May 10, 2013

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board Panel

In the Matter of	)	
	)	Docket No. 50-443-LR
NextEra Energy Seabrook, LLC	)	
	)	ASLBP No. 10-906-02-LR
(Seabrook Station, Unit 1)	)	

NEXTERA'S MOTION FOR SUMMARY DISPOSITION OF FRIENDS OF THE COAST/NEW ENGLAND COALITION CONTENTION 4B (SAMA ANALYSIS SOURCE TERMS)

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.1205 and the Atomic Safety and Licensing Board's ("ASLB" or the "Board") Initial Scheduling Order (April 4, 2011), NextEra Energy Seabrook, LLC ("NextEra") moves for summary disposition of Friends of the Coast and the New England Coalition ("Friends/NEC") Contention 4B. NextEra seeks summary disposition on the grounds that no genuine issue of material fact exists and NextEra is entitled to a decision as a matter of law. 10 C.F.R. § 2.710(d)(2). This Motion is supported by (1) a Statement of Material Facts as to which NextEra asserts that there is no genuine dispute (Attachment 1); and (2) a Declaration of Dr. Kevin R. O'Kula ("O'Kula Decl.") (Attachment 2).

II. PROCEDURAL BACKGROUND

NextEra applied to the Nuclear Regulatory Commission ("NRC", or the "Commission") for a renewed operating license ("LRA") for Seabrook Station ("Seabrook") in May 2010. Friends/NEC filed a Petition to Intervene and Request for a Hearing ("Petition") on October 20, 2010 raising four contentions, which included, *inter alia*, Contention 4, a multi-part challenge to

NextEra's analysis of severe accident mitigation alternatives ("SAMA"). On February 15, 2011, the Board admitted portions of Contention 4, including Contention 4B.[1]

As admitted by the Board, Contention 4B alleges that "[t]he SAMA analysis for Seabrook minimizes the potential amount of radioactive release in a severe accident" because it used the Modular Accident Analysis Progression ("MAAP") code to generate severe accident source terms. Pet. at 41; LBP-11-02, 73 N.R.C. at 64. Friends/NEC base this contention on three claims:

> (1) The MAAP code "has not been validated by the NRC."
>
> (2) Source terms generated by the MAAP code "are consistently smaller for key radionuclides than the release fractions specified in NUREG-1465 and its recent revision for high burnup fuel."[2]
>
> (3) A draft version of NUREG-1150[3] and a 2002 report of the Brookhaven National Laboratory[4] suggest that "MAAP generates lower release fractions than those derived and used by NRC in studies such as NUREG-1150."

Pet. at 44-45.

On appeal by NextEra, the Commission found the support for Contention 4B to be weak.[5] The Commission found no factual or expert support for the Contention's suggestion that the Seabrook SAMA analysis release fractions should be replaced with generic release fractions from NUREG-1465. Further, the Commission found that the references to the Draft NUREG-1150 and the BNL Report were a "thin reed" which did not suggest that the MAAP-generated source terms are inaccurate or that generic sources terms would be more accurate. CLI-12-5, 75

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[1] *NextEra Energy Seabrook, LLC* (Seabrook Station, Unit 1), LBP-11-02, 73 N.R.C. 28 (2011).

[2] L. Soffer et al., Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465 (Feb. 1995)).

[3] Office of Nuclear Regulatory Research, Draft for Comment, Reactor Risk Reference Document, NUREG-1150, Vol. 1 (Feb. 1987) ("Draft NUREG-1150").

[4] John R. Lehner et al., Brookhaven National Laboratory, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" (Dec. 2002) ("BNL Report").

[5] *NextEra Energy Seabrook, LLC* (Seabrook Station, Unit 1), CLI-12-5, 75 N.R.C. 301, 326 (2012).

N.R.C. at 326. Nevertheless, the Commission permitted the admission of this Contention to stand. This Motion and supporting expert declaration will now demonstrate that none of Friends/NEC's claims raises a genuine issue of material fact, and accordingly, that NextEra is entitled to a decision on Contention 4B as a matter of law.

III. STATEMENT OF LAW

A. NRC Legal Standards for Summary Disposition

In ruling on motions for summary disposition in 10 C.F.R. Subpart L proceedings, the Board applies the standards in 10 C.F.R. Subpart G. 10 C.F.R. § 2.1205(c). Subpart G states that summary disposition is appropriate where the record demonstrates that no genuine dispute exists regarding any material fact and the moving party is entitled to a decision as a matter of law. 10 C.F.R. § 2.710(d)(2).[6]

When a summary disposition motion is supported by affidavits in accordance with 10 C.F.R. § 2.710(b), the "party opposing the motion may not rest upon . . . mere allegations or denials," but must, by affidavit or as otherwise provided in the rule, set forth "specific facts showing that there is a genuine issue of fact" warranting a hearing. 10 C.F.R. § 2.710(b); *Advanced Medical Systems, Inc.* (One Factory Row, Geneva, Ohio, 44041), CLI-93-22, 38 N.R.C. 98, 102 (1993). "Bare assertions or general denials are not sufficient. Although the opposing party does not have to show that it would prevail on the issues, it must at least demonstrate that there is a genuine factual issue to be tried." *Id.* at 102 (citations omitted). "[Opponents] have to present contrary evidence that is so significantly probative that it creates a material factual issue." *Id.* at n.13 (*citing Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-92-8, 35 N.R.C. 145, 154 (1992)).

[6] *See also Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), CLI-01-11, 53 N.R.C. 370, 384 (2001).

The Commission has encouraged Boards to use the summary disposition process where the proponent of a contention has failed to establish that a genuine issue exists, so that evidentiary hearing time is not unnecessarily devoted to such issues. *Statement of Policy on Conduct of Licensing Proceedings*, CLI-81-8, 13 N.R.C. 452, 457 (1981). The summary disposition procedures "provide in reality as well as in theory, an efficacious means of avoiding unnecessary and possibly time-consuming hearings on demonstrably insubstantial issues . . . ." *Houston Lighting & Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 N.R.C. 542, 550 (1980).

B. NEPA Standards As Applied to NRC SAMA Analysis

The National Environmental Policy Act ("NEPA") demands that federal agencies contemplating major actions prepare an environmental impact statement ("EIS") addressing "any adverse environmental effects which cannot be avoided should the proposal be implemented." 42 U.S.C. § 4332(C)(ii). Pursuant to this inquiry, an EIS must "discuss the extent to which adverse effects can be avoided" by mitigation. *Robertson v. Methow Valley Citizens Council*, 490 U.S. 332, 352 (1989). A mitigation alternatives analysis need not reflect, however, a "most conservative — or worst case — analysis." *Entergy Nuclear Generation Company and Entergy Nuclear Operations, Inc.*, (Pilgrim Nuclear Power Station), CLI-12-10, 75 N.R.C. 479, 485 (2012) ("*Pilgrim*").

Accordingly, NRC's NEPA regulations require license renewal applicants perform a plant-specific, quantitative assessment of measures that could significantly mitigate the frequency-weighted consequences of radiological impacts in a severe accident — a SAMA

analysis.[7] The Commission has repeatedly stated that SAMA analyses are subject to NEPA's rule of reason.[8] Consequently, not every possible objection to an LRA's SAMA analysis gives rise to a genuine dispute regarding a material fact. For example, whether there are better models or the SAMA analysis can be refined further are not material issues. *Pilgrim*, CLI-10-11, 71 N.R.C. 287, 315 (2010). Instead, to be litigable in an NRC hearing, a SAMA contention must identify a "*significant deficiency* in the SAMA analysis — i.e., a deficiency that could credibly render the SAMA analysis altogether unreasonable under NEPA standards." *Pilgrim*, CLI-12-01, 75 N.R.C. 39, 51 (2012) (emphasis in original).

The Commission has highlighted this requirement in the instant proceeding, noting that "the proper question is not whether there are plausible alternative choices for use in the analysis, but whether the analysis that was done is reasonable under NEPA." *Seabrook*, CLI-12-05, 75 N.R.C. at 323. The Commission elaborated that:

> A contention proposing alternative inputs or methodologies must present some factual or expert basis for why the proposed changes in the analysis are warranted (e.g., why the inputs or methodology used is unreasonable, and the proposed changes or methodology would be more appropriate). Otherwise, there is no genuine material dispute with the SAMA analysis that was done, only a proposal for an alternate NEPA analysis that may be no more accurate or meaningful.

Id. at 323-24. Therefore, challenges to a SAMA analysis "must be tethered to the computer modeling and mathematical aspects of the analysis" and must demonstrate a significant defect skewing the cost benefit results. *Davis Besse*, CLI-12-8, 75 N.R.C. at 414-15. Consequently, Friends/NEC cannot avoid summary disposition merely by alleging the existence of a better source term code. Rather, they must provide evidence that NextEra's use of the MAAP source

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[7] *See* Final Rule, Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, 61 Fed. Reg. 28,467, 24,480-81 (June 5, 1996). *See also Mass. v. NRC*, 708 F.3d 63, 75-76 & n.19 (1st Cir. 2013) (endorsing the NRC's "site-specific and plant-specific" approach to performing SAMA analyses).

[8] *Pilgrim*, CLI-10-22, 72 N.R.C. 202, 208 (2010).

term code in its SAMA analysis was unreasonable by significantly skewing the cost benefit results.

IV. THERE IS NO GENUINE DISPUTE REGARDING ANY MATERIAL ISSUE

MAAP is unquestionably a reasonable code for the development of source terms and release fractions within SAMA analyses. The attached Declaration of Dr. Kevin O'Kula — an expert whose three decades of work in reactor probabilistic risk assessment ("PRA") and accident consequence analysis includes extensive experience with SAMA analyses in NRC license renewal proceedings[9] — characterizes the MAAP 4 code used in the Seabrook SAMA analysis as a contemporary code for reactor accident consequence analysis. O'Kula Decl. at ¶¶ 11, 24, 52. Dr. O'Kula's characterization of the MAAP code was endorsed by a recent decision by the Board in the *Davis-Besse* license renewal proceeding granting summary disposition of a nearly-identical contention challenging the reasonableness of the use of the MAAP within SAMA analyses.[10] Further, as discussed in more detail below, none of Friends/NEC's claims underlying Contention 4B has any validity or support.

A. The MAAP Code Has Been Extensively Validated and Benchmarked, and Has Been Repeatedly Accepted by the NRC for Use in SAMA Analyses Supporting LRAs

As an initial matter, Friends/NEC's assertion that the MAAP code has not been validated by the NRC is immaterial to whether the Seabrook SAMA analysis is reasonable under NEPA. As the *Davis-Besse* Board noted, there is no legal requirement that the MAAP code be "independently validated" by the NRC (LBP-12-26, slip op. at 17), and Friends/NEC have cited no law, regulation, or Commission precedent to the contrary.

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[9] Dr. O'Kula has worked in support of SAMA analyses in the *Pilgrim, Davis-Besse*, and *Indian Point* license renewal proceedings. O'Kula Decl. at ¶ 5.

[10] *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), LBP-12-26, 76 N.R.C. __ (Dec. 28, 2012) (slip op. at 14, 29).

Moreover, Friends/NEC's assertion that the MAAP code has not been validated lacks any factual basis. As explained in the O'Kula Declaration, the MAAP code has been benchmarked and validated for decades. The MAAP code was developed and maintained in accordance with 10 C.F.R. Part 50, Appendix B, and International Organization for Standardization ISO 9001 quality assurance requirements. O'Kula Decl. at ¶ 23. Extensive U.S. and international benchmarking and validation of the MAAP code has found it acceptable for estimating certain phenomena defined in severe accident analyses. *Id*. at ¶¶ 26-27. In particular, the MAAP code has been successfully benchmarked and validated against numerous nuclear severe accident studies and the Three Mile Island Unit-2 core melt accident. *Id*. Based on this large and comprehensive body of benchmarking and validation, the Electric Power Research Institute ("EPRI") has identified the MAAP code (Version 4.0.5 and later) as a "consensus computer code" suitable for use in PRA success criteria analyses.[11] *Id*. at ¶ 24.

Additionally, the MAAP code is frequently used by U.S. nuclear plant licensees in Level 2 PRAs supporting SAMA analyses. *Id*. at ¶¶ 27-28. Before accepting the use of a computer code in an LRA, the NRC reviews its inputs and assumptions, as well as sources of uncertainty. *Id*. The NRC has employed this approach in accepting the MAAP code as a tool for modeling severe accident phenomenology in numerous LRA SAMA analyses. *Id*.

In summary, Friends/NEC have failed to raise any genuine dispute of material fact regarding validation of the MAAP code. Not only is such validation not required and thus immaterial here, the indisputable facts demonstrate that MAAP has been extensively benchmarked and validated. Dr. O'Kula thus concludes that MAAP is appropriate and

[11] For ease of reference, this Motion will hereafter refer to MAAP Versions 4.0.5 and later as "MAAP 4".

reasonable for use in SAMA analyses. *Id*. at ¶ 29. Consequently, this basis for Friends/NEC Contention 4B raises no genuine dispute of material fact.

B. NUREG-1465 Release Fractions and Source Terms Are Inappropriate for Use in SAMA Analyses and Are Thus Immaterial to the Reasonableness of the MAAP Code

Friends/NEC's attempt to compare MAAP-generated source terms and release fractions with those specified in NUREG-1465 also raises no genuine, material dispute. There is no requirement that the Seabrook SAMA analysis (or any SAMA analysis for that matter) rely on NUREG-1465 source terms. Moreover, any comparison of a NUREG-1465 source term with a MAAP source term is akin to comparing apples to oranges. As explained further below and in the O'Kula Declaration, the NUREG-1465 source term represents a generic analysis of the radioactive release from a damaged reactor core into containment. A MAAP source term, however, represents a plant-specific analysis of a radioactive release from the plant's containment to the environment, which has been reduced by natural processes and engineered safety features (both active and passive). There simply is no genuine comparison between the two.

SAMA analyses, which are concerned with public dose and offsite economic cost risk, focus on the radioactive releases to the environment following a severe accident. O'Kula Decl. at ¶¶ 12-13, 31-32. Thus, SAMA analyses require modeling of environmental source terms and release fractions. *Id*. at ¶ 16. This is precisely the type of modeling performed by the MAAP code. *Id*. at ¶¶ 14, 33-35. In contrast, NUREG-1465 provides only *containment* source terms and release fractions, which neglect natural and engineered fission product removal mechanisms associated with any plant's containment. *Id*. at ¶¶ 36, 39-43, 50. As the O'Kula Declaration explains, NUREG-1465's neglect of those fission product removal mechanisms inevitably results in a larger source term than the MAAP-generated environmental source term because a MAAP

environmental source term has been reduced by natural and engineered fission product removal mechanisms.[12] *Id*. at ¶¶ 44, 50. Thus, use of the larger NUREG-1465 containment source term as suggested by Friends/NEC would inaccurately equate releases to Seabrook's containment as direct releases to the environment, and represent an unrealistic, worst-case analysis. *Id*. at ¶¶ 44, 48, 51, 75.

SAMA analyses are plant-specific evaluations. *See* 61 Fed. Reg. at 28,480. This is because:

> the accident sequences evaluated and their assessed probabilities are specific to the features and location of the plant, including numerous factors extending far beyond the particular design of the reactor (e.g., reactor core radionuclide inventory, physical and climate features of the site, existing equipment or hardware, relevant plant procedures). If one could simply assume that all nuclear power stations would have the same estimated radionuclide releases, caused by the same sequence of events, with the same frequency of occurrence, there would be little reason to do a site-specific probabilistic risk analysis.

Pilgrim, CLI-12-15, 75 N.R.C. 704, 716 (2012). Accordingly, the development of source terms and release fractions in a SAMA analysis must account for a spectrum of interdependent, plant-specific factors — including plant design, support systems, plant maintenance and operating procedures — influencing the frequency-weighted consequences of radiological impacts in a severe accident. O'Kula Decl. at ¶¶ 32, 45, 47. Consistent with the Commission's expectations, NextEra used the MAAP code to analyze detailed design and operational information for Seabrook to generate source terms and release fractions within the Level 2 PRA for Seabrook's SAMA analysis. *Id.* at ¶¶ 34-35, 47, 49.

[12] Dr. O'Kula's analysis is consistent with the results of the NRC's State-of-the-Art Reactor Consequence Analyses ("SOARCA") project indicating that containment source terms are consistently higher than environmental source terms for a representative PWR plant. O'Kula Decl. at ¶ 69 (citing NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis" (Jan. 2012)).

In contrast, NUREG-1465's source term is a generic, default value incorporating no information or data from Seabrook. Indeed, NUREG-1465's generic source term and its associated release fractions were derived from historical analyses of a handful of older Pressurized Water Reactors ("PWRs") — the Surry Power Station, the Zion Nuclear Power Station, and the Oconee Nuclear Station Unit 3 (a B&W PWR) and Sequoyah Nuclear Generating Station — each with distinct design features and operating procedures bearing on the mitigation of the consequences of severe accidents. *Id*. at ¶ 46. Further, because NUREG-1465 neglects the quantification of containment-related fission product removal mechanisms, its release fraction and source term are predicated on mistaken assumptions regarding the scenarios in which core damage results in radioactive release directly to the environment. *Id*. at ¶¶ 30, 40, 43-44. This markedly departs from the plant-specific approach employed in connection with the MAAP-generated source terms and release fractions in Seabrook's SAMA analysis, which account for a range of timing and containment damage scenarios materially affecting the quantification of source terms for the Seabrook facility. *Id*. Thus, use of the generic NUREG-1465 source term and its release fractions within Seabrook's plant-specific Level 2 PRA would be an unrealistic, worst-case source term scenario that fails to account for Seabrook's specific characteristics. *Id*. at ¶¶ 44, 48, 51.

In summary, use of generic NUREG-1465 source terms and release fractions in place of the plant-specific, MAAP-generated source terms and release fractions in Seabrook's SAMA analysis would neglect natural and engineered fission product removal mechanisms associated with Seabrook's containment and distort the SAMA analysis into an overly-conservative, worst-case inquiry. *Id*. The *Davis-Besse* Board similarly concluded NUREG-1465's source terms and release fractions were unsuitable for use in SAMA analyses, finding:

> a) that there is no requirement that applicants use NUREG-1465 source terms, b) that the differences between the NUREG-1465 source terms and the MAAP source terms are due in part to containment engineered safety features and passive and active fission product removal mechanisms and hence are not an appropriate comparison, and c) that the NUREG-1465 source terms are generic in nature as compared to the MAAP source terms that are Davis-Besse plant and plant-specific, the latter in accordance with NRC requirements that SAMA analyses be site-specific.

Davis-Besse, LBP-12-26, slip op. at 22-23. For these reasons, Friends/NEC's claim regarding NUREG-1465 fails to raise any genuine dispute of material fact.

C. Friends/NEC's References to NUREG-1150 and the BNL Report Are Immaterial to the Reasonableness of the MAAP Code Version Used in the Seabrook SAMA Analysis

Finally, Friends/NEC's claim that the MAAP source term and release fractions used in the Seabrook SAMA are too low when compared to the source terms and release fractions identified in a draft version of NUREG-1150 and the BNL Report is specious, having no relation to the version of MAAP used in the Seabrook analysis. Thus, this claim too fails to raise any genuine dispute concerning a material issue.

Again, as a threshold matter, there is no requirement (nor have Friends/NEC identified any) for NextEra to use the source terms and release fractions described in the Draft NUREG-1150 and the BNL Report. Furthermore, the release fraction comparisons extracted from those documents involve legacy codes or older code versions[13] and earlier versions of the MAAP code — not the contemporary, "consensus" version of MAAP used in evaluating the probabilistic risks associated with a severe accident in light of Seabrook's unique characteristics. *Id*. at ¶¶ 24, 30, 53-55, 57, 59, 61-62. The MAAP-generated source terms and release fractions in Seabrook's Level 2 PRA are more appropriately compared to the results of recent nuclear accident studies

[13] Legacy codes/code versions are integrated computer codes that are either no longer use or permanently retired because, *inter alia*, they (1) are based on old experiments and data, or (2) do not reflect contemporary understanding of severe accident phenomenology. O'Kula Decl. at 31 n.19.

employing more sophisticated computer codes and better insights into nuclear accident phenomenology than the documents identified by Friends/NEC. *Id*. at ¶¶ 62-74. For these reasons, Friends/NEC's claim regarding NUREG-1150 and the BNL Report once more fails to raise a genuine dispute on any material issue.

Within a SAMA analysis, integrated computer codes are used that incorporate updated knowledge of severe accident phenomenology with the unique characteristics of a particular plant and other information from a plant-specific PRA. *See id*. at ¶¶ 18-19, 32. For Seabrook's SAMA analysis, NextEra analyzed a series of Seabrook-specific features using a contemporary, extensively benchmarked and validated version (Version 4) of the MAAP code that incorporated many of the latest insights into severe accident phenomenology. *Id*. at ¶¶ 20, 24, 26-27.

Neither Draft NUREG-1150 nor the BNL Report evaluated release fractions generated by the MAAP 4 code. *Id*. at ¶¶ 53-55. Rather, those documents compared (1) release fractions obtained from legacy codes/code versions with (2) those generated by earlier versions of the MAAP code. Draft NUREG-1150 compares release fractions obtained from Versions 1.1 through 3 of the MAAP code against those obtained using the Source Term Code Package ("STCP") code.[14] *Id*. at ¶ 53, 55. Similarly, the BNL Report compares release fractions obtained from MAAP Version 3B against those obtained using legacy codes/code versions identified in NUREG-1150, including STCP and a decade-old version of the MELCOR code. *Id*. at ¶¶ 54-55, 62.

Further, the dated vintage of the computer codes referenced in Draft NUREG-1150 and the BNL Report ensures that those codes reflect a less complete understanding of severe accident

[14] STCP is a legacy code formerly used by the NRC in severe accident analyses. The NRC now uses an updated integrated code developed by Sandia National Laboratory — the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code — to generate severe accident source terms. O'Kula Decl. at 31 n.19.

phenomenology than the MAAP 4 code used in the Seabrook SAMA analysis. The nuclear safety community's understanding of severe accident phenomenology — in particular, reactor pressure vessel heat transfer and fission products chemistry — has advanced significantly since the publication of NUREG-1150 (in draft and final form) nearly a quarter-century ago. *Id*. at ¶¶ 57-59, 61-63. Because newer computer codes such as the MAAP 4 code used in the Seabrook SAMA analysis incorporate many of those insights, they provide far more sophisticated and realistic modeling of severe accident phenomena — and thus, markedly different release fractions — than the codes compared within Draft NUREG-1150 and the BNL Report.[15] *Id*. at ¶¶ 24, 57-59, 61-63.

Additionally, neither Draft NUREG-1150 nor the BNL Report compare release fractions obtained using Seabrook-specific information and data. The Draft NUREG-1150 comparison cited in Contention 4B involved release fractions associated with severe accident analysis for the Zion Nuclear Station.[16] *Id*. at ¶ 56. Similarly, the BNL Report compared release fractions in a Level 2 PRA for the Catawba Nuclear Station with release fractions for a "typical NUREG-1150 release" from the Sequoyah Nuclear Generation Station. *Id*. at ¶ 54. Any further comparison between these studies and the MAAP 4-generated source terms for the Seabrook SAMA analysis is belied by the significant differences in containment design between Seabrook (dry, ambient air containment) and some of the plants evaluated in Draft NUREG-1150 and the BNL Report (ice

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[15] One of the release fraction comparisons in the BNL Report is exemplary of this phenomenon. As Dr. O'Kula explains, the release fractions obtained for the Sequoyah plant using older STCP/MELCOR codes were higher than those generated for Catawba plant using the (more advanced) MAAP code that accounted for an additional mitigating assumption. O'Kula Decl. at ¶ 61.

[16] Notably, Contention 4B neglects to mention that release fraction comparison for the Zion Nuclear Station within the 1987 Draft NUREG-1150 was omitted from the final version of NUREG-1150 published in 1990. O'Kula Decl. at ¶ 56. Although NRC precedent attaches no regulatory significance to draft documents, *see, e.g., Duke Power Co.* (Catawba Nuclear Station, Units 1 & 2), ALAB-355, 4 N.R.C. 397, 416 (1976), this Motion nevertheless will address this particular element of Contention 4B.

condenser PWR containments and Mark III Boiling Water Reactor containments) that would inevitably result in markedly divergent release fractions.[17] *Id*. at ¶¶ 54, 55, 61.

Moreover, the results of more recent severe nuclear accidents studies support the reasonableness of NextEra's use of the MAAP code in the Seabrook SAMA analysis. In 2006, the NRC commissioned the SOARCA project to incorporate advances in plant operations, nuclear accident phenomenology, and computer modeling since the publication of earlier severe accident analyses such as the (final version of) NUREG-1150. *Id.* at ¶¶ 64-65. SOARCA's more realistic modeling results indicate that the predictions of those earlier severe accident studies were overly-conservative in that they estimated higher-magnitude radionuclide releases occurring earlier in an accident sequences than those predicted by SOARCA's more realistic model.[18] *Id*. at ¶¶ 67-68. Further, SOARCA's MELCOR-generated environmental source terms and release fractions for station blackout at a nuclear plant (Surry Nuclear Power Plant) with a containment design (dry, ambient air containment) similar to that of Seabrook were found to be reasonably consistent with the MAAP-generated environmental source terms and release fractions in Seabrook's SAMA analysis for station blackout events.[19] *Id*. at ¶ 70. As Dr. O'Kula

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[17] Notwithstanding the lack of comparisons between release fractions for Seabrook generated using different computer codes, Pet. at 44, the O'Kula Declaration identifies one such "apples-to-apples" comparison for another facility. The NRC Staff's Supplemental Environmental Impact Statement for the Catawba Nuclear Station license renewal compared two studies — NUREG-1150 and revision 2b of the Catawba PRA (which included that facility's IPE models) — and concluded there was "reasonable agreement" between MAAP-generated release fractions and those generated using the STCP/MELCOR codes for the closest corresponding release scenarios. O'Kula Decl. at ¶ 60.

[18] SOARCA's findings, moreover, were consistent with the results of other recent severe accident studies employing more sophisticated modeling of severe accident phenomenology than NUREG-1150. *See* O'Kula Decl. at ¶ 59 (citing Frank J. Rahn & Robert E. Henry, "Release and Dispersion of Radioactivity from Reactor Fuel Research and Analytical Results Leading to Reductions in Radiological Source Terms," American Nuclear Society Position Statement No. 65 on "Realism in the Assessment of Nuclear Technologies," Appendix 1A (June 2004) (predicting smaller magnitude and longer-delayed radioactive releases to the environment than NUREG-1150)).

[19] Dr. O'Kula notes that the results of his comparison are consistent with the conclusions of a 2004 study that the MELCOR and MAAP codes predict similar thermal-hydraulic and core degradation responses during a severe accident in a representative PWR plant. O'Kula Decl. at ¶ 72 (citing K. Vierow, Y. Liao, J. Johnson, M. Kenton,

explains, this latter finding is particularly compelling evidence of the reasonableness of the use of the MAAP code in the Seabrook SAMA analysis's Level 2 PRA, as it represents an "apples-to-apples" comparison (1) between integrated computer codes of similar vintage, (2) for reactor plants of similar core and containment design, and (3) in modeling the same type of accident scenario. *Id*. For all these reasons, Friends/NEC's claim relating to NUREG-1150 and the BNL Report raises no genuine issue of material fact.

V. CONCLUSION

For the above-stated reasons, the Board should grant NextEra's motion for summary disposition of Friends/NEC Contention 4B.

VI. CERTIFICATION

In accordance with 10 C.F.R. §2.323(b), counsel for NextEra conferred with the representatives of the other parties in a sincere effort to resolve the matters at issue in the instant Motion prior to the filing of the Motion, but was unsuccessful in doing so.

Respectfully Submitted,

/Signed (electronically) by David R. Lewis /

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Dated: May 10, 2013

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and R. Gauntt, "Severe Accident Analysis of a PWR Station Blackout with the MELCOR, MAAP 4, and SCDAP/RELAP5 Codes," *Nuclear Engineering and Design* 234, 129-45 (2004)).

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CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "NextEra's Motion for Summary Disposition of Friends of the Coast/New England Coalition Contention 4B (SAMA Analysis Source Terms)" has been served through the E-Filing system on the participants in the above-captioned proceeding, this 10th day of May 2013.

/Signed electronically by David R. Lewis/
____
David R. Lewis

ATTACHMENT 1

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STATEMENT OF MATERIAL FACTS

NextEra Energy Seabrook, LLC ("NextEra") hereby submits, in support of its Motion for Summary Disposition of Friends of the Coast/New England Coalition Contention 4B (SAMA Analysis of Source Terms), this statement of material facts as to which NextEra contends there is no genuine dispute.

A. General

1. Severe accident source terms can be generated using computer codes such as (1) the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code, or (2) the Modular Accident Analysis Program ("MAAP") code. O'Kula Decl. at ¶¶ 19, 71.

2. NextEra used Version 4 of the MAAP code ("MAAP 4") to generate source terms in connection with the SAMA analyses performed in support of the license renewal application for Seabrook ("Seabrook"). NextEra used MAAP Version 4.0.5 in its initial Severe Accident Mitigation Alternatives ("SAMA") analysis. NextEra used MAAP Version 4.0.7 in an updated SAMA analysis for Seabrook that was used with Seabrook

probabilistic risk assessment ("PRA") information to evaluate risk metrics associated with 13 postulated severe accident release categories. O'Kula Decl. at ¶¶ 20, 34.

B. Development of the MAAP Code

3. The MAAP code is used for accident analysis by a wide array of entities, including utilities, vendors, research organizations, and universities. O'Kula Decl. at ¶¶ 19, 25.

4. The MAAP code has a strong technical basis for use in PRA and severe accident analysis and has been accepted for use in numerous NRC-approved analyses. O'Kula Decl. at ¶¶ 11, 19, 21, 24, 27-29, 49, 75.

5. The MAAP code simulates thermal-hydraulic and fission product phenomena in both the primary and containment systems of pressurized water reactors ("PWRs") in connection with severe accidents. O'Kula Decl. at ¶ 19.

6. MAAP 4 incorporates updated physical models for core melt, reactor vessel lower head response, and containment response in connection with severe accidents. O'Kula Decl. at ¶ 24.

7. MAAP has been the subject of extensive benchmarking and validation studies in the areas instrumental to severe accident source term estimation. O'Kula Decl. at ¶¶ 11, 21, 26, 75.

8. The original version of MAAP and its successor versions, including MAAP 4, were developed in accordance with 10 C.F.R. Part 50, Appendix B and International Organization for Standardization ("ISO") 9001 quality assurance requirements. O'Kula Decl. at ¶ 23.

9. The Electric Power Research Institute ("EPRI") has identified the MAAP code (Version 4.0.5 and later) as a "consensus computer code" suitable for use in evaluation of PRA success criteria. O'Kula Decl. at ¶ 24.

10. Development of MAAP 4 (1) was sponsored by several organizations, including EPRI and the Department of Energy, (2) included a peer review by a committee of independent experts, and (3) involved an additional review by a Design Review Committee comprised of senior members of the nuclear safety community. O'Kula Decl. at ¶ 24.

11. The MAAP 4 code has been benchmarked and validated against the results of (1) numerous severe accident studies, as well as (2) the Three Mile Island Unit 2 ("TMI-2") core melt accident. Both EPRI and the Nuclear Energy Agency have documented the benchmarking and validation of the MAAP 4 code. O'Kula Decl. at ¶ 26.

12. The NRC Staff has found use of the MAAP code acceptable by numerous license renewal applicants in simulating severe accident phenomenology for supporting SAMA analysis. O'Kula Decl. at ¶¶ 27-28.

C. Comparison of NUREG-1465 Computer Codes with MAAP

13. The use of plant-specific source terms derived from the MAAP code is preferred over the use of generic source terms extracted from NUREG-1465 for a SAMA analysis, which evaluates plant specific design and operational changes. O'Kula Decl. at ¶¶ 12-13, 18, 31-32, 45, 75.

14. NextEra used the MAAP 4 code to integrate plant-specific information within the Level 2 PRA for the Seabrook SAMA analysis to obtain source term groups, descriptions, and release category information for 13 release categories evaluated during Seabrook's Level 2 PRA. O'Kula Decl. at ¶¶ 30, 34-35.

15. NUREG-1465 postulates generic release fractions and a single, generic source term for use in determining compliance with 10 C.F.R. Part 100 reactor siting criteria. O'Kula Decl. at ¶¶ 37-38, 45-46.

16. NUREG-1465 quantifies only the amount and types of radioactive material released into containment following a severe accident, not the environment. O'Kula Decl. at ¶¶ 36, 39-40.

17. NUREG-1465 provides data only for a single PWR release into the containment and its source term is a generic source term with no basis in plant-specific information and data from Seabrook. O'Kula Decl. at ¶ 46.

18. A SAMA analysis requires a plant-specific evaluation of the spectrum of plant-specific releases to the environment. O'Kula Decl. at ¶¶ 12-13, 27, 31-32, 45-47, 57, 75.

19. The MAAP 4-generated source terms used in Seabrook's Level 2 PRA account for the risk associated with a range of timing and containment damage scenarios for Seabrook. O'Kula Decl. at ¶¶ 43-44, 47.

20. Unlike NUREG-1465, MAAP quantifies fission product removal mechanisms (including active or passive engineered safety features, and natural processes) in modeling the release of radionuclides into the environment following a postulated severe accident. O'Kula Decl. at ¶¶ 39-41.

D. Comparison of Legacy Codes or Code Versions, and Older Versions of the MAAP Code, with the MAAP 4 Code

21. Comparisons of earlier versions of MAAP to earlier versions of MELCOR or its predecessor, the Source Term Code Package ("STCP"), are not material to NextEra's use of the current versions of MAAP today. O'Kula Decl. at ¶¶ 55-59, 61-62, 71, 74-75.

22. Neither Draft NUREG-1150[1] nor the BNL Report[2] compared release fractions obtained from the MAAP 4 code used in the Seabrook SAMA analysis. O'Kula Decl. at ¶¶ 53-55.

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[1] Office of Nuclear Regulatory Research, Draft for Comment, Reactor Risk Reference Document, NUREG-1150, Vol. 1 (Feb. 1987) ("Draft NUREG-1150").

23. Draft NUREG-1150 compares release fractions obtained from early versions of the MAAP code (Versions 1.1 through 3.0) against release fractions obtained from an alternative legacy code, the STCP code. O'Kula Decl. at ¶¶ 53, 55.

24. The BNL Report compares release fractions obtained from MAAP Version 3B against release fractions obtained from alternative legacy codes (the STCP code) and code versions (specifically, an older version of the MELCOR code) identified in NUREG-1150. O'Kula Decl. at ¶¶ 54-55, 62.

25. The BNL Report's comparison of release fractions from Catawba and Sequoyah are immaterial to the Seabrook SAMA analysis because those plants have significantly different design features for control and mitigation of radioactive release in a severe accident (*e.g.*, ice condenser containments), than Seabrook, which has a dry, ambient air containment. O'Kula Decl. at ¶ 61.

26. The NRC Staff has found that the STCP/MELCOR and MAAP codes produce consistent results when used to compare release fractions for a single plant. Specifically, the Staff reviewed MAAP-based source term estimates for the major release categories and found those predictions to be in reasonable agreement with estimates of NUREG-1150 for the closest corresponding release scenarios. O'Kula Decl. at ¶ 60.

27. The understanding of severe accident modeling has improved considerably over time. As the State-of-the-Art Reactor Consequence Analyses ("SOARCA") Project demonstrated, current modeling of severe accidents shows a much smaller and delayed radioactive release than was recognized in earlier studies and calculated with older computer code models. O'Kula Decl. at ¶¶ 11, 57, 59, 61-64, 68, 75.

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[2] John R. Lehner et al., Brookhaven National Laboratory, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" (Dec. 2002) ("BNL Report").

28. The release fractions obtained using the MAAP Version 4.0.7 code within Seabrook's Level 2 PRA are reasonably consistent with release fractions for the Surry Power Station (another PWR with a dry, ambient air containment) obtained in the SOARCA project using the MELCOR Version 1.8.6 code. O'Kula Decl. at ¶¶ 68, 70, 72.

Respectfully Submitted,

/Signed (electronically) by David R. Lewis /

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Counsel for NextEra Energy Seabrook, LLC

Dated: May 10, 2013

ATTACHMENT 2

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety And Licensing Board

In the Matter of	)	
	)	Docket No. 50-443-LR
NextEra Energy Seabrook, LLC	)	
	)	
(Seabrook Station, Unit 1)	)	
	)	May 10, 2013

DECLARATION OF KEVIN R. O'KULA IN SUPPORT OF NEXTERA'S MOTION FOR SUMMARY DISPOSITION OF FRIENDS OF THE COAST/NEW ENGLAND COALITION CONTENTION 4B (SAMA ANALYSIS SOURCE TERMS)

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Kevin R. O'Kula states as follows under penalties of perjury:

I. PROFESSIONAL QUALIFICATIONS

Dr. Kevin R. O'Kula

1. My name is Kevin R. O'Kula. I am an Advisory Engineer with URS Professional Solutions ("URS") LLC, in Aiken, South Carolina, and a consultant to NextEra Energy Seabrook, LLC ("NextEra") on source term, dispersion/consequence and severe accident mitigation alternatives ("SAMA") analysis issues.

2. My professional qualifications are provided in the attached *curriculum vitae*. In brief, I have over 30 years of experience as a technical professional and manager in the areas of safety analysis methods and guidance development, computer code validation and verification, probabilistic risk assessment ("PRA"), design basis and severe accident and consequence analyses for reactor and non-reactor nuclear facilities, source term evaluation, risk management, software quality assurance ("SQA"), and shielding analysis. I obtained my B.S. in Applied and

Engineering Physics from Cornell University in 1975, and my M.S. and Ph.D. in Nuclear Engineering from the University of Wisconsin in 1977 and 1984, respectively.

3. In addition, I have over 25 years of experience using and applying the MELCOR Accident Consequence Code System 2 ("MACCS2") computer code or its predecessor, MACCS, to support PRA and nuclear facility deterministic analyses. I co-chaired a U.S. Department of Energy ("DOE") Accident Phenomenology and Consequence evaluation program in the 1990s that evaluated applicable computer models for radiological dispersion and consequence analysis. More recently, I was a technical peer reviewer of the Sandia National Laboratories ("Sandia") and NRC State-of-the-Art Reactor Consequence Analyses ("SOARCA") Project that reviewed updated and more realistic evaluations of severe accident progression in U.S. nuclear power plants. By virtue of my training and experience, I also am familiar with the Modular Accident Analysis Program, ("MAAP") code, including the version ("MAAP 4") used in the Seabrook Station SAMA analysis.

4. I have taught MACCS2 training courses for DOE and its contractors at Lawrence Livermore National Laboratory, Los Alamos National Laboratory, Idaho National Laboratory, Oak Ridge, the Waste Isolation Pilot Plant, and the DOE Safety Basis Academy. In addition, I was the lead author of a DOE guidance document on the use of MACCS and MACCS2 for DOE safety analysis applications, and managed overall completion of equivalent reports for DOE on MELCOR (similar in function to MAAP) and GENII (similar in function to MACCS2). As part of the SOARCA Project Peer Review Committee, I provided critical review and comment to Sandia and the NRC on the use of integrated modeling of accident progression and offsite consequences from postulated severe accidents using both improved computational

analysis tools and more accurate inputs and assumptions reflecting current-day plant operations, and improved accident management/response planning.

5. In addition to supporting NextEra on SAMA analysis-related issues regarding the Seabrook Station license renewal application, I have supported other license renewal applications. These include both completed and ongoing support of SAMA activities for the license renewal applications of Pilgrim Nuclear Generating Station, Prairie Island Nuclear Power Plant, Indian Point Energy Center, and Davis-Besse Nuclear Power Station.

6. I am providing this Declaration in support of "NextEra's Motion for Summary Disposition of the Friends of the Coast/New England Coalition Contention 4B (SAMA Analysis Source Terms)" in the above-captioned proceeding. I understand that, in May 2010, NextEra submitted a license renewal application ("LRA") to the U.S. Nuclear Regulatory Commission ("NRC") seeking to renew its operating license for the Seabrook Station Unit No. 1 ("Seabrook Station") for another 20 years.[1] The SAMA analysis was originally described in Section 4.20 and Attachment F of the Environmental Report ("ER") (Appendix E to the LRA).[2]

7. I have thoroughly reviewed the various inputs and assumptions used in NextEra's SAMA analysis, as submitted in May 2010 and revised in March 2012,[3] to calculate offsite consequences associated with a postulated severe accident at the Seabrook Station, including relevant supporting technical documentation for the SAMA analysis prepared by a NextEra contractor. I also have reviewed the SAMA analysis revisions and clarifications that

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[1] *See* Letter from Paul O. Freeman, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Seabrook Station, Application for Renewed Operating License," May 25, 2010 (ADAMS Accession No. ML101590099).

[2] *See* ER § 4.20 (Severe Accident Mitigation Alternatives) & Attach. F (Severe Accident Mitigation Alternatives Analysis).

[3] *See* Letter from Paul O. Freeman, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Seabrook Station, Supplement 2 to Severe Accident Mitigation Alternatives Analysis" March 19, 2012 (ADAMS Accession No. ML12080A137).

NextEra provided in response to NRC Staff requests for additional information ("RAIs") in January, April and June 2011, and September 2012, and Supplement 2 of the SAMA analysis (March 2012).[4] I thus have personal knowledge of the modeling methods, inputs, and assumptions used in the Seabrook Station SAMA analysis, as described in the Seabrook Station ER and other related documentation. I have also reviewed the relevant portions of the Second Draft Report for Comment of NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 46 Regarding Seabrook Station (Apr. 2013) concerning SAMA analysis, specifically Section 5.3 and Appendix F.

8. In preparing this Declaration, I also reviewed relevant pleadings of the parties and Orders issued by the Atomic Safety and Licensing Board ("Board") and the Commission in this proceeding, applicable NRC regulations and guidance documents, and relevant technical reports and studies.

II. ISSUES RAISED IN CONTENTION 4B

9. I understand that, as admitted by the Board and narrowed in scope by the Commission, Friends of the Coast and the New England Coalition ("Friends/NEC") Contention 4B challenges NextEra's use of the MAAP computer code to the determine the source terms

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[4] *See* Letter from Paul O. Freeman, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Seabrook Station, Response to Request for Additional Information" SBK-L-11001, January 13, 2011 (ADAMS Accession No. ML110140810); Letter from Paul O. Freeman, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Seabrook Station, Response to Request for Additional Information" SBK-L-11067, April 18, 2011 (ADAMS Accession No ML11111A035); Letter from Paul O. Freeman, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Seabrook Station, Supplement to Response to Request for Additional Information - April 18, 2011", SBK-L-11125, June 10, 2011 (ADAMS Accession No ML11166A255); Letter from Kevin T. Walsh, Site Vice President, NextEra Energy Seabrook, LLC to Document Control Desk, U.S. N.R.C. "Supplement 3 to Severe Accident Mitigation Alternatives Analysis - Response to RAI Request dated July 16, 2012", SBK-L-12185, September 13, 2012 (ADAMS Accession No ML12262A513).

(including release fractions) used in the Seabrook Station SAMA analysis.[5] *NextEra Energy Seabrook, LLC* (Seabrook Station Nuclear Power Station, Unit 1), LBP-11-2, 73 N.R.C. 28, 64-68 (2011); *Seabrook*, CLI-12-5, 75 N.R.C. __, slip op. at 29-31 (2012). Specifically, Contention 4B alleges that the use of source terms generated by MAAP "appears to lead to anomalously low consequences when compared to source terms generated by NRC staff" because: (1) the MAAP code has not been validated by the NRC; (2) the radionuclide release fractions generated by MAAP are consistently smaller for key radionuclides than the release fractions specified in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants* (Feb. 1995) (ML041040063) and its recent revision for high-burnup fuel; and (3) it has been previously observed that MAAP generates lower release fractions than those derived and used by NRC in studies such as NUREG-1150. Petition at 41, 44-45. Each of these claims is discussed and refuted in this Declaration.

III. SUMMARY OF KEY POINTS AND CONCLUSIONS

10. This Declaration explains why NextEra's use of the MAAP[6] code is reasonable and appropriate within Seabrook Station's PRA-based, site-specific SAMA analysis. This Declaration also explains why Friends/NEC's criticisms of the MAAP code as applied in the Seabrook Station SAMA analysis are entirely without merit.

11. Friends/NEC make three principal claims in support of their contention, which, for clarity and ease of reference, are addressed as Bases 1, 2 and 3 (*see* paragraph 9). In summary, the Friends/NEC contention and three supporting arguments lack merit for the following reasons:

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[5] Source term, release fraction, and other PRA-related terms used in this Declaration are defined in Table 1 and discussed further throughout this Declaration. In this Declaration, the figures are found in the text of the Declaration. The tables are found at the end of the Declaration.

[6] Although NextEra has used MAAP Versions 4.0.5 and 4.0.7 (MAAP4) in support of the Seabrook Station SAMA analysis, the term "MAAP" will be used as appropriate for brevity and convenience.

a. The MAAP code used in Seabrook Station's SAMA analysis is a proven and validated computer model that simulates the response of nuclear power plants to postulated design basis and severe accident conditions. MAAP has been repeatedly accepted by the NRC in connection with quantitative estimates of plant-specific source terms in other light water reactor ("LWR") plant license renewal proceedings. The MAAP code has been developed and maintained in accordance with NRC quality assurance standards, extensively benchmarked, applied to different reactor designs throughout the world, identified as a consensus computer code suitable for use in PRA applications, and has long been accepted by the NRC for use in supporting environmental applications, including numerous NRC-approved SAMA analyses in operating plant license renewal proceedings.

b. MAAP generates plant-specific, quantitative source terms and release fractions appropriate for use in determining the release of radionuclides from the containment into the environment following a postulated severe accident. In the context of a SAMA analysis, the MAAP source term output is used as part of the subsequent input into the consequence analysis to quantify offsite population doses and economic costs. The MAAP code uses plant-specific information to account for the engineered and natural fission product removal mechanisms following release of radionuclides into containment for each source term. The Seabrook Station SAMA analysis used a recent version of MAAP (MAAP 4.0.7) to generate source terms based on its plant-specific PRA analysis of initiating events and accident progression. The current Seabrook Station PRA analysis includes 13 plant-specific source terms to the environment, and these are labeled release categories.

In contrast, the generic severe accident source term discussed in NUREG-1465 and recommended by Friends/NEC is inappropriate for use in SAMA analyses. NUREG-1465 presents only one set of generic pressurized water reactor ("PWR") release fractions and associated timing for release into the environment, representing an in-containment accident source term for plants, some of which have different types of containment compared to Seabrook Station. While a number of engineering safety systems and natural removal mechanisms in commonly credited in PWRs are discussed, NUREG-1465 does not quantify release fractions from containment into the environment. Applying NUREG-1465's single, generic source term to replace all 13 plant-specific release categories examined in the Seabrook Station SAMA analysis would treat all releases into the containment as releases into the environment; *i.e.*, it would treat a wide spectrum of containment failure and containment bypass events equivalently and would neglect source term reductions that would occur as a result of engineered and natural fission product removal mechanisms following release of radionuclides into containment. Consequently, the alternative NUREG-1465 source term recommended by the Friends/NEC is unrealistic, represents a worst-case source term, and is inappropriate for SAMA analysis.

c. The MAAP code applied in Seabrook Station's SAMA analysis represents a contemporary, integrated thermal-hydraulic and fission product response simulation accounting for important phenomena in determining the release of radionuclides from the containment into the environment following a postulated severe accident. Specifically, Seabrook Station PRA source terms are evaluated using MAAP code Version 4.0.7 for a large set of accident sequences within each release category. The MAAP code accounts

explicitly for source term release and depletion methods based on the current best estimate understanding of severe accident phenomenon. The MAAP code was used to generate source terms by running Seabrook Station-specific models.[7]

In contrast, the Friends/NEC's contention references comparisons between legacy computer models and older versions of the MAAP code than that used in the Seabrook Station SAMA analysis. The draft NUREG-1150 study cited by Friends/NEC (Pet. at 45 (citing NRC Office of Nuclear Regulatory Research, *Draft for Comment, Reactor Risk Reference Document*, NUREG-1150, Vol. 1, at 5-14 (Feb. 1987))[8] was published over 25 years ago and involved an assessment of severe accident risks at five commercial nuclear power plants in the United States. The Seabrook Station was not among the five plants evaluated in the NUREG-1150 study, and the source term analyses being compared in that study lacked (in comparison with the MAAP code used in the Seabrook Station SAMA analysis) a quarter-century of research and operational experience. Further, the final version of NUREG-1150 omitted the comparison between historical source terms relied on by the Friends/NEC's contention. Similarly, the 2002 Brookhaven National Laboratory (BNL) report cited by the Friends/NEC compared source terms obtained in the Level 2 PRA for different plants (Sequoyah and Catawba), using a (now) decade-old version of MAAP and the same 25-year-old source term basis described in the NUREG-1150 study. John R. Lehner et al., Brookhaven National Laboratory, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" at 17 (Dec. 2002) ("BNL report"). Lastly, recent NRC studies of the offsite radiological health effects consequences — in

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7 SBK-L-12053, Supplement 2 to Severe Accident Mitigation Alternatives Analysis, at 6 (March 19, 2012).

8 The NRC published NUREG-1150 in final form in December 1990. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Vol. 1, Tbl. 2.1 at 2-3 (Dec. 1990) ("NUREG-1150").

particular, the State-of-the-Art Reactor Consequence Analyses Report — have revised the phenomenological models and implemented more realistic, best-estimate models incorporating updated experiments and test data that were unavailable at the time of the NUREG-1150 study referenced in the BNL study. Friends/NEC provide no explanation as to why their recommendation that the use of the older, legacy computer models should supersede the contemporary, updated MAAP computer models. Furthermore, Friends/NEC provides no technical basis for why the quantitative source terms applied in the Seabrook Station SAMA analysis are unreasonable.

IV. REGULATORY AND TECHNICAL BACKGROUND

A. SAMA Analysis Is a Plant-Specific, Quantitative Evaluation

12. SAMA analysis is a site-specific, frequency-weighted assessment of the benefits and costs of mitigation alternatives ("SAMAs" or "SAMA candidates") that could reduce the risks (frequencies or consequences, or both) of potential nuclear power plant severe accidents. NEI 05-01 at 1. NRC-endorsed industry guidance on SAMA analyses states: "The purpose of the analysis is to identify SAMA candidates that have the potential to reduce severe accident risk and to determine if implementation of each SAMA candidate is cost-beneficial." NEI 05-01, Rev. A, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" at 1 (Nov. 2005) ("NEI 05-01") (ML060530203); "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses" (Aug. 2007), 72 Fed. Reg. 45,466 (Aug. 14, 2007) (endorsing NEI 05-01, Rev. A). Changes to the plant that could reduce the risk of a severe accident include plant modifications or operational changes (*e.g.*, improved procedures and augmented training of

control room and plant personnel). NEI 05-01 at 23. These potential changes are referred to as SAMAs or SAMA candidates. *Id.* at 23-26.

13. The methodology for the overall SAMA analysis approach is based on NRC guidance in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook, Final Report" (January 1997) (ML050190193) and NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4" (August 2004) (ML042820192). A SAMA analysis consists four major principal steps:

- Characterize the overall *plant-specific* severe accident risk using PRAs and other risk studies to identify the leading contributors to core damage frequency ("CDF") and offsite risk based on a plant-specific risk study;
- Identify potential plant improvements (*i.e.*, SAMA candidates) that could reduce the risk of a severe accident;
- Quantify the risk-reduction potential and the implementation cost for each SAMA candidate; and
- Determine whether implementation of the SAMA candidates may be cost-effective.

NUREG-1850, "Frequently Asked Questions on License Renewal of Nuclear Power Reactors, Final Report," (March 2006) ("NUREG-1850") at 4-32 to 4-34 (emphasis added). The SAMA evaluation of a plant is based on the numerical evaluation of severe accident risk impacts in four categories: (1) offsite exposure cost, (2) offsite economic cost, (3) onsite exposure cost, and (4) onsite economic cost. NEI 05-01 at 28. The basis for a SAMA analysis conducted for a U.S. nuclear power plant is a sequential, three-level PRA, *i.e.*, a comprehensive assessment of postulated accident sequences resulting in damage to the core and containment, radiological release, and their associated frequencies. A PRA assesses the risk from an operating nuclear power plant by answering three basic questions: (1) What can go wrong?; (2) How likely is it?; and (3) What are the consequences? As shown in Figure 1, the PRA for a commercial nuclear power plant ("NPP") is divided into three levels—Level 1, Level 2, and Level 3 — all of which

are required to perform a plant-specific SAMA analysis. Most, if not all NPPs, maintain plant-specific Level 1 and 2 PRA models. As will be summarized in paragraph 14, the Level 3 Probabilistic Safety Assessment ("PSA")-type information needed for a SAMA analysis is quantified following the NEI 05-01 guidance.[9]

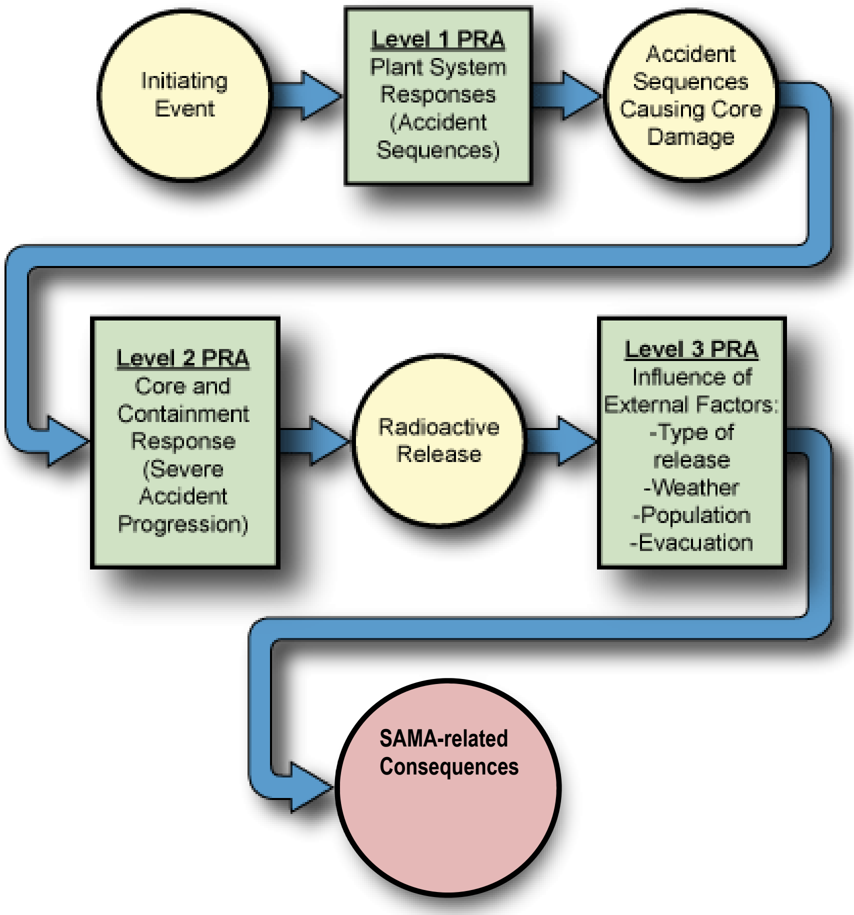

Figure 1. Three-Level Probabilistic Risk Assessment for Reactor Operation (adapted from U.S. Nuclear Regulatory Commission, *Probabilistic Risk Assessment (PRA)*. http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html)

[9] The term PSA is often used interchangeably with PRA. This Declaration uses both terms without distinction.

B. The MAAP 4 Code Provides Plant-Specific, Quantitative Environmental Source-Term Inputs for the SAMA Analyses

14. Various computer codes — including the MAAP code at issue in Contention 4B — are used in support of a SAMA analysis. These include codes used to support development of a Level 1 PRA (analysis of initiating events and ensuing accident sequences leading to core damage) and a Level 2 PRA (analysis of accident progression leading to containment failure and bypass, and release of radionuclides to the environment). The output of the Level 1 PRA is used in the Level 2 PRA. The output of the Level 2 PRA, in turn, is used in the Level 3 PRA determination of offsite consequences (offsite dose and offsite economic impacts) portion of the analysis. As identified in NEI 05-01, in many SAMA analyses the MACCS2 code is used to provide the Level 3 PRA-type consequences from postulated severe accident releases of radioactive materials to the atmosphere. NEI 05-01 at 13.

15. To calculate the consequence values for a SAMA analysis, the computer model requires information on the *core inventory* (*i.e.*, the amount of each radionuclide present in the reactor core at the time accident initiation) and characteristics of the postulated release, including the amount of each radionuclide released and their physical-chemical characteristics, the release height, release duration, and the sensible energy released.

16. The *source term* refers to the amount and isotopic composition of material released (or postulated to be released) from the reactor core during an accident. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Vol. 1, Table 2.1 at 2-3 (Dec. 1990).Two types of source term should be differentiated, the containment source term and the environmental source term. The source term that refers to the quantity of the each radionuclide group in the reactor core inventory at the start of an accident that is released to the containment is defined as the *containment source term*. The source term that refers to the

quantity of the each radionuclide group in the reactor core inventory at the start of an accident that is released to the environment is defined as the *environmental source term*. (This is an important distinction that is discussed further below in addressing the claims made in Contention 4B.) An *environmental source term* describes the physical, chemical, and radiological composition of an atmospheric release. The environmental source term is input into the atmospheric transport and dispersion module to quantify the population dose and economic cost consequences that are estimated for purposes of supporting a SAMA analysis.

17. One component of the source term is the *release fraction*, which is the fraction of the total activity of the fission products present at the start of an accident that are released from the core to the environment during the postulated accident. NUREG-1150, Vol. 1 at 10-4.

18. Generation of accurate source terms for use in SAMA analysis requires a great deal of plant-specific information to develop a detailed analytical model that includes a multitude of physical process sub-models that account for, among other things, the timing and performance of both passive and active plant safety features and human (*i.e.*, operator) actions affecting accident progression and containment conditions. Any radionuclide releases outside of containment are sequentially modeled beginning with their release from the reactor core through any release path from the containment (through partial containment failure or bypass conditions), and subsequently into the environment. Source terms depend on how rapidly the accident progresses, the path by which the radionuclides escape from the reactor into containment, the path through containment (or possibly bypassing containment altogether), and the effectiveness of both passive and active safety features, for example, sprays and filtration systems in the containment, that are intended to mitigate releases. An example of mitigation provided by sprays would be the "scrubbing" or removal of radionuclides and cooling of the

internal environment to which radionuclides have been released (which reduces the containment internal pressure driving the release).

19. In the U.S., severe accident source terms for SAMA analysis usually are calculated using one of two computer codes: (1) the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code or (2) the MAAP code. MAAP simulates the dominant thermal-hydraulic and fission product phenomena in both the primary and containment systems of PWRs. MAAP evaluates a broad spectrum of phenomena, including steam formation; core heat-up; cladding oxidation and hydrogen evolution; vessel failure; corium-concrete interactions; ignition of combustible gases; fluid entrainment by high-velocity gases; and fission-product release, transport, and deposition. It also addresses important engineered safety systems and allows a user to model operator interventions. The thermo-hydraulic and fission product phenomena simulated by the MAAP code are shown in Figure 2.

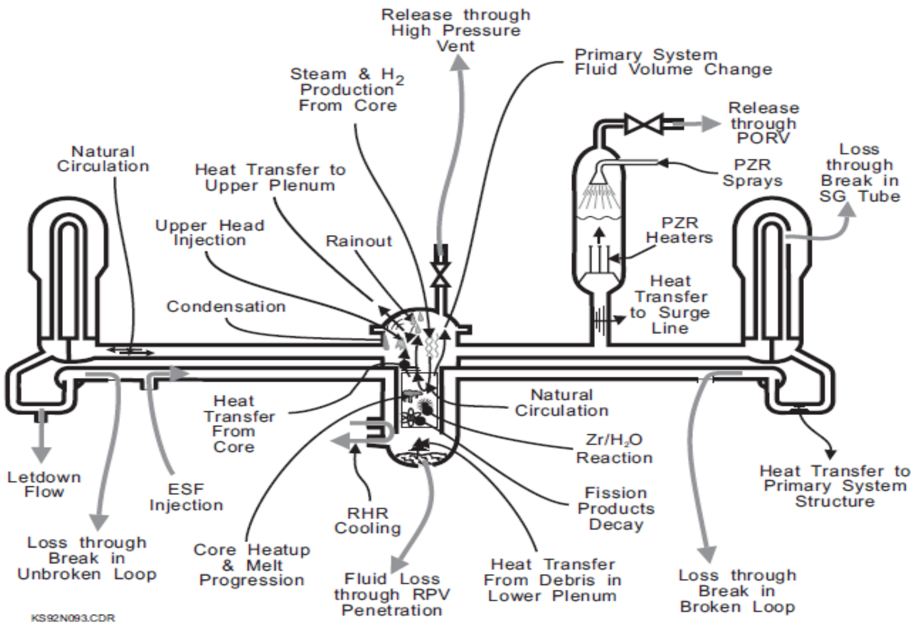

Figure 2. MAAP 4 Primary System Modeling (from Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program)*, *available at* http://www.fauske.com/pdf/MAAP.pdf) Electric Power Research Institute ("EPRI") Report 1020236, "MAAP 4 Applications Guidance: Desktop Reference for Using MAAP 4 Software, Revision 2" at 2-2 to 2-3 (2010) ("MAAP 4 Applications Guidance"); Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program)*, http://www.fauske.com/pdf/MAAP.pdf.

20. As discussed in the Environmental Report, NextEra used MAAP 4.0.5 in support of its initial SAMA analysis. Environmental Report, Appendix E - Environmental Report, Attachment F Severe Accident Mitigation Alternatives at F-47. More recently, MAAP 4.0.7 was used to support Seabrook Station's updated SAMA analysis. Specifically, source terms were further revised during the SB2011 PRA model update based on more detailed modeling using the Modular Accident Analysis Program (MAAP), Version 4.0.7. The revised Level 2 analysis models 13 postulated severe accident release categories, each with a specific annual frequency of occurrence.[10] Seabrook Station Supplement 2 to Severe Accident Mitigation Alternatives of the License Renewal Application (March 19, 2012) at 5.

[10] *See* Table 2 and Table 3, *infra*.

V. RESPONSE TO ISSUES RAISED IN CONTENTION 4B

A. The MAAP Code Has Been Extensively Validated and Benchmarked, and Repeatedly Approved by the NRC for Use in License Renewal Proceedings (Basis 1)[11].

21. Contrary to the claim of the claim of Friends/NEC (Pet. at 44), the MAAP code has been subject to extensive U.S. and international benchmarking and validation efforts, and the NRC has accepted source terms calculated by numerous licensees with MAAP for use in their respective license renewal SAMA analyses.

22. MAAP was originally developed for the Industry Degraded Core Rulemaking ("IDCOR") program in the early 1980s by Fauske & Associates, LLC (formerly Fauske & Associates, Inc.). At the completion of IDCOR, ownership of MAAP was transferred to the Electric Power Research Institute (EPRI), which was charged with maintaining and improving the code.

23. The first version of the code, MAAP 1.1, was released in 1983 to support the IDCOR program. Since then, the code has been continually improved. MAAP 3.0 was released in 1986, with MAAP 3.0B being released in 1988.[12] MAAP version 3B became widely used, first in the United States and then worldwide, to support success criteria determination, human action timing evaluations, and Level 2 analyses for Individual Plant Examinations (IPEs) required by the NRC. MAAP 3B was updated to MAAP 4 in the mid-1990s to expand its modeling capabilities. MAAP and its successor versions, including MAAP 4, were developed in accordance with 10 C.F.R. Part 50, Appendix B and International Organization for Standardization ("ISO") 9001 quality assurance requirements.

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[11] Each basis corresponds to one of the arguments summarized in paragraph 9 and then responded to as a basis in paragraph 11.

[12] Severe Accident Analysis, A Nordic Study of Codes, Final Report of the Project NKA-AKTI-130, Aro L, Finnish Centre for Radiation and Nuclear Safety Blomquist P., Studsvik Nuclear Fynbo P., Risø National Laboratory Pekkarinen E., Technical Research Centre of Finland Schougaard B., Elsam (1989) at 7.

24. MAAP 4 incorporates updated physical models for core melt, reactor vessel lower head response, and containment response that provide improved mechanistic modeling of severe accident phenomena. MAAP 4's development was sponsored by several organizations, including EPRI and the DOE. As part of the development process, a committee of independent experts reviewed MAAP 4 to ensure that it was contemporary and applicable for accident management evaluations. Further, the new software was subjected to review by a Design Review Committee, comprised of senior members of the nuclear safety community. EPRI Report 1020236, *MAAP 4 Applications Guidance: Desktop Reference for Using MAAP 4 Software, Revision 2* at 2-2 (2010) ("MAAP 4 Applications Guidance"), available at http://my.epri.com. EPRI has identified the MAAP code (Version 4.0.5 and later) as a "consensus computer code" suitable for use in evaluation of PRA success criteria. EPRI Report 1013492, *Probabilistic Risk Assessment Compendium of Candidate Consensus Models* at 2-3 (2006), *available at* http://my.epri.com.

25. EPRI licenses MAAP 4 to a wide array of entities, such as utilities, vendors, and research organizations, including universities. The majority of MAAP 4 users are members of the MAAP Users Group ("MUG"). The MUG provides direction and funding for code maintenance, enhancements, and benchmarking; facilitates information transfer through biannual meetings and the issuance of various communications on code problems and best practices; and supports industry and regulatory acceptance. *MAAP 4 Applications Guidance* at 2-2. Fauske & Associates is the current maintenance contractor for the MAAP 4 code.

26. MAAP has been sponsored by the nuclear industry in part to predict the phenomenology of severe accident progression and has been benchmarked, to the extent possible, with applicable severe accident experimental research results. For example, MAAP

has been successfully benchmarked against numerous severe accident studies and the Three Mile Island Unit-2 ("TMI-2") core melt accident. The extensive benchmarking of MAAP is documented in the Section 7 (MAAP Benchmarks) and Appendix F (Summaries of MAAP Benchmarks) of EPRI's *MAAP 4 Applications Guidance*, and also in a 2007 report issued by the Nuclear Energy Agency ("NEA"). The 2007 NEA report summarizes key MAAP benchmarking activities as follows:

> Many comparisons between the MAAP code and separate effects tests, integral experiments, actual plant transients, and accidents have been performed to illustrate the performance of individual models and to provide confidence in the MAAP integral results. The assessment matrix listed in Table 3.2 shows the experimental benchmarking status of the MAAP computer code. It is seen that the various code versions . . . have been compared to several separate effects and integral experiments. These include: CORA and PHEBUS (core damage); LOFT FP-2 (integral severe accident test); ABCOVE (aerosol behaviour); CSE (containment spray); COPO (molten pool heat transfer); FARO (debris quenching); Surtsey IET (DCH); SWISS, SURC-4, ACE, KfK BETA (core-concrete interaction); NUPEC mixing tests; Marviken, FAI, and GE vessel blowdown tests; and HDR containment experiment, among many others. The current version of the code, MAAP 4, has also been benchmarked against the TMI-2 accident. This comparison study shows that MAAP 4 provides a reasonable simulation of the TMI-2 accident in terms of the system response prior to core uncovery, during core degradation, following core reflood, and the lower head behaviour after 224 minutes. These are all severe accident processes that are essential for application of computer codes for decisions related to design, operations, emergency operating procedures, and accident management.

NEA Committee on the Safety of Nuclear Installations, NEA/CSNI/R(2007)16, *Recent Developments in Level 2 PSA and Severe Accident Management* at 36 (Nov. 2007), *available at* http://www.oecd-nea.org/nsd/ docs/2007/csni-r2007-16.pdf. MAAP's extensive benchmarking and validation demonstrates the reasonableness of its use in the calculation of severe accident source terms for the Seabrook Station SAMA analysis.

27. Furthermore, the MAAP code has long been used by nuclear plant licensees to predict the response of nuclear power plants to postulated severe accidents. To my knowledge,

MAAP is the most commonly used code in the U.S. for such purposes.[13] The use of MAAP and its successor versions in IPEs and subsequent PRA applications has been accepted by the NRC Staff for many years.[14] The NRC previously has described its approach to reviewing licensee submittals that rely on MAAP for a design basis application, (also suggesting how the approach will be extended to submittals that rely on MAAP for severe accident applications) as follows:

> For each plant-specific submittal that relies on MAAP for a design-basis application, we will review those portions of the code relevant to the application, as we would any other licensing basis code. The review will generally be limited to identifying the critical MAAP models, assumptions, and code input used in the application, verifying the validity of the models by benchmarking the code with experiments and other codes, and assessing the integration of the MAAP results (*e.g.*, containment pressure and temperature history) into the analysis package. We may supplement this review by performing audit calculations (using staff codes) to confirm the results. The approval of the analysis will be limited to that specific licensing action (*i.e.*, the approval will not be an approval of MAAP.) … *This approach will also be used for plant-specific submittals that rely on MAAP for severe accident applications*, when we consider a technical review appropriate.

Letter from Gary M. Holahan, Director, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation, U.S. N.R.C., to Theodore U. Marston, Vice-President & Chief Nuclear Officer, EPRI at 1 (Dec. 4, 2001) (emphasis added). As explained below, the use of MAAP code results have been found acceptable by the NRC Staff in Level 2 PRAs supporting SAMA analyses. In these analyses, the NRC has accepted the use of the MAAP code as a tool for modeling specific severe accident phenomenology in specific reactor systems, such as a

[13] *See also* Kenneth D. Kok, Ed., *Nuclear Engineering Handbook* at 539 (2009) ("The most commonly used Level-II PRA tools include CAFTA for fault tree analysis … and the modular accident analysis program (MAAP) for severe accident simulation.").

[14] For example, the Staff accepted the use of MAAP in its 1994 design certification approval for the Advanced Boiling Water Reactor ("ABWR") design in NUREG-1503, "Final Safety Evaluation Report Related to Certification of the ABWR Standard Design," Vol. 1 at 19-53 to 19-55 (July 1994), and has done so for other subsequent design certification approvals. *See*, *e.g.*, NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Vol. 1 at 19-61 (Sept. 2004) (finding the applicant's use of both the MAAP4 and MACCS2 codes "to be consistent with the present state of knowledge regarding severe accident modeling" and acceptable).

PWR's thermal-hydraulic response and fission product release characteristics under postulated accident conditions.

28. Numerous license renewal applicants have used the MAAP code to support NRC-approved SAMA analyses, including several very recent recipients of renewed operating licenses. *See*, *e.g.*, NUREG-1437, Supp. 47, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Columbia Generating Station – Final Report," Vol. 2, App. F at F-2, F-6 to F-7, F-27 (Apr. 2012); NUREG-1437, Supp. 45, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Hope Creek Generating Station and Salem Nuclear Generating Station, Units 1 and 2, Vol. 2, App. G at G-4, G-6, G-15 to G-16 (Mar. 2011).

29. In view of the above, NextEra's use of the validated, benchmarked MAAP 4 code in the Seabrook Station SAMA analysis is consistent with NRC-approved industry practice. Consequently, MAAP 4's use as a basis for fission product release from the core, transport into the containment, and subsequent environmental source term prediction is reasonable and appropriate. Friends/NEC have offered no credible information supporting a different conclusion.

B. A MAAP 4 - Generated Source Term Is Reasonable for Use in SAMA Analysis (Basis 2)

30. The MAAP 4 code used in the Seabrook Station SAMA to generate plant-specific, quantitative source terms (including release fractions) is appropriate for use in determining the release of radionuclides into the environment following a postulated severe accident within a SAMA analysis. In contrast, the alternative source term from NUREG-1465 recommended by the Friends/NEC is inappropriate for use in SAMA analyses because it represents a generic prescriptive evaluation of the release of radionuclides into the containment

of an allegedly representative PWR plant. Furthermore, it only provides qualitative guidance for calculating the environmental source term describing release of radionuclides from the containment to the environment.

1. **<u>SAMA Analysis Source Terms Represent Plant-Specific, Quantitative Evaluation of Releases into the Environment</u>**

31. A SAMA analysis postulates and models the release of radionuclides into the environment during a severe accident. The methodology used to develop source terms for a SAMA analysis must account for plant-unique conditions, plant design, support system dependencies, plant maintenance and operating procedures, operator training, and the interdependencies among these factors that can influence the CDF estimate for a specific plant. *See*, *e.g.*, Final Rule, Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, 61 Fed. Reg. 28,467, 28,480 (June 5, 1996) (referring to SAMA analysis as "site-specific" and stating that "the Commission expects that significant efficiency can be gained by using site-specific IPE and IPEEE results in the consideration of severe accident mitigation alternatives"); NUREG-1150, Vol. 1, Section 1.2 at 1-3 ("One of the clear perspectives from this study of severe accident risks and other such studies is that characteristics of design and operation specific to individual plants can have a substantial impact on the estimated risks."); Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 2 at 7 (May 2011) (ML100910006) (stating that the scope, level of detail, and technical acceptability of these risk-informed analyses are to "be based on the as-built and as-operated and maintained plant," and "reflect operating experience at the plant."); NEI 05-01 at 2 (directing applicants to use the "plant-specific" PRA model in their SAMA analyses).

32. Plant-specific source terms developed for SAMA analysis must consider a spectrum of probabilistically-significant accident scenarios to have any meaning from a risk quantification perspective. As discussed above, the progression from the failure of individual plant components to the determination of accident frequencies, accident progressions, and offsite consequences involves plant- and site-specific phenomena and can be separated into the three PRA levels. The Level 1 PRA establishes the plant damage states and frequency of the reactor CDF. The Level 2 PRA determines different accident progressions and a set of radioactive release conditions from the containment into similar representative groups (release categories). The Level 2 PRA defines the sequence of events resulting in a radioactive release to the environment. The source term analysis then follows and quantifies the amount of radioactivity released for a given sequence, and the frequency of occurrence (*i.e.*, release categories and their respective frequencies). The Level 3 PRA focuses on the determination of offsite public dose and economic consequences associated with those releases to the environment. A PRA analysis uses detailed design-, plant type-, and site-specific information to identify initiating events and their likelihood of potentially leading to core damage, and to establish the CDF, subsequent reactor containment release, and environmental release conditions.

2. The MAAP 4 Code Generated Plant-Specific, Quantitative Source Terms for Use in the Seabrook Station SAMA Analysis

33. The MAAP 4.0.7 calculations provide an integrated analysis of the Seabrook Station plant under postulated severe accident conditions for a variety of initiating events and include the influence of operator actions and safety system actuation on accident sequence progression. MAAP calculations predict the integrated response of the reactor core, primary system, steam generators, and primary containment building. Results include the time of core

damage and reactor vessel failure to support Level 1 PRA success criteria, as well as containment response and fission product source term characterization to support the Level 2 and Level 3 PRA assessments.

34. NextEra used the Level 1 PRA and Level 2 PRA models for Seabrook Station (as discussed in Attachment F, Sections F.3.1, F.3.2 of the ER) to estimate the CDF and release category frequencies, as well as the source terms, for use in the SAMA analysis. Table 2 lists source term groups, descriptions, and release category information for the 13 release categories developed in part using MAAP for the revised SB2011 PRA Model. Table 3 lists the release category annual frequencies for the revised model, as documented in SBK-L-12053, Supplement 2 to Severe Accident Mitigation Alternatives Analysis, March 19, 2012, ("SBK-L-12053"). Severe accident sequences were developed from internal and external initiating events. Fault tree and containment event tree ("CET") logic models, plant data, and mechanistic models of severe accident phenomena (*e.g.*, MAAP) were used as part of this process. The Level 1 PRA included initiating event ("IE") and core damage ("CD") sequence analyses and yielded a set of plant damage states ("PDS") and associated frequencies.[15] The Level 2 PRA used CET and deterministic source term models to provide a set of 13 release categories, each of which has a characteristic frequency and unique timing and fission product magnitude characteristics that represent the release to the environment.[16] The 13 release categories determined from the Level 2 Seabrook Station PRA, based in part on the MAAP analysis, were applied in the MACCS2 SAMA analysis, along with other site-specific inputs, to calculate the Seabrook Station SAMA analysis risk metrics, *i.e.*, offsite population dose risk (in units of

[15] Initiating events may include, for example, a plant trip, loss-of-coolant accident, loss of offsite power, or steam generator tube rupture.

[16] Tables 5 and 6 summarize release category characteristics and the fission product release fractions for each of the 13 plant-specific Seabrook Station release categories.

person-rem/year) and offsite economic cost risk (in units of dollars/year). While not fully a representation of the Level 1 and 2 PRA for the Seabrook Station, Figure 3 does illustrate a generic process for sequential analyses that are performed as part of a three-level PRA including the use of the MAAP and MACCS2 codes.[17]

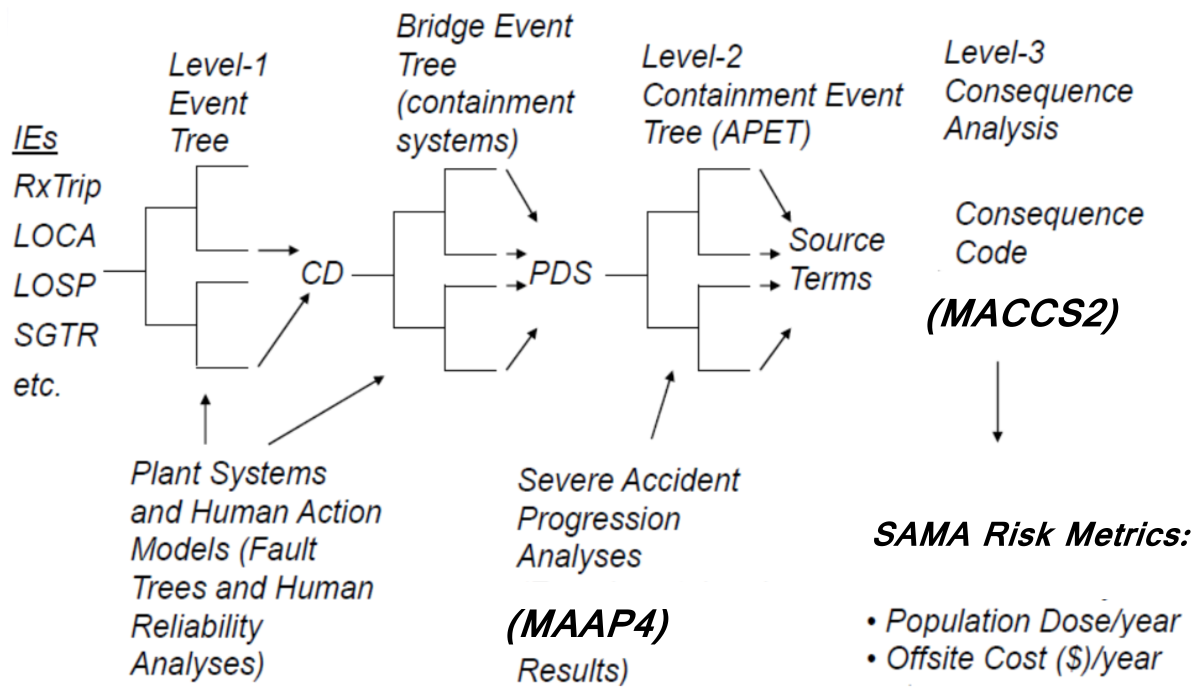

Figure 3. Sequential Analyses Performed as Part of a Three-Level PRA (Based on Electric Power Research Institute, *Basics of Nuclear Power Plant Probabilistic Risk Assessment,* Slide 10, *Fire PRA Workshop 2011*, San Diego, CA and Jacksonville, FL (2011), *available at* http://mydocs.epri.com/docs/publicmeetingmaterials/1108/J7NBS83L7MY/E236609_Module_1.pdf)

35. Thus, the PRA-based analysis using MAAP generates Seabrook Station-specific source terms for 13 different release categories that are then used in the MACCS2 analysis. This information, as summarized for each of the 13 release categories in Tables 2 through 6, is technically defensible and more reasonable for supporting the purpose of a SAMA analysis than

[17] This is a generic representation. In the Seabrook Station PRA analysis, the Level 1 and Level 2 event trees are "directly" linked eliminating the need for a bridge tree.

the prescribed single PWR containment source term from NUREG-1465. This point is discussed in greater detail below.

3. <u>The NUREG-1465 Source Term is Not Site-Specific and Represents Only Radionuclides Released into the Containment as a Result of a Core-Melt Accident</u>

36. The NUREG-1465 source term represent radionuclides released into the containment (as contrasted to the environment); *i.e.*, it assumes a "release resulting from 'substantial meltdown' of the core into the containment . . . [and assume] that the containment remains intact but leaks at its maximum allowable leak rate."[18] NUREG-1465 at 1. Indeed, NUREG-1465 states: "In *this document, a release of fission products from the core of a light-water reactor (LWR) into the containment atmosphere ("source term") was postulated* for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, 'Reactor Site Criteria.'" *Id.* at vii. (emphasis added).

37. The inapplicability of NUREG-1465's source term to SAMA analysis is evident from the origin and stated purpose of NUREG-1465. In 1962, the Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" (Mar. 1962), which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core." This "source term," the basis for NRC Regulatory Guides 1.3 and 1.4, has been used to determine compliance with the NRC's Part 100 reactor siting criteria, and to evaluate other important plant performance requirements.

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[18] Regulatory Guide 1.183 (Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, (January 2000)) provides environmental source terms to be applied for *design basis accident* consideration of the NUREG-1465 source term, and prescribes the performance of engineered safety features and the containment leak rate. It too acknowledges, "NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the *containment*." (emphasis added).

38. In the period between TID-14844 and the issuance of NUREG-1465, the knowledge base for severe LWR accidents and the associated behavior of released fission products was substantially updated and augmented. The NRC developed and issued NUREG-1465 "to provide a postulated fission product source term *released into containment* that is based on current understanding of LWR accidents and fission product behavior." NUREG-1465 at vii (emphasis added). NUREG-1465 states: "The primary objective of this report is to define a revised accident source term for regulatory application for *future* LWRs. Current LWR licensees may voluntarily propose applications based on it." *Id.* at iii (emphasis added).

39. It should be expected that MAAP produces release fractions that are different from, and generally smaller than, the release fractions specified in the NUREG-1465 source term. As discussed above, MAAP models the release of radionuclides *from the containment into the environment* following a postulated severe accident. In contrast, the NUREG-1465 source term describes only the amounts and types of radioactive material that would enter the containment. The NUREG-1465 source term does not specify the release from containment into the environment following a severe accident. Nor does it take into account the reductions in the source terms that would occur due to fission product removal mechanisms.

40. Although NUREG-1465 recognizes the importance of fission product removal mechanisms, including engineered safety features ("ESFs") and natural processes (*e.g.*, aerosol deposition and the absorption of vapors on equipment and structural surfaces), it does not attempt to quantify effects of such mechanisms. NUREG-1465 at 17-21. That is, NUREG-1465 does not provide numerical estimates of the containment source terms that account for the effects of in-containment fission product removal mechanisms (*e.g.*, containment sprays,

aerosol deposition). Rather, it directs the reader to use appropriate methodologies in crediting fission product removal or reduction within containment. NUREG-1465 at 4-5, 17-18.

41. In contrast, the MAAP code *does* model and credit these ESFs and natural processes as fission product removal mechanisms. The developer of MAAP has recognized this fact:

> Due to the strong dependence of fission product retention of plant specific features and accident sequence progression, however, NUREG-1465 source terms do not already credit retention. This is left up to the individual licensees . . . . The advantage of using [MAAP 4] is that, in a single integrated analysis, it will provide time dependent fission product release from the core, transport to the containment, leakage to the reactor or auxiliary buildings, credit for all major engineered safeguard features, and modeling of all active and passive fission product retention mechanisms.

Fauske & Associates, Inc. Technical Bulletin No. 1295-1, "BWR MSIV Leakage Assessment: NUREG-1465 vs. MAAP 4.0.2" (http://www.fauske.com/Download/Nuclear/TechBulletins/tb1295-1.pdf.).

42. Thus, it would be expected to see obvious differences between a source term into the containment and a source term from the containment into the environment, with the magnitude of the containment source term being larger than the one into the environment.

43. Further, the application of NUREG-1465 PWR source term data to all 13 release categories in the Seabrook Station SAMA analysis (*i.e.,* from containment bypass-steam generator tube rupture (LE1) source terms through no-failure, containment maintained intact with excessive containment leakage (INTACT2) source terms) essentially treats all releases into the containment as releases into the environment. However, for the Seabrook Station, a large fraction (approximately 58%) of the core damage sequences involve accidents in which the containment retains its structural integrity (*i.e.*, radiological release is limited to containment

leakage, as modeled in INTACT1 and INTACT2). The remaining 42% of core damage sequences are those sequences that involve various early and late containment failure modes (*e.g.*, containment bypass events, specifically steam generator tube rupture and interfacing system loss of coolant accidents, small/early containment bypass, and large/late release events). Additionally, early/late containment failure and containment bypass are markedly different event types, with significant differences in radionuclides released, sequence progression, timing, release pathways, and fission product deposition and removal mechanisms. These different event types logically would result in different — generally lower — source terms and release fractions than the single PWR source term described in NUREG-1465.

44. For the reasons described above, one would expect to see obvious differences between NUREG-1465's containment source term and the MAAP 4-generated environmental source terms in the Seabrook Station SAMA analysis. In contrast, use of NUREG-1465 source term for all release categories in the Seabrook Station SAMA analysis would essentially treat all releases into the containment as releases into the environment and greatly distort the results of the SAMA analysis. Indeed, the failure to credit the containment's presence and the neglect of associated passive and active engineered safety features for mitigating and delaying releases would lead to a highly conservative, worst-case source term scenario without any technically supported weighting by likelihood of occurrence.

45. PRA and SAMA analyses seek to maximize the use of plant-specific data. Use of a thermal-hydraulic and fission product release code, like MAAP, to develop plant-specific release fractions for the Seabrook Station SAMA analysis is strongly preferred to using generic inputs from other sources, such as the NUREG-1465. Use of the plant-specific inputs in the SAMA analysis allows for better resolution of data, more accurate portrayal of plant-specific

response to postulated severe accident phenomenology, and better serves the purpose of evaluating potentially cost-beneficial plant improvements.

46. NUREG-1465 is a generic analysis based in part on source term analyses from plants other than Seabrook Station, some of which have containment types different from Seabrook Station. NUREG-1465 considers past analyses of PWRs Surry, Zion, Oconee 3 and Sequoyah plant severe accident sequences to develop a single generic, default containment source term that is not associated with any likelihood of occurrence for a particular plant. NUREG-1465 presents only one set of PWR release fraction data and one set of BWR release fraction data and does not quantify the effects of plant-specific features. NUREG-1465 at 13 Table 3.13 ("PWR Releases Into Containment") and Table 3.12 ("BWR Releases Into Containment").

47. This is in contrast to a plant-specific PRA and SAMA analysis such as that performed for the Seabrook Station which used the MAAP code in part to analyze plant-specific information and features, including passive and active engineered safety features within Seabrook Station's containment for mitigating and delaying releases, to provide a representative spectrum of severe accident release category source terms. In particular, the MAAP analysis for the Seabrook Station provides information needed to support the consequence analysis (Table 4, MACCS2 Inputs Derived from MAAP 4 Results). Plant-specific information is shown in Table 5 (Release Characteristics) and Table 6 (Release Fractions) for the 13 Release Categories analyzed in the Seabrook Station SAMA analysis.

48. Use of the NUREG-1465 source term as a surrogate for the release into the environment instead of the Seabrook Station, plant-specific Level 2 PRA leads to an overly conservative estimate of radiological release to the environment and lacks technical basis. It

essentially treats all types of postulated core melt accident releases into the containment as releases into the environment; *i.e.*, the assumption would treat containment failure sequences and containment bypass events equivalently. Such an assumption of not crediting the containment's presence, and neglecting associated passive and active engineered safety features for mitigating and delaying releases specific to the plant being evaluated, leads in effect to one worst-case source term scenario. Thus, use of the NUREG-1465 source term instead of source terms generated using plant-specific information would oversimplify the SAMA cost-benefit process and lead to technically unfounded conclusions about a particular plant's offsite risks.

49. It also bears mention that an integrated computer code such as MAAP may serve multiple functions in a Level 2 PRA. NextEra used the MAAP code to support the entire Seabrook Station PRA that serves as a major input to the SAMA analysis. For example, NextEra used MAAP to support the development of plant equipment success criteria (*e.g.*, amount of flow required to meet core cooling needs at specific times) and to develop timelines for operator actions to determine human error probabilities included in the PRA. Use of alternate data for release fractions as inputs to the Level 3 analysis for SAMA purposes could lead to inconsistencies in the accident progression analysis and does not obviate the dependence of the Seabrook Station SAMA analysis on the MAAP code.

50. In summary, the distinct phenomenological bases and regulatory purposes of the NUREG-1465 and MAAP source terms explain the relative numerical differences in the amount of radionuclides and the timing for the release. Due to containment ESFs (*e.g.*, containment air coolers, containment spray) and natural depletion processes (*e.g.*, aerosol deposition and containment holdup), the source term released from the reactor coolant system into containment expectedly is different from that of the containment into the environment. Thus, the NUREG-

1465 and MAAP source terms *should* differ, with the MAAP source term being the smaller of the two.

51. Use of an overly conservative, *i.e.*, excessively higher than can be technically supported, source term from NUREG-1465 would have numerous (and unjustified) effects. Such modeled effects include exaggerated early and long-term health effects, incorrect determination of the size of the area that might become contaminated, inflated offsite economic losses, and incorrect estimates of the risk benefit value of SAMA candidates. The net effect would be to distort the SAMA process, and likely misrepresent the risk reduction effectiveness of existing plant design features and plant-specific SAMA candidates.

C. The Historical Release Fraction Comparisons Cited by Friends/NEC Involve Legacy Codes or Code Versions, and Older Versions of MAAP Not Used in Seabrook Station SAMA Analysis (Basis 3)

52. MAAP is a contemporary code for simulating the release of radionuclides into the environment following a severe accident. Contention 4B's references to comparisons between legacy codes do not support a contrary conclusion.[19]

1. NUREG-1150 and the Brookhaven Study Contain Comparisons Between Legacy Codes, and Nowhere Mention the Advanced Version of MAAP Used in the Seabrook Station SAMA Analysis

53. The first comparison is a 1987 draft of the NUREG-1150 severe accident risk study that, in examining accident risk at Zion Nuclear Station, found that "the MAAP estimates for environmental release fractions were significantly smaller" than those obtained with the

[19] Legacy codes and code versions are computer codes that may have been applied in the late 1980s through the 1990s but are no longer used. In some cases, these codes have been retired. Many of these codes are based on older vintage experiments and test data, and their computer models not based on current, realistic understanding of severe accident progression.

Source Term Code Package[20] ("STCP") computer code (the primary source term code used in the draft NUREG-1150 study). Draft NUREG-1150, Vol. 1 at 5-14.

54. The second is the 2002 BNL report (cited previously at paragraph 11) that reviewed combustible gas control availability in ice condenser and Mark III containment plants. The BNL report compared the Level 2 portion of the PRA results for the Catawba plant (obtained using the MAAP code) with a "typical NUREG-1150 release" for the Sequoyah plant (obtained using the STCP and MELCOR codes). The BNL report states that the "NUREG-1150 release fractions for the important radionuclides are about a factor of 4 higher than the ones" in the Catawba PRA, and that the "differences in the release fractions . . . are primarily attributable to the use of the different codes in the two analyses." *Id.*

55. Neither of the comparisons made by Friends/NEC is germane to NextEra's use of MAAP-generated source terms or release fractions based on a plant-specific PRA model in the Seabrook Station SAMA analysis. The comparisons involved earlier versions of the MAAP code than that applied in the Seabrook Station SAMA analysis. The 1987 NUREG-1150 draft comparison was made to the early versions of the MAAP code (1.1 through 3.0) used in the IDCOR program. *See* paragraphs 22-23. The 2002 Catawba use of MAAP, as discussed in the BNL report, apparently was performed using version 3B of MAAP. Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, February 1, 2002 at 6. As summarized earlier, NextEra initially applied MAAP

[20] As noted on the Sandia MELCOR website (http://melcor.sandia.gov/), the STCP is the predecessor to MELCOR: "MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code package."

4.0.5 and most recently MAAP 4.0.7 in its Level 2 PRA basis calculations for the Seabrook Station SAMA analysis.

56. Although it remains a primary foundational document for probabilistic safety assessment, the final NUREG-1150 study was completed over 22 years ago and involved an assessment of the risks from severe accidents at five commercial nuclear power plants in the United States. However, Seabrook Station was not one of those five plants. Furthermore, the IDCOR (MAAP) to NUREG-1150 (STCP) comparison of Zion results cited by Friends/NEC was only one of four sets of plant results compared in the February 1987 draft of NUREG-1150 (with several other comparisons in the draft report showing reasonable agreement). In addition, after extensive peer review of, and public comment on, the February 1987 draft, NUREG-1150, Volume 1, was issued as a second draft in 1989, before being published as a final report in December 1990. In summary, the report and its underlying technical analyses were substantially modified in two rounds of review before the report's final publication in December 1990. Notably, one of the changes included deleting the specific discussion comparing the MAAP and STCP results for Zion, such that the comparison cited by Friends/NEC in Contention 4B was not incorporated into the final December 1990 version of NUREG-1150.

57. As discussed previously, severe accident source term estimates from computer codes depend on user assumptions and expertise, the extent to which plant-specific passive and active design features are modeled, the degree of benchmarking,[21] and the technical accuracy provided by computer code models and their underlying algorithms. While the final 1990 NUREG-1150 report still is relevant to the nuclear safety community's understanding of severe accident progression, additional severe accident research performed in the U.S. and abroad in

[21] In this context, benchmarking refers to comparison of code predictions with experiments, other qualified codes that model the same phenomena and, in some situations, hand or spreadsheet calculations.

the 25 years since the 1987 draft of NUREG-1150 was issued has significantly improved that understanding. One of these research efforts, a recent NRC-sponsored project, is discussed later in this Declaration (paragraphs 63-70).

58. In using STCP in the NUREG-1150 study to predict complex phenomena, the study's authors noted that they used both expert elicitation and additional computer codes to augment the results of simplified STCP models. For example, with respect to the core degradation process, the NUREG-1150 authors stated the thermal-hydraulic model in the STCP "uses simplified models and assumptions for the treatment of some of the very complex steps in the core degradation process, such as fuel slumping into the lower plenum of a reactor vessel." NUREG-1150, Vol. 3, App. D at D-17. More realistic models such as MELCOR and MAAP were used to adjust the thermal-hydraulic estimates affecting core degradation, ultimately leading to differences in the source term. *Id.*

59. Consistent user inputs and assumptions, modeling the same plant and containment type, and the use of more realistic computer models are important in the comparisons of computer code results from severe accident sequences. For example, in accounting for the effect of more realistic models evaluating the same sequences and for the same type of containment as the Seabrook Station, more contemporary analyses show that there are substantial processes (*e.g.*, heat transfer and fission product chemistry) within the reactor coolant system and the containment that extend the interval before releases to the environment occur and also substantially limit the magnitude of releases to the environment for severe core damage accidents. Frank J. Rahn and Robert E. Henry, "Release and Dispersion of Radioactivity from Reactor Fuel Research and Analytical Results Leading to Reductions in Radiological Source Terms," American Nuclear Society Position Statement No. 65 on "Realism

in the Assessment of Nuclear Technologies," Appendix 1A (June 2004). In this comparison, the MAAP-based results for station blackout sequences calculated releases that occurred later and extended over a longer period of time than were calculated in the 1990 NUREG-1150 or 1975 WASH-1400[22] studies. This type of comparison is more valid than those cited by Friends/NEC where different plants were compared applying codes of different vintage, and in which user inputs and assumptions in modeling the accident sequences are dissimilar or unclear.

60. Regarding the 2002 BNL report's comparison of the Catawba and Sequoyah plants' release fractions, the NRC Staff in its Supplemental Environmental Impact Statement for Catawba license renewal compared similar sequences between the two studies — NUREG-1150 and Revision 2b of the Catawba PRA,[23] which included the plant's IPE models — and concluded there was "reasonable agreement" for the closest corresponding release scenarios. NUREG-1437, Supp. 9, at 5-9 to 5-10. Specifically, the Staff provided the following summary:

> The Staff reviewed the process used by Duke to extend the containment performance (Level 2) portion of the IPE to the offsite consequence (Level 3) assessment. This included consideration of the source terms used to characterize fission product releases for each containment release category and the major input assumptions used in the offsite consequence analyses. This information is provided in Section 6.3 of Duke's IPE submittal. Duke used the Modular Accident Analysis Program (MAAP) code to analyze postulated accidents and develop radiological source terms for each of 29 containment release categories used to represent the containment end-states. These source terms were incorporated as input to the MACCS2 analysis. The staff reviewed Duke's source term estimates for the major release categories and found these predictions to be in reasonable agreement with estimates of NUREG-1150 (NRC 1990) for the closest corresponding release scenarios. The staff concludes that the assignment of source terms is acceptable.

Id. at 5-10.

61. The state of the art for source term analysis has improved considerably since the NUREG-1150 study was performed over twenty years ago. This is a well-known fact that the

[22] U.S. Nuclear Regulatory Commission; "Reactor Safety Study An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), December 1975.

[23] Letter from Gary R. Peterson, Vice President, Duke Energy Corporation, to Document Control Desk, U.S. N.R.C., Attach. 1, "Catawba PRA Revision 2b Summary Results" (Apr. 18, 2001).

faulty comparisons of Friends/NEC fails to consider. For example, the comparison between the Catawba Level 2 PRA release fractions and the NUREG-1150 Sequoyah release fractions represents a difference of more than ten years in terms of severe accident modeling (~2002 versus ~1990). Also, the comparison uses a release category that represents an early containment failure in which the Catawba source term is based on an "early containment failure without ex-vessel release" assumption that may not have been applied in the Sequoyah source term.[24] Early containment failure, in this case, seems to be associated with high pressure in the Reactor Coolant System at reactor vessel failure, with the resulting blowdown dispersing the corium into the lower containment area. With the debris bed spread over a large area, the debris bed will be coolable, preventing the ex-vessel release of fission products, such as that due to molten core-concrete interactions, and subsequently leading to a smaller source term to the environment. This assumption and modeling was apparently not applied in the earlier NUREG-1150 analysis for Sequoyah. It should also be noted that although Seabrook Station also is a PWR, it has a dry, ambient air containment type, whereas both Catawba and Sequoyah are typical of ice condenser containment PWR plants. Thus, the relevance of this comparison to the Seabrook Station PRA and SAMA analysis use of MAAP is highly questionable.

62. Since the issuance of NUREG-1150, better understanding of heat transfer and removal from the reactor pressure vessel ("RPV") during severe accident sequences; improved insights into iodine, cesium, and other fission product group chemistry from contemporary research; and modeling improvements suggest that the early containment failure releases

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[24] *See* Memorandum from Asimios Malliakos, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, to Marc A. Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, "Telecommunication with Duke Energy Corporation in Support of Generic Safety Issue (GSI) 189, 'Susceptibility of Ice Condenser and BWR Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident,'" (Oct. 8, 2002).

potentially could be smaller than previously concluded. *See* paragraph 61, *supra*. Thus, the BNL report's comparison of MAAP-based source terms with those estimated over ten years earlier with the simpler STCP code and an earlier version of MELCOR — and for different plants — is expected to show differences. A more logical and meaningful approach is to compare contemporary, severe accident computer code models and compare their predictions to experimental data or to postulated reactor conditions for the same scenario. In other words, the better comparison is of the predictions of computer models at the same point in time with the same inputs and data available to the code analysts.

2. <u>Contemporary Severe Accident Modeling Supports the Realism of the Version of MAAP Used in the Seabrook Station SAMA Analysis</u>

63. The NRC, the nuclear power industry, and the international nuclear energy research community have extensively researched and studied accident phenomena and offsite consequences of severe reactor accidents over the last 25 years. As part of an initiative to assess plant response to security-related events following the terrorist attacks of 2001, the NRC completed updated analyses of severe accident progression and offsite consequences. Those analyses incorporate a wealth of accumulated research data as well as more detailed, integrated, and realistic modeling methods than previous analyses. One insight gained from these security assessments was that updated analyses of severe reactor accidents were needed to reflect realistic estimates of the more likely accident outcomes given the current state of plant design and operation and advances in our understanding of severe accident behavior. NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," at xi (Nov. 2012) ("NUREG-1935").

64. Consequently, the NRC initiated the SOARCA project in 2006 to develop revised best estimates of the offsite radiological health effect consequences of severe reactor

accidents. The project's principal objective was to develop updated and more realistic severe accident analyses by including significant plant changes and reactor safety research updates not reflected in earlier NRC assessments. SOARCA included consideration of plant system improvements, improvements in training and emergency procedures, offsite emergency response, and security-related improvements, as well as plant changes such as power uprates and lengthened operating times.

65. The SOARCA analyzed two plants that are typical of the two U.S. commercial reactor types, *i.e.*, a BWR plant, the Peach Bottom Atomic Power Station in Pennsylvania, and a PWR plant, Surry Power Station in Virginia. These two plants also took part in earlier accident analyses performed by the NRC, including the seminal WASH-1400 PRA study (1975), the Sandia Siting Study (1982),[25] and the NUREG-1150 (1990) study. The SOARCA analysis considered one plant unit at each site.

66. The SOARCA project used computer-modeling techniques to understand how a reactor might behave under severe accident conditions, and how a release of radioactive material from the plant might affect the public. Specifically, it used MELCOR to model the severe accident scenarios within the plant and MACCS2 to model the offsite health effect consequences of any atmospheric releases of radioactive material.

67. In 2012, the NRC published the results of its assessment and plant-specific reports for Peach Bottom and Surry. *See* NUREG-1935; NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis" (Jan. 2012); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis" (Jan. 2012). The NRC found that, in addition to delayed

[25] NUREG/CR-2239, "Technical Guidance for Siting Criteria Development" (1982).

radiological releases, the magnitude of the radionuclide release, especially with respect to the key radionuclide (iodine and cesium) groups, is much smaller than estimated in prior studies, such as the 1982 Sandia Siting Study, and the accident at Chernobyl. This comparison is shown in Figure 4. For reference purposes, the iodine and cesium release from the accident at Three Mile Island is also shown.

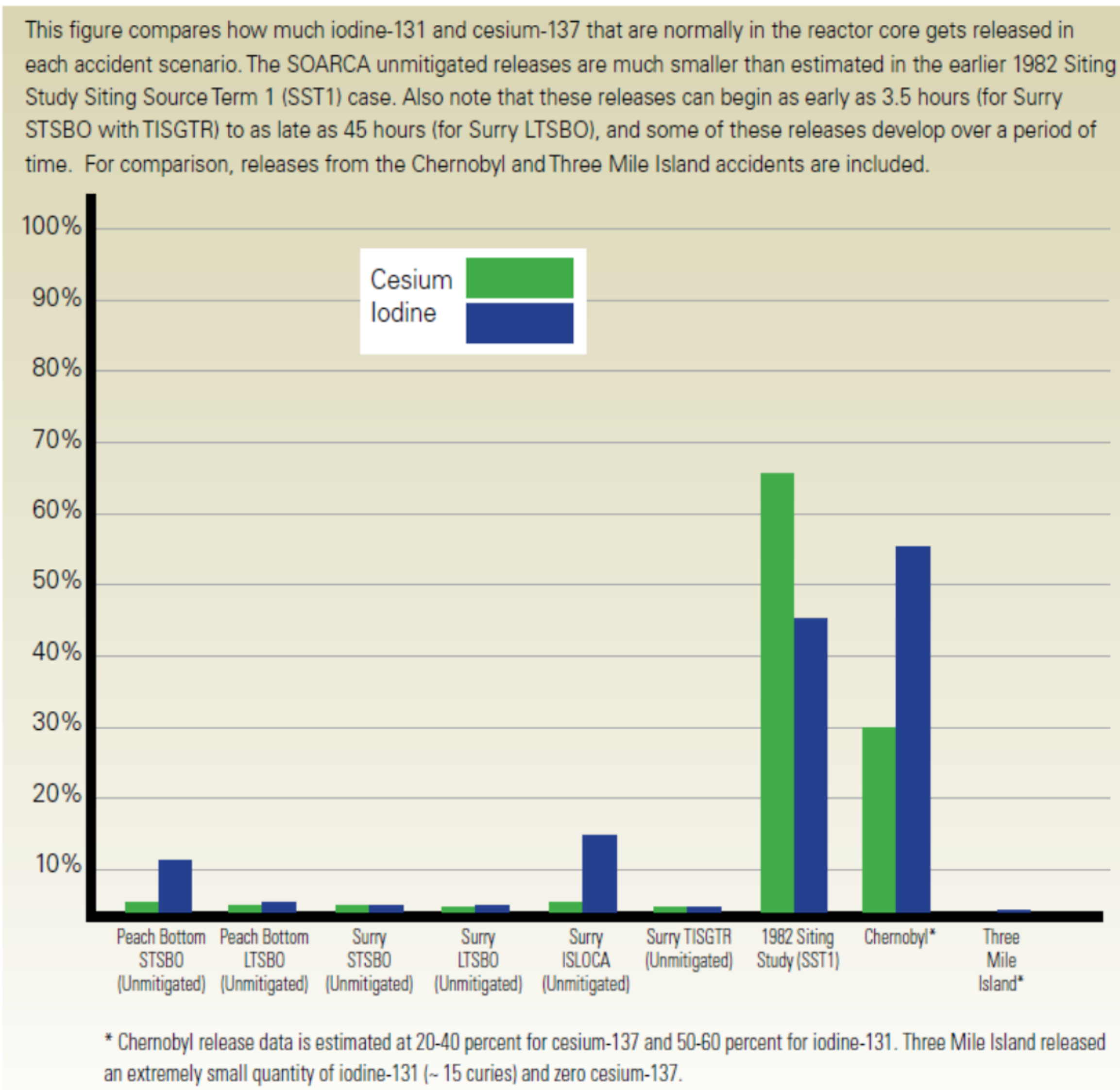

Figure 4. Percentages of Iodine and Cesium Released to the Environment During the First 48 Hours of the Accident for SOARCA Unmitigated Scenarios, 1982 Siting Study (SST1), and Historical Accidents (from NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences," 24 Fig. 4.1 (Jan. 2012).

68. In comparison to the 1990 final NUREG-1150 analyses, the 2012 SOARCA study, in general, estimated delayed releases, and smaller source terms for the same type of accident sequences. For example, the Surry Power Station containment failure release categories RSUR1 and RSUR2 (due primarily to station blackout initiating event) from NUREG-1150[26] can be compared to the SOARCA long-term station black out ("LTSBO") and the short term station blackout ("STSBO") from the SOARCA Surry analysis (NUREG/CR-7110).[27] The comparison shows smaller release fractions for the principal groups of interest, specifically, noble gas, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium. (Table 7). This comparison has higher validity as the same type of PWR containment, *i.e.*, a large dry containment, for the same type of plant (Surry), and similar types sequences are considered with MELCOR (or STCP) than the comparisons proffered by Friends/NEC.

69. In addition, the SOARCA analyzed accident sequences would be lower, as expected, for releases into the *environment* than the single generic NUREG-1465 PWR source term to the *containment*. One such comparison that demonstrates this difference is shown in Table 8. Table 8 is based on NUREG-1465's Table 3.13 (PWR Releases Into Containment) and shows the release fraction for each of the four phases used in NUREG-1465 and the summed total of the four phases. The total release fraction into the containment is higher than either of the SOARCA SBO releases into the environment (repeated from Table 7 and the last two rows of the table). This difference is especially large for SOARCA cesium and iodine environmental release fractions compared to the total release fraction (0.75) in NUREG-1465 for both cesium and iodine. All scenarios from SOARCA for Surry are shown in Figure 4 for cesium and iodine

[26] U.S. NRC, Advisory Committee on Reactor Safeguards, Presentation by Hossein P. Nourbakhsh, "Historical Perspectives and Insights on Reactor Consequence Analyses", (November 2008).

[27] *State-of-the Art Reactor Consequence Analyses Project*, NUREG/CR-7110, Vol. 2: Surry Integrated Analysis (January 2012).

and (*see* Figure 4 above for Surry accident sequences - the third, fourth, fifth, and sixth scenarios). It may be concluded that PWR environmental releases in SOARCA based on the Surry PWR plant are significantly below the containment release fraction from NUREG-1465.

70. Comparisons between the Seabrook Station SAMA analysis use of MAAP 4.0.7 and the SOARCA use of MELCOR 1.8.6 for the Surry Power Station demonstrate reasonable consistency between the calculated environmental source terms when contemporary versions of these codes are applied to the same type of PWR with (large dry) containment, using the same basis scenario. For Seabrook Station, two representative station blackout ("SBO") cases from the SAMA analysis, release categories LL5 and SELL, are compared with the mean release fractions obtained for Surry for LTSBO and STSBO scenarios. In this comparison, the SOARCA MELCOR-calculated release fractions are about the same or smaller than the MAAP-calculated release fraction for the Seabrook Station SAMA analysis for SBO as is shown in Table 9. The barium and strontium radionuclide group is the only release fraction that is appreciably larger in SOARCA but this group was found to be a negligible contributor to severe accident risk in either the short-term or long-term phase of the accident (Section 7.3 of NUREG/CR-7110, Vol. 2). This comparison more correctly limits the potential for different inputs and assumptions being applied, and allows a more meaningful code comparison because the reactor types, accident type and modeling changes over the last 20 years are controlled.

71. Sandia developed MELCOR for the NRC, and it is the current software tool in a line of evolutionary, severe accident progression computer models used by the NRC. As noted above, Sandia and the NRC used MELCOR in support of the SOARCA project. Both MAAP and MELCOR are now used world-wide and are much more advanced than predecessor versions or simpler models, such as the STCP applied more than 22 years ago in the NUREG-

1150 study. They are both integral plant models codes that allow the calculation of accident sequences from the initiating event while taking into account important inter-related phenomena (*e.g.*, reactor coolant system, containment thermal-hydraulics, in-vessel core degradation, molten core concrete interaction, fission product release and transport into the environment).

72. A 2004 comparison using MELCOR and MAAP for a PWR accident sequence demonstrates that the two codes provide similar calculated results for thermal-hydraulic and core degradation response of the plant, with minor differences in various timings of phenomena. The authors indicated that these minor differences in results were within the uncertainties of the code numerical computations and the physics models. K. Vierow, Y. Liao, J. Johnson, M. Kenton, and R. Gauntt, "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP 4, and SCDAP/RELAP5 Codes," *Nuclear Engineering and Design* 234, 129-145 (2004). Although this documented comparison did not specifically address the calculation of release fractions to the environment, this comparison supports the use of either code for supporting PRAs. In my judgment, the results of this and previous comparisons show that use of MAAP is reasonable for the purposes of supporting a best-estimate SAMA analysis.

73. Notably, both MAAP and MELCOR have been used to study the March 2011 Fukushima Dai-ichi nuclear power plant accident in Japan. Tokyo Electric Power Company, the operating utility for the six-unit station, has used MAAP to inform its understanding of the accident progression in Units 1-3 during the earthquake and subsequent tsunami event. International Atomic Energy Agency, *IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami*, at 33-35 (June 2011). Sandia has applied MELCOR in modeling the Station Blackout sequence for the NRC and other government agencies in support of the Japanese Government.

74. The contemporary applications and comparisons of MAAP discussed above demonstrate the current-day use and value of MAAP (and MELCOR) with respect to modeling and simulation of severe reactor accident conditions. In my professional opinion, they are better indicators of the MAAP code's fitness for simulating severe accident conditions and estimating environmental source terms than the references cited by Friends/NEC, which are not as relevant, outdated and immaterial, and which fail to show any flaw in the MAAP code or related inputs used by NextEra in its SAMA analysis.

VI. CONCLUSION

75. The claims in Contention 4B have been thoroughly evaluated against information in the Seabrook Station ER and its supporting supplementary technical documentation, the applicable accepted standards for performing PRAs and SAMA analyses, and the reports discussed above. Based on this evaluation, all of the Friends/NEC claims are concluded to lack a technical foundation and provide no reason to conclude that the source terms used in the Seabrook Station SAMA analysis are invalid or unreasonable. Specifically:

- The MAAP code has a strong technical basis for use in PRA and severe accident analysis and has been accepted for use in numerous NRC-approved analyses. MAAP has been the subject of extensive benchmarking and validation studies in the areas instrumental to severe accident source term estimation. Use of the MAAP code is reasonable for a SAMA analysis performed under NEPA.
- The use of plant-specific source terms derived from MAAP is preferred over the use of generic source terms extracted from NUREG-1465 for a SAMA analysis, which evaluates plant specific design and operational changes.

- The primary purpose of NUREG-1465 source terms is for defining releases into containment, not to the environment. A SAMA analysis requires a plant-specific evaluation of releases to the environment. A NUREG-1465 containment source term is not consistent with the requirements of SAMA analysis, which is based on a representative spectrum of environmental source terms.
- NUREG-1465 provides data only for a single PWR release into the containment. A SAMA analysis requires an evaluation of the spectrum of plant-specific environmental releases. Use of the NUREG-1465 prescriptive release for the entire spectrum of specific releases would result in distorted SAMA results without a supporting technical basis to the plant being considered.
- Comparisons of earlier versions of MAAP to earlier versions of MELCOR or its predecessor, the STCP, are not material to NextEra's use of the current versions of MAAP today. The understanding of severe accident modeling has improved considerably over time. As the SOARCA Project demonstrated, current modeling of severe accidents shows a much smaller and delayed radioactive release than was recognized in earlier studies and calculated with older computer code models.

Per 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

<u>*Executed in accord with 10 C.F.R. § 2.304(d)*</u>
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Table 1. Definitions of Key Severe Accident and PRA Terms

Term	Definition
Accident Progression Bin	A group of postulated accidents that has similar characteristics with respect to the timing of containment building failure and other factors that determine the amount of radioactive material released. Accident progression bins are sometimes referred to as containment failure modes in older PRAs.
Core Damage Frequency	The frequency of combinations of initiating events, hardware failures, and human errors leading to core uncovery with reflooding of the core not imminently expected.
Core Inventory	The amount (in units of activity) of each radionuclide present in the reactor core at the time accident initiation.
External Initiating Events	Events occurring away from the reactor site that result in initiating events in the plant. In keeping with PRA tradition, some events occurring within the plant during normal power plant operation, *e.g.*, fires and floods initiated within the plant, are included in this category.
Initiating Event	A challenge to plant operation from which there can be numerous accident sequences. The various accident sequences result regardless of whether plant systems operate properly or fail and what actions operators take. Some accident sequences will result in a safe recovery and some will result in reactor core damage. PRA normally consider internal and external initiating events.
Internal Initiating Events	Initiating events involving components internal to the plant (*e.g.*, transient events requiring reactor shutdown, pipe breaks) occurring during the normal power generation of a nuclear power plant. In keeping with PRA standard practice, loss of offsite power is considered an internal initiating event.
Plant Damage State	A group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.
Release Fraction	The fraction defining the portion of the radionuclide inventory by radionuclide group in the reactor at the start of an accident that is released through a containment barrier(s), such as the reactor coolant system, to the primary containment, or from the primary containment to the environment.
Severe Accident	Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. A severe accident is often described as a beyond design-basis accident involving multiple failures of equipment or function. Although severe accidents generally have lower likelihoods than design-basis accidents, they may have greater consequences.
Source Term	The fractions of the core inventory released to the atmosphere, and the timing and other release information needed to calculate the offsite consequences. Specifically, the information includes the fractions of the radionuclide groups in the inventory in the reactor at the start of an accident that are released to the containment (*i.e.*, the containment source term), or to the environment (*i.e.*, the environmental source term). The source term to the environment also includes the initial elevation, heat or energy content of the plume, and timing of the release (time after accident initiation or shutdown, and duration of release).

Table 2. Seabrook Station Source Term Release Categories (From SBK-L-12053, page 6, Seabrook Station Source Term Release Categories)

* Release category IDs ending in "a" are "dry" scenarios while "b" release categories are "wet" scenarios.

No.	Source Term Group	Source Term Title	Related Release Categories*
1	LE1	Large/Early Containment Bypass - SG Tube Rupture	LE11a, LE12a, LE13a
2	LE2	Large/Early Containment Bypass - ISLOCA	LE21 a, LE21 b, LE22a
3	LE3	Large/Early Containment Penetration Failure to Isolate (Containment Online Purge valve failure)	LE3a, LE3b
4	LE4	Large Containment Basemat Failure with Delayed Evacuation	LE4a
5	SE1	Small/Early Containment Bypass - SG Tube Rupture with Scrubbed Release	SE11b, SE12b
6	SE2	Small/Early Containment Bypass - ISLOCA with with Scrubbed Release	SE2b
7	SE3	Small/Early Containment Penetration Failure to Isolate	SE3b
8	LL3	Large/Late Containment Venting	LL3b
9	LL4	Large/Late Containment Overpressure Failure	LL4b
10	LL5	Large/Late Containment Basemat Failure	LL5a
11	SELL	Small/Early Containment Penetration Failure to Isolate and Large/Late Containment Basemat Failure	SELL3b, SELL4b, SELL5a
12	INTACT1	Nominal Containment Leakage	INTACT1
13	INTACT2	Excessive Containment Leakage	INTACT2

Table 3. Release Category Frequencies

(From Table "Seabrook Station Release Category Public Dose and Economic Risk Results - Level 3 Model SEABRK" page 27 of 96 SBK-L-12053**)**

Release Category	Frequency (/year)*
LE-1	5.19E-08
LE-2	1.81E-08
LE-3	8.61E-10
LE-4	2.11E-08
SE-1	5.08E-07
SE-2	2.79E-08
SE-3	9.97E-07
LL-3	1.75E-07
LL-4	5.79E-08
LL-5	3.10E-06
SELL	9.84E-08
INTACT1	7.07E-06
INTACT2	6.90E-08
TOTAL = Mean Core Damage Frequency	**1.23E-05**

*Table results are Level 2 release category frequencies from SBK model SB2011 (base case).

Table 4. Description of MACCS2 Inputs Derived from MAAP 4 Results

MACCS2 Input Parameter	Release Category Description Containment Failure Type	MACCS2 Code User's Guide*	Seabrook Station SAMA Analysis
1. OALARM	Defines the time at which notification is given to off-site emergency response officials to initiate protective measures for the surrounding population. This time is a function of the accident sequence. It is measured from accident initiation (scram time) and is given in units of seconds.	p. 5-24	Pages 20-26 of SBK-L-12053.
2. PLHEAT	Specifies the rate of release of sensible heat in each plume segment. This quantity should be calculated as the amount of sensible heat in the plume segment divided by the duration of the plume segment. The value specified here is used to determine the amount of buoyant plume rise that will occur.	p. 5-25	Pages 20-26 of SBK-L-12053.
3. PLHITE	Specifies the height above ground level at which each plume segment is released.	p. 5-26	Pages 20-26 of SBK-L-12053.
4. PLUDUR	Specifies the duration in seconds of each plume segment.	p. 5-26	Pages 20-26 of SBK-L-12053.
5. PDELAY	Specifies the start time of each plume segment in seconds from the time of accident initiation, *e.g.,* reactor scram.	p. 5-26	Pages 20-26 of SBK-L-12053.
6. RELFRC	Defines the release fractions for each of the plume segments. One set of values is supplied for each plume and it contains as many values as there are element groups. All components of an element group are released from the facility in the same fraction.	p. 5-28	Pages 20-26 of SBK-L-12053.

* NUREG/CR-6613, "Code Manual for MACCS2: User's Guide," Sandia National Laboratories (May 1998).

Table 5. Release Characteristics of the Seabrook Station Release Categories for the SAMA Analysis (Based on Seabrook Station Severe Accident Release Parameters for Seven-Day Release, SBK-L-12053, pages 20-26).

Release Category	Time After Scram when General Emergency is Reached, (s)	Plume segment	Release time from shutdown, (s)	Plume duration*, (s)	Plume Height, (m)
LE 1	9,328	1	9,328	3,982	54.64
		2	13,309	86,400	
		3	99,709	73,091	
		4	172,800	86,400	
LE 2	40,162	1	40,162	7,898	54.64
		2	48,060	38,340	
		3	86,400	86,400	
		4	172,800	86,400	
LE 3	2,984	1	4,262	6,113	54.64
		2	10,375	76,025	
		3	86,400	86,400	
		4	172,800	86,400	
LE 4	1,984	1	3,125	71,633	54.64
		2	74,758	11,642	
		3	86,400	86,400	
		4	172,800	86,400	
SE 1	75,413	1	75,416	41,393	54.64
		2	116,809	742	
		3	117,551	55,249	
		4	172,800	86,400	
SE 2	42,865	1	42,869	23,638	54.64
		2	66,506	19,894	
		3	86,400	86,400	
		4	172,800	86,400	
SE 3	3,445	1	5,108	7,024	54.64
		2	12,132	74,268	
		3	86,400	86,400	
		4	172,800	86,400	
LL 3	2,988	1	4,280	5,875	54.64
		2	10,156	76,244	
		3	86,400	86,400	
		4	172,800	86,400	
LL 4	2,988	1	4,280	5,875	54.64
		2	10,156	86,400	
		3	96,556	76,244	
		4	172,800	86,400	
LL 5	63,529	1	71,219	10,249	54.64
		2	81,468	10,037	
		3	91,505	86,400	
		4	177,905	81,295	
SELL	63,378	1	71,035	10,829	54.64
		2	81,864	4,536	
		3	86,400	86,400	

Release Category	Time After Scram when General Emergency is Reached, (s)	Plume segment	Release time from shutdown, (s)	Plume duration*, (s)	Plume Height, (m)
		4	172,800	86,400	
INTACT1	6,527	1	8,957	11,858	54.64
		2	20,815	65,585	
		3	86,400	86,400	
		4	172,800	86,400	
INTACT2	6,523	1	8,953	12,035	54.64
		2	20,988	65,412	
		3	86,400	86,400	
		4	172,800	86,400	

* MAAP code duration output longer than 8.64E+04 seconds was run for the longest plume duration input permitted in MACCS2 (8.64E+04 seconds, or 24 hours).
**The energy release rate in all plume segments was input as zero.

Table 6. Radionuclide Release Fractions of the Seabrook Station Release Categories (Total Release Summed Over Four Plume Segments)

Number	Release Category	Release Group									
		NG	I	Cs	Te	Sr	Mo	La	Ce	Ba	Sb
1	LE-1	9.99E-01	2.67E-01	2.83E-01	2.14E-01	1.26E-03	4.57E-04	2.77E-05	6.86E-04	7.94E-04	1.67E-01
2	LE-2	1.00E+00	4.17E-01	4.40E-01	4.35E-01	2.12E-02	5.07E-02	3.50E-04	3.50E-04	3.36E-02	4.90E-01
3	LE-3	1.00E+00	3.08E-01	2.93E-01	2.39E-01	1.92E-03	5.82E-03	4.09E-05	1.67E-04	4.44E-03	2.76E-01
4	LE-4	1.00E+00	3.32E-01	1.90E-01	1.22E-01	2.68E-05	1.96E-05	2.45E-06	2.97E-05	5.29E-05	3.96E-01
5	SE-1	6.32E-02	3.34E-04	2.01E-04	6.21E-05	3.21E-07	7.19E-06	1.66E-08	6.63E-08	2.75E-06	2.16E-05
6	SE-2	1.00E+00	9.44E-02	5.76E-02	3.36E-02	1.79E-03	4.74E-03	1.93E-05	8.97E-05	2.70E-03	1.21E-01
7	SE-3	1.87E-01	2.43E-03	2.26E-03	2.68E-03	2.40E-05	4.45E-04	1.09E-06	8.41E-06	8.07E-05	6.48E-04
8	LL-3	9.99E-01	9.41E-03	1.03E-02	5.71E-03	2.66E-05	5.41E-04	1.24E-06	2.54E-06	1.15E-04	8.67E-03
9	LL-4	1.00E+00	9.18E-02	9.13E-02	6.97E-02	2.56E-04	6.06E-05	3.18E-06	1.19E-04	1.60E-04	2.11E-01
10	LL-5	1.00E+00	5.20E-01	4.76E-01	2.28E-01	1.55E-05	1.61E-06	1.34E-06	1.67E-05	4.24E-05	3.21E-01
11	SELL	1.00E+00	4.94E-01	3.49E-01	1.48E-01	5.45E-04	8.30E-04	4.08E-05	4.64E-04	5.02E-04	4.19E-01
12	Intact1	7.76E-03	1.41E-06	5.00E-07	9.89E-07	1.13E-07	8.72E-07	1.64E-08	2.01E-08	2.32E-07	1.10E-06
13	Intact2	7.40E-02	1.37E-05	4.89E-06	8.55E-06	1.85E-07	3.99E-06	4.89E-09	1.01E-08	5.82E-07	6.11E-06

Table 7. Release fractions for Environmental Source Term from NUREG-1150 Study for SBO-Type Sequences Compared to the LTSBO and STSBO Sequences from SOARCA Study for Surry.

Source Term	Release Group								
	NG	I	Cs	Te	Sr* (Note 2)	Ru	La	Ce	Ba	Sb
NUREG-1150: RSUR1 (Note 1)	1.0E+00	3.5E-01	3.1E-01	1.8E-01	6.0E-02	6.0E-03	6.0E-03	1.0E-02	6.0E-02	Note 3
NUREG-1150: RSUR2	1.0E+00	6.0E-02	3.0E-02	9.0E-02	3.0E-03	1.0E-03	4.0E-04	4.0E-04	3.0E-03	Note 3
SOARCA: Surry LTSBO	8.0E-01	6.0E-03	8.0E-04	2.3E-02	8.0E-04	1.5E-05	1.0E-06	2.0E-05	8.0E-04	6.0E-04
SOARCA: Surry STSBO	9.2E-01	1.0E-02	4.0E-03	1.7E-02	1.2E-03	3.0E-05	2.0E-06	3.0E-05	1.2E-03	7.5E-03
	Note 1.	The release fractions RSUR1 are summed over two release periods.							
	Note 2.	Included in the SOARCA Ba group (Ba group fraction repeated here)							
	Note 3.	This group was not explicitly calculated in NUREG-1150							

Table 8. NUREG-1465 PWR Releases into Containment for Each Time Phase and Total Release (Based on Table 3.13 of NUREG-1465) Compared to SOARCA SBO Environmental Releases

NUREG-1465 Radionuclide Groups	Noble gases	Halogens	Alkali Metals	Tellurium Group	Ba, Sr	Noble Metals	Lanthanides	Ce,Pu,Np	Ba, Sr	Tellurium Group
Elements in Group	Xe, Kr	I, Br	Cs, Rb	Te, Sb, Se	Ba, Sr	Ru, Rh, Pd, Mo, Tc, Co	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm,	Ce,Pu,Np	Ba, Sr	Note 1.
Gap release	5.0E-02	5.0E-02	5.0E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Early in-vessel	9.5E-01	3.5E-01	2.5E-01	5.0E-02	2.0E-02	2.5E-03	2.0E-04	5.0E-04	2.0E-02	5.0E-02
Ex-Vessel	0.0E+00	2.5E-01	3.5E-01	2.5E-01	1.0E-01	2.5E-03	5.0E-03	5.0E-03	1.0E-01	2.5E-01
Late in-vessel	0.0E+00	1.0E-01	1.0E-01	5.0E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	5.0E-03
Total	**1.0E+00**	**7.5E-01**	**7.5E-01**	**3.1E-01**	**1.2E-01**	**5.0E-03**	**5.2E-03**	**5.5E-03**	**1.2E-01**	**3.1E-01**
SOARCA: Surry LTSBO	**8.0E-01**	**6.0E-03**	**8.0E-04**	**2.3E-02**	**8.0E-04**	**1.5E-05**	**1.0E-06**	**2.0E-05**	**8.0E-04**	**6.0E-04**
SOARCA: Surry STSBO	**9.2E-01**	**1.0E-02**	**4.0E-03**	**1.7E-02**	**1.2E-03**	**3.0E-05**	**2.0E-06**	**3.0E-05**	**1.2E-03**	**7.5E-03**

Table 9. Seabrook Station SAMA (MAAP 4.0.7) and NUREG/CR-7110 Surry/SOARCA (MELCOR 1.8.6) SBO Source Terms

Release Category	Release Group									
	NG	I	Cs	Te	Sr	Mo	La	Ce	Ba	Sb
Seabrook Station MAAP:LL-5 (Note a.)	1.0E+00	5.20E-01	4.76E-01	2.28E-01	1.55E-05	1.61E-06	1.34E-06	1.67E-05	4.24E-05	3.21E-01
Seabrook Station MAAP: SELL (Note b.)	1.0E+00	4.94E-01	3.49E-01	1.48E-01	5.45E-04	8.30E-04	4.08E-05	4.64E-04	5.02E-04	4.19E-01
Surry MELCOR: LTSBO (Note c.)	8.00E-01	6.00E-03	8.00E-04	2.30E-02	**8.00E-04**	1.50E-05	1.00E-06	2.00E-05	**8.00E-04**	6.00E-04
Surry MELCOR: STSBO (Note d.)	9.20E-01	1.00E-02	4.00E-03	1.70E-02	**1.20E-03**	3.00E-05	2.00E-06	3.00E-05	**1.20E-03**	7.50E-03

Notes for Table 9 Release Categories:

NextEra Energy Seabrook License Renewal Application Seabrook Station, *Supplement 2 to Severe Accident Mitigation Alternatives Analysis***,** SBK-L-12053,March 19, 2012, and *State-of-the Art Reactor Consequence Analyses Project*, NUREG/CR-7110, Vol. 2: Surry Integrated Analysis (January 2012).

1. Source Term LL5, Large, Late Containment Basemat Failure (Dry): Source term LL5 is used for release categories containing core damage sequences with no RWST injection – dry containment. The containment fails by basement erosion from core-concrete interaction before long term over-pressure failure of the containment. This results in a large, late release from the containment to the environment. The representative MAAP case model is a station blackout with Emergency Feedwater (EFW) success for 12 hours and with intact containment initially but with no power recovery. Thus, basemat melt through occurs at about 49 hours.

2. Source Term SELL, Small, Early, Large Late Containment Failure: This is a new release category introduced as part of the source term revision. Source term SELL is used for release categories containing core damage sequences with initial small containment isolation failure but with failure of long-tern containment cooling. The results of the RCP seal return line isolation valves failing to close is a small release from the containment to the environment.

However, in the long term, the containment fails due to overpressure or basemat melt-through resulting in a large, late release from the containment to the environment.

3. Based on Table 2 of Affidavit of Kyle W. Ross concerning the motion for Summary Disposition of Contention 4, FirstEnergy Nuclear Operating Company, September 14, 2012, and unmitigated SOARCA scenario for Surry, MELCOR Long-Term Station Blackout, NUREG/CR-7110, Vol. 2, Rev. 0, Figure 5-8.

4. Based on Table 2 of Affidavit of Kyle W. Ross concerning the motion for Summary Disposition of Contention 4, FirstEnergy Nuclear Operating Company, September 14, 2012, and unmitigated SOARCA scenario for Surry, MELCOR Short-Term Station Blackout, NUREG/CR-7110, Vol. 2, Rev. 0, Figure 5-36.

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KEY AREAS:

- **Probabilistic Risk Analysis and Assessment**
- **Regulatory Standard & Guidance Development**
- **Computer Model Verification and Validation**
- **Accident and Consequence Analysis for Design Basis Accident Support**
- **Level 3 PRA Standard Development**
- **MACCS2 Code Applications**
- **New Reactor Design Accident Analysis and PRA Suppo**
- **Severe Accident and Quantitative Risk Analysis**
- **Tritium Dispersion and Consequence Analysis**

PROFESSIONAL SUMMARY:

Dr. O'Kula has over 30 years of experience as a manager and technical professional in the areas of commercial and production reactor probabilistic risk assessment (PRA) and severe accident analysis, accident and consequence analysis, source term evaluation, safety software quality assurance (SQA), safety analysis standard and guidance development, computer code evaluation and verification, risk management, hydrogen safety, reactor materials dosimetry, shielding, and tritium safety applications. He is currently the lead for the PRA technical area in the Risk Assessment and Analysis Group. Dr. O'Kula is a member of the American Nuclear Society (ANS) Standard working group ANS 58.25 on Level 3 Probabilistic Safety Assessment, and recently concluded activities as a member of the Peer Review Committee for the Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program. Kevin was part of the Department of Energy (DOE) team writing DOE G 414.1-4, *Safety Software Guide*. He coordinated technical support for the DOE Office of Environment, Safety, and Health (EH) in addressing Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 2002-1 on Software Quality Assurance (SQA), and is a technical consultant to DOE/Health, Safety and Security (HS-31) in completing a DOE Accident Analysis Handbook.

He is supporting, or has supported, Atomic Safety Licensing Board (ASLB) relicensing issue resolution for several commercial plants including Seabrook Station, Davis-Besse, Indian Point, Prairie Island, and Pilgrim Nuclear Power Station, on MACCS2 and severe accident mitigation alternatives (SAMA) analysis. He was also part of the accident analysis and PRA/severe accident teams supporting the Design Certification Document for the U.S. Advanced Pressure Water Reactor (US-APWR) a joint effort with URS Power Division and Mitsubishi Heavy Industries (MHI). This project also included severe accident mitigation design alternatives (SAMDA) analysis. He has provided similar support for an alternative reactor technology, the Pebble Bed Modular Reactor (PBMR) prior to the project's cessation.

Dr. O'Kula was a member of the Partner, Assess, Innovate, and Sustain (PAIS) Safety Case team for the Sellafield Sites in the United Kingdom in the early 2009 period. The PAIS team identified and began implementation of improvement opportunities in nuclear safety and related areas for Sellafield. Recommendations were documented in comprehensive reports to the Site's Nuclear Management Partners consortium in March 2009.

Kevin is currently coordinating the Failure Mode, Effects, and Criticality Analysis (FMECA) for URS PS support to BNI to the Tank Waste Immobilization and Treatment Plant (WTP) design at Hanford. He also coordinated URS SMS support to the Quantitative Risk Analysis (QRA) for evaluation of hydrogen events to risk-inform the WTP design, including fault tree analysis and reliability data, and human factors areas. He is also a contributor to the DOE response on the use of risk assessment methodologies as part of the DNFSB Recommendation 2009-1 implementation action for Risk Assessment. He led work in reviewing EIS food pathway consequence analysis performed on assumed accident conditions from the Mixed Oxide Fuel Fabrication Facility (MFFF), sited at the Savannah River Site. This project compared and evaluated the impacts calculated from three computer models, including MACCS2, GENII, and UFOTRI.

He is past chair of the American Nuclear Society (ANS) Nuclear Installations Safety Division (NISD), and the Energy Facility Contractors Group (EFCOG) Accident Analysis Subgroup, and has recently completed a term as the ANS NISD Program Committee Chair. He is a member of the Nuclear Hydrogen Production Technical Group under the ANS's Environmental Sciences Division, and is chair for the EFOCG Hydrogen Safety Interest Group. He was the Technical Program Chair for two ANS embedded topical meetings on Operating Nuclear Facility Safety (Washington, D.C., 2004) and the Safety and Technology of Nuclear Hydrogen Production, Control and Management (Boston, MA, 2007). He is the Assistant Technical Program Committee Chair for the Probabilistic Safety Assessment (PSA) Meeting in Columbia, SC, scheduled for September 22-26, 2013.

Dr. O'Kula was PRA group manager for K Reactor at the time of restart in the early 1990s. He led a successful effort demonstrating Savannah River Site (SRS) K-Reactor siting compliance to 10 CFR 100, and tritium facility compliance with SEN-35-91.

He was the project leader for independent Verification and Validation (V&V) of urban dispersion software for the Defense Threat Reduction Agency (DTRA) and is the current V&V project manager for the evaluation of several chemical/biological software tools for the U.S. Army Test and Evaluation Command (ATEC) and Chemical-Biological Program (Dugway Proving Ground (Utah) and Edgewood Chemical/Biological Center in Maryland.

EDUCATION:

Ph.D., Nuclear Engineering, University of Wisconsin, 1984
M.S., Nuclear Engineering, University of Wisconsin, 1977
B.S., Applied and Engineering Physics, Cornell University, 1975

TRAINING:

Conduct of Operations (CONOPS), 1994
Harvard School of Public Health, Atmospheric Science and Radioactivity Releases, 1995
Consequence Assessment, (Savannah River Site, 1995)
U.S. DOE Risk Assessment Workshop (Augusta, GA, 1996)
MELCOR Accident Computer Code System (MACCS) 2 Computer Code, 1997, 2005, 2011
MCNPX Training Class (ANS Meeting, 1999)

CLEARANCE:

Inactive DOE "L"

PROFESSIONAL EXPERIENCE:

URS Safety Management Solutions LLC **2000 to Present**
Advisory Engineer and Senior Fellow Advisor

Dr. O'Kula is a lead technical consultant to the Department of Energy's response plan to Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 2010-1, Safety Analysis Requirements for Defining Adequate Protection for the Public and the Workers, in the area of completion of a DOE Accident Analysis Handbook. This project will complete an accident analyst's desktop guide for performing consistent, standardized analysis to support Documented Safety Analysis for DOE facilities.

Kevin recently concluded activities as a member of the NRC-Sandia National Laboratories State-of-the-Art Reactor Consequence Analysis (SOARCA) Project Peer Review Committee. The SOARCA team provided recommendations on applying MACCS2 for modeling accident phenomena and subsequent off-site consequences from postulated severe reactor accidents. This activity supported the efforts of Sandia National Laboratories (SNL) and the Nuclear Regulatory Commission (NRC) to provide more realistic assessment of severe accidents.

Dr. O'Kula is also part of the Level 3 PRA Standard working group charged with developing an ANSI/ANS standard for Level 3 PRA analysis. He participated in a team that conducted an SQA gap analysis on the bioassay code [Integrated Modules for Bioassay Analysis (IMBA)] based on DOE G 414.1-4 requirements. He identified safety analysis codes that were designated as DOE "toolbox" codes, and oversaw production of the first documents (QA criteria and application plan, code guidance reports, and gap analysis) for six accident analysis codes designated for the DOE Safety Software Toolbox. He provided support to DOE/EH-31 (now DOE/HSS) for addressing SQA issues for safety analysis software. He was a contributor to DOE G 414.1-4, *Safety Software Guide* on SQA practices, procedures, and programs.

Kevin continues to support MACCS2 and related Severe Accident Mitigation Alternatives (SAMA) analysis issue resolution as a subject matter expert for a number of commercial nuclear plant license renewal applications. Included are the Pilgrim Nuclear Power Station, Indian Point Energy Center (Units 2 and 3), Davis-Besse Nuclear Plant, and Seabrook Nuclear Power Plants.

He was part of tritium environmental release analysis team that supported evaluation of tritium control and management areas for the Braidwood plant. He supported an initial SAMDA document for the Mitsubishi Heavy Industries (MHI) US-APWR (1610 MW_e evolutionary PWR), as well as complete a control room habitability study for postulated toxic chemical gas releases.

Kevin was part of a Washington Group team that developed a Design Control Document (DCD) for the MHI US-APWR using input information from MHI. He was Chapter lead on Chapter 15 (Transient and Accident Analysis), and later transitioned to severe accident evaluation and documentation support to Chapter 19 (PRA and Severe Accidents). He was the Chapter 19 lead for PRA and Severe Accident for COLA development for the Pebble Bed Modular Reactor (PBMR).

Dr. O'Kula developed the outline, coordinated contributors, and assembled the first draft of the DOE *Accident Analysis Guidebook*, a reference guide for hazard, accident, and risk analysis of nuclear and chemical facilities operated in the DOE Complex. He is also the primary author and coordinator for the *Accident Analysis Application Guide* for the Oak Ridge contractor. Dr. O'Kula also developed a one-day course and exam for the guide, which he later presented to the Oak Ridge, Paducah, and Portsmouth staff.

Dr. O'Kula also led an independent V&V review for the DTRA of the U.K.-developed Urban Dispersion Model (UDM) software for predicting chemical and biological plume dispersion in city environments, and is leading projects to verify and validate chemical/biological simulation suite software applications for the Dugway Proving Ground (Utah), and the Edgewood Chemical Biological Center (ECBC) in Maryland.

Managing Member, Consequence Analysis 1997-2000

Dr. O'Kula was responsible for the consequence analysis associated with accident analysis sections of Documented Safety Analysis (DSA) reports and other safety basis documents for SRS, Oak Ridge, and other DOE nuclear facilities. He also developed the methodology and identified appropriate computer models for this purpose. Additionally, Dr. O'Kula developed training to enhance consistency and standardize analyses in the consequence analysis area. He was project manager for environmental assessment support to SRS on a transportation safety analysis using the RADTRAN code.

Dr. O'Kula coordinated development of a DOE Accident Analysis Guidebook involving over 10 sites and organizations. He also led the effort to produce Computer Model Recommendations for source term (fire, spill, and explosion), in-facility transport, and dispersion/consequence (radiological and chemical) areas.

Westinghouse Savannah River Company 1989 to 1997
Group Manager

Dr. O'Kula managed consequence analyses associated with accident analysis sections of DSA reports and other safety basis documents. He also developed the associated methodologies and identified appropriate computer models. He was a member of the management team supporting Criticality Safety Evaluation preparation assisting Safe Sites of Colorado and the dispositioning of final criticality safety issues for the decommissioning and decontamination of nuclear facilities at the Rocky Flats Environmental Technology Site.

In a teaming arrangement with Science Applications International Corporation, Kevin initiated discussions that led to development of an emergency management enhancement tool to risk inform likely source terms. Applied this approach to a Savannah River nuclear facility (K Reactor), and was part of the team to provide this methodology for use on the British Advanced Gas-Cooled Reactors (AGRs) (for the United Kingdom's Nuclear Installation Inspectorate). Model was knowledge-based, and required development of an Accident Progression Event Tree (APET) for the facility in question.

Dr. O'Kula managed the completion of the SRS K Reactor PRA program. He was the lead for development of the K Reactor Source Term Predictor Model and assisted with the core technology lay-up program to preserve competencies in reactor safety. He coordinated a 25-person group responsible for K Reactor probabilistic and deterministic dose analyses, and led the examination of reduced power cases at project termination. He developed risk and dose management applications to cost-effectively prioritize facility modifications.

Kevin interfaced with DOE Independent and Senior Review teams to finalize study acceptance, and transitioned the risk assessment team to risk management functions for nuclear and waste processing facilities. In addition, he successfully prepared a 10 CFR 100 Siting white paper to resolve issues raised by the DNFSB, and teamed with DOE/HQ legal support to document resolutions. He led the development of a position paper demonstrating SRS Replacement Tritium Facility compliance with DOE Safety Policy (SEN-35-91).

Staff Engineer

Dr. O'Kula led an analytical team quantifying the tritium source term during a Loss of River Water design basis accident. He evaluated airborne tritium levels with multi-cell CONTAIN model, interfaced with a multidisciplinary team to resolve Operational Readiness Review concerns, developed an SRS-specific methodology for applying MACCS as a tool for Level 3 PRA Applications, and applied CONTAIN code for K Reactor source term analysis.

E.I. du Pont de Nemours & Company **1982 to 1989**
Principal Engineer, Research Engineer

Dr. O'Kula performed risk analysis duties for the Savannah River Laboratory (SRL) Risk Analysis Group, after earlier conducting research activities for the Reactor Materials and Reactor Physics Groups. He performed initial planning for offsite irradiation of test specimens to evaluate remaining reactor lifetime for Savannah River reactor components.

Westinghouse Electric Corporation **1975**
Summer Student, Reactor Licensing
Monroeville, PA

American Electric Power Corporation **1973 to 1974**
Co-op Student, Reactor Physics and Reactor Licensing
New York, NY

Long Island Lighting Company **1972**
Summer Intern
Riverhead, NY

PARTIAL LIST OF PUBLICATIONS (2000-2012):

K. R. O'Kula, M. G. Wentink, and C.R. Lux , *Early Lessons Learned from Risk Applications of DOE Nonreactor Nuclear Facilities***,** 2012 EFCOG Safety Analysis Workshop, May 5-10, 2012 (Santa Fe, NM).

M. G. Wentink, K. R. O'Kula (Primary and Presenting Author), H. A. Ford, C.R. Lux, and H. C. Benhardt, *Operational Frequency Analysis Model Supporting the QRA for Risk-Informing the Design of a Waste Processing Facility***,** American Nuclear Society Winter Meeting, October 30 - November 3, 2011 (Washington, D.C.).

K. R. O'Kula, D. C. Thoman, J. Lowrie, and A. Keller, *Perspectives on DOE Consequence Inputs for Accident Analysis Applications*, American Nuclear Society 2008 Winter Meeting and Nuclear Technology Expo, November 9-13, 2008 (Reno, NV).

K. R. O'Kula, F. J. Mogolesko, K-J Hong, and P. A. Gaukler, *Severe Accident Mitigation Alternative Analysis Insights Using the MACCS2 Code*, American Nuclear Society 2008 Probabilistic Safety Assessment (PSA) Topical Meeting, September 7-11, 2008 (Knoxville, TN).

K. R. O'Kula and D. C. Thoman, *Modeling Atmospheric Releases of Tritium from Nuclear Installations*, American Nuclear Society Embedded Topical Meeting on the Safety and Technology of Nuclear Hydrogen Production, Control and Management, June 24-28, 2007 (Boston, MA).

K. R. O'Kula and D. C. Thoman, *Analytical Evaluation of Surface Roughness Length at a Large DOE Site (U)*, American Nuclear Society Winter Meeting, November 12-16, 2006 (Albuquerque, NM).

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Analysis Workshop, April 30 – May 5, 2005, Los Alamos, NM (2005).

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K. R. O'Kula, D. C. Thoman, J. A. Spear, R. L. Geddes, *Assessing Consequences Due to Hypothetical Accident Releases from New Plutonium Facilities (U)*, American Nuclear Society Embedded Topical Meeting on Operating Nuclear Facility Safety, November 14 – 18, 2004 (Washington, D.C.).

K. O'Kula and J. Hansen, *Implementation of Methodology for Final Hazard Categorization of a DOE Nuclear Facility (U)*, Annual Meeting of the American Nuclear Society, June 13-17, 2004, (Pittsburgh, PA).

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PUBLICATIONS (1988-1999):

Dr. O'Kula authored or co-authored more than 20 publications between 1988 and 1999. Details are available upon request.

PROFESSIONAL SOCIETIES AND STANDARDS COMMITTEES

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