July 15, 2013

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

DECLARATION OF RANDY GAUNTT CONCERNING NEXTERA'S MOTION FOR SUMMARY DISPOSITION OF FRIENDS OF THE COAST/NEW ENGLAND COALITION CONTENTION 4B

I, Randy Gauntt, hereby state as follows:

1. I have a Ph.D. in Nuclear Engineering from Texas A&M University and more than 30 years of experience performing diverse nuclear power plant related research activities for the Department of Energy (DOE) and Nuclear Regulatory Commission ("NRC") including analyses of severe accident progression in commercial power plants. I am the manager of the Severe Accident Analysis Department at Sandia National Laboratories (Sandia) and have maintained this position since 2004. Prior to my management position, I was a Distinguished Member of Technical Staff at Sandia. Currently, I manage a team of researchers and code developers who develop and apply the MELCOR severe accident analysis code and the MELCOR Accident Consequence Code System ("MACCS") atmospheric transport and consequence assessment code primarily for the NRC and Department of Energy ("DOE"). Under my direction, my group has recently completed the State-of-the-Art Reactor Consequence Analyses ("SOARCA") project and analysis of the Fukushima accidents.

2. From March 23, 2011, through April 27, 2011, I was on temporary assignment in the U.S. Embassy in Tokyo, Japan, providing support to the DOE efforts related to the Fukushima response. I provided in-country technical support and consultation to numerous stakeholders following the accidents at the Fukushima power plants, including support to the U.S. Military, U.S. NRC, State department and DOE as well as Japanese organizations including Tokyo Electric Power Company and the Japanese Nuclear Energy Safety Organization. I supervised initial source term estimations for use in ongoing estimation of what releases from the accidents could have been. I performed MELCOR analyses evaluating the potential damage states of each of the Fukushima reactors. I also performed MELCOR analyses focused on assessing the potential for subsequent damage to the already crippled Fukushima cores in the event of strong seismic aftershocks and to evaluate potential consequences of continued core damages.

3. As the Technical Leader of the MELCOR Code development project from 1995 to 2004, I led a team of engineers in the development of the MELCOR severe accident computer code for the NRC. I performed expert analyses of severe accident behavior using MELCOR aimed at characterizing anticipated fission product releases from accidents at nuclear power plants. I have performed numerous code validation and assessment efforts aimed at developing and improving physical models of core melt progression and fission product release and transport behavior. In addition, I served as the Technical Leader of the Ex-Reactor Severe Accident Experiments Team; Technical Leader of the Annular Core Research Reactor (ACRR) In-Pile “MP” Late Phase LWR Fuel Damage Experiments; Technical Leader of the ACRR In-Pile “DFR”(Damaged Fuel and Relocation) Early Phase LWR Fuel Damage Experiments; and Technical Leader of the TRAN-GAP Experiment Project.

4. In 2011, I received the DOE Secretarial Honors Award for Japan Earthquake and Tsunami Disaster Response Teams – which is the highest honor awarded to DOE Laboratory contractors, awarded for ongoing support of DOE and Japan during the Fukushima post-

accident crisis management period. I maintain membership in American Nuclear Society ("ANS") and American Society of Mechanical Engineers ("ASME"), regularly supporting organization and reviews for technical conferences of the ANS and ASME, including Nuclear Reactor Thermal Hydraulics and the International Conference on Nuclear Engineering. A copy of my professional qualifications is attached as Attachment ("Att.") 4B-N.

5. I have thoroughly reviewed the various inputs and assumptions used in the NextEra Energy, Inc. ("NextEra") severe accident mitigation alternatives ("SAMA") analysis, as submitted in May 2010 and revised in March 2012 to calculate offsite consequences associated with a postulated severe accident at the Seabrook Station ("Seabrook"). I have reviewed relevant supporting technical documentation for Seabrook's SAMA analysis. I have also reviewed the SAMA analysis revisions and clarifications provided in response to NRC Staff requests for additional information ("RAIs"). I thus have personal knowledge of the modeling methods, inputs, and assumptions used in the Seabrook SAMA analysis, as described in the Seabrook Station ER and other related documentation. I have also reviewed the relevant portions, including Section 5.3 and Appendix F of NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 46 regarding Seabrook.

6. In preparation for preparing this declaration, I also reviewed the 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 information and studies. I have reviewed the applicable portions of the following: Fauske and Associates, LLC Completes Development of MAAP5 notification regarding the MAAP code; NUREG-1935. "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report;" NUREG-7110 Volume 1 " State of the Art Reactor Consequence Analyses (SOARCA) Project: Peach Bottom Integrated Analysis;" NUREG-7110 Volume 2 " State of the Art Reactor Consequence Analyses (SOARCA) Project: Surry Integrated Analysis;" MAAP4 Applications Guidance. Desktop Reference for Using MAAP4 Software, Revision 2, (EPRI

1020236); In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI, S.R. Kinnersly, et.al; A Direct Comparison of MELCOR 1.8.3 and MAAP4 Results for Several PWR & BWR Accident Sequences," SAND—96-2053C; "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4 and SCDAP/RELAP5 codes," K. Vierow, *et. al*.; Severe Accident Code Analysis – The Nuclear News Interview, Nuclear News, K. Vierow; "Progression of Severe Accidents in the U.S. EPR," Z. Yuan and M. Khatib-Rahbar; NUREG-1465 "Accident Source Terms for Light-Water Nuclear Power Plants;" Draft NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Draft for comment (1987) and the final NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants;" and "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report," Brookhaven National Laboratory.

7. I have also reviewed applicable portions of the "Seabrook Station Applicant's Environmental Report Operating License Renewal Stage" (SAMA Analysis), "Seabrook Station Supplement 2 to Severe Accident Mitigation Alternatives Analysis," dated March 19, 2012; Supplement 3 to Severe Accident Mitigation Alternatives Analysis Response to RAI Request dated July 16, 2012" dated September 13, 2012; Friends of the Coast and New England Coalition Petition for Leave to Intervene, Request for Hearing, and Admission of Contentions; ASLB Memorandum and Order (Ruling on Petitions for Intervention and Requests for Hearing) (LPB-11-02) dated February 15, 2011; and the Commission Ruling (CLI-12-05), Memorandum and Order.

8. In this declaration, I present my views with respect to the issues addressed in NextEra's Motion for Summary Disposition of Contention 4B (SAMA Analysis Source Terms). In Contention 4B, the intervenor, Friends of the Coast and New England Coalition ("Friends/NEC"), alleges that NextEra's SAMA analysis "minimizes the potential amount of radioactive release in a severe accident". As admitted, the Commission and the Board

narrowed Contention 4B to three issues: (a) the Modular Accident Analysis Program ("MAAP") code has not been validated by the NRC; (b) the radionuclide release fractions generated by MAAP "are consistently smaller for key radionuclides than the release fractions specified in NUREG-1465" and result in "anomalously low" accident consequences; and (c) it previously has been observed that MAAP generates lower release fractions than those derived and used by NRC in other severe accident studies.

<u>Overview of the MAAP Code</u>

9. The MAAP code is a severe accident analyses computer code typically used by licensees in support of SAMA analyses.

10. MAAP was developed by Fauske & Associates originally under the sponsorship of Industry Degraded Core Rule-Making ("IDCOR") Program. Press Release, Fauske & Associates, LLC, Fauske & Associates, LLC, Completes Development of MAAP5 ("Fauske Press Release"), Attachment ("Att.") 4B-A at 1 (Jan. 27, 2009).

11. The sponsorship of MAAP was transferred to the Electric Power Research Institute ("EPRI") in 1985. *See id.*

12. MAAP3B was updated to MAAP4 in the mid-1990s to expand its modeling capabilities. MAAP4 incorporates updated physical models for core melt, reactor vessel lower head response, and containment response that provide improved mechanistic modeling of severe accident phenomena.

13. MELCOR is another severe accident analyses computer code typically used by the NRC and its contractors to perform severe accident analyses.

14. MELCOR was the code used in the recent SOARCA project. NUREG/CR-1935 State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, ("SOARCA Report"), Att. 4B-B, at xiii (2012).

<u>Validation and Benchmarking of the MAAP4 Code</u>

15. The MAAP code, while not explicitly validated by the NRC, is certainly accepted by the industry as a capable severe accident code of choice and has received rigorous validation and benchmarking during its development and update cycles. The validation and benchmarking have been documented repeatedly including:

a) An extensive list of benchmarks used in the review process of MAAP4 is presented in the "MAAP4 Applications Guidance: Desktop Reference for Using MAAP4 Software, Revision 2." MAAP4 Applications Guidance. Desktop Reference for Using MAAP4 Software, Revision 2, 1020236 ("MAAP Application Guide"), Att. 4B-C, at 7-1 (2010). The review includes compilations of the degree of agreement for the major code models in Table 7-3, "Compilation of the Agreement for the Major Code Models Related to Level 1 Phenomena." *Id.* at 7-16. The code has been developed and maintained under Fauske and Associates quality assurance program which is in compliance with 10 CFR 50 Appendix B and ISO 9001 quality assurance requirements. *Id.* at 2-2.

b) The report, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, identifies the MAAP code as one of the codes used to "… assess the capability of the severe accident analysis methods (e.g., benchmark the relevant codes)..". S.R. Kinnersly, et.al, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI - January 1991," NEA/CSNI/R(91)12 ("Kinnersly Report"), Att. 4B-D, at 6.18 (Nov. 1991).

c) A code comparison analysis was done in 1996 between MELCOR 1.8.3 and MAAP4.0.2 where 9 accident sequences were studied with both codes. M.T. Leonard, *et.al*, "A Direct Comparison of MELCOR 1.8.3 and MAAP4 Results for Several PWR & BWR Accident Sequences," SAND—96-2053C, Att. 4B-E, at 1 (1996). The researchers in this study concluded that the differences in the timing of events between the two codes was primarily due to variations in core physics models and that the general characteristics of ex-vessel behavior of core debris are in reasonably good agreement between the codes. *Id.* at 9.

d) Another code comparison study of MELCOR 1.8.5, MAAP4.0.5 and SCDAP/RELAP5 MOD3.3 by Vierow showed that the calculation results of the codes are very similar in terms of thermal-hydraulic and core degradation response. K. Vierow, *et. al*., Nuclear Engineering and Design 234, "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4 and SCDAP/RELAP5 codes" ("Vierow Comparative Analysis"), Att. 4B-F, at 129 (2004). *See also* K. Vierow, Nuclear News, "Severe Accident Code Analysis – The Nuclear News Interview" ("Vierow Interview"), Att. 4B-G, at 23 (2005). The Vierow study stated that there are minor discrepancies in various timings of phenomena, which are within the uncertainties of the code numerical computation and the physics models (in terms of thermal-hydraulics and core degradation) for MELCOR 1.8.5, MAAP4.0.5 and SCDAP/RELAP5 MOD3.3 in a hypothetical nuclear power plant accident. Vierow Comparative Analysis , Att. 4B-F at 129. The results showed that the thermal hydraulic phenomena and major in-vessel severe accident phenomena are in good agreement between the three codes. *Id.* at 144; Vierow Interview, Att. 4B-G at 1.

e) A more recent paper on modeling severe accidents for the U.S. EPR design identified a number of differences between MELCOR 1.8.6 and MAAP4. Z. Yuan and M. Khatib-Rahbar, 102 Transactions of the American Nuclear Society, "Progression of Severe Accidents in the U.S. EPR" ("Yuan Article"), Att. 4B-H, at 2 (2010). For example, there are differences in the modeling of molten core concrete interactions between the two codes that influence the calculated retention period of core debris by the cavity melt plug. In terms of the prediction of in-vessel accident progression, both codes are generally consistent, except that MAAP predicted event progression faster than MELCOR, leading to earlier vessel breach. *Id.* at 1. The authors concluded that MAAP and MELCOR predicted fission product releases for scenarios involving intact or partially intact containment with reasonable agreement, although MELCOR predicted a significantly higher release of volatile fission products in a steam generator tube rupture scenario. *Id.* at 2.

<u>The Use of NUREG-1465 Source Terms for a SAMA Analysis is Fundamentally In Error</u>

16. Contention 4B makes a fundamental error in comparing NUREG-1465 release fractions to those calculated by MAAP for SAMA analyses. The source terms identified in NUREG-1465 are releases *to the containment* while the source terms identified in NextEra's SAMA are releases *to the environment*. Substantial amounts of the radionuclides released from a reactor during a severe accident could be expected to be captured and confined within the containment structure that is designed to isolate the reactor from the environment. The itemized descriptions below address prominent mechanisms by which radionuclides could be captured within containment. Current severe accident codes such as MAAP and MELCOR account for these mechanisms.

a) The capture of radionuclides within containment could be by active systems or passive (natural) processes. Active systems could include sprays and filtered recirculation ventilation systems and would normally require AC electrical power to operate. Under extraordinary circumstances, containment sprays might be alternatively supplied by gasoline or diesel engine-driven pumps. Passive processes, such as gravitational settling and scrubbing, occur without any dependence on electrical power.

b) Substantial amounts of the radionuclides released from a reactor core during a severe accident could gravitationally settle onto equipment and structures residing within the containment. These radionuclides would be much less susceptible to release to the outside environment.

c) Substantial amounts of the radionuclides released from a reactor core during a severe accident could become stably waterborne in containment when washed down to the floor by operating containment sprays.

d) Containment spray operation in a severe accident would serve two functions. The first would be to cool and reduce the pressure within containment. This could reduce normal leakage by reducing driving pressure or delay or prevent containment failure caused by over-pressurization. The second function would be to wash aerosols from the containment atmosphere making them waterborne and not susceptible to release to the environment. PWR containment spray systems are capable of reducing the concentration of airborne activity in containment by two orders of magnitude within 30 minutes. NUREG-1465 "Accident Source Terms for Light-Water Nuclear Power Plants" ("NUREG-1465"), Att. 4B-I, at 18.

e) In a severe accident where reactor core debris has melted through the bottom of the reactor vessel and fallen to the concrete floor of containment, the material may encounter a pool of water. The pool would have formed from the initial water inventory of the reactor system and water sprayed through an operating containment spray system collecting at this lowest elevation of containment. The pool would overlie the core debris and would serve to scrub radionuclides releasing from the debris. The efficiency of the scrubbing would increase with increasing pool depth.

17. There is the possibility of an accident that bypasses containment. For a PWR, the bypass accidents most often envisioned are an induced steam generator tube rupture (“SGTR”) (induced by a station blackout (“SBO”) event, for example), and an interfacing system loss-of-coolant accident (“ISLOCA”). An SGTR would allow radionuclides to pass from the primary coolant system to the steam and power conversion system and eventually to the environment. The ISLOCA is characteristically envisioned to be caused by the failure of two serial check valves, leading to a pipe over-pressurization rupture and subsequent leakage of coolant from the reactor vessel into buildings outside of containment.

18. Releases to the environment estimated for containment bypass scenarios are typically relatively high. Even so, the releases in a bypass accident would be through a tortuous path serving to capture significant amounts of radionuclides such that the releases to the environment would be smaller than the releases to containment addressed by NUREG-1465. For example, the SOARCA project calculated both a spontaneous SGTR and a thermally induced SGTR (“TI-SGTR”) included in a short term SBO (“STSBO”) sequence for the Surry reactor. Surry is a PWR that is similar in design to Seabrook. SOARCA radionuclide releases to the environment for the STSBO with TI-SGTR are found to be significantly lower than the release fractions to containment contained in NUREG-1465. *See* NUREG-7110, Volume 2, " State of the Art Reactor Consequence Analyses (SOARCA) Project: Surry Integrated Analysis" (“NUREG-7110”), Att. 4B-J, at 160-161 (2012). For the spontaneous SGTR scenario, it was concluded that there was a high likelihood that operator mitigative actions would be successful, therefore no radionuclide results were reported for the unmitigated case. The mitigated case successfully kept fuel failures from occurring and did not result in radionuclide releases to the environment.

19. SOARCA also included an ISLOCA sequence for Surry, which proved to have much lower releases to the environment than release fractions in NUREG-1465. *See* NUREG-

7110, Att. 4B-J, at 210. Additionally, the frequency of occurrence estimated for induced SGTR or ISLOCA bypass events are typically quite low.

20. The Seabrook SAMA addresses multiple containment bypass scenarios, including SGTR and ISLOCA.

21. Intervenors state that "MAAP generates lower release fractions than those derived and used by NRC in studies such as NUREG-1150," and referenced a BNL Report which identified NUREG-1150 release fractions were higher than MAAP generated values. Friends Of The Coast and New England Coalition Petition for Leave to Intervene, Request for Hearing, and Admission of Contentions ("Friends/NEC Petition") at 44 (Oct. 20, 2010). Friends/NEC next reference the draft NUREG-1150 where comparisons between STCP results and MAPP results were made. Friends/NEC Petition at 45. Such comparisons to these historical documents which used the outdated STCP model are inappropriate. The understanding of severe accident phenomenology and physics modeling capabilities have advanced significantly since the 1990 publication of the NUREG-1150. This understanding has been integrated into severe accident codes such as MAAP4, which was used in the Seabrook analysis, making comparisons to such dated analyses as the draft NUREG-1150 of little value.

Draft NUREG-1150's[1] Conclusions Have Been Overcome by Improved
State of the Art Accident Sequence Code Modeling During the Intervening 20 Years

22. Relatively limited severe accident computing abilities existed at the time NUREG-1150 was prepared. As such, comparisons between results obtained from modern codes such as MELCOR or MAAP4 to early codes such as those used in NUREG-1150 are of limited value. The realism afforded by both of these codes has advanced substantially since the publication of NUREG-1150. The work of NUREG-1150 utilized an assortment of computer codes collectively

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[1] Contention 4B includes arguments related to the final NUREG-1150 published in 1990 and the Draft NUREG-1150 published for comment in 1987. In this declaration, reference to the draft version of NUREG-1150 will be identified as "Draft NUREG-1150 and the final version as "NUREG-1150."

called the Source Term Code Package (“STCP”), which included MELCOR, and expert elicitation to estimate radionuclide releases.

23. At the time when NUREG-1150 was being published, the MELCOR code was predominantly parametric with respect to modeling complicated physical processes (in the interest of quick code execution time and a general lack of understanding of reactor accident physics). Over the years, however, as phenomenological uncertainties have been reduced and user expectations and the number of applications for using MELCOR have increased, the models implemented into MELCOR have become increasingly best-estimate in nature. The increased speed (and decreased cost) of modern computers (including PCs) has eased many of the perceived constraints on MELCOR code development. Today, most MELCOR models are mechanistic, with capabilities approaching those of the most detailed codes of a few years ago. The use of models that are strictly parametric is limited, in general, to areas of high phenomenological uncertainty where there is no consensus concerning an acceptable mechanistic approach.

24. The differences suggested as part of Contention 4B between releases calculated by MAAP and STCP as reported in Draft NUREG-1150 are summarized in Figure 5.5, “Comparison of results for station blackout scenarios at Zion.” Draft NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Draft for comment, (“Draft NUREG-1150”) at 5-15 (1987). Figure 1 reproduces part of Figure 5.5 from Draft NUREG-1150. Figure 1 compares the release fractions for Zion’s station blackout scenarios experiencing early containment failure between draft NUREG-1150 distributions and STCP and IDCOR/MAAP point estimates.[2]

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[2] The full Figure 5.5 from the draft NUREG-1150 also compares the releases for late containment failure which were exhibiting the same relative trends

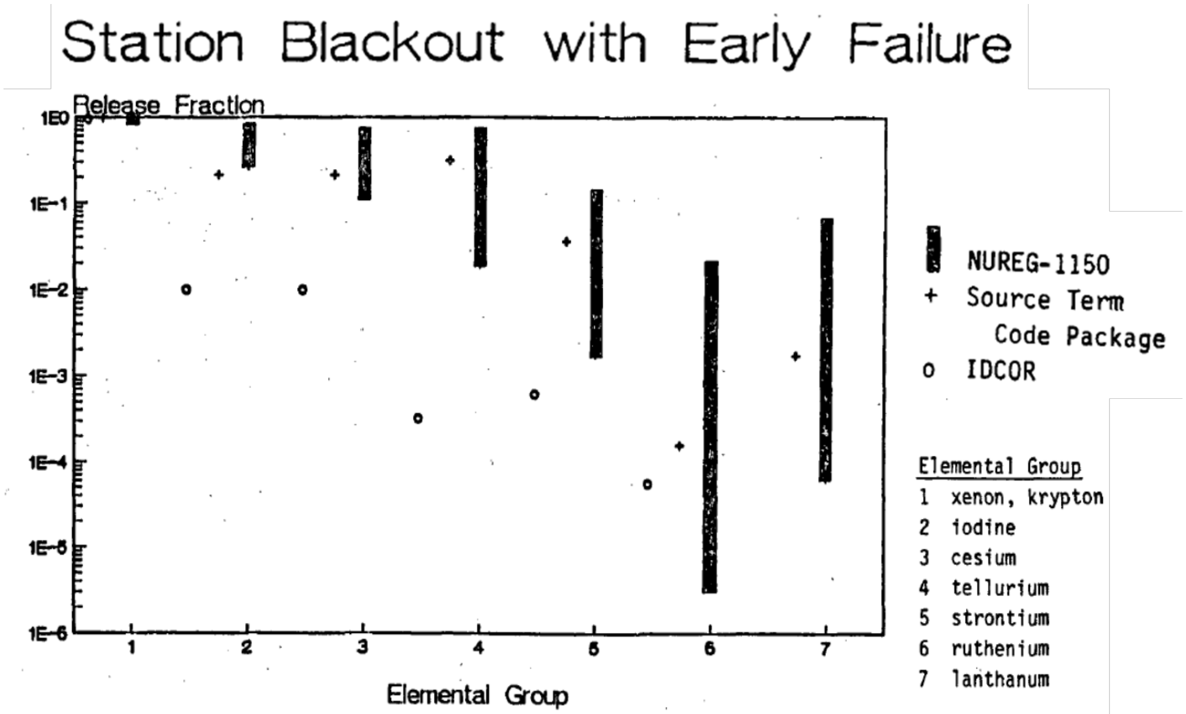

Figure 1: Comparison of NUREG-1150 and MAAP (IDCOR) Release Fractions

Draft NUREG-1150, Att. 4B-K, at 5-15.

25. As can be seen in Figure 1, MAAP release fractions, labeled IDCOR[3] in the figure, for the iodine, cesium and tellurium groups are one to two orders of magnitude less than the STCP and NUREG-1150 values. Draft NUREG-1150 does not attribute these differences to any characteristic deficiencies in the physical models in MAAP. Rather, it concludes that these differences are "indicative of technical disagreements in the source term models," *i.e.* due certainly at least in part to differences in user input to the codes as opposed to only differences between the codes themselves. *See* Draft NUREG-1150, Att. 4B-M, at 5-14. Noteworthy is that this comparison between codes presented in Draft NUREG-1150 is not part of the final version of the NUREG-1150. Additionally, these results do not reflect improvements in state-of-the-art modeling as reflected in the current Seabrook analysis.

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[3] MAAP was originally developed for the IDCOR program in the early 1980s.

The BNL Report Does Not Make a Proper Comparison
Between NUREG-1150 and MAAP

26. Contention 4B identifies a BNL report comparing radionuclide releases for the Catawba plant predicted with MAAP to releases for Sequoyah predicted with the STCP and reported in the final version of NUREG-1150. The BNL report, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" ("BNL Report"), contains Table 9, which summarizes the comparisons. Brookhaven National Laboratory (BNL). "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" ("BNL Report"), Att. 4B-L, at 17 (2002). The authors of the BNL report contend that the Sequoyah/NUREG-1150 releases are about four times greater than the MAAP/Catawba releases, and state that "apparently the differences in the release fractions in the above table are primarily attributable to the use of the different codes in the two analyses". BNL Report, Att. 4B-L, at 17. Although one may infer that the "use of the different codes" encompasses differences in user input as well as inherent phenomenological modeling differences, the report does not discuss how important the differences in user input are when comparing the results from different computer codes. The BNL report also does not discuss the limitations of comparing releases calculated with computer codes many years apart in their development. The Catawba release category used for comparison to Sequoyah NUREG-1150 results is described as an early containment failure with no ex-vessel release[4]. The Sequoyah results are for a collection of scenarios which may or may not include scenarios with no ex-vessel release, making direct comparisons between the Catawba and Sequoyah results questionable.

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[4] *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).

27. Prominently missing from the BNL report is an accounting of which results for Sequoyah from NUREG-1150 were used in the comparison with MAAP/Catawba and why they were identified as being the closest comparison.

28. It is important to note that the release fractions presented in NUREG-1150 are in the form of distributions resulting from many accident scenarios. For the Sequoyah plant, NUREG-1150 presents distributions for two release categories – one for all scenarios involving early containment failure and one for all scenarios involving late containment failure. Each distribution has a minimum, maximum, median, and mean value associated with it as shown in Figure 5.6 "Source term distributions for early containment failure at Sequoyah." NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" ("NUREG-1150"), Att. 4B-M, at 5-13 (1990).

29. Table 9 in the BNL report apparently limited the comparison to the maximum releases for each radionuclide class from the early containment failure category for Sequoyah. Given the conservative approach of selecting the maximum release values for early containment failure at Sequoyah, it would be expected that MAAP/Catawba releases are smaller than the NUREG-1150 Sequoyah releases. That type of comparison provides very little meaningful information as to the acceptability of the MAAP code. A more meaningful comparison would involve plotting the MAAP/Catawba release fractions alongside the Sequoyah/NUREG-1150 release distributions from Figure 5.6, or constructing a table with the MAAP/Catawba values juxtaposed with Sequoyah/NUREG-1150 minimum, median, mean, and maximum values. Table 1, below, shows the Sequoyah/NUREG-1150 mean and maximum values for early containment failure scenarios. As this table shows, the MAAP release fractions are actually in fairly good agreement with the mean release fractions calculated with STCP for NUREG-1150. This is a more proper comparison.

Table 1. MAAP/Catawba versus Sequoyah/NUREG-1150 Release Fractions

RN class	Duke MAAP Catawba	NUREG-1150 Sequoyah maximum	NUREG-1150 Sequoyah mean	Ratio of Sequoyah maximum to Catawba	Ratio of Sequoyah mean to Catawba
Xe	1.0E+00	1.0E+00	1.0E+00	1.00	1.00
I	5.5E-02	2.9E-01	1.0E-01	5.27	1.82
Cs-Rb	4.8E-02	2.6E-01	8.0E-02	5.42	1.67
Te-Sb	3.0E-02	2.1E-01	3.5E-02	7.00	1.17

NextEra's SAMA Source Terms are Higher than NUREG-1150 Source Terms

30. The radionuclide releases presented in NextEra's supplement to its SAMA analysis include 13 release categories, most of which contain multiple accident sequences. [1] Letter from Paul O. Freeman, Site Vice President, dated March 19, 2012, transmitting Seabrook Station Supplement 2 to Severe Accident Mitigation Alternatives Analysis, ("SAMA Supplement SBK-L-12053") at 6 (ADAMS Accession No. ML12080A137). The release fractions associated with these sequences were developed using MAAP4.0.7. *Id*. at 6. The 5%, 95%, median, and mean values of fractional releases of key radionuclide groups were calculated with information provided in 11 of the 13 release categories presented in the revised SAMA. *Id.* at 20-26. Figure 2, "Source Term Comparison – Seabrook SAMA Versus NUREG-1150," compares these source terms from the Seabrook SAMA analysis and NUREG-1150 source terms.[5] The NUREG-1150 source terms are taken from the accident sequences involving containment bypass at Surry and early containment failures at Sequoyah and Zion. NUREG-1150, Att. 4B-M, at 3-15, 5-13, 7-10. The operating reactors at Surry and Sequoyah and the retired reactors at Zion are the same type as the reactor at Seabrook, *i.e.*, all of these reactors are pressurized water reactors. In particular, Zion is a Westinghouse 4-loop large dry containment like Seabrook. *Id.* at 7-1.

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[5] Release categories INTACT1 and INTACT2 of the Seabrook SAMA analysis have been excluded from this comparison because they do not involve either containment failure or containment bypass. Category INTACT1 only allows nominal containment leakage at the maximum Technical Specification allowable limits while INTACT2 allows 10x this amount.

These comparisons are between reactors of the same type, for accident sequences involving some type of containment failure leading to large releases to the environment. The ranges of calculated release fraction are grouped together on the x-axis by radionuclide class, and the comparison is done on a logarithmic scale due to a large spread in the results. The figure shows that the 5%-95% range is comparable between Seabrook and the NUREG-1150 plants. In addition, the mean and median for iodine, cesium and tellurium are all higher for Seabrook than for the NUREG-1150 plants, showing that the SAMA analysis actually includes a level of conservatism when compared to previous reactor studies, in direct contradiction to the claims of the interveners.

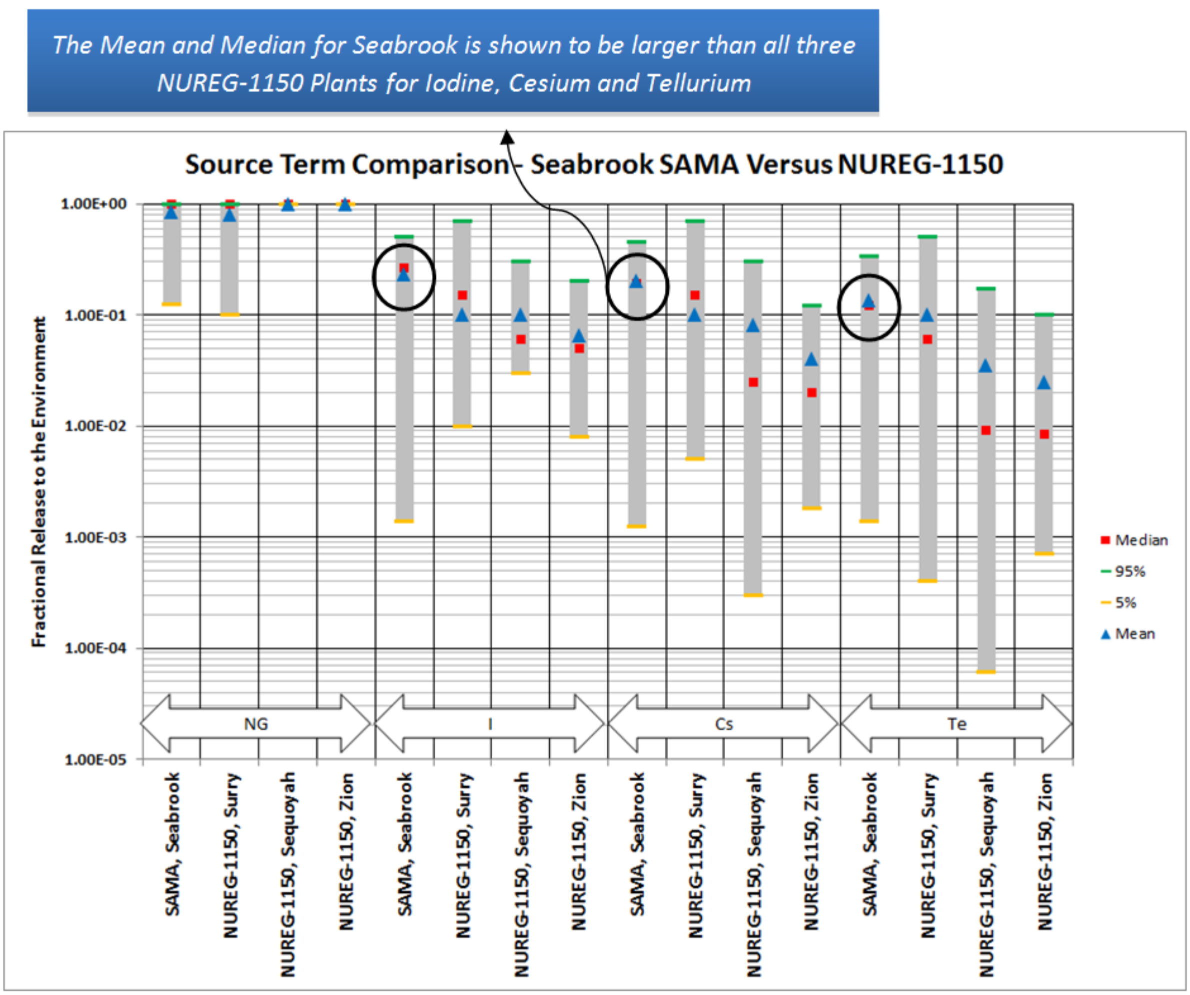

Figure 2: Source Term Comparison – Seabrook SAMA Versus NUREG-1150.

31. The conservatism in the Seabrook release fractions appear related to the use of multiple plume segments. The Seabrook SAMA analysis used four plume segments to represent releases over the seven days for which the accident sequences were modeled. In many of the 13 accident sequences, releases during the fourth (last) plume segment dominated the release or were a significant contributor. For all categories, the fourth plume release does not begin until at least 48 hours after the accident initiating event. Releases this late in the accident sequence would be considered a late release. Because the selected NUREG-1150 plant results only included containment bypass and early containment failure sequences (which had the highest release fractions), Figure 3 was created to compare the same NUREG-1150 release fractions to the sum of the first three plume releases for Seabrook, which would include any early releases but exclude late releases. This approach allowed us to compare more similar accident sequences. Figure 3 shows that with the exception of noble gases, the mean and medians for Seabrook compare very well with the NUREG-1150 plants. These two figures identify that the release fractions for key radionuclide groups calculated with MAAP4 and reported in NextEra's SAMA analysis for Seabrook are comparable and slightly conservative when compared to the release fractions presented in NUREG-1150 for the same type of reactor as the Seabrook reactor.

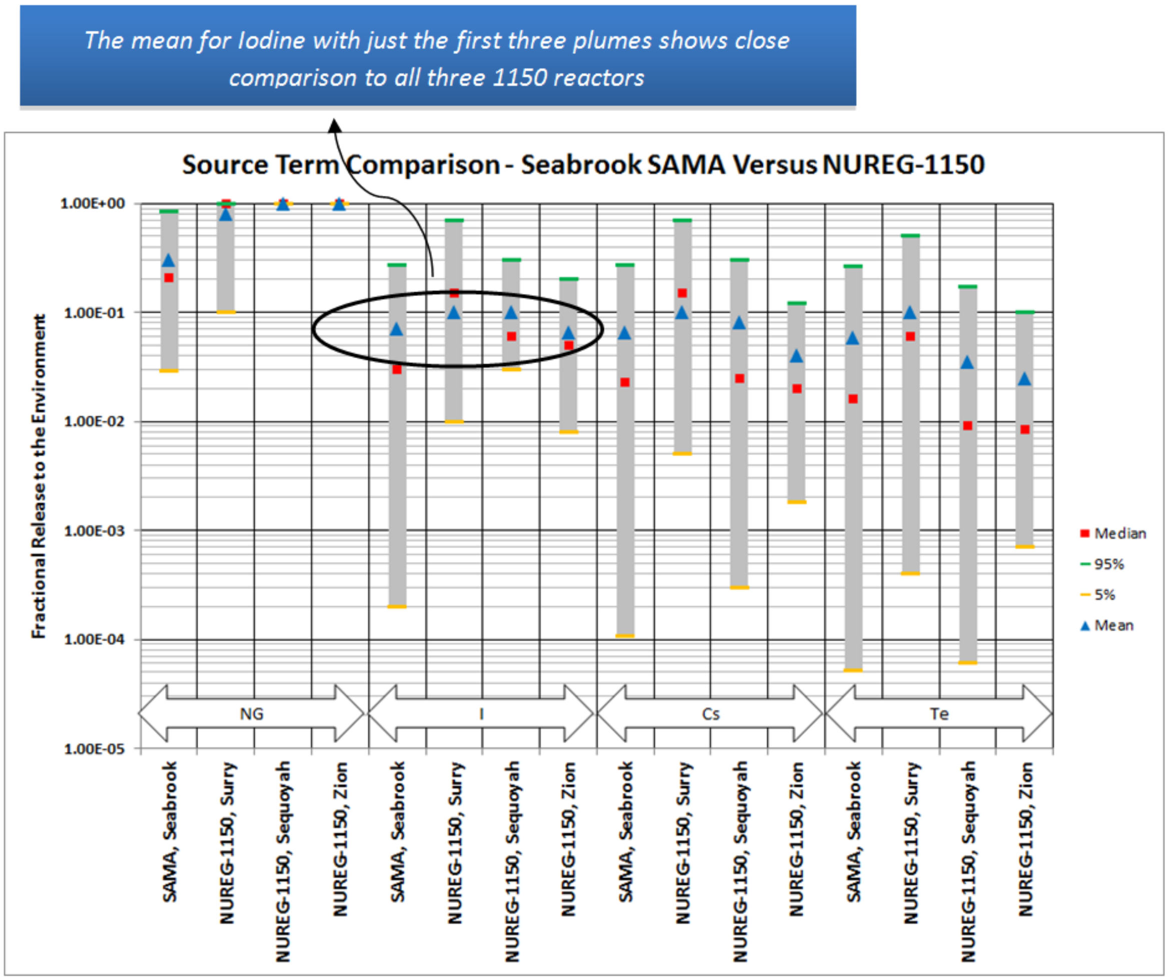

Figure 3: Source Term Comparison – Seabrook SAMA Excluding Plume 4 Releases Versus NUREG-1150

<u>NextEra's SAMA Source Terms are Consistent with SOARCA Source Terms</u>

32. A meaningful comparison with respect to the reasonableness of the Seabrook SAMA source terms is to the results in recently published SOARCA reports. The SOARCA project was commissioned by the NRC to develop best estimates of the offsite radiological health consequences of conceivable severe reactor accidents. NUREG/CR-1935, SOARCA report, excerpt attached as Att. 4B-B, describes the SOARCA project and NUREG-7110, excerpt attached as Att. 4B-J, describes the PWR related analyses which are compared to Seabrook. The SOARCA project used insights gathered from previous accident studies including NUREG-1150 and WASH-1400. It also benefitted from advancements in computer

modeling and increased understanding of the physical and chemical processes that occur during a reactor accident. A suite of accident sequences was determined, informed by the probability of fuel damage (otherwise known as core damage frequency ("CDF")), and the sequences were simulated with the MELCOR code, Version 1.8.6, which was updated to include improved phenomenological understanding of such processes as core melt progression, fission product release and transport, and molten core concrete interactions. NUREG-7110, Att. 4B-J, at 37. This was accomplished for two nuclear power plants – Peach Bottom (a BWR) and Surry (a PWR). The calculations included plant specific data wherever possible.

33. Two comparisons between Seabrook's SAMA analysis and the SOARCA project were completed. In Figures 4 and 5 below, accident category LL5 from Seabrook's SAMA analysis is compared to the long term station blackout ("LTSBO") from the SOARCA analysis as reported for Surry Power Station in NUREG/CR-7110 Volume 2. Figure 6 compares the SE2 category from Seabrook to the interfacing systems loss-of-coolant accident ("ISLOCA") from SOARCA. Category LL5 is defined as a "Large/Late Containment Basemat Failure" which encompasses a number of possible accident sequences, however the representative case is a station blackout with emergency feedwater success for 12 hours. This makes a comparison to the Surry LTSBO reasonable. Category SE2 is a "Small/Early Containment Bypass – ISLOCA with Scrubbed Release". Category LE2 also contains ISLOCA scenarios, but the LE2 scenarios do not include scrubbing prior to release to the environment. In SOARCA, the ISLOCA release pipe is underwater by the time releases begin, making it also a somewhat scrubbed release, therefore the SE2 category is more appropriate for comparison than the LE2 category. The source terms in SOARCA were evaluated using MELCOR version 1.8.6. Both Seabrook and Surry are pressurized water reactors with similar large, dry containments, although Seabrook's containment is at atmospheric pressure, while Surry's is subatmospheric. The comparisons then, between the MAAP generated source terms for Seabrook and the MELCOR generated

source terms for Surry, are more meaningful given the type of accident and reactor, and reflect changes in accident modeling over the last two decades.

34. The comparison between LL5 and the SOARCA LTSBO was taken in two parts. The one major difference between the two sequences is the containment failure mechanism. Category LL5 states that failure occurred by base mat erosion, while in SOARCA the failure was due to over-pressurization of containment. The reason is likely due to a combination of differences in system availabilities and timings, containment type, and modeling choices, and is not indicative of differences between the two codes. Due to the divergence in containment failure modes, the first comparison, seen in Figure 4, compares releases up to the point of containment failure. The release fractions between the two codes are in very good agreement up to this point. The only class for which SOARCA had a significantly larger release than Seabrook was the molybdenum class. This could be due to the fact that in SOARCA a large fraction of the molybdenum class combines with cesium to form Cs_2MoO_4 which is tracked separately. It is unknown if the modelers at Seabrook used a similar class combination. However, as can be seen in Figure 4, cesium, iodine and tellurium, radionuclides observed to be of highest importance in consequence analyses, are slightly higher for Seabrook as compared to SOARCA. These radionuclides are of highest importance because they are more volatile than many other classes, leading to typically higher release fractions. Additionally, the extremely short half-lives of iodine and tellurium (high specific activity) make them strong contributors in the early, emergency, phase of an accident, while the very long half-life and relative large inventory of cesium leads to a domination of risk during the late phases of an accident and during the recovery phase. Figure 5 shows the comparison of release fractions at the end of the modeled transient, which progresses down very different paths after the divergent containment failures. Most classes have higher releases for Seabrook than for SOARCA showing a level of conservatism and the only classes that are significantly larger in SOARCA than Seabrook’s SAMA are the barium, strontium and molybdenum classes that are shown in a

SOARCA Volume 2 class importance study found in Figures 7-12 to 7-20 to have very little contribution to risk for either the emergency or long-term phases of the accident. NUREG-7110 Att. 4B-J, at 334 to 339. This comparison shows that MELCOR and MAAP produce very similar release fractions while on the same general accident progression and in this specific instance, Seabrook results can be seen as reasonable and even slightly conservative.

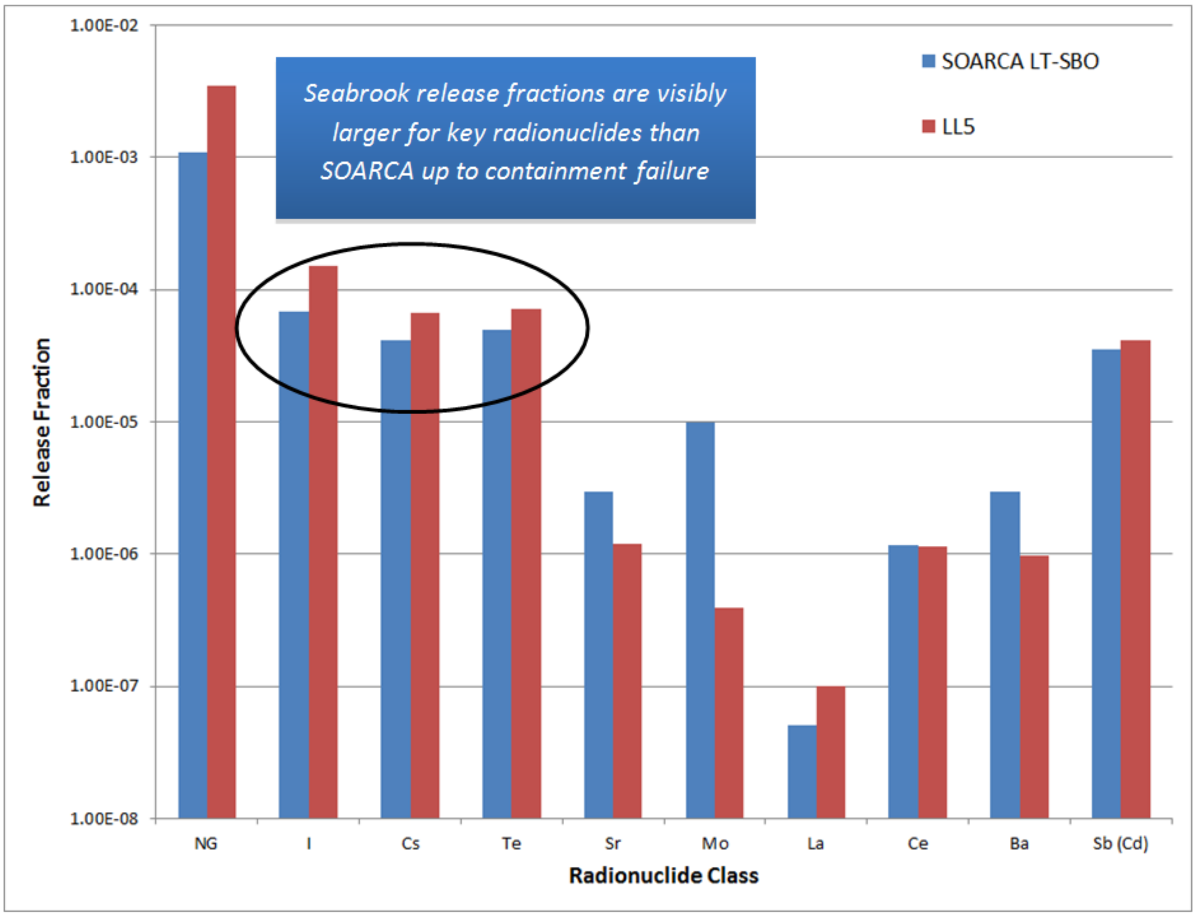

Figure 4: Source Term Comparison – Seabrook SAMA LL5 Versus Surry LTSBO Up To Containment Failure.

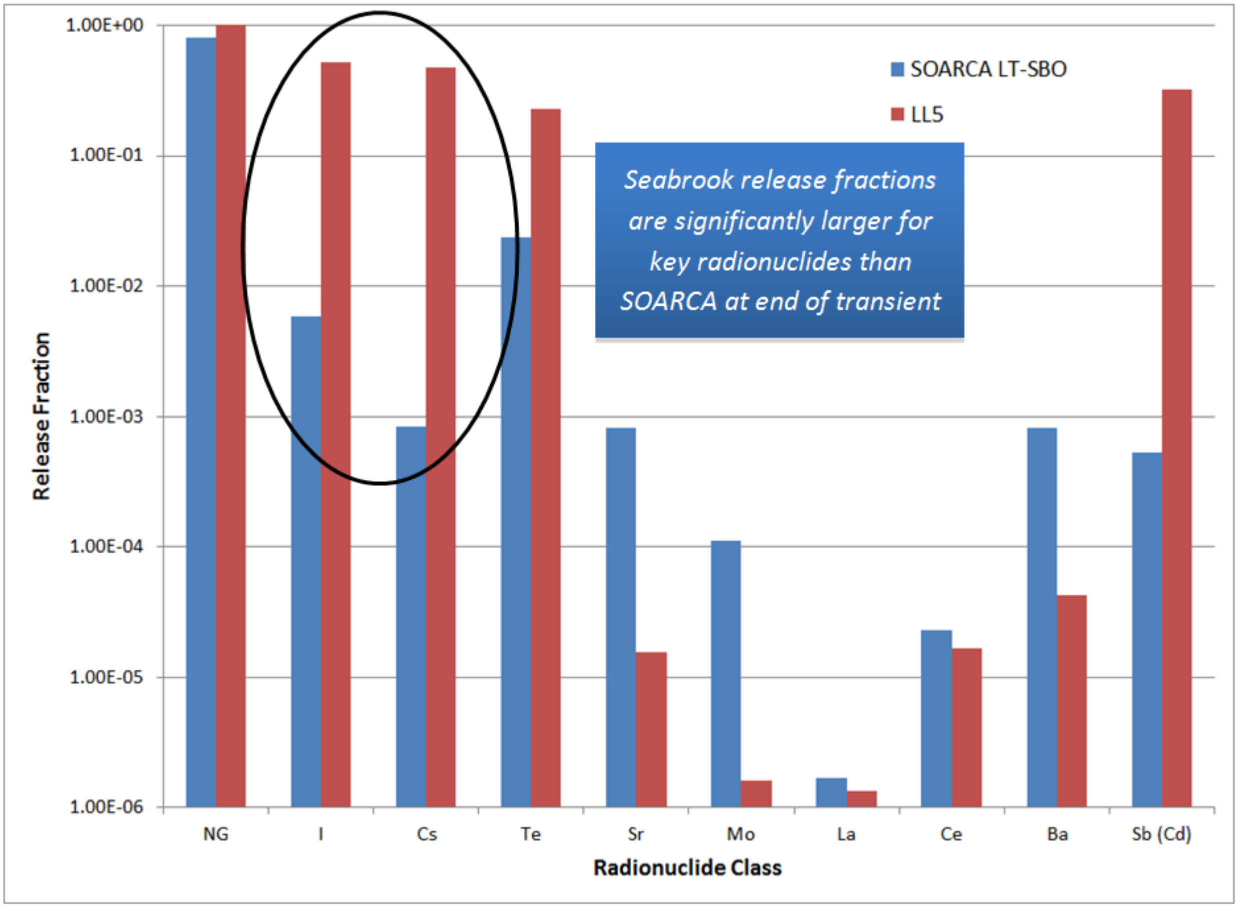

Figure 5: Source Term Comparison – Seabrook SAMA LL5 Versus Surry LTSBO Total Releases.

35. A second comparison was undertaken, between ISLOCA sequences from the Seabrook SAMA analysis and the Surry SOARCA project. Although the entire release path cannot be confirmed between the two codes, both release points (outside of primary containment) are flooded at the time releases begin, allowing for some scrubbing of radionuclides before the ultimate environmental release. This combination of scrubbing and early containment bypass makes the two sequences similar. For SOARCA, this ISLOCA was only run out to 48 hours, which corresponds to the first three plume segments for Seabrook. Therefore, in Figure 6, the SOARCA ISLOCA release fractions at 48 hours are compared to both the sum of the first three plume segments from Seabrook's SAMA analysis as well as the total release from all four plume segments. While remaining differences in the accident sequences prevent a completely quantitative comparison, it is important to note that release fractions are very comparable for most radionuclide classes, and for any classes that exhibit large magnitude differences, the Seabrook releases are larger and therefore, conservative. This comparison between similar accident sequences using modern versions of the MAAP and MELCOR codes gives additional support to the appropriateness of Seabrook SAMA release fractions.

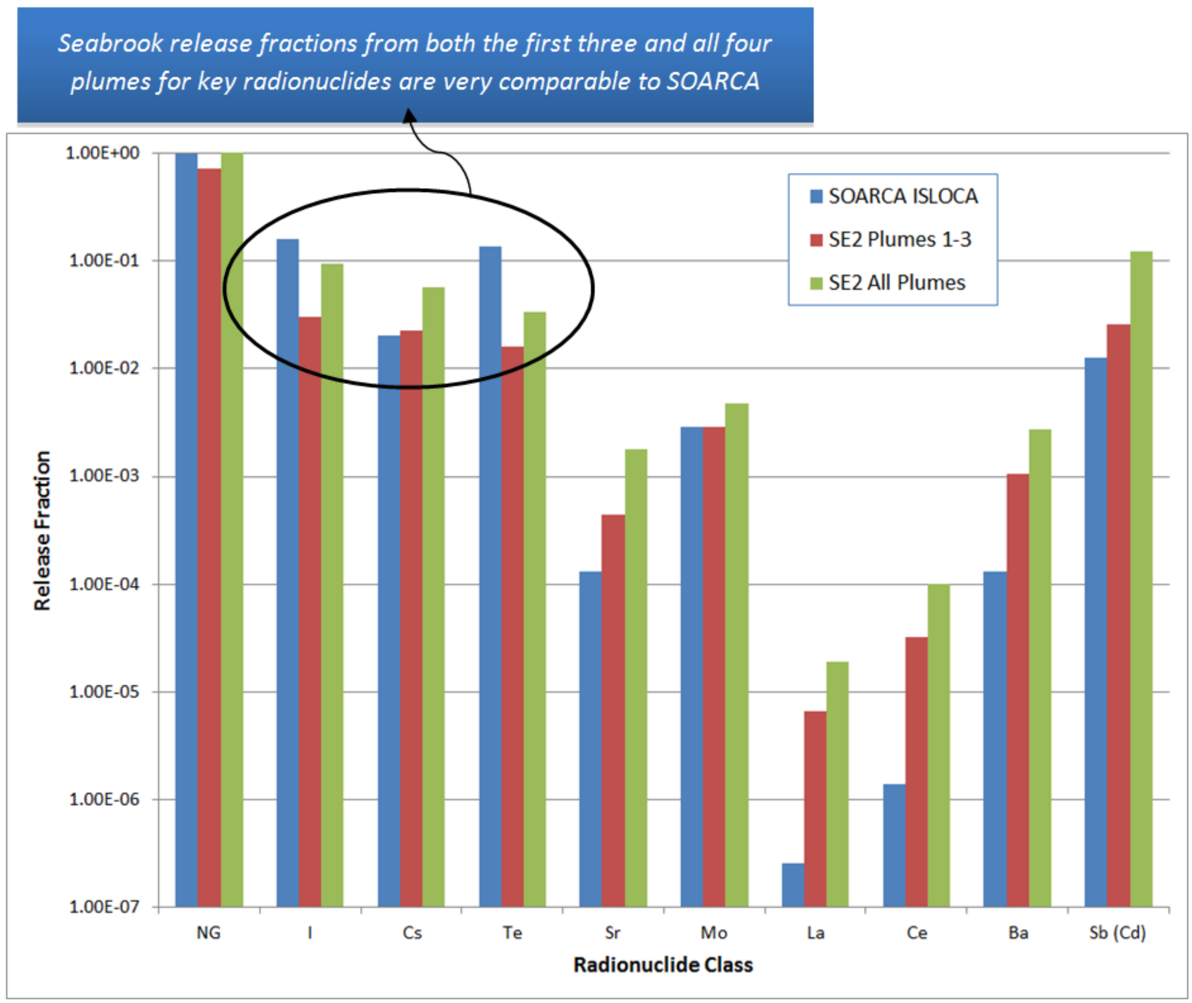

Figure 6: Source Term Comparison – Seabrook SAMA SE2 Versus Surry ISLOCA.

<u>Executed in accordance with 10 C.F.R. § 2.304(d).</u>
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Date of signature: July 15, 2013