprehensive site Level 3 PRA proposed as Option 3; the use of this specific PRA would ultimately depend on its scope and applicability to the larger population of NPP sites.

Confirm the Acceptability of the Agency's Current Use of PRA in Risk-informed Regulatory Decisionmaking

Future Level 3 PRAs could be used to assess the agency's current use of PRA in risk-informed regulatory decisionmaking. Examples include the use of Level 1 and limited-scope Level 2 PRAs to support regulatory applications, and the use of RG 1.174 subsidiary numerical objectives based on the reactor-specific risk metrics CDF and LERF that were originally developed and validated using NUREG-1150 information.

Verify or Revise Regulatory Requirements and Guidance

Future Level 3 PRAs could be used to either verify or revise regulatory requirements and guidance, particularly those based on the last NRC-sponsored Level 3 PRAs that were conducted as part of the NUREG-1150 study. In addition to the previously discussed RG 1.174 acceptance guidelines based on CDF and LERF that are used by the staff in regulatory decisionmaking involving plant-specific applications, this would include the regulatory analysis guidelines[18] and technical evaluation handbook[19] used by the staff to evaluate proposed backfits[20] to determine, among other things, whether the benefits associated with a proposed regulatory action are commensurate with the cost. The NRC performs regulatory analyses to support numerous NRC actions that affect its reactor licensees. The regulatory analysis guidelines and handbook contain a number of policy decisions that have broad implications for the NRC and its licensees, including the use of safety goal evaluations, a $2000 per person-rem conversion factor, and criteria for the treatment of individual requirements.

Support Specific Risk-Informed Regulatory Applications

Future Level 3 PRAs could be used to provide support for a variety of specific risk-informed regulatory applications. Examples include providing the technical basis for risk-informing the regulation of spent fuel storage and handling, siting, and emergency preparedness; and focusing the Reactor Oversight Process (ROP), including the NRC's inspection program.

Develop and Pilot Test PRA Technology, Standards, and Guidance

Future Level 3 PRAs could be used to develop and pilot test new PRA technology (e.g., methods, models, and tools) developed to obtain new and improved risk insights; consensus PRA standards; and regulatory guidance to ensure requirements are clear, understandable, and achieve consistency.

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[18] NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (September 2004).

[19] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook" (January 1997).

[20] Title 10 Section 50.109, of the *Code of Federal Regulations* (10 CFR 50.109). Backfitting.

Prioritize Generic Issues and Nuclear Safety Research Programs

Future Level 3 PRAs could also be used to inform the prioritization and resolution of GIs and the prioritization of nuclear safety research programs by focusing limited agency resources on issues most directly related to the agency's mission to protect public health and safety. This was one of the identified uses of the NUREG-1150 PRA results and risk insights.

Develop In-House PRA Technical Capability and Support PRA Knowledge Management and Risk Communication Activities

Future Level 3 PRAs could be used to support a variety of PRA staffing, knowledge management, and risk communication activities. Development of future Level 3 PRAs within the NRC could help develop first-hand knowledge about PRA and the technical skills needed for performing and reviewing PRAs. To support knowledge management activities, they could provide the technical basis for updating training materials for PRA developers, reviewers, and users. In addition, by using modern information technology to document the relevant assumptions, decisions, methods, models, tools, and data, future Level 3 PRAs can provide readily accessible information to support potential future needs. In addition to improving internal risk communication by improving PRA training and making PRA information more accessible, future Level 3 PRAs with improved documentation can be used to enhance external risk communication by facilitating external stakeholder understanding of not only the relative importance of various risk contributors to public risk, but also the underlying assumptions and limitations affecting the results and risk insights.

Support Future Risk-Informed Licensing of New and Advanced Reactor Designs

Future Level 3 PRAs could be used to support the future risk-informed licensing of new and advanced reactor designs. First, in its Policy Statement on the Regulation of Advanced Reactors[21], the Commission stated its intention to "improve the licensing environment for advanced nuclear power reactors to minimize complexity and uncertainty in the regulatory process." The staff noted in its Advanced Reactor Research Plan (ML082530184) that a risk-informed regulatory structure applied to license and regulate advanced reactors, regardless of their technology, could enhance the effectiveness, efficiency, and predictability of future plant licensing. In NUREG-1860[22], the staff documented the results of a study that was conducted to establish the feasibility of developing a risk-informed and performance-based regulatory framework for the licensing of future NPPs that could be used to develop a set of regulatory requirements that would serve as an alternative to 10 CFR 50. This framework was envisioned to have the following potential advantages:

(1) It would require a broader use of design-specific risk information in establishing the licensing basis, thus better focusing on those items most important to safety for that design;

(2) It would stress the use of performance as the metric for acceptability; and

(3) It could be written to be applicable to any reactor technology ("technology neutral'), thus avoiding the time consuming and less predictable process of reviewing non-light water

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[21] 73 FR 60612, "Policy Statement on the Regulation of Advanced Reactors" (October 14, 2008).

[22] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing" (December 2007).

reactor (LWR) designs against the LWR-oriented regulations in 10 CFR 50 and making decisions on a case-by-case basis.

In this concept of a “technology neutral framework,” a design-specific, full-scope Level 3 PRA would be used to identify licensing basis events (LBEs) by comparing the frequencies and consequences of all possible event scenarios with a frequency-consequence (F-C) curve established by various site boundary radiation dose limits. The LBEs, whose purpose is principally similar to that of the design basis accidents in the current regulatory framework, are selected from those PRA event sequences whose frequencies and consequences approach the F-C curve. This process is further used to inform defense-in-depth (including safety margin) requirements and the safety categorization of structures, systems, and components (SSCs).

In the SRM to SECY-07-0101[23], the Commission stated that the staff should publish the technology-neutral framework and its concepts should be tested on an actual design. Although the Commission indicated in this SRM that the Pebble Bed Modular Reactor (PBMR) design review would be a logical choice, the testing of the framework has not yet occurred. Future Level 3 PRAs could be used for this purpose.

In addition to pilot testing the “technology neutral framework,” future Level 3 PRAs could be used to inform the staff’s follow-on activities related to (1) resolving issues with small modular reactor (SMR) designs[24], (2) using risk insights to enhance the safety focus of SMR reviews[25], and (3) modifying risk-informed regulatory guidance for new reactors[26]. Although future Level 3 PRAs would not be developed in time to inform the staff’s current activities related to these efforts, they could be used to inform related follow-on activities.

[23] SECY-07-0101, “Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50 (RIN 3150-AH81)” (June 14, 2007).

[24] SECY-10-0034, “Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs” (March 28, 2010).

[25] SECY-11-0024, “Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews” (February 18, 2011).

[26] SECY-10-0121, “Modifying the Risk-Informed Regulatory Guidance for New Reactors” (September 14, 2010).

OPTIONS FOR PROCEEDING WITH FUTURE LEVEL 3 PRA ACTIVITIES
This section presents detailed descriptions of the three primary options deemed by the staff to best frame the choices from a feasibility and cost-benefit perspective and their relative advantages and disadvantages.

Examples of other options that were considered by the staff through participation in the previously discussed scoping study activities are provided in the notation vote SECY paper to which this document is enclosed.

<u>Option 1: Maintain Status Quo – Continue Evolutionary Development of PRA Technology</u>
This option maintains the status quo in ongoing activities related to the development and implementation of PRA technology and risk-informed regulation. Ongoing and planned research to develop and improve upon existing PRA methods, models, tools, and data would continue on a resource-available basis as driven by program office user need requests (UNRs) and the agency's long term research plan (LTRP). The staff also would continue to monitor relevant developments within both the United States and international nuclear industries.

As part of its strategic LTRP efforts, the staff has identified Level 2 and Level 3 PRA as areas that would benefit from examination of advanced methods, and is performing limited research in these areas. For example, under this program, a scoping study to evaluate both methodological and implementation-oriented issues associated with the advancement of Level 2 and Level 3 PRA modeling techniques was recently completed. This study resulted in the development of a spectrum of modeling approaches, which included: modified traditional approaches, hybrid event tree approaches, dynamic event tree approaches, and sampling-based simulation approaches[27]. As a result, a new phase of work that focuses on the dynamic event tree approach using the MELCOR Severe Accident Analysis Code in conjunction with a dynamic operator response model has begun. The initial methods development, including application of the approach to a demonstration problem, is scheduled to be completed by the end of calendar year 2011.

Advantages
- Is consistent with the current fiscal climate by focusing limited staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.

Disadvantages
- Insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking would not be realized.

- Can result in inconsistent and more costly treatment of potential future issues by developing the necessary PRA technology on an ad-hoc basis.

[27] Helton, D. "Scoping Study on Advancing Modeling Techniques for Level 2/3 PRA" (May 2009). Available in ADAMS at ML091320454.

<u>Option 2: Conduct Focused Research to Address Identified Gaps in Existing PRA Technology Before Performing a Full-Scope Comprehensive Site Level 3 PRA</u>
This option involves near-term focused research aimed at addressing identified gaps in existing PRA technology over the next 2 years. These technical gaps are related to the expanded scope and the differing degrees of sophistication in the existing PRA technology used to analyze the risk from various risk contributors.

This option was developed with the understanding that reallocating resources to develop a new full-scope comprehensive site Level 3 PRA would be particularly challenging because of mission-critical work already assigned to a limited number of qualified risk analysts. Moreover, a decision on whether to proceed with such a study would benefit from better understanding the recommendations and Commission tasking from multiple task forces (e.g., the Chairman's task force to develop options for a more holistic risk-informed, performance-based regulatory approach and the near-term task force to conduct methodical and systematic reviews of our current processes and regulations in response to the recent events in Japan[28]).

Objective
The primary objective of this research would be to ensure that important technical gaps related to the expanded scope and the differing degrees of sophistication in the existing PRA technology used to analyze the risk from various risk contributors are addressed before developing a new full-scope comprehensive site Level 3 PRA.

[28] Tasking Memorandum – COMGBJ-11-0002, "NRC Actions Following the Events in Japan" (March 23, 2011).

Advantages

- Focuses limited available staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.

- Focuses additional staff and contract support resources that have already been requested to support future Level 3 PRA activities on research needed to address identified gaps in existing PRA technology.

- Produces results and insights that would advance the state-of-practice in specific PRA technical elements and thereby enhance NRC's PRA capability in those technical areas.

Disadvantages

- Delays insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking.

Scope

Examples of gaps in existing PRA technology that need to be addressed either prior to or in parallel with conducting future Level 3 PRAs include:

Modeling of Consequential (Linked) Multiple Initiating Events

Current PRA models do not include scenarios in which multiple, linked initiating events occur either simultaneously or close in time with respect to overall mission time such that a second initiating event occurs while the plant is still responding to the first. Methods for incorporating these types of scenarios into current PRA models need to be investigated.

Modeling of Multi-Unit Dependencies

Because the Commission's safety goals, QHOs, and subsidiary numerical objectives are applied on a per reactor basis, most PRAs developed to date do not explicitly consider multi-unit accidents in which initiating events lead to reactor core damage in multiple units at the same site. Current PRA models therefore do not appropriately identify and address dependencies between systems at multi-unit sites, particularly those with highly convoluted support system dependencies involving systems and subsystems that are shared by multiple units.

To understand the contribution of these multi-unit effects to the risk associated with a NPP, PRA models need to be enhanced to include both initiating events that might simultaneously impact multiple units and equipment and human action dependencies in responding to multi-unit accidents.

Post-Core Damage and External Events HRA Modeling

Severe Accident Management Guidelines (SAMGs)

In response to the TMI accident, the nuclear industry developed and implemented SAMGs to provide tools and strategies for managing the in-plant aspects and mitigating the results of a severe accident. The overall goal of SAMGs is to terminate emergency conditions by (1) returning the reactor core to controlled and stable state, (2) maintaining or returning the containment to a controlled and stable state, and (3) terminating any fission product releases.

The following three groups of individuals use SAMGs in the event of a severe accident leading to core damage:

(1) **Evaluators.** Evaluators are members of the plant evaluation team that are responsible for diagnosing plant conditions, evaluating the impacts of potential strategies, and assessing the effectiveness of implemented strategies.

(2) **Implementors.** Implementors are typically plant operators that are responsible for monitoring plant indications, operating equipment, and communicating with evaluators and decision makers.

(3) **Decision Makers.** Decision makers are typically plant managers or technical directors who are responsible for analyzing information and recommendations provided by both implementors and evaluators and for deciding which strategies to implement.

From a post-core damage HRA modeling perspective, the use of SAMGs presents a unique challenge. In Rasmussen's Cognitive Taxonomy, the emergency operating procedures (EOPs) used by operators to prevent core damage are "rule-based," and therefore allow for identification of the best course of action for any set of conditions by simply following procedures, However, SAMGs are “knowledge-based,” and therefore require evaluators to use their knowledge and problem solving skills to identify the least-bad course of action in unfamiliar severe accident conditions. In addition, almost all of the SAMG strategies to mitigate the effects of one problem result in adverse effects on another problem. Evaluators must therefore make risk-benefit decisions when considering different strategies. Since the most appropriate response to a given condition cannot be determined in advance, the definition of what constitutes a failure and the identification of post-core damage human failure events or recovery actions that can be credited in the PRA model presents a unique challenge that needs to be addressed if a site Level 3 PRA model is to be developed.

Extensive Damage Mitigation Guidelines (EDMGs)
Following the terrorist attacks on September 11, 2011, the NRC required licensees to implement EDMGs described in Title 10, Section 50.54(hh). Much like SAMGs, the definition of what constitutes a failure and the identification of human failure events or recovery actions that can be credited in the PRA model presents a unique challenge that needs to be addressed if a site Level 3 PRA model is to be developed.

External Events HRA
In addition to addressing challenges associated with the modeling of SAMGs and EDMGs, research into the modeling of human actions in response to various external events (e.g., seismic events, external flooding) is needed.

<u>Spent Fuel PRA Technology</u>
Process areas not related to reactor core operations that can contribute to nuclear site accident risk include those associated with onsite nuclear spent fuel handling and storage. Although limited PRA models for quantitatively evaluating the risk of accidents involving spent fuel pools and dry cask storage exist, additional or significantly improved PRA technology must be developed to enable a meaningful comparison and relative ranking of these process area risk contributors as part of a comprehensive site Level 3 PRA. Example spent fuel PRA areas for improvement include: success criteria determination, HRA, accident phenomena, and source term analysis.

<u>Modeling of Aqueous Transport and Dispersion of Radioactive Materials</u>
As demonstrated by the recent events in Japan, certain accident scenarios can result in large volumes of contaminated water being generated by emergency measures to cool the reactor cores and SFPs, with yet to be determined offsite radiological consequences. To determine the relative risk significance of these types of scenarios, a Level 3 PRA must be capable of modeling and analyzing the aqueous transport and dispersion of radioactive materials through surface water, sediments, soils, and groundwater. Existing PRA analytical tools do not have this capability. Research is therefore needed to identify or develop methods, models, and tools that can be used to simulate geochemical speciation and transport of dissolved radionuclides in surface water, sediments, soils, and groundwater.

<u>Level 2 and Level 3 PRA Uncertainty Analysis</u>
Although guidance on the process for identifying and characterizing key sources of uncertainty exists[29], research is needed to identify the key sources of uncertainty in Level 2 and Level 3 PRA analyses and to develop specific methods for propagating uncertainty through the Level 2 and Level 3 PRA analyses.

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[29] NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making” (March 2009).

<u>Option 3: Conduct a Full-Scope Comprehensive Site Level 3 PRA</u>
This option involves planning for and performing a new full-scope comprehensive site Level 3 PRA for an operating NPP. Research identified in Option 2 also would be conducted as part of this option, but on an accelerated schedule to support the completion of a full-scope comprehensive site Level 3 PRA within 3 years. Option 2 and Option 3 differ only in terms of timing, sequencing, near-term use of resources, and relative advantages and disadvantages. In addition, selection of Option 2 would require separate Commission direction in the future before proceeding with a new full-scope comprehensive site Level 3 PRA.

Objectives
The staff has identified the following four high-level objectives for a new full-scope comprehensive site Level 3 PRA for an operating NPP:

(1) Extract new and improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety by:

   a. expanding the PRA scope to include an assessment of the risk from accidents involving spent fuel and multiple units,

   b. incorporating technical advances since NUREG-1150, and

   c. using a more integrated and consistent analysis approach to enable a meaningful comparison and relative ranking of all analyzed site risk contributors

(2) Enhance PRA capability, expertise, and documentation by improving upon existing analytical tools, by:

   a. providing training opportunities for staff and contractors, and

   b. using improved documentation practices and current information technology to make PRA information more accessible, retrievable, and understandable.

(3) Demonstrate the technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, where appropriate, rather than developing entirely new models.

Advantages

- Provides new and improved risk insights earlier to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

- Enhances PRA capability, expertise, and documentation earlier to address potential future issues.

Disadvantages

- Is resource-intensive, requiring more staff and contract support resources than currently budgeted.

- Requires reallocation of qualified risk analysts from other ongoing important activities, potentially resulting in delays to reviews of National Fire Protection Association (NFPA) Standard 805[30] license amendments, refinement of Standardized Plant Analysis Risk (SPAR) models, and reviews of PRAs in support of combined operating license applications. A more detailed resource discussion, including the potential implications of selecting Option 3, is provided in the notation vote SECY paper to which this document is enclosed.

[30] NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

PRA Scope

The scope of this PRA would include (1) site radiological sources—all reactor cores, spent fuel pools, and dry storage casks on site; (2) initiating event hazards—internal events, fires, flooding, seismic events, and other site-specific external hazards; (3) POSs—at-power and low-power/shutdown. The only factors specifically excluded from the scope would be radiological sources involving fresh nuclear fuel and radiological waste, and initiating events involving deliberate malevolent acts (e.g., terrorism and sabotage). The table below illustrates the options included for each factor considered in the analysis.

Factor	Options Included in Full-Scope Comprehensive Site Level 3 PRA
Radiological hazards	Reactor core(s) Spent nuclear fuel (spent fuel pools and dry storage casks)
Population exposed to hazards	Offsite population
Initiating event hazard groups	Internal hazards • Internal events (transients, loss-of-coolant accidents) • Internal floods • Internal fires
	External hazards • Seismic events (earthquakes) • Other site-specific external hazards (e.g., high winds, external flooding)
Plant operating states	At-Power Low-Power/Shutdown
End state/Risk metrics	Level 1 PRA: Core damage frequency* Level 2 PRA: Large early release frequency* Level 3 PRA: Number of early fatalities Number of early injuries Number of latent cancer fatalities Population dose (person-rem) at various locations Individual early fatality risk defined in QHO Individual latent cancer fatality risk defined in QHO Economic costs of mitigation actions**

* Although the Level 3 PRA will be used to estimate the public risk in terms of a variety of consequence measures, it is envisioned that the CDF and LERF risk metrics will be computed in intermediate steps to obtain near-term benefit in support of existing risk-informed regulatory applications.

** Although the staff previously considered the possibility of developing additional safety goals based on the risk of land contamination and overall societal impact, based on significant weaknesses in the analytical tools at the time, the staff recommended not pursuing this effort[31]. If the staff were to perform a new full-scope comprehensive site Level 3 PRA, it would plan on estimating economic risk associated with mitigation actions such as land interdiction, condemnation, and decontamination. These calculations could easily be performed by existing analytical tools described in more detail below at relatively little, if any, additional cost. By doing so, the staff would not be proposing to use this information for regulatory decisionmaking; instead, it would be used as an additional source of risk insights.

[31] SECY-00-0077, “Modifications to the Reactor Safety Goal Policy Statement” (March 30, 2000).

PRA Technology
Consistent with the above objectives to enhance PRA capability and to demonstrate the technical feasibility and evaluate the cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, the staff envisions using the following existing PRA technology as part of a new full-scope comprehensive site Level 3 PRA:

<u>SPAR Models</u>
The staff uses SPAR models in support of risk-informed activities related to the inspection program, incident investigation program, license amendment reviews, performance indicator verification, accident sequence precursor (ASP) program, GIs, and special studies. These tools also support and provide rigorous and peer-reviewed evaluations of operating experience, thereby demonstrating the agency's ability to analyze operating experience independently of licensees' risk assessments and enhancing the technical credibility of the agency.

The SPAR models integrate systems analysis, accident scenarios, component failure likelihoods, and HRA into a coherent model that reflects the design and operation of the plant. The SPAR model gives risk analysts the capability to quantify the expected risk of a NPP in terms of CDF and the change in that risk given an event, an anomalous condition, or a change in the design of the plant. More importantly, the model provides the analyst with the ability to identify and understand the attributes that significantly contribute to the risk and insights into how to manage that risk.

Currently, 78 SPAR models representing the 104 operating U.S. commercial NPPs are used for analysis of reactor core damage risk (Level 1 PRA) from internal events at-power. The Level 1 SPAR model includes an assessment of reactor core damage risk resulting from general transients (including anticipated transients without scram), transients induced by loss of a vital alternating current or direct current bus, transients induced by a loss of cooling (service) water, loss-of-coolant accidents, and loss of offsite power (LOOP). The SPAR models use a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance, although these input data may be modified to be more plant- and event-specific, when needed. The system fault trees contained in the SPAR models are generally not as detailed as those contained in licensees' PRA models.

To more accurately model plant operation and configuration and to identify the significant differences between the licensee's PRA and SPAR logic, the staff performed detailed cut-set level reviews on all 78 models. In addition to the internal event at-power models, the staff developed 15 integrated external event models based on the licensee responses to the Individual Plant Examination of External Events (IPEEE) Program[32]; seven integrated low-power/shutdown models; and three extended Level 1 models supporting LERF and Level 2 modeling. The external event models were recently used to identify and evaluate severe accident sequences for the Consequential Steam Generator Tube Rupture Project in support of the NRC's Steam Generator Action Plan (ML003770259).

One significant upcoming activity is the incorporation into the SPAR models of internal fire

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[32] Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (November 23, 1988).

Supplement 4 to Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (June 28, 1991).

scenarios from the NFPA 805 pilot applications. In addition, the staff continues to provide technical support for SPAR model users and risk-informed programs. The staff also completes about a dozen routine SPAR model updates annually.

The NRC implemented a formal SPAR model quality assurance plan in September 2006. Limited-scope validation and verification is accomplished by comparisons to licensee PRA models (as available) and to NRC NUREGs and analyses. Limited-scope peer reviews consist of internal quality assurance reviews by NRC contractors, NRC PRA staff, and regional SRAs (as available). Improvements to the models on a continuing basis result from staff user feedback, peer reviews from licensees, and insights gained from special studies, such as identification of threshold values during Mitigating Systems Performance Index (MSPI) reviews. In 2007, the NRC began a cooperative effort with the Electric Power Research Institute (EPRI) to improve PRA quality and address several key technical issues common to both the SPAR models and industry models. This cooperation resulted in the joint publication of a report that documents current methods to identify and quantify support system initiating events using PRAs[33]. Other cooperative projects include improvements to LOOP modeling and emergency core cooling system performance following boiling-water reactor (BWR) containment failure. In addition, the staff, with the cooperation of industry experts, performed a peer review of a representative BWR SPAR model and pressurized-water reactor (PWR) SPAR model in accordance with the industry consensus PRA standard for internal events, at-power Level 1/LERF PRAs[34] and RG 1.200[35]. The staff reviewed the peer review comments and initiated projects to address these comments where appropriate. The staff is also reevaluating certain success criteria in the SPAR models using state-of-the-art thermal-hydraulic modeling tools such as the MELCOR severe accident analysis code, which is described in more detail below.

<u>Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 8</u>

SAPHIRE 8 is a software application developed by the NRC for performing PRAs. SAPHIRE can be used to model a plant's response to initiating events, to quantify associated CDFs, and to identify important contributors to core damage (Level 1 PRA). It can also be used to evaluate containment failure and release models for severe accident conditions, given that core damage has occurred (Level 2 PRA). It can also be used in a limited manner to quantify risk in terms of release consequences to the public and environment (Level 3 PRA). It can be used for a PRA assuming that the reactor is operating at-power or in a low-power/shutdown POS. Furthermore, it can be used to analyze both internal and external initiating events, and it has special features for transforming models built for internal event analysis to models for external event analysis.

SAPHIRE 8 contains improved editors or options for creating event trees and fault trees, defining accident sequences and basic event failure data, solving system fault trees and accident sequence event trees, quantifying cut sets, performing sensitivity and uncertainty analyses, documenting the results, and generating reports.

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[33] EPRI Report 1016741, "Support System Initiating Events: Identification and Quantification Guideline" (2008).

[34] American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (February 2, 2009).

[35] Regulatory Guide 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (March 2009).

SAPHIRE 8 is designed to easily handle larger and more complex models than previous versions. Applications of previous versions indicated the need to build and solve models with a large number of sequences. In addition, the complexity of the models has increased since PRAs evaluate both potential internal and external event initiators, as well as different POSs in which the initiating event may occur. Special features have been designed into SAPHIRE 8 to help create and run integrated models that may be composed of a number of different model types (e.g., models with different types of initiating events or POSs). External events models can be built more expeditiously through the use of automation tools. Any combination of model types can be solved, and a powerful graphical editor allows examination of the underlying logic.

New modeling and calculation methods have also been implemented. For example, phase mission time analysis capability was incorporated in support of the NRC's "extended Level 1" and limited-scope Level 2/LERF SPAR models; however, it may also be useful for low-power/shutdown models, which may consider multiple POSs. For CCF modeling, the Risk Assessment Standardization Project method has been incorporated, with CCF probabilities now automatically adjusted to account for the impact of sequence flag sets. In addition, SAPHIRE 8 offers an improved sequence solving algorithm which addresses limitations in the previous solving algorithm related to application of sequence recovery rules.

The uncertainty analysis functions in SAPHIRE 8 estimate the variability (due to the uncertainties in the basic event probabilities) of a fault tree top event probability, an event tree sequence frequency, and end state frequency, or any of the importance measures. In an uncertainty analysis, SAPHIRE 8 samples the user-specified distributions for each basic event in a group of cut sets, and then quantifies these cut sets using the sampled values.

One of the strengths of SAPHIRE 8 lies in its computational capabilities, which can easily be leveraged by non-expert users via an improved graphical user interface. SAPHIRE 8 has become a powerful and easy to use PRA tool. Its relational database structure and editing rules offer the capability for sophisticated modeling of accident progression and, therefore, offer the means for a more accurate and efficient analysis. Several other features, many constructed from feedback by users dealing with large-scale PRA models, make SAPHIRE 8 among the fastest and most sophisticated PRA codes available today.

<u>MELCOR Severe Accident Analysis Code</u>
The MELCOR severe accident analysis code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of postulated accidents in both LWRs and in non-reactor systems such as SFPs and dry storage casks.

MELCOR is a modular code consisting of three general types of packages:

(1) basic physical phenomena (e.g., hydrodynamics, heat and mass transfer to structures, gas combustion, aerosol and vapor physics),

(2) reactor-specific phenomena (e.g., decay heat generation, core degradation and relocation, ex-vessel phenomena, engineering safety systems), and

(3) support functions (e.g., thermodynamics, equations of state, material properties, data-handling utilities, equation solvers).

These packages model the major systems of a NPP and their associated interactions, including:

- Thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and the confinement buildings,
- Core uncovering (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation,
- Heatup of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,
- Core-concrete attack and ensuing aerosol generation,
- In-vessel and ex-vessel hydrogen production, transport, and combustion,
- Fission product release (aerosol and vapor), transport, and deposition,
- Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling, and
- Impact of engineered safety features on thermal-hydraulic and radionuclide behavior

Initially, in the interest of quick code execution time and a general lack of understanding of reactor accident physics, the MELCOR code was envisioned as being predominantly parametric with respect to modeling complicated physical processes. However, over the years as phenomenological uncertainties have been reduced and user expectations and demands from MELCOR have increased, the models implemented into MELCOR have become increasingly best-estimate in nature. The increased speed and decreased cost of modern computers has eased many of the perceived constraints on MELCOR code development. Today, most MELCOR models are mechanistic, with capabilities approaching those of the most detailed codes of a few years ago. The use of models that are strictly parametric is limited, in general, to areas of high phenomenological uncertainty where there is no consensus concerning an acceptable mechanistic approach.

Current uses of MELCOR often include uncertainty analyses and sensitivity studies. To facilitate these uses, many of the mechanistic models have been coded with optional adjustable parameters. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient. Parameters of this type, as well as such numerical parameters as convergence criteria and iteration limits, are coded in MELCOR as sensitivity coefficients, which may be modified through optional code input.

<u>MELCOR Accident Consequence Code System, Version 2 (MACCS2)</u>
MACCS2 represents a major enhancement of its predecessor MACCS, which was developed in 1990 to evaluate the impacts of severe accidents at NPPs on the surrounding public as part of the NUREG-1150 study. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short- and long-term mitigation actions and exposure pathways, deterministic and stochastic health effects, and economic costs. No other U.S. code that is publicly available at present offers all these capabilities.

MACCS2 was developed as a general-purpose tool applicable to diverse reactor and non-reactor facilities licensed by the NRC or operated by the Department of Energy or the Department of Defense. The MACCS2 package includes three primary enhancements (1) a more flexible emergency-response model, (2) an expanded library of radionuclides, and (3) a semi-dynamic food-chain model. Other improvements are in the areas of phenomenological modeling and new output options.

MACCS2 requires a substantial amount of supporting site-specific information, including, for example: meteorology, demography, land use, and property values. In addition, MACCS2 requires analysts make assumptions about the values of several parameters related to the implementation of mitigation actions following a severe accident. Examples include time needed to warn the public and initiate emergency response actions, effective evacuation speed, fraction of the offsite population that effectively participates in the emergency response actions, the degree of radiation shielding afforded by buildings, and projected dose limits. Uncertainty analyses would therefore be needed to understand the impact of parameter assumptions on the results.

An important limitation of MACCS2 is that it does not currently model and analyze the aqueous transport and dispersion of radioactive materials through surface water, sediments, soils, and groundwater. As demonstrated by the recent events in Japan, certain accident scenarios can result in large volumes of contaminated water being generated by emergency measures to cool the reactor cores and SFPs, with yet to be determined offsite radiological consequences. To determine the relative risk significance of these types of scenarios, a Level 3 PRA must be capable of modeling and analyzing the aqueous transport and dispersion of radioactive materials. This has therefore been identified as an important technical gap to be addressed as part of Option 2.

Site Selection Considerations
Although the objective of the Level 3 PRA scoping study was not to select a specific site for participation in a new full-scope comprehensive site Level 3 PRA, the staff identified a number of site selection considerations that can influence both the quality and availability of relevant information, as well as the resources needed to complete the study. These site selection considerations are presented below. Since licensee willingness to cooperate would be critical to success, it is important to recognize that the staff would have to engage with industry to identify and select the appropriate licensee for participation in the proposed study, should the Commission direct the staff to proceed with Option 3 or any other option requiring licensee cooperation.

Multi-Unit
Development of a full-scope comprehensive site Level 3 PRA model that can be used to understand the relative contribution of multi-unit effects to risk obviates the need for a multi-unit site.

SPAR Model Capability
Consistent with the proposed study objectives to enhance PRA capability and to demonstrate the technical feasibility and evaluate the cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, the staff would use an existing SPAR model as the starting point for developing the proposed full-scope comprehensive Level 3 PRA model. Sites with SPAR models that have been enhanced to incorporate external initiating event hazards, low-power/shutdown POSs, and/or Level 2 PRA technical elements are therefore good candidates for participation in the proposed study.

The potential enhancements that will be needed to the site-specific SPAR model will be driven primarily by what is needed to ensure the primary objective of obtaining new and improved risk insights is met. Due consideration will be given to requirements specified in industry consensus PRA standards and RG 1.200 to ensure technical adequacy of PRA results for risk-informed applications.

NFPA 805 Transition
To obtain credible results and insights from a fire PRA, a complete electronic cable raceway database and circuit analyses are needed. Development of these elements is extremely resource intensive and therefore cost prohibitive for the NRC. Sites participating in the voluntary transition to NFPA 805 implementation that have developed a state-of-the-art fire PRA that includes both of these elements are therefore good candidates for participation in the proposed study.

MELCOR Input Decks
Detailed MELCOR input decks that can support success criteria and accident progression calculations are both costly and time-consuming to develop. Sites participating in either the SOARCA project or in other ongoing research to investigate success criteria associated with specific Level 1 PRA sequences would already have detailed MELCOR input decks, and are therefore good candidates for participation in the proposed study.

Site-Specific External Hazards
The external initiating event hazards that are included in the scope of a PRA are determined by the site-specific hazards. The applicability of the insights that can be gained from one full-scope comprehensive site Level 3 PRA will depend in part on how representative the analyzed site is of the larger population of NPPs. Selecting a site with a representative set of external hazards

may therefore be desirable. Alternatively, selecting a site with greater than normal external hazards of interest (e.g., seismic events, external flooding) can provide a different set of useful insights.

Spent Fuel Pool Storage Configuration
In attempting to understand the relative contribution of spent nuclear fuel to risk, another attribute to consider is the site-specific spent nuclear fuel storage configuration. For example, some sites use a common SFP for all reactors on the site, whereas others use separate SFPs for each reactor. Since the risk depends on the inventory of spent fuel that can be threatened, the site-specific storage configuration can have important risk implications.

Independent Spent Fuel Storage Installations (ISFSIs)
Sites with ISFSIs still have 4-5 cycles of spent nuclear fuel in their SFPs, which may be too hot to load into dry casks for storage. Because the risk associated with dry cask storage has been estimated to be lower than that for SFPs, this can have important risk implications. Performing the full-scope comprehensive site Level 3 PRA on a site that has an ISFSI can provide useful risk insights for other sites that also have ISFSIs. Alternatively, if a site without an ISFSI is selected, the PRA model can be used to obtain additional insights by assessing the risk significance of adding an ISFSI to the site.