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BENEFITS AND USES OF THE STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES (SOARCA) PROJECT

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Research Information Letter
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EXECUTIVE SUMMARY

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project was a major research study undertaken over the last decade by the U.S. Nuclear Regulatory Commission (NRC) and its contractor, Sandia National Laboratories. The project objectives were to develop an updated body of knowledge on the realistic outcomes of severe reactor accidents and update the assessment of severe accidents in previous NRC studies that were believed to be conservative. The project's scope evolved over time and ultimately includes detailed accident progression and source term calculations using the MELCOR computer code and detailed accident consequence calculations using the MELCOR Accident Consequence Code System for three pilot plants: Peach Bottom Atomic Power Station, a boiling-water reactor with a Mark I containment in Pennsylvania; Surry Power Station, a pressurized-water reactor with a large dry subatmospheric containment in Virginia; and Sequoyah Nuclear Plant, a pressurized-water reactor with an ice condenser containment in Tennessee. For each plant, the project team conducted a set of deterministic "best estimate" calculations and a detailed uncertainty analysis for one accident scenario. These calculations were completed in 2017. The results of these analyses consistently predict essentially zero individual early fatality risk for the modeled scenarios, very low long-term cancer fatality risks, and smaller radiological releases than those predicted from previous studies. The project has involved extensive communication through technical documentation, presentations, and public meetings with diverse stakeholders.

In addition to satisfying the project objectives, the SOARCA project has (1) developed staff expertise in a variety of important technical areas, including accident progression and source term analysis, offsite consequence analysis, parametric uncertainty analysis, and risk communication; (2) identified improvements in NRC analytical tools, such as computer codes and associated severe accident analysis methodologies; (3) provided readily available, detailed site- and plant-specific computer code models that could be used for additional analyses; and (4) been used to support risk-informed decisions that in turn supported safe and economical operating decisions. The improvement of tools, methodologies, and technical expertise has enhanced the NRC's ability to efficiently and effectively carry out its mission to protect public health and safety and the environment.

The project's results, insights, computer code models, and modeling best practices have supported NRC rulemaking, licensing, and oversight efforts and facilitated international cooperation and knowledge management. For example, computer code models and modeling best practices from SOARCA enabled the NRC to perform timely calculations to support its technical basis for issues identified as a result of the accident at the Fukushima Daiichi nuclear power plant in Japan. These include filtered containment venting for boiling-water reactors with Mark I and Mark II containments (see NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments," issued March 2018) and expedited transfer of spent fuel from pools to dry cask storage (see NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," issued September 2014). SOARCA results and insights were used during the accident at Fukushima to support rapid, time-sensitive emergency response. The outcomes of NRC and external stakeholders' uses of SOARCA include a substantial number of risk-informed decisions that have enhanced safety and security, while also supporting operational flexibilities, in the U.S. and abroad. This research information letter (RIL) describes these and many other applications of SOARCA project insights, models, and methodologies.

The SOARCA project has produced nine NUREG-series publications and has been used or cited in over 325 publications in the open literature, including technical reports, conference papers, journal articles, and dissertations. These publications cover a broad range of research areas, including but not limited to accident-tolerant fuel, reactor safety (including advanced designs), societal risk, and spent fuel storage and transportation, demonstrating the diverse areas in which researchers have referenced or used aspects of the SOARCA project.

The purpose of this RIL is to formally document the numerous benefits and uses of the project beyond its original objectives as well as its uses by the NRC, reactor licensees and applicants, domestic and international regulatory and research organizations, academia, and other stakeholders. This RIL focuses on the benefits and uses of SOARCA to date, and although there are many potential additional future benefits, the RIL does not speculate on them. It summarizes the SOARCA project, including motivation, approach, and results, followed by a high-level description of how SOARCA information has supported various projects and analyses related to nuclear power safety. The appendix lists publications from diverse research areas that have used or cited the SOARCA project.

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1 OVERVIEW OF THE SOARCA PROJECT

The U.S. Nuclear Regulatory Commission (NRC) initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to provide a state-of-the-art, more realistic evaluation of severe accident progression, radionuclide release, and offsite consequences for risk-significant severe reactor accident scenarios. The analyses leveraged insights from several decades of research on severe accident phenomenology and radiation health effects, information that was captured in two modern codes: MELCOR, an integrated severe accident progression code, and the MELCOR Accident Consequence Code System (MACCS), an accident consequence analysis code. One of the objectives was to update the quantification of offsite consequences found in earlier NRC publications, particularly NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," issued December 1982, as well as WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," issued October 1975, and NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued October 1990. The project team did so by incorporating (1) significant plant improvements and changes that were not reflected in earlier assessments, including security-related enhancements issued in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2),[1] in the wake of the terrorist attacks of September 11, 2001, (2) emergency response, (3) reactor power uprates, and (4) higher core burnup. The MELCOR and MACCS models used the most up-to-date site- and plant-specific information. An additional objective of the SOARCA project was to enable the NRC to more effectively communicate severe accident-related aspects of nuclear safety to diverse stakeholders, including the public; Federal, State, and local authorities; and nuclear power plant licensees.

The SOARCA project analyzed severe accidents for Peach Bottom Atomic Power Station (Pennsylvania), which is a U.S. boiling-water reactor (BWR) with a Mark I containment, and Surry Power Station (Virginia), a U.S. pressurized-water reactor (PWR) with a large dry (subatmospheric) containment. The three historical studies mentioned above also evaluated these reactors. The project team analyzed two groups of reactor accident scenarios: (1) long-term station blackouts (LTSBOs) and short-term station blackouts (STSBOs) for both Peach Bottom and Surry, and (2) containment bypass scenarios, including thermally induced steam generator tube rupture (SGTR) and interfacing systems loss-of-coolant accident, for Surry only. The project evaluated all scenarios, with and without the successful implementation of 10 CFR 50.54(hh)(2) mitigation equipment and procedures.

The process used to perform these evaluations included five steps:

(1) Select accident scenarios to model. The project used core damage frequencies (CDFs) to select accident scenarios. The SOARCA project analysts selected accident scenarios with a CDF higher than 10^{-6} per reactor year to allow them to analyze the most likely, yet very remotely possible, severe accident scenarios. These include the station blackout (SBO) scenarios described above. They also selected some lower probability accident scenarios (e.g., containment bypass scenarios) because of their potential to result in higher consequences. These accident scenarios used a lower screening criterion of 10^{-7} per-reactor-year.

[1] SOARCA did not evaluate FLEX, since the FLEX strategies were not yet formulated at the time of the Peach Bottom and Surry studies and were still under development at the start of the study for Sequoyah Nuclear Plant.

(2) Model accident progression and mitigation measures. The team used MELCOR to analyze accident progression, plant response, and mitigation measures for each of the scenarios described above.

(3) Model offsite release of radioactive material. The MACCS code used site-specific weather conditions to model atmospheric transport and dispersion of released radionuclides and site-specific population and land use data to model radiation exposure to the population.

(4) Model emergency response. In conjunction with step 3, the team modeled the evacuation of the public using site-specific emergency plans and evacuation time estimate studies.

(5) Model health effects. The team used MACCS to calculate radiation exposure to the population. It subsequently used the code to determine the resulting early and latent cancer fatality risks to the public. Analysts used multiple dose-response models to calculate individual latent cancer fatality risks.

The NRC completed the Peach Bottom and Surry SOARCA studies in 2012 and documented them in NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA)," issued November 2012; NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 1: Peach Bottom Integrated Analysis," Revision 1, issued May 2013; and "State-of-the-Art Reactor Consequence Analyses Project: Volume 2: Surry Integrated Analysis," Revision 1, issued August 2013. The NRC's Advisory Committee on Reactor Safeguards reviewed the methodology, assumptions, and results published in these studies, which were peer reviewed by an independent panel of external scientific and technical experts in the fields covered by the analysis. In addition, the staff received feedback from the public when it released draft NUREG-1935 for public comment. The staff addressed these comments before the final publication of NUREG-1935. To facilitate dissemination of severe accident consequence results to the general public, the NRC issued NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences," Revision 1, in December 2012.

The Peach Bottom and Surry SOARCA studies (2012 publications) had the following summary results:

- Radiological releases are considerably smaller than those reported in NUREG/CR-2239 for its siting source term 1 case[2], which led to the highest consequences in that study.

- Successful implementation of existing mitigation measures can prevent core damage or delay or reduce offsite releases of radionuclides.

- The individual early fatality risk for the modeled scenarios is essentially zero for both sites.

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2 One of five source terms evaluated in NUREG/CR-2239, siting source term 1 is the source term resulting from a postulated severe accident scenario in which there is severe core damage, loss of all installed safety features, and a severe breach of containment.

- The calculated individual long-term cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk from all causes.

Before the conclusion of the Peach Bottom and Surry evaluations, the staff began an uncertainty analysis (UA) of the SOARCA unmitigated LTSBO severe accident scenario for Peach Bottom with the following objectives:

- Assess the overall sensitivity of SOARCA results to uncertainties in inputs.
- Identify the input parameters that most strongly influence releases and consequences.
- Demonstrate a UA methodology that could be used for subsequent studies.

This analysis used the same SOARCA model and software that were used for the deterministic[3] analyses documented in NUREG-1935 and NUREG/CR-7110, but it varied a set of key uncertain MELCOR and MACCS input parameters. The specific parameters chosen captured important influences on potential releases of radioactive materials and on offsite consequences. Importantly, the results for the Peach Bottom UA (NUREG/CR-7155, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," issued May 2016) corroborated the conclusions obtained from the Peach Bottom deterministic SOARCA study.

After completing the Peach Bottom and Surry SOARCA studies, the staff recommended to the Commission in SECY-12-0092, "State-of-the-Art Reactor Consequence Analyses—Recommendation for Limited Additional Analysis," dated July 5, 2012, that it perform a UA for Surry and a consequence analysis for Sequoyah focused on issues unique to the ice condenser containment design and limited to SBO scenarios. In Staff Requirements Memorandum (SRM)-SECY-12-0092, "State-of-the-Art Reactor Consequence Analyses (SOARCA)—Recommendation for Limited Additional Analysis," dated December 6, 2012, the Commission approved the staff's recommendation and stated that the analyses should complement and support the ongoing Site Level 3 probabilistic risk assessment (PRA) project (referred to hereafter as the Site Level 3 PRA project; see Section 2.4.2) and regulatory activities following the accident at the Fukushima Daiichi nuclear power plant in Japan in March 2011. The staff performed a Surry UA of an unmitigated STSBO accident using the same approach as for the Peach Bottom UA. Notably, the UA results corroborated the results obtained from the Peach Bottom and Surry SOARCA studies. The NRC summarized the models, results, and insights in the draft report[4], "State-of-the-Art Reactor Consequences Analyses Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station," issued August 2015 (ADAMS Accession No. ML15224A001).

As noted above, the Sequoyah deterministic (and uncertainty) analyses differed from the Peach Bottom and Surry analyses in that it focused only on LTSBO and STSBO accidents, specifically

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[3] The term "deterministic" is commonly used to refer to an analysis in which the model form is fixed, such that the same set of initial and boundary conditions will always lead to the same output. Even though SOARCA used selected accident sequences based on plant probabilistic risk analyses, the accident progression and consequences were not modeled probabilistically, with the exception that a range of potential weather conditions is sampled even in deterministic MELCOR Accident Consequence Code System calculations. NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking," issued November 2013, defines "best estimate" as "…used for deterministic calculations, in which best estimate designates inputs or results obtained by using the most realistic assumptions available to the analyst (i.e., not biased by conservatism or optimism)."

[4] The updated final analyses will be published in the forthcoming NUREG/CR-7262 report.

on issues unique to ice condenser containments, including hydrogen generation and combustion. Moreover, the Sequoyah SOARCA analysis included an integrated UA, whereas the project team conducted the Peach Bottom and Surry UAs after the deterministic analyses. The Sequoyah UA evaluated an unmitigated STSBO using insights from the Surry UA for the MELCOR analysis as well as the same input parameters that were varied in the Surry UA MACCS analysis. The results from the Sequoyah UA with respect to radionuclide release, individual early fatality risk, and individual latent cancer fatality risk also support those from previous SOARCA analyses for Peach Bottom and Surry. NUREG/CR-7245, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses," published October 2019, documents the Sequoyah SOARCA UA. The table below summarizes the SOARCA studies described above.

Report No.	Title	Reactor Type	Scenarios Evaluated
NUREG-1935 (2012)	State-of-the-Art Reactor Consequence Analyses (SOARCA) (Parts 1 and 2)	BWR-4 (Mark I containment), 3-loop PWR (large dry containment)	STSBO, LTSBO, thermally induced SGTR, interfacing systems loss of coolant accident*
NUREG/BR-0359, Revision 2 (2016)	Modeling Potential Reactor Accident Consequences	BWR-4 (Mark I containment), 3-loop PWR (large dry containment), 4-loop PWR (ice condenser containment)	STSBO, LTSBO, thermally induced SGTR, interfacing systems loss of coolant accident
NUREG/CR-7110, vol. 1, Revision 1 (2013)	State-of-the-Art Reactor Consequence Analyses Project: Volume 1: Peach Bottom Integrated Analysis	BWR-4 (Mark I containment)	STSBO, LTSBO*
NUREG/CR-7110, vol. 2, Revision 1 (2013)	State-of-the-Art Reactor Consequence Analyses Project: Volume 2: Surry Integrated Analysis	3-loop PWR (large dry containment)	STSBO, LTSBO, thermally induced SGTR, interfacing systems loss of coolant accident
NUREG/CR-7155 (2016)	State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station	BWR-4 (Mark I Containment)	STSBO
NUREG/CR-7245 (2019)	State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses	4-loop PWR (ice condenser containment)	STSBO, LTSBO
NUREG/CR-7262 (Draft)	State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station	3-loop PWR (large dry containment)	STSBO

*A third scenario, loss of vital AC bus E-12, was evaluated for Peach Bottom but was determined not to lead to core damage.

2 BENEFITS AND USES OF THE SOARCA PROJECT

The SOARCA studies have increased the technical expertise of the staff in several areas, including UA, accident progression and accident consequence analyses, and atmospheric transport and dispersion. In accordance with its original objectives, the SOARCA project has also enabled enhanced communication of severe nuclear accident safety to external stakeholders. In addition to the numerous SOARCA reports referenced in the previous section, the NRC published a brochure, NUREG/BR-0359, with the specific intention of communicating up-to-date information about realistic severe accident consequences to the public. Other ways in which the staff has communicated SOARCA results include hosting public meetings near the plants studied in the SOARCA project to solicit public comments, as documented in Appendix C, "SOARCA Public Comments Summary," of NUREG-1935, and presenting at NRC Regulatory Information Conference sessions, Cooperative Severe Accident Research Program (CSARP) meetings, International MACCS User Group (IMUG) meetings, and special SOARCA UA sessions organized for international Probabilistic Safety Assessment and Analysis and Probabilistic Safety Assessment and Management conferences. Additionally, the staff has communicated SOARCA insights to interested Federal partners, including the Federal Emergency Management Agency and the Federal Bureau of Investigation.

The SOARCA project has also identified improvements in NRC analytical tools, such as computer codes and associated severe accident analysis methodologies, including parametric UA. The MELCOR and MACCS computer codes and their modeling best practices have evolved significantly throughout the SOARCA project. NUREG/CR-7008, "MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," and NUREG/CR-7009, "MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," both issued August 2014, document the MELCOR and MACCS best practices from the Peach Bottom and Surry deterministic studies, respectively. These reports document model improvements, modeling approaches, and parameter selection and explain the significance of the modeling improvements and approaches, as of 2012. The best practices further evolved during the later SOARCA studies and post-Fukushima severe accident risk evaluations that relied on SOARCA methods. These improvements are reflected in subsequent reports such as NUREG/CR-7245 for the Sequoyah SOARCA study. This documentation of best practices has helped provide the basis for the modeling choices made for accident analyses that followed, including the Site Level 3 PRA project. Further, a compilation of results and insights derived from the collective UAs is under development and should provide a useful reference for risk-informed regulatory activities. It should be noted that the MELCOR modeling best practices are continually being updated and communicated to code users at workshops as information becomes available from modeling improvements and code assessment.

The NRC has used insights, models, methodologies, and results from the SOARCA studies to support potential rulemaking activities, licensing activities, oversight, and regulatory decisionmaking, particularly in response to the accident at the Fukushima Daiichi nuclear reactors. The SOARCA project has also helped to risk-inform current NRC programs and projects such as the Site Level 3 PRA project, which further enhances the regulatory process. The staff at the NRC and Sandia National Laboratories (SNL) involved with the SOARCA project have trained other NRC staff and international partners for knowledge management purposes and to help support the correct application of SOARCA tools and information. Lastly, other regulatory and research institutions, both domestic and international, have used or cited aspects of the SOARCA project in diverse research areas such as accident-tolerant fuel and dynamic PRA. The appendix of this RIL lists more than 325 citations of the SOARCA project in these and other areas. This section summarizes several ways in which the NRC, reactor licensees and applicants, regulatory and research organizations, academia, and other external stakeholders have used the SOARCA studies.

2.1 Regulatory Support for NRC Post-Fukushima Activities

The SOARCA studies for Peach Bottom and Surry were still ongoing when the Great Tōhoku earthquake and subsequent tsunami in Japan caused extensive damage to the nuclear reactors at the Fukushima Daiichi nuclear power plant, a BWR with a similar containment design (Mark I) as Peach Bottom. In the U.S. response to this accident, the NRC established the Near-Term Task Force (NTTF) to determine whether the NRC should make any near- or long-term improvements to its regulatory system and make recommendations to the Commission for its policy direction. The NTTF issued its "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated July 12, 2011, providing recommendations to the Commission that were intended to clarify and strengthen the regulatory framework for protecting against natural disasters, for mitigation and emergency preparedness (EP), and for improving the effectiveness of existing NRC programs. The NTTF recommendations and additional issues identified by the staff included a Tier 1 recommendation[5] related to reliable hardened vents for BWR Mark I and Mark II containments and Tier 3 recommendations[6] related to hydrogen control, reliable hardened vents for containment designs other than BWR Mark I and Mark II, and expedited transfer of spent fuel to dry cask storage. The Commission later directed the staff to evaluate these recommendations to inform separate potential rulemaking efforts. The NRC used SOARCA insights and models as inputs for evaluating these recommendations, as described below.

2.1.1 *Evaluation of Filtered Containment Venting and Containment Protection and Release Reduction Strategies for Boiling-Water Reactors with Mark I and Mark II Containments*

Following the Fukushima Daiichi accident, one of the major issues identified by NRC staff was whether to require licensees of BWRs with Mark I and Mark II containments to add capabilities for containment protection and release reduction following a potential loss of power accident. These capabilities included installing filtered containment venting systems, as many other countries had done, but also included severe accident water addition and water management strategies. In SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012, the staff recommended adding filtration to reliable hardened vents. In SRM-SECY-12-0157, "Staff Requirements – SECY-12-0157 – Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated March 19, 2013, the Commission directed the staff to develop the technical bases and rulemaking for filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments in support of NTTF Tier 1

[5] Tier 1 recommendations include NTTF recommendations and additional issues that should be addressed without delay.

[6] Tier 3 recommendations include NTTF recommendations and additional issues identified by the staff that require further staff study to support a regulatory action, have an associated shorter term action that needs to be completed to inform the longer-term action, or depend on the resolution of a NTTF Tier 1 recommendation.

Recommendation 5.1. This effort was known as the containment protection and release reduction (CPRR) rulemaking.

The NRC staff were able to conduct timely technical analyses to inform the CPRR rulemaking effort by leveraging insights from the SOARCA project and using updated versions of plant-specific MELCOR and MACCS models from SOARCA in its analyses. If the SOARCA MELCOR and MACCS models were not available, the staff would have required significantly more time to conduct sufficiently detailed analyses to inform the CPRR rulemaking effort. The CPRR technical analyses included accident sequence analyses, accident progression and source term analyses, and offsite consequence analyses. The BWR Mark I accident progression and source term analysis used an updated version of the Peach Bottom SOARCA MELCOR model, and the offsite consequence analysis used an updated version of the Peach Bottom SOARCA MACCS site model. The calculated offsite consequences were weighted by accident frequency to assess the relative public health risk reduction associated with the various CPRR alternatives. The CPRR analyses showed the risk reduction benefits of having severe-accident-capable hardened vents, adding water to cool core debris, and adding filtered containment vents to Mark I and II containments.

Based in part on the CPRR technical analyses, the staff modified its risk-informed recommendations in SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities," dated June 18, 2015, to no longer require external filters (as was done in SECY-12-0157) but instead make the requirements of Order EA-13-109, "Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013, generically applicable. In SRM-SECY-15-0085, "Staff Requirements – SECY-15-0085 – Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR PART 50) (RIN-3150-AJ26)," dated August 2015, the Commission terminated the CPRR rulemaking and directed the staff to continue implementation of Order EA-13-109 for severe-accident-capable vents, with no additional regulatory actions. It also directed the staff to leverage the draft regulatory basis to support the resolution of Tier 3 issues related to containments of other designs (i.e., NTTF Recommendation 5.2). The staff issued JLD-ISG-2015-01, "Compliance with Phase 2 of Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions, Revision 0," dated April 2015, to assist nuclear power reactor licensees with the identification of methods that could be used to comply with the requirements of Order EA-13-109.

The CPRR technical analyses were published in detail in NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments," issued March 2018. As illustrated above, these analyses, which leveraged elements of the ongoing SOARCA project, supported risk-informed decisions that led to the continued safe operation of nuclear power plants, enhanced operational flexibilities, and offset costs associated with expensive plant modifications. These analyses serve as an example of evidence-based policymaking as discussed in the Foundations for Evidence-Based Policymaking Act of 2018. NUREG-2206 discusses the data, methods, and analytical approaches that were used to ensure NRC's actions on containment filtered venting, containment protection, and release reduction were based on evidence and sound methods and approaches.

2.1.2 *Evaluation of Tier 3 Issues Related to Containment Venting for Other than Mark I and Mark II Containments and to Hydrogen Control and Mitigation*

NTTF Tier 3 Recommendation 5.2 suggested that the staff assess whether to require installation of reliable, hardened venting systems for containments other than Mark I and II designs (i.e., BWR Mark III containments and PWR ice condenser and large dry containments). In addition, NTTF Tier 3 Recommendation 6 suggested that the staff assess the need to strengthen requirements associated with hydrogen control and mitigation inside and outside reactor containment buildings. In SECY-15-0137, Enclosure 4, "Proposed Resolution Plans for Tier 3 Recommendations 5.2 and 6: Reliable Hardened Vents for Other Containment Designs and Hydrogen Control and Mitigation Inside Containment and Other Buildings," dated October 29, 2015, the staff relied, in part, on results and insights from the Surry and Sequoyah SOARCA analyses to conclude that additional study of these topics would be unlikely to identify any regulatory actions beyond those already taken that would provide a substantial safety improvement for large dry and ice condenser containments. The abovementioned draft regulatory basis for the CPRR rulemaking (NUREG-2206) provided additional insights in support of this conclusion. In SECY-16-0041, "Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation," dated March 31, 2016, the staff informed the Commission of its plan to formally close NTTF Recommendation 5.2 and 6 activities.

2.1.3 *Evaluation of Expedited Transfer of Spent Fuel to Dry Cask Storage*

The Fukushima Daiichi nuclear accident renewed international interest in the safety of spent nuclear fuel stored in spent fuel pools (SFPs) under prolonged loss-of-cooling conditions. Although the SFPs and spent fuel assemblies stored in the pools remained safe after the accident, it led the NRC to consider whether it should require the expedited transfer of spent fuel from spent fuel pools to dry cask storage at U.S. nuclear power plants. Shortly after the NTTF's report was released, the staff initiated a project to evaluate SFP accident consequences from a beyond-design-basis earthquake for a reference U.S. BWR with a Mark I containment. As part of NRC's post-September 11 security assessments, SFP models that used detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and used to assess the realistic heatup of spent fuel under various pool draining conditions. The Peach Bottom MACCS model developed for the SOARCA studies was adapted for use in the SFP offsite consequence analysis. The results of the study, documented in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," issued September 2014, and in SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated October 9, 2013, indicated that expedited transfer of spent fuel to dry cask storage did not provide a substantial safety enhancement for the reference plant. This study was later expanded to inform a broader regulatory analysis (enclosed with COMSECY-13-0030,"Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 12, 2013) for an NTTF Tier 3 issue of whether expedited transfer of spent fuel to dry cask storage at all U.S. nuclear power plants substantially enhances public health and safety. The staff concluded that the expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit and that its expected implementation costs were not warranted. In SRM-COMSECY-13-0030, "Staff Requirements – Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 23, 2014, the Commission approved the staff's recommendation to close the

Tier 3 Japan lessons-learned activities for expedited transfer and perform no further generic assessments.

2.2 Licensing and Design Certification Application Reviews

One of the primary regulatory functions of the NRC is licensing (and renewing) operating reactors and certifying new reactor designs. As the current fleet ages, operating reactors are required to provide severe accident evaluations as part of the renewal process, while reactor designers who seek NRC approval of their new designs provide similar evaluations.[7] Operating reactor licensees and advanced reactor applicants have employed SOARCA computational tools and insights to support assumptions and methodologies in their respective renewal and design certification applications, and NRC technical reviewers have used these insights to review these applications. This section summarizes how the NRC has used SOARCA insights and computational tools for operating license renewals and for domestic and international light-water reactor design certification applications and reviews.

2.2.1 *NuScale Application and the NRC's Review*

In December 2016, NuScale Power, LLC (NuScale), submitted to the NRC a design certification application that used SOARCA model improvements and insights for NuScale's small modular reactor design. The NRC is using SOARCA model improvements and insights to perform its technical reviews. The following documents NuScale's use of SOARCA insights:

- NuScale Standard Plant Design Certification Application Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," Revision 2. NuScale used the MELCOR and MACCS codes, methods, and assumptions containing SOARCA insights to evaluate accident progression and source term for the PRA and severe accident mitigation.

- NuScale Standard Plant Design Certification Application Environmental Report, Revision 2. NuScale used the MACCS codes, methods, and assumptions to calculate severe accident offsite consequences to evaluate the averted offsite costs and exposures to identify whether any SAMDA candidates are potentially cost beneficial.

- NuScale Licensing Topical Report No. TR-0915-17772, Revision 0, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," issued December 2015. NuScale used the MELCOR and MACCS codes, methods, and assumptions to evaluate accident progression and source terms and to inform the selection of the required distance over which an emergency planning zone may be needed.

Because the SOARCA studies represent the state of the art in severe accident analysis, the accident progression and offsite consequence analysis methodologies used in the NuScale

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7 The environmental report that is submitted as part of a license renewal application includes a severe accident mitigation alternative (SAMA) analysis. It identifies potentially cost-beneficial enhancements that could further reduce nuclear power plant risk. A design certification application for a new reactor includes analyses of severe accident mitigation design alternatives (SAMDAs), which are similar in scope to SAMA analyses.

application and NRC license review process may also benefit other new reactor application processes and contribute to knowledge management.

2.2.2 *APR1400 Design Certification Review and SOARCA-Like Analysis*

Korea Electric Power Corporation and Korea Hydro and Nuclear Power (KHNP) submitted their APR1400 design to the NRC for review in December 2014. The NRC staff used SOARCA insights in its review of the APR1400 design certification application, particularly when reviewing the applicant's models. In Chapter 19 of the NRC's safety evaluation report, dated September 28, 2018, the agency referenced these insights when it approved the APR1400 design certification application.

In 2016, KHNP, with SNL support, also initiated SOARCA-like analyses of the APR1400 and Canada Deuterium Uranium (CANDU) reactors. The analysis leverages SOARCA project methodologies and MACCS best practices to perform realistic accident progression and consequence analyses for two pilot plants, Shin-Kori (APR1400) and Wolsung (CANDU). KHNP gave an update on its SOARCA project at the NRC's 2018 Regulatory Information Conference. The NRC staff is following the research and collaborative efforts to identify any areas of interest and consideration of model updates based on these designs.

2.2.3 *Contentions on Severe Accident Mitigation Alternatives during Indian Point Operating License Renewal Proceedings*

Part of the NRC's and Entergy's expert testimonies in Atomic Safety and Licensing Board Panel proceedings for renewal of the operating license for Indian Point Nuclear Generating used insights from the 2012 SOARCA studies. For example, as part of NRC staff testimony on SAMA contentions submitted by the State of New York, experts referred to the peer-reviewed SOARCA study as a reference for realistic modeling of aerosol particle sizes and corresponding deposition velocities in offsite consequence modeling.

2.2.4 *AREVA European Pressurized Reactor Application*

Prior to the SOARCA project, it was thought that a severe-accident-induced SGTR would lead to a large release because airborne fission products would flow from the reactor coolant system to the environment through the ruptured tube with little or no deposition in the reactor coolant system or steam generator. However, the SOARCA project showed that hot leg rupture would occur after a SGTR. Hot leg rupture results in most airborne fission products depositing in the containment, implying that the release of airborne fission products to the environment would be less than originally assumed.

AREVA NP submitted a design certification application for its European Pressurized Reactor to the NRC in 2007. During the staff's review of the application, AREVA NP developed an update to its application to reflect the SOARCA insight regarding hot leg failure subsequent to a SGTR. This insight showed an additional margin in the European Pressurized Reactor's ability to mitigate the severe accident source term. Nevertheless, AREVA NP requested that the NRC suspend the EPR application in 2015.

2.3 Domestic and International Cooperation in Response to the Fukushima Daiichi Accident

The Great Tōhoku earthquake and subsequent tsunami led to core melting and radioactive material release from Units 1, 2, and 3 at the six-unit Fukushima Daiichi nuclear power plant. In addition, hydrogen production from zirconium-cladding reactions with water led to hydrogen explosions in Units 1, 3, and 4. The NRC began immediate activities to support the Federal Government response to the event. In the aftermath of the earthquake and tsunami, domestic and international efforts were initiated to reconstruct the accident, assess severe accident modeling capabilities, and enhance offsite protective action decisionmaking. This section explains how SOARCA models, tools, and insights supported these activities.

2.3.1 *Support for Federal Government Response to Fukushima*

The SNL staff, under contract to the U.S. Department of Energy (DOE), used the SOARCA thermal-hydraulic analysis of the Peach Bottom LTSBO scenario to estimate the accident progression at each affected Fukushima reactor. SNL used Peach Bottom SOARCA source terms adjusted for thermal power differences at Fukushima to perform MACCS offsite consequence analyses for DOE and the NRC. DOE's consequence management group also received SOARCA source terms and provided them to Lawrence Livermore National Laboratory's National Atmospheric Release Advisory Center. The Center used the modified SOARCA source terms for consequence analyses of the Fukushima accident to support the overall U.S. Federal Government's response to the accident.

2.3.2 *Fukushima Incident Response*

One of the key objectives of the SOARCA studies was to enhance communication of severe-accident-related aspects of nuclear safety, including severe accident consequences, to the public and other stakeholders. Because of the similarities between the Fukushima Daiichi accident and Peach Bottom SOARCA calculations (both SBOs at BWRs with Mark I containments), Peach Bottom SOARCA insights were used in the immediate response to Fukushima. Specifically, the NRC Operations Center used Peach Bottom SOARCA MELCOR and MACCS models as starting points to generate source terms for the reactor and estimate doses around Fukushima. The staff also used preliminary SOARCA insights to prepare for the congressional hearings immediately after the accident at Fukushima, specifically noting that Peach Bottom had hardened vents and some pre-staged portable equipment, as required by Section B.5.b of Order EA-02-026, "Order for Interim Safeguards and Security and Compensatory Measures," issued February 2002, as well as the resulting severe accident mitigation those plant features provide. A few months after the accident, the New York Times referenced the SOARCA study in its article, "N.R.C. Lowers Estimate of How Many Would Die in Meltdown."

2.3.3 *Fukushima Forensic Analysis*

Peach Bottom SOARCA calculations were still being analyzed when the Fukushima accident occurred. After the incident response phase of the accident, the NRC and DOE jointly sponsored an accident reconstruction study as a means of (1) better understanding the Fukushima accident progression and (2) assessing the severe-accident modeling capability of MELCOR. Using knowledge gained from the Peach Bottom SOARCA studies, SNL used MELCOR and existing information to predict how the accident may have progressed. SNL published the results from this study in SAND2012-6173, “Fukushima Daiichi Accident Study (Status as of April 2012),” issued July 2012, which is publicly available on SNL’s website.

2.3.4 *Fukushima Uncertainty Analyses*

SNL conducted UAs of the Fukushima Unit 1 reactor using the Peach Bottom SOARCA SBO UA model as a starting point. The identification of important uncertain parameters was also informed by the Peach Bottom SOARCA UA of the LTSBO scenario, which had many similarities to the Fukushima accident progression. The goals of the Fukushima UAs were to evaluate uncertainty in core damage progression behavior and its effect on key figures of merit such as hydrogen production, fraction of intact fuel, and vessel lower head failure; to characterize the range of predicted damage states in the reactor considering uncertainties associated with MELCOR modeling; and to help inform the decommissioning activities available to Japanese decisionmakers to defuel the damaged reactors. SAND2016-0232, “Fukushima Daiichi Unit 1 Accident Progression Uncertainty Analysis and Implications for Decommissioning of Fukushima Reactors—Volume I,” issued January 2016, summarizes this work.

2.3.5 *Benchmark Study of the Accident at Fukushima*

In 2012, the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) established a joint research project known as the Benchmark Study of the Accident at Fukushima (BSAF) to improve severe accident codes and analyze the accident progression of Fukushima Daiichi Units 1, 2, and 3. Sixteen organizations, including the NRC and the Electric Power Research Institute (EPRI), from eight countries joined the project, which was conducted in two phases. The first phase of the BSAF project focused on the first 6 days of the accident progression and estimated the current status inside the reactor pressure vessels and primary containment vessels for Units 1–3. Phase 1 of the BSAF project was completed in 2015. Phase 2 of the project expanded the work scope to include fission product behavior outside primary containment vessels, which lengthened the timespan for analyses of accident events to about 3 weeks after the earthquake. The Peach Bottom SOARCA MELCOR models were used as the starting point for the BSAF analyses to compare MELCOR against other accident progression computer codes such as the Modular Accident Analysis Program (MAAP), Accident Source Term Evaluation Code (ASTEC), and SAMPSON codes. In addition, SOARCA-initiated developments to the MACCS code have been used to help analyze atmospheric dispersion and land contamination to help benchmark efforts to simulate the Fukushima accident progression. Phase 2 of the project was completed in 2018.

2.3.6 *Benchmarking of Fast-Running Software Tools to Inform Offsite Decisionmaking*

After the accident at Fukushima Daiichi, it was observed that protective measures recommended to citizens occasionally differed by country, especially during the initial stages of the accident. Such differences could impact the projected radiological dose to members of the public. Because of this observation, a group within the OECD/NEA recommended that the agency “analyse the comparison of source-term methodologies utilised by countries and determine if or why the dose prediction differed for Fukushima.” To that end, the OECD/NEA sponsored a research project to benchmark fast-running software tools that are used to model offsite releases during a nuclear accident to support protective action decisionmaking. Twenty organizations, including the NRC, participated in the study. The study used Peach Bottom and Surry SOARCA source terms for two of the five scenarios that were selected to benchmark different fast-running software tools. The results of this study were used to explain why calculated source terms using the various codes and methodologies differed.

NEA/CSNI/R(2015)19, “Benchmarking of Fast-running Software Tools Used to Model Releases During Nuclear Accidents,” issued January 2016, summarizes this work and provides information to participants to inform their own severe accident analysis codes.

2.4 Risk-Informing NRC Programs and Projects

The NRC currently has several activities and initiatives to integrate risk information and performance measures into its regulations, guidance, and oversight processes in the regulatory arenas of reactor safety, material safety, and waste management. The NRC used risk insights from the SOARCA studies to improve or develop tools for risk-informed oversight activities and to support other risk-informing projects such as the Site Level 3 PRA project. The Nuclear Energy Institute (NEI) has also used SOARCA results in discussions on quantitative health objective (QHO) margins. This section highlights the ways in which the NRC has used SOARCA studies to risk-inform its current oversight programs and projects and summarizes NEI’s use of SOARCA to discuss margins.

2.4.1 *Probabilistic Risk Assessment Technical Development*

The NRC used the Peach Bottom SOARCA MELCOR model, in conjunction with other analytical tools, to investigate a longstanding PRA technical issue associated with the degree of credit warranted for emergency core cooling system injection following either containment venting or failure in BWR Mark I and II designs, specifically for PRA sequences in which adequate reactor pressure vessel injection is available but suppression pool heat removal is unavailable. Additionally, the NRC employed the Peach Bottom and Surry SOARCA MELCOR models for a second project to investigate specific thermal-hydraulic success criteria and sequence timing issues as documented in NUREG-1953, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom,” issued September 2011. The agency used insights gained from MELCOR calculations to establish the technical bases for a subset of the success criteria used in the Standardized Plant Analysis Risk (SPAR) models for these reactor designs, which increased the realism of the models. The NRC uses SPAR models in its significance determination process for findings in the Reactor Oversight Process and other risk-informed oversight activities.

2.4.2 *Site Level 3 PRA Project*

The Site Level 3 PRA project has leveraged SOARCA knowledge and insights to enhance staff capability and extract new risk insights that may be used to enhance regulatory decisionmaking. For example, the NRC used insights from the draft Surry SOARCA UA for a STSBO to help identify and determine the appropriate alternative treatment of many model uncertainties, and some parameter uncertainties, in the Level 2 analysis portion of its Level 3 PRA. In some cases, the MELCOR model parameter distributions that were developed for the Surry SOARCA UA helped define alternative treatments (MELCOR sensitivity runs). Having MELCOR sensitivity cases informed by the Surry SOARCA UA made the resulting sensitivities more informative. The documentation for the Site Level 3 PRA project refers the reader to the Surry SOARCA UA results for insights into the potential effects of model uncertainties for which alternative MELCOR sensitivity runs were not completed for the Site Level 3 PRA project.

The Site Level 3 PRA project also benefitted more broadly from the MELCOR and MACCS analytical improvements initiated by the SOARCA project. Examples of MACCS enhancements prompted by the SOARCA project include the ability to more realistically model early phase protective actions by including multiple population cohorts and the ability to capture the effect of shifts in wind directions by implementing a 64-sector spatial grid. Finally, the work carried out on documenting the technical bases for MACCS input parameter values in SOARCA was valuable for developing the documentation for the Level 3 PRA offsite consequence analyses, which assisted in fulfilling the knowledge management objectives of the Site Level 3 PRA project.

2.4.3 *Emergency Preparedness Significance Quantification Process*

The NRC staff initiated a project to develop a decision process for use in regulatory oversight that would help determine the risk significance of EP planning elements. This process, known as the DedUctive Quantification Index (DUQI) method, quantifies the value of EP program elements in terms of dose avoided, a value obtained from performing consequence analyses. The NRC used the DUQI method in a proof-of-concept application to two representative nuclear power plant sites: a PWR at a site with high population density and a BWR at a site with medium population density. As part of this application, detailed consequence analyses were performed for two accident scenarios at each site using MELCOR and MACCS models from the Peach Bottom and Surry SOARCA studies. This work successfully illustrated one approach to risk-informing EP oversight at nuclear power plants and demonstrated that the DUQI method may be adapted to determine the risk significance of mitigating actions. Such risk information could help prioritize resources while enhancing overall safety, increasing public confidence, and reducing unnecessary regulatory burden. The study is documented in NUREG/CR-7160, "Emergency Preparedness Significance Quantification Process: Proof of Concept," issued June 2013.

2.4.4 *Seismic Probabilistic Risk Assessment Relief Request*

The NRC NTTF's Recommendation 2.1 identified the need for nuclear power plant licensees to reevaluate the seismic hazards at their sites against current NRC requirements and guidance and, if necessary, update the design basis and structures, systems, and components important to safety to protect against the updated hazards. In response, the NRC issued a 10 CFR 50.54(f) letter asking licensees to reevaluate the seismic hazards at their respective sites by March 31, 2014. The staff also developed a screening process to identify which plants needed to conduct a seismic PRA. Two plants that screened in were Duke Energy's Catawba Nuclear Station and McGuire Nuclear Station, both of which have ice condenser containments. Duke Energy submitted a letter requesting seismic PRA relief and referenced the Sequoyah SOARCA study, which was done for a PWR with an ice condenser containment, to support its request. Sequoyah SOARCA conclusions generally confirmed conclusions of previous ice condenser studies and helped the NRC make a timely risk-informed decision when responding to Duke's seismic PRA relief request. Ultimately, the NRC approved Duke's seismic PRA relief request.

2.4.5 *Identification of Missing Emergency Action Level*

The SOARCA project led staff to identify an inconsistency in the emergency action level[8] (EAL) schemes for BWRs and PWRs. The EAL scheme for Peach Bottom would trigger a declaration of a general emergency for an event that led to an immediate loss of alternating current and

8 An EAL is a predetermined, site-specific, observable threshold for a plant condition that places the plant in an emergency class.

direct current power, which was not the case for Surry. The EAL scheme was updated to address this, which enhanced the emergency preparedness of this plant.

2.4.6 *NEI Use of SOARCA in Discussions on Margins*

In October of 2018, NEI sent a letter and white paper (which relied on an earlier EPRI white paper) to the NRC's Executive Director of Operations with the subject, "Facilitating Regulatory Transformation through an Understanding of the Current Levels of Safety." The letter used results from recent NRC analyses (including SOARCA) to examine the perceived margin between the core damage frequency and large early release frequency surrogate objectives and the QHOs. In comments at a public meeting, the NRC staff cautioned about extrapolating results from studies such as SOARCA, which were intended for other purposes, to draw conclusions on QHO margins. At this public meeting, NRC staff also emphasized that focusing solely on margins between QHOs and surrogate objectives overlooks the part of the Commission's 1986 safety goal statement "to continue to pursue a regulatory program that has as its objective providing reasonable assurance, while giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur."

2.5 Enhancement of Existing Computer Models and Methodologies for Technical Analyses

The Peach Bottom, Surry, and Sequoyah SOARCA studies provided detailed MELCOR and MACCS models of these sites that could be used as a starting point for other studies. The NRC has also used these models to enhance its incident response capabilities. This section summarizes the applications of SOARCA MELCOR and MACCS models and insights by the NRC, SNL, DOE, EPRI, and Argentina's Nuclear Regulatory Authority for other projects.

2.5.1 *Analysis of Cyber-Attack Vectors on Nuclear Power Plants*

As a laboratory-directed research and development project, SNL developed a model for evaluating how cyber-attack vectors could cause core damage at U.S. nuclear power plants. SNL developed MELCOR models to simulate accident progression using Peach Bottom and Surry SOARCA MELCOR models as starting points. This project provided important insights about how cyber-attack vectors could cause core damage. However, the insights, results, and reports generated from this work are proprietary and are, therefore, not publicly available.

2.5.2 *Atucha II Level 2 and 3 Probabilistic Risk Assessment Technical Review*

Argentina's Nuclear Regulatory Authority contracted SNL to conduct an independent technical review of the Level 2 and 3 PRAs for the Atucha Unit 2 pressurized heavy-water nuclear power plant to support the safety of these plants. SNL's technical review used accident progression and offsite consequence insights from the SOARCA project, which improved the quality of the review. The insights, results, and reports generated from this work are proprietary and are, therefore, not publicly available.

2.5.3 *Boiling-Water Reactor Owners Group Severe Accident Guidance*

DOE and EPRI are jointly preparing Revision 4 of the BWR Owners Group's severe accident guidelines. They are using the Peach Bottom SOARCA UA model and performing a dynamic analysis, with respect to time and equipment failures, as they review the generic severe accident management guidelines applicable to Peach Bottom. The guidance generated from this work is proprietary and not publicly available.

2.5.4 *Integrated System Response Modeling*

The Light Water Reactor Sustainability Program, sponsored by DOE's Office of Nuclear Energy, has begun a physical security initiative to link force-on-force modeling with reactor system response modeling. The goal of the effort is to develop modeling and simulations for existing nuclear power plant security regimes by using identified target sets to link dynamic assessment methodologies. This is being accomplished by leveraging nuclear power plant system-level modeling (i.e., SOARCA best practices MELCOR models) with force-on-force modeling and three-dimensional (3D) visualization to develop security tabletop scenarios. The impact of this effort is to create an integrated force-on-force and nuclear power plant system response framework that enables a holistic approach to determining security-related events as they relate to the potential onset of significant core damage.

To date, the MELCOR model (based on Three Mile Island Nuclear Station, Unit 2) has been adapted for physical security scenarios and is based on the open source Lone Pine Nuclear Power Plant notional facility. The MELCOR model uses SOARCA insights and SOARCA best practices and has been linked with a dynamic event/fault tree scheduler, ADAPT. The force-on-force scenarios have been developed within the SCRIBE 3D software, and this software is currently being linked with ADAPT. The report SAND2019-12015, issued in October 2019, summarizes the proof-of-concept integrated system response model.

2.5.5 *MELCOR Integration with the Emergency Response Data System*

The MELCOR Accident Simulation Trainer (MAST), a MELCOR postprocessing tool, was developed to feed MELCOR outputs to the Emergency Response Data System. The NRC used the Peach Bottom MELCOR SOARCA model as input to train its emergency responders about what to expect during station blackouts at a Mark I BWR.

2.5.6 *Enhancement of the RASCAL Code for Incident Response*

At the time of the accident at Fukushima, the Radiological Assessment System for Consequence AnaLysis (RASCAL) code, the NRC's primary code for incident response, could not calculate source terms for LTSBOs. Using information obtained from SOARCA studies, RASCAL was modified to include a LTSBO option that incorporates core release fractions and accident progression timings for both PWRs and BWRs. This addition enabled LTSBO scenarios to be modelled in real time during an event. NUREG-1940, Supplement 1, "RASCAL 4.3: Description of Models and Methods," issued May 2015, provides more details about specific modifications to RASCAL as well as comparisons between RASCAL and SOARCA results.

2.5.7 *Modern Radiological Source Terms for MACCS Model Testing and Benchmarking*

Prior to SOARCA, radiological source terms used for consequence modeling and testing were highly simplified representations of complex patterns of radiological releases under hypothetical severe accident conditions. Radiological source terms developed in SOARCA provide more realistic examples of the potentially complex time-dependent radiological release patterns from hypothetical severe accidents. These source terms are being used to test MACCS model performance using more realistic release patterns. Current examples include the use of SOARCA source terms to examine the potential differences in consequence estimates obtained using the MACCS simplified, fast-running Gaussian plume segment model and those using the more modern HYSPLIT atmospheric transport and dispersion code developed and maintained by the National Oceanic and Atmospheric Administration. To assist NRC staff in preparing to evaluate potential requests for reduced emergency planning zone sizes for new and advanced reactor technologies, SOARCA source terms are also being used by NRC staff to examine the sensitivity of radiological dose estimates to different consequence modeling choices.

2.6 Knowledge Management

One of the primary objectives of the SOARCA project was to enhance the communication of severe accident aspects of nuclear safety to internal (NRC) and external stakeholders. In addition to publishing SOARCA reports and delivering presentations at conferences, staff members at NRC and SNL have also provided training to domestic and international stakeholders for knowledge management purposes. This section describes the training sessions provided to diverse stakeholders about the SOARCA studies.

2.6.1 *Cooperative Severe Accident Research Program*

The Cooperative Severe Accident Research Program (CSARP) is an international program on severe accident phenomenological research and code development activities organized by the NRC since 1988. The objective of CSARP is the exchange of data and analyses on experimental and analytical research on severe accidents. Examples of information exchanged include the SOARCA studies, Fukushima lessons learned, the Phebus Fission Product Program, NEA's Behavior of Iodine Project, mixed-oxide and high-burnup fuel fission product releases, QUENCH experiments, and the OECD/NEA Melt Coolability and Concrete Interaction project. Several other entities operate within CSARP, including the European MELCOR and MACCS Users Group, the Asian MELCOR and MACCS Users Group, and IMUG. NUREG/BR-0524, "Cooperative Severe Accident Research Program (CSARP)," issued November 2015, provides more information about CSARP and lists SOARCA as an example of information exchanged through the program.

2.6.2 *NRC-Sponsored Training*

The NRC and DOE National Laboratories teach SOARCA insights through their Accident Progression Analysis, Accident Consequence Analysis, and Perspectives on Reactor Safety training courses. These courses are part of several NRC formal qualification programs, such as those for reliability and risk analysts, senior reactor analysts, and MACCS analysts. Several training classes and workshops offered during CSARP and IMUG meetings used material from SOARCA analyses.

2.6.3 *Boiling-Water Reactor Owners Group and Nuclear Energy Expert Group Training*

The BWR Owners Group has a technical support guidance workshop, and SNL has been invited to discuss severe accident phenomenology insights, some of which come from

SOARCA. The Center for Strategic and International Studies hosts Nuclear Energy Expert Group meetings once or twice a year in Southeast Asia. These meetings include attendees from Southeast Asian countries who are interested in building nuclear reactors in their countries. The SNL staff gave a presentation that included SOARCA project information at a Center for Strategic and International Studies meeting in 2016 on nuclear accidents and incident response.

2.6.4 *Sandia National Laboratories Training to Korea Hydro and Nuclear Power*

KHNP contracted with SNL to provide MELCOR and MACCS training to support its SOARCA-like consequence analysis of the APR1400. The training materials used SOARCA modeling and insights.

3 CONCLUSIONS

The SOARCA project leveraged decades of research on severe accident phenomenology and radiation effects and used site- and plant-specific data to evaluate the consequences arising from realistic severe accident scenarios at Peach Bottom, Surry, and Sequoyah. In addition, the studies incorporated significant plant improvements and changes not reflected in earlier reactor assessments and evaluated the potential benefits of security-related enhancements aimed at preventing core damage and reducing or delaying any potential offsite releases.

To achieve these objectives, the NRC used a five-step process to identify selected scenarios and model accident progression and mitigative measures, offsite release of radiological material, emergency response, and health effects. The results of the analyses consistently predict essentially zero individual early fatality risk for the modeled scenarios, very low long-term cancer fatality risks, and smaller radiological releases than those predicted from previous studies. The results also highlighted the importance of successful implementation of mitigative measures for preventing core damage. Additional UAs were conducted for these plants to help ensure the robustness of the models used and to identify key uncertain parameters that strongly influenced results. Importantly, the results from these studies support the results obtained from the deterministic evaluations of the three plants.

In addition to providing a more realistic estimate of severe accident consequences, the SOARCA project helped develop staff expertise in UA, accident progression and source term analysis, and consequence analysis. The project also led to improvements in NRC severe accident computer codes and accident analysis methodologies and provided detailed site- and plant-specific models ready for use by the NRC and other stakeholders for additional analyses. In accordance with the objectives of the project, the NRC has communicated safety aspects of severe accidents to its staff and external stakeholders by holding public meetings, presenting the project at internal seminars and external conferences, and publishing nine SOARCA NUREG-series publications, including a plain-language color brochure summarizing the project.

The NRC and external stakeholders have used SOARCA computer models, methodologies, and insights in several ways. For example, they were used to support risk-informed rulemaking efforts resulting from NTTF Report recommendations, which have enhanced safety and security, while also supporting operational flexibilities, of operating nuclear power plants in the U.S. Operating reactor licensees and advanced reactor designers have used SOARCA insights to support assumptions and methodologies in the respective renewal and design certification applications, and the NRC staff has used them in reviewing these applications. During and after the accident at Fukushima, SOARCA was used for emergency response and for domestic and international efforts to reconstruct the accident, assess severe accident modeling capabilities, and enhance offsite protective action decisionmaking. Other countries have leveraged SOARCA models and insights to improve their severe accident progression and consequence analysis capabilities. Ongoing risk-informed projects such as the Site Level 3 PRA project and NRC oversight programs have also benefited from risk insights leveraged by the SOARCA studies. Lastly, the up-to-date, detailed SOARCA computer models for Peach Bottom and Surry have been used as starting points for additional model development and to enhance real-time emergency response capabilities.

At least 325 conference papers, journal articles, presentations, and technical reports have used or cited the SOARCA project for scholarly research in diverse areas such as accident-tolerant fuels, advanced reactors, and dynamic PRA. The SOARCA project has proven to be useful in many ways beyond its original objectives and has been instrumental in ensuring the NRC staff has the technical capabilities to analyze the nuclear power safety issues of the future.

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28. U.S. Nuclear Regulatory Commission, NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," Washington, DC, September 2014. ADAMS Accession No. ML14255A365.

29. U.S. Nuclear Regulatory Commission, NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments," Washington, DC, March 2018. ADAMS Accession No. ML18065A048.

30. U.S. Nuclear Regulatory Commission, NUREG-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 1: Peach Bottom Integrated Analysis," Revision 1, Washington, DC, May 2013. ADAMS Accession No. ML13150A053.

31. U.S. Nuclear Regulatory Commission, NUREG-7110, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Volume 2: Surry Integrated Analysis," Revision 1, Washington, DC, August 2013. ADAMS Accession No. ML13240A242.

32. U.S. Nuclear Regulatory Commission, NUREG/BR-0359, Revision 1, "Modeling Potential Reactor Accident Consequences," Washington, DC, December 2012. ADAMS Accession No. ML12347A049.

33. U.S. Nuclear Regulatory Commission, NUREG/BR-0359, Revision 2, "Modeling Potential Reactor Accident Consequences," Washington, DC, April 2016. ADAMS Accession No. ML16083A086.

34. U.S. Nuclear Regulatory Commission, NUREG/BR-0524, "Cooperative Severe Accident Research Program (CSARP)," Washington, DC, November 2015. ADAMS ML15329A041.

35. U.S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," Sandia National Laboratories, Albuquerque, NM, December 1982. ADAMS Accession No. ML072320420.

36. U.S. Nuclear Regulatory Commission, NUREG/CR-7008, "MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," Sandia National Laboratories, Albuquerque, NM, August 2014. ADAMS Accession No. ML14234A136.

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38. U.S. Nuclear Regulatory Commission, NUREG/CR-7155, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," Sandia National Laboratories, Albuquerque, NM, May 2016. ADAMS Accession No. ML16133A461.

39. U.S. Nuclear Regulatory Commission, NUREG/CR-7160, “Emergency Preparedness Significance Quantification Process: Proof of Concept,” Sandia National Laboratories, Albuquerque, NM, June 2013. ADAMS Accession No. ML13164A285.

40. U.S. Nuclear Regulatory Commission, NUREG/CR-7245, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses,” Sandia National Laboratories, Albuquerque, NM, October 2019. ADAMS Accession No. ML19296B786.

41. U.S. Nuclear Regulatory Commission, Order EA-02-026, “Order for Interim Safeguards and Security Compensatory Measures,” Washington, DC, February 2002.

42. U.S. Nuclear Regulatory Commission, Order EA-13-109, “Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions,” Washington, DC, June 6, 2013. ADAMS Accession No. ML13143A321.

43. U.S. Nuclear Regulatory Commission, public meeting presentation, "QHO Margins: Progress and Path Forward,” Washington, DC, November 28, 2018. ADAMS Accession No. ML18331A363.

44. U.S. Nuclear Regulatory Commission, “Safety Goals tor the Operations of Nuclear Power Plants; Policy Statement; Republication,” *Federal Register*, Vol. 51, No. 162, August 4, 1986, pp. 30028-30033.

45. U.S. Nuclear Regulatory Commission, Safety Evaluation Report Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” Washington, DC, September 28, 2018. ADAMS Accession No. ML18215A239.

46. U.S. Nuclear Regulatory Commission, SECY-11-0093, Enclosure, “Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” Washington, DC, July 12, 2011. ADAMS Accession No. ML111861807.

47. U.S. Nuclear Regulatory Commission, SECY-12-0157, “Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments (REDACTED VERSION),” Washington, DC, November 26, 2012. ADAMS Accession No. ML12345A030.

48. U.S. Nuclear Regulatory Commission, SECY-12-0092, “State-of-the-Art Reactor Consequence Analyses—Recommendation for Limited Additional Analysis,” Washington, DC, July 5, 2012. ADAMS Accession No. ML12132A203.

49. U.S. Nuclear Regulatory Commission, SECY-13-0112, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor,” Washington, DC, October 9, 2013. ADAMS Accession No. ML13256A334.

50. U.S. Nuclear Regulatory Commission, SECY-15-0085, “Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26),” Washington, DC, June 18, 2015. ADAMS Accession No. ML15022A218.

51. U.S. Nuclear Regulatory Commission, SECY-15-0137, Enclosure 4, "Proposed Resolution Plans for Tier 3 Recommendations 5.2 and 6: Reliable Hardened Vents for Other Containment Designs and Hydrogen Control and Mitigation Inside Containment and Other Buildings," Washington, DC, October 29, 2015. ADAMS Accession No. ML15254A016.

52. U.S. Nuclear Regulatory Commission, SECY-16-0041, "Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation," Washington, DC, March 31, 2016. ADAMS Accession No. ML16049A088.

53. U.S. Nuclear Regulatory Commission, SRM-COMSECY-13-0030, "Staff Requirements—COMSECY-13-0030—Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," Washington, DC, May 23, 2014. ADAMS Accession No. ML14143A360.

54. U.S. Nuclear Regulatory Commission, SRM-SECY-12-0092, "Staff Requirements—State-of-the-Art Reactor Consequence Analyses (SOARCA)—Recommendation for Limited Additional Analysis," Washington, DC, December 6, 2012. ADAMS Accession No. ML12341A349.

55. U.S. Nuclear Regulatory Commission, SRM-SECY-12-0157, "Staff Requirements—SECY-12-0157—Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," Washington, DC, March 19, 2013. ADAMS Accession No. ML13078A017.

56. U.S. Nuclear Regulatory Commission, SRM-SECY-15-0085, "Staff Requirements—SECY-15-0085—Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26)," Washington, DC, August 19, 2015. ADAMS Accession No. ML15231A471.

57. U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station" (Draft Report), Sandia National Laboratories, Albuquerque, NM, August 2015. ADAMS Accession No. ML15224A001.

58. U.S. Nuclear Regulatory Commission, WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Washington, DC, October 1975.

59. Wald, M., "N.R.C. Lowers Estimate of How Many Would Die in Meltdown," *The New York Times*, New York, NY, July 29, 2011. Website accessed 10/14/2019.

APPENDIX A LIST OF TECHNICAL LITERATURE CITING THE SOARCA PROJECT

This appendix lists the results of an extensive literature search for citations of the SOARCA project. Analysts identified more than 325 citations by using the following search terms in Google, Primo, and other search engines: SOARCA, NUREG-1935, NUREG/BR-0359, NUREG/CR-7110, NUREG/CR-7008, NUREG/CR-7009, and NUREG/CR-7155. While the list is not exhaustive, it serves to illustrate the volume of citations and variety of SOARCA applications to research that were identified as of May 2019. The list groups the citations by topic to help the reader identify areas of interest, and links to each citation are provided where possible. To move to a subject of interest, click on the topic heading from the list below.

Research Topic
Severe Accident Progression Analysis
Offsite Consequence Analysis
Accident-Tolerant Fuel
Advanced Reactor Analysis
Emergency Preparedness and Response
Dynamic PRA
Multi-Unit PRA
Fukushima Forensic Analysis
Fukushima Lessons Learned
Severe Accident Management
Level 1 PRA
Nuclear Safety/Societal Risk
Sensitivity and Uncertainty Analysis
Storage and Transportation Safety

I. Severe Accident Progression Analysis

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Heat Up and Failure of BWR Upper Internals During a Severe Accident	Robb, K. R.	April 2017	Nuclear Engineering and Design, vol. 314, p.293-306
2	Terry Turbopump Analytical Modeling Efforts in Fiscal Year 2016 - Progress Report	Ross, K.; Cardoni, J. N.; Osborn, D.	April 2018	SAND2018-4337
3	Terry Turbopump Expanded Operating Band	Osborn, D.	July 2017	SAND2017-6715C
4	Consequences of Degraded Containment in a Severe Nuclear Power Plant Accident	Jankovsky, Z.; Jones, C.; Kalinich, D.	November 2014	Transactions of the American Nuclear Society, vol. 111, no. 1 (2014 ANS Winter Meeting)
5	MELCOR Code Source Term Characteristics for Fast SBO Scenario of OPR1000 Plant	Han, S. J. et al.	October 2012	Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, October 25-26, 2012
6	Effects of Source Term Characteristics on Off-Site Consequence	Han, S. J.; Ahn, K. I.	October 2012	Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, October 25-26, 2012
7	Development of the SharkFin Distribution for Fuel Lifetime Estimates in Severe Accident Codes	Denman, M. R.	November 2016	Transactions of the American Nuclear Society, vol. 115, no. 1 (2016 ANS Winter Meeting)
8	Simplified Method for Assessing the Risk Associated with Consequential Steam Generator Tube Rupture Events	Azarm, M. A. et al.	September 2013	ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis, Columbia, SC, September 22-26, 2013
9	Models and Methods Related to Severe Accidents and Source Term Evaluation	Gauntt, R. O.	April 2015	SAND2015-3101PE
10	Comparative Analysis on the Influence of the MAAP4 Phenomenological Model Parameters for the Severe Accident Source Term for Different Plant Designs and Accident Scenarios	Kang, S. W. and Yim, M.-S.	July 2018	Annals of Nuclear Energy, vol. 117, p. 98-108
11	MAAP-MELCOR Crosswalk Phase 1 Study	Luxat, D. L. et al.	December 2016	Nuclear Technology, vol. 196, p. 684-697

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
12	Accident Consequences and Analysis	Gauntt, R. O.	December 2017	International Atomic Energy Agency Workshop on Severe Accident Management Guidelines, December 11-15, 2017
13	Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis	Farmer, M. T. et al.	January 2015	ANL/NE-15/4
14	A Simple Assessment Scheme for Severe Accident Consequences using Release Parameters	Silva, K. and Okamoto, K.	August 2016	Nuclear Engineering and Design, vol. 305, p. 688-696
15	Station Blackout Mitigation Strategies Analysis for Maanshan PWR Plant using TRACE	Lin, H.-T. et al.	March 2016	Annals of Nuclear Energy, vol. 89, p. 1-18
16	External Flooding Event Analysis in a PWR-W with MAAP5	Fernandez-Cosials, M. K. et al.	February 2015	Annals of Nuclear Energy, vol. 76, p.226-236
17	A Reassessment of Low Probability Containment Failure Modes and Phenomena in a Long-Term Station Blackout	Brunett, A.; Denning, R.; Aldemir, T.	May 2014	Nuclear Technology, vol.186, p. 198-215
18	The Ultimate Response Guideline Simulation and Analysis using TRACE, MAAP5, and FRAPTRAN for the Chinshan Nuclear Power Plant	Wang, T.-C. et al.	May 2017	Annals of Nuclear Energy, vol. 103, p. 402-411
19	U.S. Nuclear Regulatory Commission's State-of-the-Art Reactor Consequence Analyses (SOARCA) Project	Osborn, D. M.	October 2017	SAND2017-10631PE
20	Conceptual Design Enhancement for Prevention and Mitigation of Severe Accidents	Song, J. H.	November 2014	Nuclear Technology, vol. 188, p. 113-122
21	Ongoing SNL International Severe Accident Activities	Andrews, N.C.	August 2018	SAND2017-9327R
22	Advanced Seismic Probabilistic Risk Assessment Methodology: Development of Beta 1.0 MASTODON Toolset	Bolisetti, C. et al.	August 2017	INL/EXT-17-43148

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
23	Thermal-Hydraulic Study of Air-Cooled Passive Decay Heat Removal System for APR+ under Extended Station Blackout	Kim, D. Y. et al.	February 2019	Nuclear Engineering and Technology, vol. 51, p. 60-72
24	Integral Analyses of Fission Product Retention at Mitigated Thermally-Induced SGTR using ARTIST Experimental Data	Rýdl, A.; Lind, T.; Birchley, J.	February 2016	Nuclear Engineering and Design, vol. 297, p. 175-187
25	An Analysis of Radiological Releases during a Station Black Out Accident for the APR1400	Vo, T. H.; Kim, D. H.; Song, J. H.	June 2018	Nuclear Engineering and Design, vol. 332, p.22-30
26	Estimating Safety Valve Stochastic Failure-to-Close Probabilities for the Purpose of Nuclear Reactor Severe Accident Analysis	Ghosh, S. T. et al.	July 2017	ASME/NRC 2017 13th Pump and Valve Symposium
27	Safety Research Opportunities Post-Fukushima: Initial Report of the Senior Expert Group	Organisation for Economic Co-operation and Development/ Nuclear Energy Agency	February 2017	NEA/CSNI/R(2016)19
28	SRV Modeling	Phillips, J.	August 2017	SAND2017-8365PE
29	Refined Boiling Water Reactor Station Blackout Simulation with RELAP-7	Zhao, H. et al.	September 2014	INL/EXT-14-33162
30	Developing Fully Coupled Dynamical Reactor Core Isolation System Models in RELAP-7 for Extended Station Black-Out Analysis	Zhao, H. et al.	April 2014	INL/CON-13-29971
31	RCIC Governing Equation Scoping Studies for Severe Accidents	Cardoni, J. N. and Ross, K.	August 2015	SAND2015-6996C
32	Demonstration of Fully Coupled Simplified Extended Station Black-out Accident Simulation with RELAP-7	Zhao, H. et al.	October 2014	INL/CON-13-30940
33	Post-Fukushima Critical Safety Equipment Evaluation	Osborn, D.	November 2015	SAND2015-10295C

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
34	A Strongly Coupled Reactor Core Isolation Cooling System Model for Extended Station Black-Out Analyses	Zhao, H. et al.	September 2015	16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-16), Chicago, USA
35	Modeling of the Reactor Core Isolation Cooling Response to Beyond Design Basis Operations - Interim Report	Ross, K. et al.	December 2015	SAND-2015-10662
36	Critical Equipment Performance	Osborn, D.	September 2015	SAND2015-8178PE
37	RCIC Governing Equation Scoping Studies	Cardoni, J. N.; Ross, K.; Gauntt, R. O.	August 2015	SAND2015-6995C
38	Probabilistic Economic Valuation of Safety Margin Management	Riley, T. et al.	July 2018	DOE contract NE0008295
39	SARNET Benchmark on Phébus FPT3 Integral Experiment on Core Degradation and Fission Product Behaviour	Di Giuli, M. et al.	July 2016	Annals of Nuclear Energy, vol. 93, p. 65-82
40	State-of-the-Art Report on Molten Corium Concrete Interaction and Ex-Vessel Molten Core Coolability	OCED/NEA	January 2017	NEA/CSNI/R(2016)15
41	Modeling of Water Ingression Mechanism for Corium Cooling with MELCOR	Sevón, T.	February 2017	Nuclear Technology, vol. 197, no. 2, p. 171-179
42	Exercises in Severe Accident Analysis using MELCOR: Accident Walkthrough	Gauntt, R. O.	April 2015	SAND2015-3100PE
43	Sensitivity Analysis of Debris Properties in Lower Plenum of a Nordic BWR	Galushin, S. and Kudinov, P.	June 2018	Nuclear Engineering and Design, vol. 332, p. 374-382
44	SOARCA Modeling	Phillips, J.	August 2017	SAND2017-8638PE
45	Nuclear Severe Accident Modeling and Analysis	Humphries, L. L.	November 2015	SAND2015-10296PE
46	Overview of Reactor Safety and Severe Accident Analysis Technologies at Sandia National Laboratories	Gauntt, R. O.	June 2007	SAND2007-4143C
47	Modeling of Aerosol Fission Product Scrubbing in Experiments and in Integral Severe Accident Scenarios	Rýdyl, A.; Fernandez-Moguel, L.; Lind, T.	May 2019	Nuclear Technology, vol. 205, no. 5, p. 655-670

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
48	Investigation of the Recriticality Potential during Reflooding Phase of Fukushima Daiichi Unit-3 Accident	Darnowski, P.; Potapczyk, K.; Świrski, K.	January 2017	Annals of Nuclear Energy, vol. 99, p. 495-509
49	Severe Accident Modeling for Cyber Scenarios	Cardoni, J.; Denman, M. Wheeler, T.	November 2016	Transactions of the American Nuclear Society, vol. 115, no. 1, p. 837-840
50	Updated Peach Bottom Model for MELCOR 1.8.6: Description and Comparisons	Robb, K. R.	September 2014	ORNL/TM-2014/207
51	Analisi di un Incidente Non Mitigato di Tipo LOCA in un Reattore PWR Mediante il Codice MELCOR 2.1 (Dissertation)	Pescarini, M.	June 2016	LM-DM270
52	MELCOR 2.1 Analysis of Melt Behavior in a BWR Lower Head during LOCA and SBO Accident	Li, G. et al.	April 2016	Annals of Nuclear Energy, vol. 90, p. 195-204
53	Heat Up and Potential Failure of BWR Upper Internals during a Severe Accident	Robb, K. R.	January 2015	OSTI ID 1213335
54	Enhanced MELCOR 2.1 Models for Standard PWR-Westinghouse Design/Modelo Detallado para un Diseño Estándar PWR-Westinghouse con el Código MELCOR 2.1	Ruiz Zapatero, M. et al.	March 2017	Nuclear Espana, vol. 382, p. 49-54
55	MELCOR Workshop OJP	Phillips, J. and Humphries, L.	August 2017	SAND2017-8243PE
56	Quicklook Overview of Model Changes in Melcor 2.2: Rev 6342 to Rev 9496	Humphries, L. L.	May 2017	SAND2017-5599
57	Modular Accident Analysis Program (MAAP) - MELCOR Crosswalk: Phase II Analyzing a Partially Recovered Accident Scenario	Andrews, N. et al.	October 2017	SAND2017-11975
58	Evaluation of JRC Source Term Methodology using MAAP5 as a Fast-Running Crisis Tool for a BWR4 Mark I Reactor	Vela-García, M. and Simola, K.	October 2016	Annals of Nuclear Energy, vol. 96, p. 446-454

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
59	Long Term SBO with Selected Mitigative Measures: MELCOR Parametric Calculations for a 2-Loop PWR	Rydl, A. et al.	July 2014	2014 22nd International Conference on Nuclear Engineering (ICONE22), paper no. ICONE22-30351, p. V006T15A012
60	Benchmarking Simulation of Long Term Station Blackout Events	Kim, S. K.; Lee, J. C.; Fynan, D. A.	May 2013	Proceedings of the Korean Nuclear Society 2013 spring meeting
61	The Role of Severe Accident Management in the Advancement of Level 2 PRA Modeling Techniques	Helton, D. et al.	October 2009	OECD/NEA Workshop on Implementation of Severe Accident Management Measures – Oct 2009 – Switzerland
62	LEVEL 2 PRA: Explicit and Seamless Approach (Example of a PWR with Large Dry Containment)	Azarm, M. A.	March 2016	n/a
63	Level-2/Level-3 Interface in a PSA Analysis	Bixler, N. E.	April 2015	SAND2015-3245PE
64	Development of a Fully-Coupled, All States, All Hazards Level 2 PSA at Leibstadt Nuclear Power Plant	Zvoncek, P.; Nusbaumer, O.; Torri, A.	March 2017	Nuclear Engineering and Technology, vol. 49, no. 2, p. 426-433
65	Overview of Revised Level 2 PRA Standard in Japan	Nakamura, K.; Narumiya, Y.; Abe, Y.	June 2016	ICONE24-61070
66	Focus Areas for a Level 2 PSA That Supports a Site NPP Risk Analysis	Helton, D. M.; Zavisca, M.; Khatib-Rahbar, M.	September 2014	Proceedings of the 2014 European Safety and Reliability Conference
67	Some International Efforts to Progress in the Harmonization of L2 PSA Development and Their Applications (European (ASAMPSA2), U.S. NRC, OECD-NEA and IAEA Activities)	Raimond, E. et al.	October 2009	OECD/NEA Workshop on Implementation of Severe Accident Management Measures – Oct 2009 – Switzerland
68	BWR MARK I Pressure Suppression Pool Mixing and Stratification Analysis using GOTHIC Lumped Parameter Modeling Methodology	Ozdemir, O. E. and George, T. L.	November 2015	Annals of Nuclear Energy, vol. 85, p. 532-543
69	In-Plant Fission Product Behavior in SGTR Accident with Long-Term SBO	Kim, T. W.; Han, S. J.; Ahn, K. I.	May 2015	Proceedings of the 2015 Korean National Society Spring Meeting

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
70	OECD/NEA/CSNI Status Report on Filtered Containment Venting	Jacquemain, D. et al.	July 2014	NEA-CSNI-R--2014-7
71	Assessment of Primary and Secondary Bleed and Feed Procedures during a Station Blackout in a German Konvoi PWR using ASTECV2.0	Gómez-García-Toraño, I. et al.	March 2018	Annals of Nuclear Energy, vol. 113, p. 476-492
72	Analysis of Flammability in the Attached Buildings to Containment under Severe Accident Conditions	de la Rosa, J. C. and Fornós, J.	November 2016	Nuclear Engineering and Design, vol. 308, p. 154-169
73	Risk-Informed External Hazards Analysis for Seismic and Flooding Phenomena for a Generic PWR	Parisi, C. et al.	July 2017	INL/EXT-17-42666
74	Total Loss of AC Power Analysis for EPR Reactor	Darnowski, P. et al.	August 2015	Nuclear Engineering and Design, vol. 289, p. 8-18
75	Development of Core Relocation Surrogate Model for Prediction of Debris Properties in Lower Plenum of a Nordic BWR	Galusin, S. et al.	October 2016	NUTHOS-11: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety, Gyeongju, Korea, October 9-13, 2016, Paper N11P1234
76	Ex-Vessel Core Melt Modeling Comparison between MELTSPREAD-CORQUENCH and MELCOR 2.1	Robb, K. R.; Farmer, M.; Francis, M. W.	March 2014	ORNL/TM--2014/1
77	HCVS-WP-02: Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping Revision 0	Nuclear Energy Institute	October 2014	ADAMS Accession No. ML14309A588
78	Effects of Degradation on the Severe Accident Consequences for a PWR Plant with a Reinforced Concrete Containment Vessel	Petti, J. P. et al.	June 2013	NUREG/CR-7149 ADAMS Accession No. ML13172A089
79	Effect of Cesium-Molybdate on Cs Behavior for Source Term Estimation	Han, S. J. and Ahn, K. I.	October 2013	Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, October 23-25, 2013
80	BWR Station Blackout: A RISMC Analysis using RAVEN and RELAP5-3D	Mandelli, D. et al.	January 2016	INL/JOU-14-33448

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
81	Influence of the Wet-Well Nodalization of a BWR3 Mark I on the Containment Thermal-Hydraulic Response during an SBO Accident	Herranz, L. E. et al.	December 2015	Nuclear Engineering and Design, vol. 295, p. 138-147
82	External Cooling of the BWR Mark I and II Drywell Head as a Potential Accident Mitigation Measure – Scoping Assessment	Robb, K. R.	August 2017	ORNL/TM-2017/457
83	Industry Guidance for Compliance with Order EA-13-109: BWR Mark I & II Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions	Nuclear Energy Institute	March 2015	NEI 13-02 [Rev. 0F4] ADAMS Accession No. ML15084A454
84	Benchmarking of Fast-running Software Tools Used to Model Releases During Nuclear Accidents	Nuclear Energy Agency	January 2016	NEA/CSNI/R(2015)19
85	ATLAS Program for Advanced Thermal-Hydraulic Safety Research	Song, C.-H.; Choi, K.-Y.; Kang, K.-H.	December 2015	Nuclear Engineering and Design, vol. 294, p. 242-261
86	Analysis of Primary Bleed and Feed Strategies for Selected SBLOCA Sequences in a German Konvoi PWR using ASTEC V2.0	Gómez-García-Toraño, I. et al.	December 2017	Annals of Nuclear Energy, vol. 110, p. 818-832
87	A Study on Fission Product Behavior during a Severe Accident at APR1400 Nuclear Power Plants	Kim, H.-C. and Cho, S.-W.	October 2015	Proceedings of the 2015 Fall meeting of the Korean Nuclear Society
88	A Study on Scenario Selection for Evaluation of Fission Product Behavior during a Severe Accident at APR 1400 Nuclear Power Plants	Yoon, E. S.; Kim, H.-C.; Cho, S.-W.	May 2015	Proceedings of the 2015 Spring meeting of the Korean Nuclear Society
89	Updated analysis of Fukushima Unit 3 with MELCOR 2.1. Part 2: Fission product release and transport analysis	Fernandez-Moguel, L.; Rydl, A.; Lind, T.	August 2019	Annals of Nuclear Energy, vol. 130, p. 93-106
90	An analysis on the consequences of a severe accident initiated steam generator tube rupture	Song, J.-H. et al.	July 2019	Nuclear Engineering and Design, vol. 348, p. 14-23
91	Development of MELCOR thermal hydraulic model of AP1000 and its verification for a DECL break	Malickia, M. et al.	June 2019	Annals of Nuclear Energy, vol. 128, p. 44-52

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
92	Analysis of the Effect of Severe Accident Scenario on Debris Properties in Lower Plenum of Nordic BWR Using Different Versions of MELCOR Code	Galushin, S. and Kudinov, P.	April 2019	Science and Technology of Nuclear Installations, vol. 2019, Article ID 5310808, 18 pages
93	Effect of Molten Corium Behavior Uncertainty on the Severe Accident Progress	Choi, W. et al.	August 2018	Science and Technology of Nuclear Installations, vol. 2018, Article ID 5706409, 9 pages
94	Analysis of PWR SBO sequences with RCP passive thermal shutdown seals	Mena-Rosell, L. et al.	June 2018	Journal of Nuclear Science and Technology, vol. 55, no. 6, p. 649-662
95	Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes	Sancaktar, S. et al.	May 2018	ADAMS Accession No. ML18122A012
96	MELCOR Code Development Status - EMUG 2018	Humphries, L. L.	April 2018	European MELCOR User Group Meeting, Zagreb, Croatia, April 18-20, 2018
97	Severe Accident Context Quantification for Long-Term Station Blackout in Boiling Water Reactor Nuclear Power Plants	Petkov, G. I. and Vela-Garcia, M.	March 2018	ASME Journal of Nuclear Engineering and Radiation Science, vol. 4, no. 2, 020913
98	Tools and Methods for Assessing the Risk Associated with CSGTR	Azarm, M. A. and Sancaktar, S.	March 2018	ADAMS Accession No. ML18074A025
99	Severe Accident Phenomena	Wagner, K. C.	March 2018	PWR Owners Group SAMG Meeting, Denver, CO, United States, March 27-28, 2018
100	MELCOR Best Practices in SOARCA AMUG 2017	Humphries, L.	November 2017	SAND2017-12028C
101	IVR Phenomena Modeling of Lower Plenum and Modifications of RN Package (1.8.6 to 2.2)	Humphries, L. L.	November 2017	Asian MELCOR User Group Meeting, Daejeon, South Korea, November 5-8, 2017
102	MELCOR Code Development Status MCAP 2017	Humphries, L. L.	October 2017	MELCOR Code Assessment Program Meeting, Bethesda, MD, USA, September 14-15, 2017

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
103	Terry Turbopump Expanded Operating Band Program Experimental and Modeling Efforts	Osborn, D.	September 2017	SAND2017-10566PE
104	An Improvement of Estimation Method of Source Term to the Environment for Interfacing System LOCA for Typical PWR Using MELCOR code	Han, S.-J.; Kim, T.-W.; Ahn, K.-Il.	June 2017	Journal of Radiation Protection and Research, vol. 42, no. 2, p. 106-113
105	Analysis of unmitigated large break loss of coolant accidents using MELCOR code	Pescarini, M. et al.	June 2017	Journal of Physics: Conference Series, vol. 923, Conference 1

II. Offsite Consequence Analysis

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Article, "NRC Keeps Its Focus on Safety," in "Meetings" section	n/a	May 2008	Nuclear News, vol. 51, no. 6 p. 53
2	A Sensitivity Study on Population Segmentation Effects in the MACCS Code	Song, W. et al.	June 2018	Transactions of the American Nuclear Society, vol. 118, no. 1, p.643-646
3	Article, “Assessing SOARCA,” in “Meetings” section	n/a	January 2013	Nuclear News, vol. 56, no. 1, p. 59
4	Nuclear Energy Related Capabilities at Sandia National Laboratories	Pickering, S. Y.	February 2014	SAND2014-1345
5	Technical Basis Development for Filtered Containment Venting System Requirements	Basu, S. et al.	November 2013	Transactions of the American Nuclear Society, vol. 109, no. 1, p. 971-974
6	MELCOR/MACCS2 Analysis for BWR Mark I Filtered Containment Venting	Osborn, D. M. et al.	November 2013	Transactions of the American Nuclear Society, vol. 109, no. 1, 2103-2106
7	MACCS-HYSPLIT Atmospheric Transport and Dispersion Model Benchmarking	Clayton, D. J.	September 2017	SAND2017-9714PE
8	MACCS-HYSPLIT Atmospheric Transport and Dispersion Model Benchmarking	Clayton, D. J.; Bixler, N. E.; Compton, K.	June 2017	SAND2017-6184C
9	Software Regression Quality Assurance for MACCS2: Version 2.5.0.0 through Version 2.5.0.9	Eubanks, L. L. et al.	July 2012	SAND2012-6333
10	Effect on the Offsite Consequence of the MACCS Non-Site-Specific Best Modeling Practices Used in the US SOARCA Project	Jin, D.-S.; Han, S.-K.; Lim, C.-K.	May 2017	Transactions of the Korean Nuclear Society Spring Meeting, May 2017, Jeju, Korea
11	Atmospheric Transport and Dose Calculations: Concepts and Implementation in the MACCS Code	Bixler, N. E.	July 2015	SAND2015-5545PE
12	Level-3 Consequence Analysis Part 1 Atmospheric Transport and Dispersion	Bixler, N. E.	April 2015	SAND2015-3244PE

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
13	Emergency Planning Zones Estimation for Karachi-2 and Karachi-3 Nuclear Power Plants using Gaussian Puff Model	Şahin, S. and Ali, M.	October 2016	Science and Technology of Nuclear Installations, vol. 2016
14	Avoidable Dose and Total Dose Radiological Assessments in Support of Public Protection Decisions	Kraus, T. D.	June 2013	SAND2013-5158C
15	Importance of Accounting for the Partitioning of Iodine Released During Nuclear Power Plant Accidents	Kraus, T. D.	June 2013	SAND2013-4833C
16	MACCS2 Consequence Analysis for BWR Mark I and Mark II Filtered Containment Venting	Osborn, D. M. et al.	October 2012	SAND2012-9533
17	Error Analysis of CM Data Products Input Distributions	Hunt, B. D. et al.	March 2017	SAND2017-3298R
18	Effects of Source Term on Off-site Consequence in LOCA Sequence in a Typical PWR	Han, S.-J.; Kim, T.-W.; Ahn, K.-I.	June 2014	Probabilistic Safety Assessment and Management PSAM 12, June 2014, Honolulu, Hawaii
19	Article, “The NRC Staff is Calling for Additional SOARCA Studies," in “Late News" section	n/a	August 2012	Nuclear News, vol. 55, no. 9, p. 175
20	Article, “Reactor accident study finds ‘essentially zero’ fatalities,” in “Power” section	n/a	March 2012	Nuclear News, vol. 55, no. 3, p. 25-26
21	Best-Estimate Calculations of Unmitigated Severe Accidents in State-of-the-Art Reactor Consequence Analyses	Schaperow, J. H. et al.	October 2009	OECD/NEA Workshop on Implementation of Severe Accident Management Measures – Oct 2009 – Switzerland
22	Consequence Analysis for Nuclear Reactors, Yongbyon	Kang, T. and Jae, M.	February 2017	Journal of Nuclear Science and Technology, vol. 54, no. 2, p. 223-232
23	Applicability of 100 TBq Cesium 137 Release into Environment as a Safety Criterion for Consequence Assessment at Reactor Design Approval Stage	Silva, K. and Okamoto, K.	December 2015	Journal of Nuclear Science and Technology, vol. 52, no. 12, p. 1530-1539

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
24	Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses	Schaperow, J. H. et al.	October 2009	OECD/NEA Workshop on Implementation of Severe Accident Management Measures – Oct 2009 – Switzerland
25	Risk and regulatory considerations for small modular reactor emergency planning zones based on passive decontamination potential	Carless, T. S.; Talabi, S. M.; Fischbeck, P. S.	January 2019	Energy, vol. 167, p. 740-756
26	Status of Practice for Level 3 Probabilistic Safety Assessments	OECD/NEA	November 2018	NEA/CSNI/R(2018)1
27	Response Technical Tools User Guide	Andrews, N.; Fu, C.; Kaberlein, A. M.	August 2018	SAND2018-9593
28	Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments	Barr, J. A. et al.	March 2018	NUREG-2206 ADAMS Accession No. ML18065A048
29	U.S. Nuclear Regulatory Commission's State-of-the-Art Reactor Consequence Analyses (SOARCA) Project	Osborn, D. M.	October 2017	4th Arab Forum on the Prospects of Nuclear Power Meeting, Amman, Jordan, October 9-12, 2017
30	Recommendations for Future Research on Nuclear Accident Consequence Analysis	Kampanart, S. and Vechgama, W.	July 2017	International Conference on Risk Analysis, Decision Analysis, and Security, Tsinghua University, Beijing, China, July 21-23, 2017
31	MELCOR Accident Code System (MACCS)	Sharp, A. E.	April 2017	NUREG/BR-0527 ADAMS Accession No. ML17102A912
32	Dose assessment in level 3 PRA - a rewiew of recently used methods	Karanta, I.	February 2017	VTT-R-00738-17
33	A Study of Base Technology of Korean Specific Level 3 PSA Code Development	Han, S. et al.	February 2017	KAERI/RR--4232/2016

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
34	Probabilistic risk assessment from potential exposures to the public applied for innovative nuclear installations	Dvorzhak, A.; Mora, J. C.; Robles, B.	August 2016	Reliability Engineering and System Safety, vol.152, p.176-186

III. Accident-Tolerant Fuel

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Accident Tolerant Fuels (ATF) Coating and Cladding Thermal Hydraulic Properties Evaluation by MELCOR YU 1.8.6: Benchmark for SURRY Short Term Station Black Out	Wang, J.; Corradini, M. L.; Jo, H.	June 2018	Transactions of the American Nuclear Society, vol. 118, no. 1, p. 1259-1262
2	Potential Recovery Actions from a Severe Accident in a PWR: MELCOR Analysis of a Station Blackout Scenario	Wang, J.; Jo, H.J.; Corradini, M. L.	October 2018	Nuclear Technology, vol. 204, p. 1-14
3	Estimation of Core Recovery Time with TRACE	Gurgen, A. and Shirvan, K.	July 2017	Transactions of the American Nuclear Society, vol. 116, no.1, p. 422-425
4	State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels	OECD/NEA	October 2018	NEA No. 7317
5	Estimation of Coping Time in Pressurized Water Reactors for Near Term Accident Tolerant Fuel Claddings	Gurgen, A. and Shirvan, K.	October 2018	Nuclear Engineering and Design, vol.337, p. 38-50
6	Development of Accident Tolerant Fuel Options for Near Term Applications: Fuel Performance Modeling under Transient/Severe Accidents by MELCOR: PART I: Benchmark for SURRY: Short Term Station Black Out (STSBO)	Wang, J.; Corradini, M. L.; Haskin, T.	November 2016	Transactions of the American Nuclear Society, vol. 115, no. 1, 1799-1802
7	Severe Accident Scoping Simulations of Accident Tolerant Fuel Concepts for BWRs	Robb, K. R.	August 2015	ORNL/SPR-2015/347
8	Accident Tolerant Clad Material Modeling by MELCOR: Benchmark for SURRY Short Term Station Black Out	Wang, J. et al.	March 2017	Nuclear Engineering and Design, vol. 313, p. 458-469
9	Accident-Tolerant Fuel Valuation: Safety and Economic Benefits	Hess, S. et al.	March 2019	EPRI TR-3002015091 (not publicly available)
10	Analysis of the FeCrAl Accident Tolerant Fuel Concept Benefits during BWR Station Blackout Accidents	Robb, K. R.	January 2015	16th International Topical Meeting on Nuclear Reactor Thermalhydraulics, Chicago, IL, USA

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
11	System Code Evaluation of Accident Tolerant Claddings During BWR Station Blackout Accident	Wu, X. and Shirvan, K.	November 2018	Transactions of the American Nuclear Society, vol. 119, no. 1, p. 444-447
12	Plant-Level Scenario-Based Risk Analysis for Enhanced Resilient PWR - SBO and LBLOCA	Ma, Z. et al.	September 2018	INL/EXT-18-51436-Rev000
13	State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels	OECD/NEA	January 2018	NEA No. 7317
14	Status Report on Activities of the Systems Assessment Task Force, OECD-NEA Expert Group on Accident Tolerant Fuels for LWRs	Bragg-Sitton, S. M.	September 2017	INL/EXT-17-43389-Rev000
15	Accident tolerant clad material modeling by MELCOR: Benchmark for SURRY Short Term Station Black Out	Wang, J. et al.	March 2017	Nuclear Engineering and Design, vol. 313, p. 458-469

IV. Advanced Reactor Analysis

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Risk Management for Sodium Fast Reactors	Denman, M. R. et al.	January 2015	SAND2015-0542
2	Regulatory Technology Development Plan - Sodium Fast Reactor: Mechanistic Source Term – Trial Calculation	Grabaskas, D. et al.	October 2016	ANL-ART-49-Vol1-Vol2
3	Overview of Key Computer Codes for the PGSFR Safety Analysis	Chang, W.-P.; Lee K.-L.; Yoo, J.	October 2016	Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, October 26-28, 2016
4	Advanced Small Modular Reactor (SMR) Probabilistic Risk Assessment (PRA) Technical Exchange Meeting	Smith, C.	September 2013	INL/EXT-13-30170
5	Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites	NuScale Power	December 2015	NuScale TR-0915-17772-NP ADAMS Accession No. ML15356A842
6	Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone (White Paper)	Nuclear Energy Institute	December 2013	ADAMS Accession No. ML13364A345
7	Advance Liquid Metal Reactor Discrete Dynamic Event Tree/Bayesian Network Analysis and Incident Management Guidelines (Risk Management for Sodium Fast Reactors)	Denman, M. R. et al.	April 2015	SAND2015--2484
8	Feasibility and Safety Assessment for Advanced Reactor Concepts Using Vented Fuel	Klein, A. et al.	January 2015	DOE/NEUP-11-3097 OSU-NE-VF-1204
9	A Methodology for Accident Analysis of Fusion Breeder Blankets and Its Application to Helium-Cooled Lead-Lithium Blanket	Panayotov, D. et al.	October 2016	IEEE Transactions on Plasma Science, vol. 44, no. 10, p. 2511-2522
10	Design and Licensing Strategies for the Fluoride-Salt-Cooled, High-Temperature Reactor (FHR) Technology	Scarlat, R. O. et al.	November 2014	Progress in Nuclear Energy, vol. 77, p. 406-420

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
11	Fluoride-Salt-Cooled, High-Temperature Reactor (FHR) Subsystems Definition, Functional Requirement Definition, and Licensing Basis Event (LBE) Identification White Paper	Allen, T.R. et al.	August 2013	n/a
12	Regulatory cross-cutting topics for fuel cycle facilities	Denman, M. R. et al.	October 2013	SAND2013-9367
13	Advanced Reactor Safety Program – Stakeholder Interaction and Feedback	Szilard, R. H. and Smith, C. L.	August 2014	INL/EXT-14-32928
14	Final White Paper on Small Advanced Reactor Design Reviews Rev 7 2017/05/08	Holahan, G. M.	May 2017	ADAMS Accession No. ML17213A849 (not publicly available)
15	Mechanistic Source Term Modeling for Sodium Fast Reactors	Clark, A. J.; Denman, M. R.; Grabaskas, D.	January 2017	SAND2017-0454C

V. Emergency Preparedness and Response

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	RASCAL 4.3	Kowalczik, J.	April 2014	ADAMS Accession No. ML14112A349
2	RASCAL 4.3: Description of Models and Methods	Ramsdell, Jr, J.V.; Athey, G. F.; Rishel, J. P.	May 2015	NUREG-1940, Supplement 1
3	Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor. Date Effective: May 2013	International Atomic Energy Agency, Incident and Emergency Centre, Vienna (Austria)	May 2013	EPR-NPP-PPA-2013
4	Operational Intervention Levels for Reactor Emergencies and Methodology for Their Derivation. March 2017	International Atomic Energy Agency, Incident and Emergency Centre	March 2017	EPR-NPP-OILS-2017
5	Alternative evacuation strategies for nuclear power accidents	Hammond, G. D. and Bier, V. M.	March 2015	Reliability Engineering & System Safety, vol. 135, p. 9-14
6	Significance Quantification Process for Emergency Preparedness Oversight	Jones, J. A.	January 2014	SAND2014-0677C
7	Emergency Preparedness Significance Quantification Process: Proof of Concept	Sullivan, R. et al.	June 2013	INL/LTD-12-27648, Rev. 1
8	Emergency Preparedness	Osborn, D.	June 2015	SAND2015-4657PE
9	Inspection of Emergency Arrangements	OECD/NEA Committee on Nuclear Regulatory Activities WGIP	December 2013	NEA-CNRA-R-2013-2
10	Emergency Preparedness. NRC Needs to Better Understand Likely Public Response to Radiological Incidents at Nuclear Power Plants	Government Accountability Office	March 2013	GAO-13-243
11	Rethinking Nuclear Emergency Planning, Preparations, and Response	Goble, R. and Bier, V.	November 2013	Transactions of the American Nuclear Society vol. 109 no. 1, p. 1959-1961

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
12	A new approach to quantify safety benefits of disaster robots	Kim, I. S.; Choi, Y.; Jeong, K. M.	October 2017	Nuclear Engineering and Technology, vol. 49, no. 7 p. 1414-1422
13	The Risk of Extended Power Loss and the Probability of Emergency Restoration for Severe Events and Nuclear Accidents	Duffey, R. B.	May 2019	ASME Journal of Nuclear Engineering and Radiation Science, vol. 5, no. 3, 031601
14	About the development of a national system of response to the nuclear emergency for agriculture	Mikailova, R. A. and Shubina, O. A.	March 2019	IOP Conference Series: Material Science and Engineering 487 012013
15	GIS-Based Integration of Social Vulnerability and Level 3 Probabilistic Risk Assessment to Advance Emergency Preparedness, Planning, and Response for Severe Nuclear Power Plant Accidents	Pence, J. et al.	November 2018	Risk Analysis, vol. 39, no. 6, p. 1262-1280

VI. Dynamic PRA

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Risk-Informed Safety Margin Characterization Methods Development Work	Smith, C. L. et al.	September 2014	INL/EXT-14-33191
2	System Theoretic Frameworks for Mitigating Risk Complexity in the Nuclear Fuel Cycle	Williams, A. D. et al.	September 2017	SAND2017-10243
3	Safety Relief Valve Cyclic Failure Analysis for Use in Discrete Dynamic Event Trees	Denman, M. R.	May 2013	SAND2013-3684C
4	Seamless Level 2/Level 3 Probabilistic Risk Assessment Using Dynamic Event Tree Analysis (Dissertation)	Osborn, D. M.	January 2013	n/a
5	Development and Application of a Dynamic Level 1 and 2 Probabilistic Safety Assessment Tool	LaChance, J. L. et al.	January 2012	SAND2012-0651C
6	Nuclear Power Plant Cyber Security Discrete Dynamic Event Tree Analysis (LDRD 17-0958) FY17 Report	Wheeler, T. A. et al.	September 2017	SAND2017-10307
7	Dynamical systems probabilistic risk assessment	Denman, M. R. and Ames, A. L.	March 2014	SAND2014-4037
8	Safety Relief Valve Cyclic Failure Analysis for use in Discrete Dynamic Event Trees (slidedeck)	Denman, M. R.	September 2013	SAND2013-8217C
9	Discrete Dynamic Event Tree Analysis of Small Modular Reactor Severe Accident Management (slidedeck)	Denman, M. R. et al.	September 2013	SAND2013-8096C
10	Discrete Dynamic Event Tree Analysis of Small Modular Reactor Severe Accident Management	Denman, M. R. et al.	May 2013	SAND2013-3680C
11	The Assessment of Low Probability Containment Failure Modes using Dynamic PRA (Dissertation)	Brunett, A. J.	January 2013	n/a

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
12	Discrete Dynamic Probabilistic Risk Assessment Model Development and Application	LaChance, J. et al.	October 2012	SAND2012-9346 ADAMS Accession No. ML12305A351
13	An Assessment of Low Probability Containment Failure in a Long-Term Station Blackout Using Dynamic PRA	Brunett, A.; Denning, R.; Aldemir, T.	November 2013	Transactions of the American Nuclear Society, vol. 109, no. 1, 954-957
14	Preliminary Cyber-Informed Dynamic Branch Conditions for Analysis with the Dynamic Simplified Cyber MELCOR Model	Denman, M. R. et al.	November 2016	Transactions of the American Nuclear Society, vol. 115, no. 1, 787-790
15	How to ADAPT	Jankovsky, Z. K.; Haskin, T. C.; Denman, M. R.	June 2018	SAND2018-6660
16	Uncertainty Quantification for External Events Analysis of LWRS/RISMC Project	Parisi, C. et. al.	June 2017	Transactions of the American Nuclear Society, vol. 116, no. 1, 795-797
17	Quantitative risk reduction by means of recovery strategies	París, C. et al.	February 2019	Reliability Engineering & System Safety Meeting, vol. 182, p. 13-32
18	Level 2 Probabilistic Risk Assessment Using Dynamic Event Tree Analysis	Osborn, D. et al.	June 2018	Chapter 5 of Advanced Concepts in Nuclear Energy Risk Assessment and Management
19	Development of Computational and Data Processing Tools for ADAPT to Assist Dynamic Probabilistic Risk Assessment	Jankovsky, Z. K.	January 2018	Dissertation
20	A Dynamic Assessment of an Interfacing System Loss of Coolant Accident	Jankovsky, Z. K. ; Denman, M. R. ; Aldemir, T.	September 2017	SAND2017-10141C
21	A Dynamic Assessment of Auxiliary Building Contamination and Failure Due to a Cyber-Induced Interfacing System Loss of Coolant Accident	Jankovsky, Z. ; Denman, M. R. ; Aldemir, T.	June 2017	International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants, VIC, Vienna, 6-9 June 2017

VII. Multi-Unit PRA

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Construction of Multi-Path Event Tree for Station Blackout Events	Kim, S. et al.	October 2017	Transactions of the American Nuclear Society, vol. 117, no. 1, p. 951-953
2	Multi-Unit Level 3 Probabilistic Safety Assessment: Approaches and Their Application to a Six-Unit Nuclear Power Plant Site	Kim, S.-Y. et al.	December 2018	Nuclear Engineering and Technology, vol. 50, no. 8, p. 1246-1254
3	Multi-Unit Accident Contributions to Quantitative Health Objectives: A Safety Goal Policy Analysis	Hudson, D. W. and Modarres, M.	March 2017	Nuclear Technology, vol. 197, no. 3, p. 227-247
4	Holistic Approach to Multi-Unit Site Risk Assessment: Status and Issues	Kim, I. S.; Jang, M.; Kim, S. R.	March 2017	Nuclear Engineering and Technology, vol. 49, no. 2, p. 286-294
5	Initiating Events for Multi-Reactor Plant Sites	Muhlheim, M. D.; Flanagan, G. F.; Poore, III, W. P.	September 2014	ORNL/TM-2014/533
6	Scoping Estimates of Multiunit Accident Risk	Stutzke, M. A.	June 2014	Probabilistic Safety Assessment and Management PSAM 12, Honolulu, Hawaii, June 2014
7	A Methodology for Performing Consequence Analysis for Multi-Unit/Spent Fuel Pool Source Terms	Bixler, N. E.	November 2014	SAND2014-20038C
8	Multi-Unit Accident Contributions to US Nuclear Regulatory Commission Quantitative Health Objectives: A Safety Goal Policy Analysis using Models from State-of-the-Art Reactor Consequence Analyses (Dissertation)	Hudson, D. W.	January 2016	n/a
9	Multi-Unit Dynamic PRA	Mandelli, D. et al.	May 2019	Reliability Engineering & System Safety, vol. 185, p. 303-317
10	Multi-Unit Level 3 Probabilistic Safety Assessment: Approaches and Their Application to a Six-Unit Nuclear Power Plant Site	Kim, S.-Y. et al.	December 2018	Nuclear Engineering and Technology, vol. 50, no. 8, p. 1246-1254

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
11	The Current Research Status and Technical Development Framework of Multi-Reactor Probabilistic Consequence Assessment	Ding, H. et al.	July 2017	25th International Conference on Nuclear Engineering (ICONE25), Shanghai, China, July 2-6, 2017

VIII. Fukushima Forensic Analysis

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	MELCOR Applications to SOARCA and Fukushima	Gauntt, R. O.	March 2014	SAND2014-1879C
2	A MELCOR Model of Fukushima Daiichi Unit 1 Accident	Sevón, T.	November 2015	Annals of Nuclear Energy, vol. 85, p. 1-11
3	A MELCOR Model of Fukushima Daiichi Unit 3 Accident	Sevón, T.	April 2015	Nuclear Engineering and Design, vol. 284, p. 80-90
4	A Review of Recent SNL MELCOR Fukushima Accident Analyses	Kalinich, D. A.	March 2015	SAND2015-2172PE
5	Overview of Sandia National Laboratories MELCOR Fukushima Analyses	Kalinich, D. A.	July 2015	SAND2015-6035PE
6	Fukushima Daiichi Unit 1 Accident Progression Uncertainty Analysis and Implications for Decommissioning of Fukushima Reactors - Volume I	Gauntt, R. O. and Mattie, P. D.	January 2016	SAND2016-0232
7	Insight from Fukushima Daiichi Unit 3 Investigations Using MELCOR	Robb, K. R.; Francis, M. W.; Ott, L. J.	May 2014	Nuclear Technology, vol. 186, no. 2, p. 145-160
8	MELCOR Simulations of the Severe Accident at the Fukushima Daiichi Unit 1 Reactor	Gauntt, R. et al.	May 2014	Nuclear Technology, vol. 186, no. 2, p. 161-178
9	MELCOR Simulations of the Severe Accident at Fukushima Daiichi Unit 3	Cardoni, J. et al.	May 2014	Nuclear Technology, vol. 186, no. 2, p. 179-197
10	Sensitivity Study of 1F1 Type Accident by MELCOR Code	Saito, K. and Yamaji, A.	October 2015	Transactions of the American Nuclear Society, vol. 113, no. 1, p. 1411-1414
11	Fukushima Daiichi unit 1 Uncertainty Analysis--Preliminary Selection of Uncertain Parameters and Analysis Methodology	Cardoni, J. N. and Kalinich, D. A.	February 2014	SAND2014-1170
12	Seismically-Induced Reactor Coolant Leakage as an Allegedly-Possible Cause of Accident at Unit 1 of Fukushima Daiichi Nuclear Power Station	Kukita, Y. and Watanabe, N.	November 2014	JAEA-TECHNOLOGY--2014-036
13	MELCOR 2.1 Simulations of Fukushima Unit 3	Cardoni, J. N.	November 2011	SAND2012-9742C

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
14	Severe Accident Progression Analyses in SOARCA and Comparisons to Fukushima	Gauntt, R. O.	July 2013	SAND2013-6058C
15	Fukushima Daiichi - A Case Study for BWR Instrumentation and Control Systems Performance during a Severe Accident Rev 0	Clayton, D. A. and Poore, III, W. P.	April 2013	ORNL/TM-2013/154
16	Fukushima Daiichi Nuclear Plant Accident: Atmospheric and Oceanic Impacts over the Five Years	Hirose, K.	June 2016	Journal of Environmental Radioactivity, vol. 157, p. 113-130
17	Analysis of the Accident in the Fukushima Daiichi Nuclear Power Station Unit 3 with MELCOR_2.1	Fernandez-Moguel, L. and Birchley, J.	September 2015	Annals of Nuclear Energy, vol. 83, p. 193-215
18	Fukushima Daiichi - A Case Study for BWR Instrumentation and Control Systems Performance during a Severe Accident Rev 1	Clayton, D. A. and Poore, III, W. P.	June 2014	ORNL/TM-2013/154 R1
19	ATHLET-CD/COCOSYS Analyses of Severe Accidents in Fukushima Daiichi Units 2 and 3: German Contribution to the OECD/NEA BSAF Project, Phase 1	Sonnenkalb, M. and Band, S.	November 2016	Nuclear Technology, vol. 196, no. 2, p. 211-222
20	Fukushima Daiichi Unit 1 Uncertainty Analysis-Exploration of Core Melt Progression Uncertain Parameters-Volume II	Denman, M. R. and Brooks, D. M.	August 2015	SAND2015-6612
21	Presentation of Fukushima Analyses to U.S. Nuclear Power Plant Simulator Operators and Vendors	Osborn, D.; Kalinich, D. A.; Cardoni, J. N.	February 2015	SAND2015-1177R
22	Insights Gained from Forensic Analysis with MELCOR of the Fukushima-Daiichi Accidents	Andrews, N. C.; Gauntt, R. O.	October 2017	SAND2017-10811R
23	Fukushima Daiichi Accident Study: Status as of April 2012	Gauntt, R. O. et al.	July 2012	SAND2012-6173
24	Historical Overview of Fukushima Forensics Work	Kelly, J. E.	March 2018	ADAMS Accession No. ML18090A006

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
25	Team Performance Comparison in Core-Melt Units of Fukushima Daiichi NPS Based on Dynamic Context Quantification of Accident	Petkov, G. and Petkov, I.	June 2017	PSAM Topical Conference on Human Reliability, Quantitative Human Factors, and Risk Management, Munich, Germany, June 2017
26	Studies on the Recriticality Potential during Fukushima Unit-3 Core Reflooding	Darnowskia, P.; Potpaczyka, K.; Świrski, K.	May 2017	The 8th European Review Meeting on Severe Accident Research - ERMSAR-2017 Warsaw, Poland, 16-18 May 2017
27	Fukushima Reactor Building Model	Andrews, N. et al.	January 2017	4th Meeting of the OECD/NEA BSAF Project Phase 2, Paris, France, January 9-13, 2017

IX. Fukushima Lessons Learned

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Nuclear Power Plant Severe Accidents	Osborn, D.	March 2015	SAND2015-1591PE
2	Health Physics: Radiation-Generating Devices, Characteristics, and Hazards (Ch. 7: Regulatory Considerations)	Bevelacqua, J. J.	April 2016	n/a
3	Nuclear Regulation in the United States and a Possible Framework for an International Regulatory Approach	Bevelacqua, J. J.	February 2013	International Nuclear Safety Journal, vol. 2, no. 1, p. 52-57
4	Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants	National Research Council	January 2014	Contract No. NRC-HQ-12-G-03-0002
5	Lessons Learned from the Fukushima Accident for Improving Safety and Security of U.S. Nuclear Plants. Phase 2	National Research Council	January 2016	Contract No. NRC-HQ-12-G-03-0002
6	The Nuclear Regulatory Commission and NEPA Review	Eccleston, C. H.	September 2012	Environmental Quality Management, vol. 22, no.1, p. 43-58
7	Fukushima Lessons Learned	Gauntt, R. O.	February 2012	SAND2012-0893C
8	The Canary, the Ostrich, and the Black Swan: A Historical Perspective on our Understanding of BWR Severe Accidents and Their Mitigation	Greene, S. R.	May 2014	Nuclear Technology, vol. 186, no. 2, p. 115-138
9	SNL BSAF Phase 2 Activities	Andrews, N. et al.	December 2015	SAND2015-10448PE

X. Severe Accident Management

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario	Rempe, J. L.; Knudson, D. L.; Lutz, R. J.	September 2015	INL/EXT-15-35940
2	Technical Basis for Severe Accident Mitigating Strategies: Volume 1	Duvall, A. et al.	April 2015	EPRI 3002003301
3	Verification of SAMGs in SBO Sequences with Seal LOCA. Multiple Damage Domains	Queral, C. et al.	December 2016	Annals of Nuclear Energy, vol. 98, p. 90-111
4	Preliminary Assessment for the Mitigative Effectiveness of External Injection during Extended SBO	Park, S. Y. and Ahn, K. I.	October 2013	Proceedings of the 2013 fall Korean Nuclear Society meeting
5	The Relative Importance of Mitigation, Early Phase, Intermediate Phase, and Late Phase Response	Denning, R. et al.	November 2013	Transactions of the American Nuclear Society, vol. 109, no. 1
6	External Cooling of the BWR Mark I and II Drywell Head as a Potential Accident Mitigation Measure – Expanded Scoping Assessment	Robb, K. R.	July 2018	ORNL/TM-2018/901
7	Key Parameters for Operator Diagnosis of BWR Plant Condition during a Severe Accident	Clayton, D. A. and Poore, III, W. P.	January 2015	ORNL/LTR-2014/320
8	Evaluation of an Accident Management Strategy of Emergency Water Injection using Fire Engines in a Typical Pressurized Water Reactor	Park, S.-Y. and Ahn, K.-I.	October 2015	Nuclear Engineering and Technology, vol. 47, no. 6, p. 719-728
9	Post-Severe Accident Environmental Conditions for Essential Instrumentation for Boiling Water Reactors	Clayton, D. and Poore, M.	September 2015	ORNL/TM-2015/278
10	Regulatory Perspective and Accident Management Procedure Influences on Accident Monitoring Instrumentation Design Criteria	Rahn, D. L. and Cowdrey, C. B.	September 2012	ADAMS Accession No. ML12243A196 IAEA/JNESO Workshop on Accident Monitoring Instrumentation, Tokyo, Japan, September 4, 2012

11	Dealing with Beyond-Design-Basis Accidents in Nuclear Safety Decisions	Nourbakhsh, H. P.	June 2014	Probabilistic Safety Assessment and Management PSAM 12, Honolulu, Hawaii, June 2014
12	iROCS: Integrated Accident Management Framework for Coping with Beyond-Design-Basis External Events	Kim, J. et al.	March 2016	Nuclear Engineering and Design, vol. 298, p. 1-13
13	APR1400 Design Certification Severe Accident Mitigation Design Alternatives Technical Analysis in Support of the Environmental Assessment	Palmrose, D. E.	September 2018	ADAMS Accession No. ML18096A697
14	FLEX Strategy Implementation for LOCA Sequences in PWR-Westinghouse	Ruiz-Zapatero, M.; Bocanegra, R.; Queral, C.	September 2017	26th International Conference Nuclear Energy for New Europe, Bled, Slovenia, September 11-14, 2017
15	Dynamic Human Performance Context Comparison for Severe Accident Management during Long Term Station Blackout in Light Water Reactors	Petkov, G. and Petkov, I.	June 2017	2017 European Safety and Reliability Conference (ESREL2017), Portoroz, Slovenia, June 18-22, 2017
16	Development of Site Risk Assessment and Management Technology including Extreme External Events	Yang, J. E. et al.	March 2017	KAERI/RR--4225/2016

XI. Level 1 PRA

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Use and Development of Probabilistic Safety Assessment	Nuclear Energy Agency	December 2012	NEA/CSNI/R(2012)11
2	The NRC's SPAR Models: Current Status, Future Development, and Modeling Issues	Buell, R. F.	September 2008	INL/CON-08-14484
3	SPAR Integrated Capabilities Model (SPAR-ICM) Project	Ma, Z. et al.	July 2013	INL/LTD-12-27648, Rev. 1 (not publicly available)
4	U.S. NRC Confirmatory Level 1 PRA Success Criteria Activities	Helton, D.; Esmaili, H.; Buell, R.	March 2011	INL/CON-11-21026
5	Development Strategy of Optimized Level 1&2 PSA Model for APR1400 NPPs Reflecting New Severe Accident Mitigating System and Its Application	Hwang, S.-W. et al.	December 2018	Annals of Nuclear Energy, vol. 122, p. 256-269
6	Lessons Learned from Applying a New HRA Method for the Petroleum Industry	Taylor, C.; Øie, S.; Gould, K.	Oct 2018	Reliability Engineering & System Safety, available online 5 October 2018

XII. Nuclear Safety/Societal Risk

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Insights on Risk Margins at Nuclear Power Plants: A Technical Evaluation of Margins in Relation to Quantitative Health Objectives and Subsidiary Risk Goals in the United States	Electric Power Research Institute	May 2018	EPRI Report No. 3002012967
2	The Three Mile Island, Chernobyl, and Fukushima Daiichi Accidents and Their Radiological Impacts	Bevelacqua, J. J.	June 2016	International Nuclear Safety Journal, vol. 5, no.1, p. 21-79
3	Software Verification and Validation: Examples from the Safety Arena	Siu, N.	September 2015	ADAMS Accession No. ML15253A634 INMM Workshop on VA Tools, Boston, MA; September 14-16, 2015
4	Development of an Updated Societal-Risk Goal for Nuclear Power Safety	Bier, B. et al.	July 2014	INL/CON-13-30391
5	The Societal Risk of Severe Accidents in Nuclear Power Plants	Denning, R. and McGhee, S.	June 2013	Transactions of the American Nuclear Society, vol. 108, no. 1, p. 521-525
6	Insights into the Societal Risk of Nuclear Power Plant Accidents	Denning, R. and Mubayi, V.	January 2017	Risk Analysis, vol. 37, no. 1, p. 160-172
7	Understanding the Nature of Nuclear Power Plant Risk	Denning, R. S.	June 2012	ICAPP '12: 2012 International Congress on Advances in Nuclear Power Plants, Chicago, IL (United States), June 24-28, 2012
8	Critical Reflections on Nuclear and Renewable Energy: Environmental Protection and Safety in the Wake of the Fukushima Nuclear Accident. (Chapter 1: Reliability and Nuclear Power and Postscript)	Kuo, W.	March 2014	n/a (Book)
9	Preliminary Study for Development of Technical Basis of Quantitative Safety Goals for NPPs Operations in UAE	Shehhi, O. A. A. et al.	June 2017	Transactions of the American Nuclear Society, vol. 116, no. 1, p. 801-803

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
10	Impact of Probabilistic Risk Assessment and Severe Accident Research in Reducing Reactor Risk	Denning, R. S. and Budnitz, R. J.	January 2018	Progress in Nuclear Energy, v. 102 p. 90-102
11	Nuclear Weapon “Pit” Production: Options to Help Meet A Congressional Requirement	Medalia, J. E.	May 2015	CRS Report No. R44033
12	Nuclear Renaissance, Public Perception and Design Criteria: An Exploratory Review	Goodfellow, M. J.; Williams, H. R.; Azapagic, A.	October 2011	Energy Policy, vol. 39, no. 10, p. 6199-6210
13	Reliability and Nuclear Power	Kuo, W.	June 2011	IEEE Transactions on Reliability, vol. 60 no. 2, p. 365-367
14	International Conference on Topical Issues in Nuclear Installation Safety: Defence in Depth — Advances and Challenges for Nuclear Installation Safety. Proceedings of an International Conference held in Vienna, Austria, 21-24 October 2013	International Atomic Energy Agency, Safety Assessment Section, Vienna (Austria)	October 2014	IAEA-TECDOC-CD--1749
15	Handbook of Advanced Nuclear Hydrogen Safety. 1st edition	Hino, R. et al.	January 2017	JAEA-REVIEW-2016-038
16	Nuclear Facility Accident (NFAC) Unit Test Report for HPAC Version 6.4	Lee, R. W. and Sulfredge, C. D.	March 2017	ORNL/TM-2017/94
17	Nuclear Energy: Overview of Congressional Issues. Updated November 16, 2018	Holt, M.	November 2018	CRS Report No. R42853
18	Environmental Impact Statement for the Combined License (COL) for Enrico Fermi Unit 3. Final Report. Appendix E	NRC and U.S. Army Corps of Engineers	January 2013	NUREG-2105 v.3 ADAMS Accession No. ML12307A177
19	Comparative Radioecological Assessment of Serious-Accident Scenarios in NPP on the Basis of the Risk for Natural Communities	Spiridonov, S. and Mikailova, R.	January 2019	Atomic Energy, vol.125, no. 3, p. 198-203

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
20	System Studies for Global Nuclear Assurance & Security: 3S Risk Analysis for Small Modular Reactors (Volume I) - Technical Evaluation of Safety Safeguards & Security	Williams, A. D. et al.	October 2018	SAND2018-12447
21	Integrating Thermal Energy Storage and Nuclear Reactors: A Technical and Policy Study	Abel, C. R.	May 2018	Dissertation
22	Irradiation Dose of the Woody Tier of a Coniferous Forest Due to Accidental Emissions from NPP	Mikailova, R. A. and Spiridonov, S. I.	January 2018	Atomic Energy, vol. 123, no. 3, p. 202–208
23	Uncertainties in Estimating Health Risks Associated with Exposure to Ionising Radiation	Preston, R. J. et al.	June 2013	Journal of Radiological Protection, vol. 33, no. 3, p. 573-588

XIII. Sensitivity and Uncertainty Analyses

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Proposed Analyses for xLPR 2.0 Outputs	Sallaberry, C. J.-M. et al.	June 2015	SAND2015-5118PE
2	Bootstrapped-Ensemble-Based Sensitivity Analysis of a Trace Thermal-Hydraulic Model Based on a Limited Number of PWR Large Break LOCA Simulations	Di Maio, F. et al.	September 2016	Reliability Engineering & System Safety, vol. 153, p. 122-134
3	SOARCA Surry Power Station Uncertainty Analysis: Parameter Methodology and Insights	Jones, J. et al.	June 2014	Probabilistic Safety Assessment and Management PSAM 12, Honolulu, Hawaii, June 2014
4	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis MELCOR Parameters and Probabilistic Results	Osborn, D. et al.	April 2013	SAND2013-3107C
5	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis MACCS2 Aleatory Weather Effects	Osborn, D. et al.	April 2013	SAND2013-3102C
6	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis MACCS2 Dose Truncation Sensitivity	Osborn, D. et al.	April 2013	SAND2013-3105C
7	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis MACCS2 Parameters and Probabilistic Results	Osborn, D. et al.	April 2013	SAND2013-3106C
8	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Contributions to Overall Uncertainty	Bixler, N. E. et al.	June 2014	Probabilistic Safety Assessment and Management PSAM 12, Honolulu, Hawaii, June 2014
9	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Knowledge Advancement	Gauntt, R. O. et al.	February 2014	SAND2014-1344C

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
10	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis Probabilistic Methodology and Regression Technique	Osborn, D. et al.	April 2013	SAND2013-3104C
11	SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Convergence of the Uncertainty Results	Bixler, N. E. et al.	February 2014	SAND2014-1346C
12	Uncertainty Analysis of Consequence Management (CM) Data Products	Hunt, B. D. et al.	January 2018	SAND2018-0329
13	Phenomenological Uncertainties in SOARCA Study	Gauntt, R. O.	March 2011	SAND2011-2024C
14	The SOARCA Surry Power Station Short-Term Station Blackout Uncertainty Analysis: MACCS Parameter Development	Bixler, N. E. et al.	June 2015	Transactions of the American Nuclear Society, vol. 112, no. 1, p. 515-516
15	A Systematic Framework for Effective Uncertainty Assessment of Severe Accident Calculations; Hybrid Qualitative and Quantitative Methodology	Hoseyni, S. M. et al.	May 2014	Reliability Engineering & System Safety, vol. 125, p. 22-35
16	Sensitivity Analysis of Successful Evacuee Proportion in Hypothetical NPP Accident by Earthquake	Kim, S.-Y. and Lim, H.-G.	June 2017	Transactions of the American Nuclear Society, vol. 116, no. 1, p. 808-811
17	Development of a Technical Basis for Component Integrity Assessment Using Probabilistic Methods	Raynaud, P. and Kirk, M.	August 2018	ADAMS Accession No. ML18235A101 (*not publicly available)
18	Sequoyah SOARCA Uncertainty Analysis of a STSBO Accident	Bixler, N. E. et al.	March 2018	Probabilistic Safety Assessment and Management Conference (PSAM 14), Los Angeles, CA, September 16-21, 2018
19	Using Deterministic and Probabilistic Methods for MELCOR Severe Accident Uncertainty Analysis	Mattie, P. D. et al.	October 2017	ADAMS Accession No. ML17278A919

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
20	Findings from Uncertainty Studies Evaluating Severe Accident Phenomena and Off-Site Consequences	Ghosh, S. T. et al.	October 2017	ADAMS Accession No. ML17278A920
21	Sensitivity Analysis for XLPR Acceptance Testing	Sallaberry, C. J. et al.	July 2017	ASME 2017 Pressure Vessels and Piping Conference, Volume 6B: Materials and Fabrication, Waikoloa, Hawaii, USA, July 16–20, 2017

XIV. Storage and Transportation Safety

#	Title	Author	Date	Journal/Technical Report/Conference Citation or Report Number
1	Example of Integration of Safety Security and Safeguard Using Dynamic Probabilistic Risk Assessment Under a System-Theoretic Framework	Kalinina, E. A. et al.	April 2017	SAND2017-3572C
2	Advancing US Public Acceptance of Spent Fuel Storage and Transport: Proposed Outreach Services for Ionising Radiation Education Support	Pennington, C. W.	2013	Packaging, Transportation, Storage & Security of Radioactive Material, vol. 24, No. 3, p. 95-107
3	Regulatory Analysis for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel	U.S. NRC	November 2013	COMSECY 13-0030, Enclosure 1 ADAMS Accession No. ML13273A628
4	Analysis of Dose Consequences Arising from the Release of Spent Nuclear Fuel from Dry Storage Casks	Durbin, S. G. and Morrow, C.	January 2013	SAND2013-0533
5	Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor	Barto, A. et al.	June 2013	NUREG-2161 ADAMS Accession No. ML14255A365