September 14, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
FIRSTENERGY NUCLEAR OPERATING COMPANY	)	Docket No. 50-346-LR
	)	
	)	
(Davis-Besse Nuclear Power Station, Unit 1)	)	
	)	
	)	

AFFIDAVIT OF KYLE W. ROSS CONCERNING
THE MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4

I, Kyle W. Ross, hereby states as follows:

1. I am a Principal Member of the Technical Staff in the Severe Accident Analysis Department at Sandia National Laboratories ("Sandia") with more than 20 years of continuous involvement in safety analysis of nuclear reactor systems, thermal-hydraulic analysis, severe accident modeling, and real-time computer simulation. I have been on the staff at Sandia for two years. Prior to joining the Sandia staff, I worked for Los Alamos National Laboratory on-site at Sandia in support of severe accident analyses beginning in 2006. I was a lead Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") analyst for Sandia on the State-of-the-Art Reactor Consequence Analyses ("SOARCA") Project. I have a lead technical role on numerous computer modeling efforts associated with simulating severe accidents in varied nuclear reactor systems. I have performed transient analyses (thermal hydraulic and core melt progression) of various postulated severe accidents in pressurized water reactor ("PWR") nuclear power plants with the MELCOR computer code, with emphasis on validating code enhancements and developing improved modeling strategies. I was responsible for developing, interfacing, and installing real-time engineering grade software in nuclear power plant

simulators.[1] I had the lead responsibility for simulator projects at the Palo Verde Nuclear Generating Station, Comanche Peak Steam Electric Station, Cooper Nuclear Station, and the Salem nuclear power plant. A statement of my professional qualifications is attached hereto.

2. In this declaration, I present my views with respect to the issues addressed in FirstEnergy's Motion for Summary Disposition of Contention 4 (SAMA Analysis Source Terms). Contention 4 alleges that the SAMA analysis "underestimates the true cost of a severe accident at Davis-Besse." The contention, as narrowed by the Atomic Safety and Licensing Board and the Commission, raises 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>

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

4. MAAP was developed by Fauske & Associates originally under the sponsorship of Industry Degraded Core Rule-Making ("IDCOR") Program. *See* FirstEnergy's Motion, Attachment 20, at 2-2.

5. The sponsorship of MAAP was transferred to the Electric Power Research Institute (EPRI) in 1985. *Id.*

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[1] The power plant simulators included two-phase thermal-hydraulic calculations coupled with space- and time-dependent neutronic calculations running in real-time with valve positions, control rod insertions, and boron concentrations (for example), manipulated by operators in a control identical to the plant's actual control room.

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

7. MELCOR was the code use in the recent SOARCA project. *See, e.g.,* FirstEnergy's Motion, Attachment 41 and 42.

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

8. 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 has 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." FirstEnergy's Motion, Attachment 20, at 7-3. 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.

b) The 1991 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, at 6.18, (1991), (Attachment ("Att.") A).

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, at 1

(1996) ("Att. B"). 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," at 129 (2004) ("Att. C"). *See also* K. Vierow, Nuclear News, "Severe Accident Code Analysis – The Nuclear News Interview," at 23, (2005) ("Att. D"). 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. Att. C 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

e) A more recent paper on modeling severe accidents for the U.S. EPR design identified a number of significant 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," at 2, (2010) ("Att. E"). 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>

9. Contention 4 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 he source terms identified in FENOC'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 could include gravitational settling and (in a boiling water reactor) suppression pool scrubbing. Passive processes/systems occur/operate 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 not be 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 by operating containment sprays down to the floor.

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 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. FirstEnergy's Motion, Attachment 8 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.

10. 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.

11. 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. The probabilities estimated for induced SGTR or ISLOCA bypass events are typically quite low.

12. The Davis-Besse SAMA addresses SGRT and ISLOCA containment bypass scenarios.

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

13. Relatively limited severe accident computing abilities existed at the time NUREG-1150 [Attachment 10] 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 called the Source Term Code Package ("STCP"), which included MELCOR, and expert elicitation to estimate radionuclide releases.

14. 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 demands from MELCOR have increased, the models implemented into MELCOR have become increasingly best-estimate in nature. The increased speed (and decreased cost) of modern computers (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.

15. 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.

FirstEnergy's SAMA Source Terms Compare
Reasonably Well with NUREG-1150's Source Terms

16. Contention 4 identifies a Brookhaven National Laboratory ("BNL") report comparing radionuclide releases for the Catawba plant predicted with MAAP to releases for Sequoyah predicted with the STCP and reported in 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", contains Table 9, which summarizes the comparisons. *See* FirstEnergy's Motion, Attachment 34 at 17. The authors 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". FirstEnergy's Motion, Attachment 34 at 17. Although one may infer that the "use of 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, dated 2002, also does not discuss the limitations of comparing releases calculated with computer codes many years apart in their development. The Catawba scenario used for comparison to Sequoyah NUREG-1150 results is described as an early containment failure with no ex-vessel release. 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.

17. Predominately 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. 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." FirstEnergy's Motion, Attachment 10 at 5-13. 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 nature of the maximum release values for early containment failure in NUREG-1150, it would be expected that MAAP/Catawba would compare poorly, with MAAP/Catawba releases showing to be too small. 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 is such a table considering Sequoyah/NUREG-1150 mean and maximum values for early containment failure scenarios. As this table shows, the MAAP release fractions are actually in fair 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

18. The differences suggested as part of Contention 4 between releases calculated by MAAP and STCP as reported in NUREG-1150 is summarized by Figure 5.5, “Comparison of results for station blackout scenarios at Zion” in the cited draft NUREG-1150 report. Part of Figure 5.5 is reproduced below in Figure 1. The figure compares the release fractions for Zion station blackout scenarios experiencing early containment failure between NUREG-1150 distributions and STCP and IDCOR/MAAP point estimates. (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).

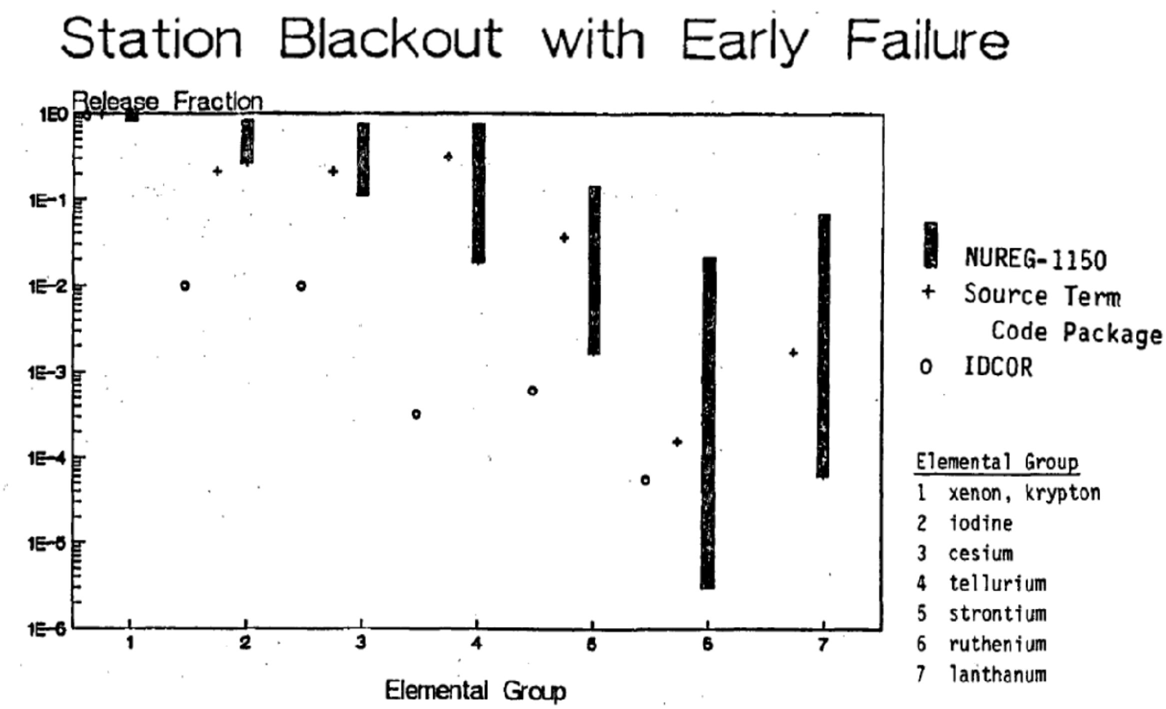

Figure 1: Comparison of NUREG-1150 and MAAP (IDCOR) Release Fractions [Draft NUREG-1150 (FirstEnergy's Motion, Attachment 9 at 5-15)].

As can be seen in the figure, MAAP release fractions, labeled IDCOR 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. (MAAP was originally developed for the IDCOR program in the early 1980s.) The draft NUREG-1150 report 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," FirstEnergy's Motion, Attachment 9 at 5-14, *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. Noteworthy is that this comparison between codes presented in the draft version of NUREG-1150 is not part of the final version of the NUREG.

19. The radionuclide releases presented in FENOC's SAMA for Davis-Besse include 34 accident sequences. The release fractions associated with these sequences were developed using MAAP4.0.6. FirstEnergy's Motion, Attachment 5, at page 1 of Attachment 1.

MAAP4 is the latest released version of the MAAP code. The 5%, 95%, median, and mean values of fractional releases of key radionuclide groups are considering 32 of the 34 accident sequences presented in the SAMA. Davis-Besse Environmental Report, Appendix E, Table E.3-13 at E-87 – E-93 (Agency Document Access & Management System ("ADAMS") Accession No. ML102450568). "Source Term Comparison – Davis-Besse SAMA Versus NUREG-1150," (Figure 2, below) compares the source terms from the Davis-Besse SAMA analysis and NUREG-1150 source terms.[2] The NUREG-1150 source terms are taken from the accident sequences involving containment bypass at Surry and containment failures at Sequoyah and Zion. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," at 3-15, 5-13, 7-10 (Dec. 1990). The operating reactors at Surry and Sequoyah and the retired reactors at Zion are the same type as the reactor at Davis-Besse, *i.e.*, all of these reactors are pressurized water reactors. The comparisons then, in Figure 2, 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. While median and 5% values for Davis-Besse are skewed low because of the inclusion of accident sequences with small, late containment failures, it is noteworthy that the mean and 95% of the release fractions are comparable, and in some cases, larger than those for the NUREG-1150 plants. The figure identifies that the release fractions for key radionuclide groups calculated with MAAP4 and

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[2] Sequences for source term ST91 and ST92 of the Davis Besse SAMA analysis have been excluded from this comparison because they do not involve either containment failure or containment bypass. The designation "ST" here refers to source term while the digit "9" refers to a particular class of accident. Subsequent digits are specific accident identifiers within the class.

reported in FENOC's SAMA for Davis-Besse are comparable to the release fractions presented in NUREG-1150 for the same type of reactor as the Davis-Besse reactor.

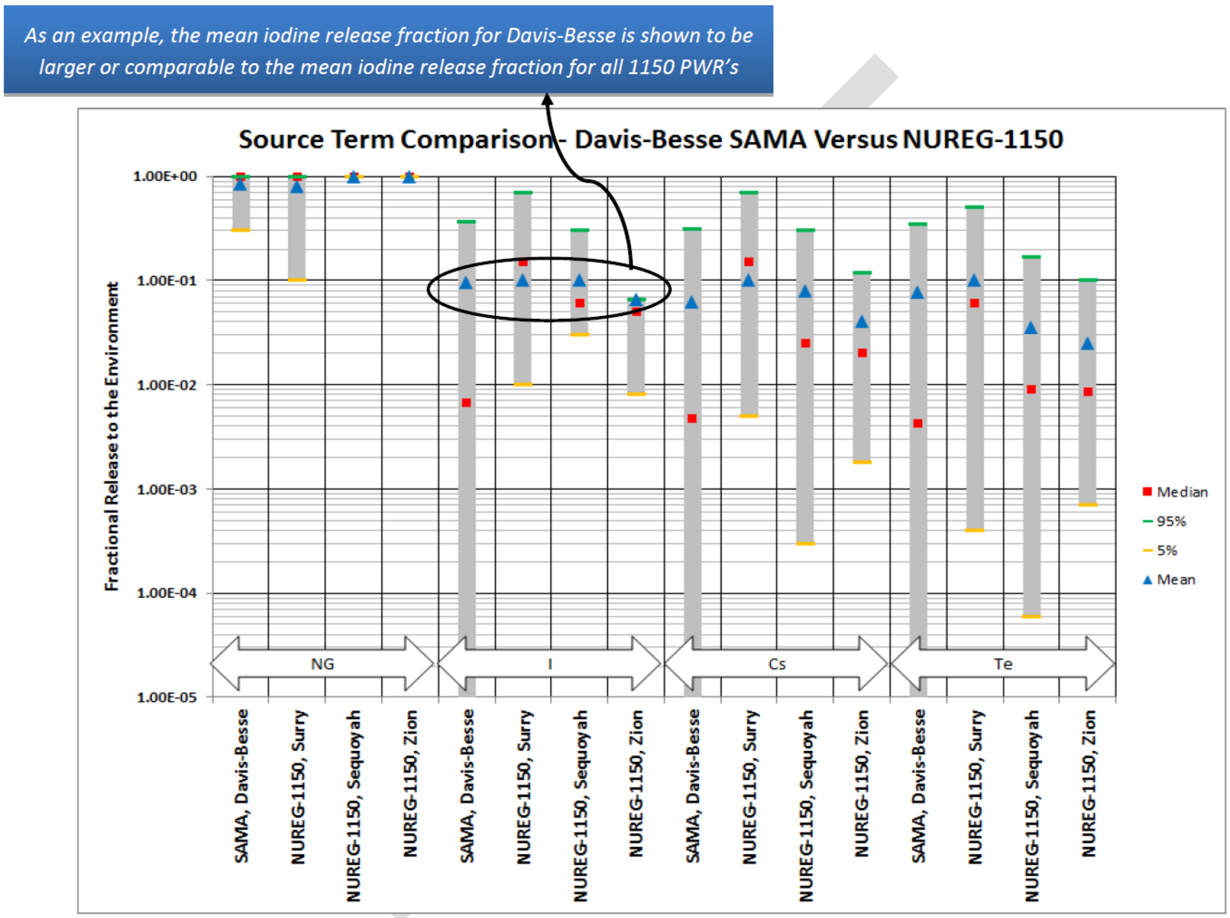

Figure 2: Source Term Comparison – Davis-Besse SAMA Versus NUREG-1150[3].

[3] In FENOC's MAAP analysis, cesium and tellurium are both considered to be released in two different radionuclide groups. Cesium is released as both CsI and CsOH. SOARCA Volume II, Table 5-15, "LHSI Piping Aerosol Capture in the ISLOCA Separate-effects Calculation and Associated DFs" identifies that ~10% of the available cesium in a reactor core would be released (from the core) combined with iodine in the form of CsI, so an appropriate weighting of the reported CsI and CsOH releases was made in determining a combined release fraction for Cs for inclusion in Figure 2. Tellurium is considered to be released both elementally and as TeO2 in the FENOC analysis. However, because the release fraction for elemental Te is substantially lower than TeO2 in all but 2 of the 34 considered accident sequences, only release fractions for TeO2 are presented in Figure 2 for conservatism.

FirstEnergy's SAMA Source Terms are Consistent with SOARCA Source Terms

20. A meaningful comparison with respect to the reasonableness of the SAMA/Davis-Besse 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. FirstEnergy's Motion, Attachments 41 and 42. This SOARCA project used insights gathered from previous accident studies including NUREG-1150 and WASH-1400. It also profited from advancements in computer modeling and our increased understanding of the physical and chemical processes occurring 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 in-vessel steam explosions and molten core concrete interactions. FirstEnergy's Motion, Attachment 42, at 2. This was accomplished for two nuclear power plants - Peach Bottom (a BWR) and Surry (a PWR). The calculations included plant specific data wherever possible.

21. In Table 2 below, ST78, a long term station black out ("LTSBO") for Davis-Besse is compared to the LTSBO and short term station blackout ("STSBO") from the SOARCA analysis as reported for Surry Power Station in NUREG/CR-7110. As the Surry LTSBO credits 8 hours of battery life while the Davis-Besse LTSBO only credits 2, the source term for the Davis-Besse LTSBO should be relatively comparable to both SOARCA sequences. The source terms in SOARCA were evaluated using MELCOR version 1.8.6. Both Davis-Besse and Surry are pressurized water reactors with similar large, dry containments. The releases of radionuclides into the environment result from a long-term over-pressurization of containment. The comparison then, between the MAAP generated source terms for Davis-Besse and the MELCOR generated source terms for Surry, is more correctly controlled for the type of accident, reactor, and changes in accident modeling over the last two decades. As shown in Table 2,

the MAAP generated source terms are very consistent with the SOARCA source terms for iodine and cesium, the two radionuclides typically observed to be of highest importance in consequence analyses. The only class that is significantly smaller in SOARCA than Davis-Besse's SAMA is the Barium/Strontium class that is shown in a SOARCA Volume II class importance study found in Tables 7-12 to 7-20 to have essentially no contribution to risk for either the emergency or long-term phases of the accident.

Table 2: Davis-Besse SAMA and NUREG/CR-7110 (SOARCA) Source Terms.

Radionuclide Group	FENOC Davis-Besse, ST78 [Davis-Besse Environmental Report, Appendix E, Table E.3-13]	Unmitigated SOARCA Scenarios	
		SOARCA Surry MELCOR LTSBO [NUREG/CR-7110, Volume 2, Rev. 0, Figure 5-8]	SOARCA Surry MELCOR STSBO [NUREG/CR-7110, Volume 2, Rev. 0, Figure 5-36]
NG	9.41E-01	8.0E-01	9.2E-01
I	1.43E-02	6.0E-03	1.0E-02
Cs	1.91E-03	8.0E-04	4.0E-03
Te	9.70E-03	2.3E-02	1.7E-02
Sr	1.58E-05	included in Ba	included in Ba
Ru	-	1.5E-05	3.0E-05
La	1.43E-05	< 1.0E-06	2.0E-06
Ce	4.93E-05	2.0E-05	3.0E-05
Ba	8.15E-06	8.0E-04	1.2E-03
Sb (Cd)	1.78E-02	6.0E-04	7.5E-03

Executed in accordance with 10 C.F.R. § 2.304(d).

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