September 14, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

THE STAFF'S RESPONSE TO FIRSTENERGY'S STATEMENT OF MATERIAL FACTS

I. REGULATORY AND TECHNICAL BACKGROUND

A. Submittal of the Original and Revised Davis-Besse SAMA Analyses

1. On August 27, 2010, FirstEnergy submitted a license renewal application ("LRA"), requesting that the NRC renew the operating license for Davis-Besse Nuclear Power Station, Unit 1 ("Davis-Besse") for 20 years (*i.e.*, through April 22, 2037). Letter from Barry S. Allen, FirstEnergy, to NRC Document Control Desk, "License Renewal Application and Ohio Coastal Zone Management Program Consistency Certification" (ADAMS Accession No. ML102450572 (package)).

Response: Admitted.

2. As required by 10 C.F.R. § 51.53(c)(3)(ii)(L), FirstEnergy prepared an analysis of severe accident mitigation alternatives ("SAMAs") as part of its LRA. The Davis-Besse SAMA analysis is documented in Section 4.20 and Attachment E of the Environmental Report ("ER"). ER § 4.20 (Severe Accident Mitigation Alternatives) & Attach. E (Severe Accident Mitigation Alternatives Analysis).

Response: Admitted.

3. On July 16, 2012, FirstEnergy submitted to the NRC certain revisions to the SAMA analysis documented in ER Section 4.20 and ER Attachment E. Among other things, the revised SAMA analysis accounts for FirstEnergy's use of updated Modular Accident Analysis Program ("MAAP") code runs that, consistent with MAAP User's Group recommendations, are based on core inventory radionuclide masses instead of radionuclide activities. Letter from John C. Dominy, Director, Site Maintenance, FirstEnergy, to NRC Document Control Desk, "Correction of Errors in the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 29" (July 16, 2012) ("Revised SAMA Analysis Submittal") (Attach. 5).

Response: Admitted.

B. SAMA Analysis Requirements and Guidance

4. SAMAs, by definition, pertain to severe accidents; *i.e.*, accidents in which substantial damage is done to the reactor core, whether or not there are serious offsite consequences. Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants, 50 Fed. Reg. 32,138 (Aug. 8, 1985) (Attach. 11); Joint Declaration of Kevin O'Kula and Grant Teagarden in Support of FirstEnergy's Motion for Summary Disposition of Contention 4 (SAMA Analysis Source Terms) at ¶ 15 (July 26, 2012) ("Joint Decl.") (Attach. 2).

Response: Admitted.

5. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Vol. 1, at 5-1 to 5-20 (May 1996) ("GEIS") (Attach. 12), provides an evaluation of severe accident impacts that applies to all U.S. nuclear power plants. Based on the GEIS evaluation of severe accident impacts, 10 C.F.R. Part 51 concludes that the "[t]he probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water, and societal and economic impacts from severe accidents are small for all plants." 10 C.F.R. Part 51, Subpart A, App. B, Table B-1 (Postulated Accidents; Severe accidents); Joint Decl. ¶ 15.

Response: Admitted.

6. 10 C.F.R. Part 51 states that if the Staff has not previously considered SAMAs for a license renewal applicant's plant in an EIS or in an environmental assessment, then the applicant must complete an evaluation of alternatives to mitigate severe accidents. 10 C.F.R. § 51.53(c)(3)(ii)(L); *see also* 10 C.F.R. Part 51, Subpart A, App. B, Table B-1; Joint Decl. ¶ 16.

Response: Admitted.

7. SAMA analysis is a site-specific, probability-weighted assessment of the benefits and costs of mitigation alternatives that might be used to reduce the risks (frequencies or consequences or both) of potential nuclear power plant severe accidents. It estimates annual average impacts for the entire 50-mile radius region surrounding a nuclear power plant. Joint Decl. ¶ 17.

Response: Admitted.

8. The Nuclear Energy Institute ("NEI") has issued a guidance document, NEI 05-01, Revision A, to assist NRC license renewal applicants in preparing SAMA analyses. NEI 05-01, Rev. A, "Severe Accident Mitigation Alternatives (SAMA) Analysis, Guidance Document," at i (Nov. 2005) ("NEI 05-01") (Attach. 14); Joint Decl. ¶ 18.

Response: Admitted.

9. The Staff has approved and recommended the use of NEI 05-01 by license renewal applicants. "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses" (Aug. 2007) (Attach. 15); Joint Decl. ¶ 18.

Response: Admitted.

10. NEI 05-01 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 at 1 (Attach. 15); Joint Decl. ¶ 18.

Response: Admitted.

11. A SAMA analysis identifies potential changes to a nuclear power plant, or its operations, that could reduce the already low risk (frequency and/or the consequence) of a severe accident for which the benefit of implementing the change may outweigh the cost of implementation. Changes to the plant that could reduce the risk of a severe accident include plant modifications or operational changes (*e.g.*, improved procedures, augmented training of control room and plant personnel). NEI 05-01 at 1, 23 (Attach. 15); Joint Decl. ¶ 18.

Response: Admitted.

12. A SAMA analysis, broadly speaking, involves four major sequential steps: (1) using probabilistic risk assessments (“PRAs”) and other risk studies to characterize the overall plant-specific severe accident risk by identifying and characterizing the leading contributors to core damage frequency (“CDF”) and offsite risk based on a plant-specific risk study; (2) identifying potential plant improvements (*i.e.*, SAMA candidates) that could reduce the risk of a severe accident; (3) quantifying the risk-reduction potential and the implementation cost for each SAMA candidate; and (4) determining whether implementation of the SAMA candidates may be cost-effective. NEI 05-01 at 2 (Attach. 15); Joint Decl. ¶ 19.

Response: Admitted.

13. 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. This methodology for the overall SAMA analysis approach is based on methods found in NRC guidance. NEI 05-01 at 28 (Attach. 15); NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” Rev. 4 (Jan. 1997) (Attach. 16); NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4” (August 2004) (Attach 17); Joint Decl. ¶ 19.

Response: Admitted.

C. Use of Plant-Specific Probabilistic Risk Assessment (“PRA”) in SAMA Analyses

14. The basis for a SAMA analysis conducted for a U.S. nuclear power plant is a sequential, three-level probabilistic risk assessment or PRA. All three PRA levels are required to perform a SAMA analysis. Joint Decl. ¶ 21.

Response: Admitted to the extent that many SAMA analyses conducted for U.S. nuclear power plants are based on a Level 3 PRA.

15. The Level 1 PRA establishes the plant damage states and frequency of reactor core damage frequency or CDF. Joint Decl. ¶¶ 47-48.

Response: Admitted.

16. The Level 2 PRA determines different accident progressions and a set of radioactive release conditions from the containment that are assigned to 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). Joint Decl. ¶¶ 47-48.

Response: Admitted.

17. The Level PRA 3 combines the Level 2 PRA results with site-specific parameters (*e.g.*, population distribution, meteorological data, land use data, and economic data) for the Level 3 PRA to calculate offsite public dose and offsite economic consequences of those releases to the environment. Joint Decl. ¶¶ 47-48.

Response: Admitted.

D. Use of the MAAP Code to Develop Source Term Inputs to the MACCS2 Code

18. Various computer codes are used in support of a SAMA analysis. These codes include, among others, the MELCOR Accident Consequence Code System Version 2 ("MACCS2") and the Modular Accident Analysis Progression ("MAAP") codes. Joint Decl. ¶ 20.

Response: Admitted.

19. As part of the Level 3 PRA, MACCS2 calculates the radiological doses, health effects, and economic consequences that result from postulated releases of radioactive materials to the atmosphere. MACCS2 performs these calculations based on plant- and site-specific, regional, and standardized regulatory inputs. NEI 05-01 at 13 (Attach. 15); Joint Decl. ¶ 20.

Response: Admitted.

20. MACCS2 executes three modules (ATMOS, EARLY, and CHRONC) in sequence to calculate consequence values necessary for a SAMA analysis, and models atmospheric transport and dispersion and subsequent deposition in a radial-polar grid (*i.e.*, 16 compass sectors over a 50-mile radius). NUREG/CR-6613, "Code Manual for MACCS2: User's Guide," Vol. 1 at 2-1 to 2-3 (May 1998) (Attach. 19); Joint Decl. ¶ 23.

Response: Admitted.

21. ATMOS, in particular, performs calculations pertaining to atmospheric transport, dispersion, and deposition of radioactive material, and to radioactive decay of that material both before and after its release into the atmosphere. NUREG/CR-6613 at 2-2 (Attach. 19); Joint Decl. ¶ 23.

Response: Admitted.

22. The ATMOS input parameters include, among other things, plant-specific source term information, including the core inventory (*i.e.*, the amount of each radionuclide present in the reactor core at the time of accident initiation), and the physical, chemical, and radiological composition of an atmospheric release. NUREG/CR-6613 at 5-23 to 5-28 (Attach. 19); Joint Decl. ¶ 24.

Response: Admitted.

23. The source term is the amount and isotopic composition of material released (or postulated to be released) from the core of a nuclear power reactor during an accident. NUREG- 1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Vol. 1, 2-3 tbl. 2.1 (Dec. 1990) (Attach. 10); Joint Decl. ¶ 24.

Response: Admitted.

24. The source term may refer to radionuclide groups in the reactor core inventory at the start of an accident that are released to the containment (*i.e.*, the containment source term) or that are released to the environment (*i.e.*, the environmental source term). Joint Decl. ¶ 25.

Response: Admitted.

25. An environmental source term describes the physical, chemical, and radiological composition of an atmospheric release. The environmental source term description includes: (1) the quantity of each important radionuclide released into the atmosphere, (2) the initial time of the release relative to the start of the accident, (3) the duration of the release, (4) the elevation of the release, (5) the sensible heat released, and (6) the particle size of the released material. Joint Decl. ¶¶ 24-25.

Response: Admitted.

26. The release fraction, *i.e.*, the fraction of the total activity of the fission products released to the environment during the accident, is one component of the source term. It defines the portion of the radionuclide inventory, by radionuclide group, in the reactor core at the start of an accident that is ultimately released to the environment. Joint Decl. ¶ 26.

Response: Admitted.

27. 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 that are intended to mitigate releases. Joint Decl. ¶ 27.

Response: Admitted.

28. The evaluation of source terms for a SAMA analysis requires use of 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. Joint Decl. ¶ 27.

Response: Admitted.

29. Source terms commonly are estimated in the U.S. using one of two computer codes: the MAAP code and the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code. Joint Decl. ¶ 28.

Response: Admitted.

30. FirstEnergy used MAAP4 (Version 4.0.6) in support of the Davis-Besse SAMA analysis. Joint Decl. ¶ 28.

Response: Admitted.

31. MAAP simulates the dominant thermal-hydraulic and fission product phenomena in both the primary and containment systems of pressurized water reactors ("PWRs") and boiling water reactors ("BWRs"). MAAP thus 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. MAAP4 also addresses important engineered safety systems and allows a user to model operator interventions. EPRI Report 1020236, "MAAP4 Applications Guidance: Desktop Reference for Using MAAP4 Software, Revision 2" at 2-2 to 2-6 (2010) ("MAAP4 Applications Guidance") (Attach. 20); Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program)* (Attach. 21); Joint Decl. ¶ 29.

Response: Admitted.

E. Use of Plant-Specific PRAs and MAAP4 Code in the Davis-Besse SAMA Analysis

32. 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. For its SAMA analysis, FirstEnergy used the results from updated Davis-Besse Level 1 and Level 2 PRA models as input to a Level 3 PRA model developed specifically to support the consequence quantification needed for the Davis-Besse SAMA analysis. Joint Decl. ¶¶ 47-48.

Response: Admitted.

33. The Level 1 PRA included initiating event and core damage sequence analyses and yielded a set of plant damage states and associated frequencies. Joint Decl. ¶ 48.

Response: Admitted.

34. The Level 2 PRA used containment event tree ("CET") and deterministic source term modeling to provide a set of 34 release categories, each of which has a characteristic frequency and unique timing and fission product magnitude characteristics, depicting the release to the environment. The release categories are defined in terms of similar properties, each with a frequency-weighted mean source term. Joint Decl. ¶ 48.

Response: Admitted.

35. The Level 2 PRA-defined release categories were characterized using the MAAP4 code. The MAAP4 calculations provided a deterministic analysis of the plant under postulated severe accident conditions for a variety of initiating events, and included the influence of operator actions and safety system actuation on accident sequence progression. The MAAP4 calculations predicted the integrated response of the reactor core, primary system, steam generators, and primary containment building. Results included 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 assessments. Joint Decl. ¶ 51.

Response: Admitted.

36. Six MACCS2 input parameters came from the output of the Davis-Besse MAAP4 runs: (1) the time after accident initiation that the offsite alarm is initiated (OALARM), (2) the heat content of release segment (PLHEAT), (3) the height of the plume segment at release (PLHITE), (4) the duration of release (PLUDUR), (5) the time of release for each plume (PDELAY), and (6) the release fraction for each radioisotopic group (RELFRC). Joint Decl. ¶ 52.

Response: Admitted to the extent that the six MACCS2 input parameters came from the output of the Davis-Besse MAAP4 runs. FirstEnergy's License Renewal Application for Davis-Besse Nuclear Power Station, Unit 1, Appendix E, Davis-Besse Environmental Report, Attachment E at E-87. The Staff would note that the definition of OALARM as the time after accident initiation that the offsite alarm is initiated differs from the MACCS2 definition of OALARM, which is the time at which notification to the public has been initiated. FirstEnergy's definition refers to when the offsite response agencies are notified rather than the public. This difference in definitions, however, would not impact evacuation timing sufficiently to affect the conclusions of the SAMA analysis.

37. The core inventory for the Davis-Besse Level 3 PRA was obtained from plant specific calculations performed using the ORIGEN-2 code. For conservatism, the Davis-Besse core inventory was evaluated at the 24-month end-of-cycle for all 177 fuel assemblies. This assumption is conservative because at the end-of-cycle, the radionuclide quantities in the core would be at their peak levels for the 24-month cycle. In total, 58 radionuclides were evaluated in the MACCS2 reactor core inventory for Davis-Besse and are represented in nine fission product groups. Joint Decl. ¶ 52.

Response: Admitted.

38. The release category frequencies and characterizations developed using Level 2 PRA information and MAAP4 were used as inputs to the Level 3 PRA. The Level 2 PRA results were then combined with Davis-Besse site-specific parameters (*e.g.*, population distribution, meteorological data, land use data, and economic data) for the Level 3 PRA to estimate the Davis-Besse Plant offsite population dose risk (in units of person-rem/year) and offsite economic cost risk (in units of dollars/year), the key risk metrics in a SAMA analysis. Joint Decl. ¶¶ 47-48.

Response: Admitted.

II. ISSUES RAISED IN INTERVENORS' CONTENTION 4

39. Contention 4 alleges that FirstEnergy's SAMA analysis "underestimates the true cost of a severe accident at Davis-Besse." Beyond Nuclear, Citizens Environment Alliance of Southwestern Ontario, Don't Waste Michigan, and the Green Party of Ohio Request for Public Hearing and Petition for Leave to Intervene at 100, 104, 108 (Dec. 27, 2010) ("Petition") (Errata filed Jan. 5, 2011); *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), LBP-11-13, slip op. at 50-54, 64 (Apr. 26, 2011), *aff'd in part and rev'd in part*, *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), CLI-12-08, (Mar. 27, 2012).

Response: Admitted.

40. Contention 4 further alleges that FirstEnergy has minimized the potential amount of radioactive material released in a severe accident by using MAAP-derived source terms that are smaller for key radionuclides than the release fractions specified in NRC guidance. Petition at

108. Intervenors make three principal claims in support of their contention (which, for clarity and ease of reference, FirstEnergy refers to as Bases 1, 2 and 3):

> 1. The MAAP code "has not been validated by the NRC." *Id.* (Basis 1)
>
> 2. 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. *Id.* at 108, 112, 114 (Basis 2)
>
> 3. It previously has been observed that MAAP generates lower release fractions than those derived and used by NRC in other severe accident studies. *Id*. at 113. (Basis 3)

Response: Admitted.

III. UNDISPUTED FACTS SHOWING LACK OF GENUINE MATERIAL DISPUTE

A. Validation of the MAAP Code (Basis 1 of Contention 4)

41. 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. Fauske & Associates, LLC is the current maintenance contractor for the MAAP code. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶¶ 31, 33;

Response: Admitted.

42. Starting in the late 1980s, the MAAP3B version 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"). MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

Response: Admitted.

43. MAAP3B 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. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

Response: Admitted.

44. Several organizations, including EPRI and the DOE, sponsored the development of MAAP4. As part of the development process, a committee of independent experts reviewed MAAP4 to ensure that it is state-of-the-art 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. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

Response: Admitted to the extent that the MAAP4 Applications Guidance states that several organizations, including the Electric Power Research Institute ("EPRI") and the Department of

Energy ("DOE"), sponsored the development of MAAP4. The Staff would note that the guidance states that as part of the development process, a committee of independent experts reviewed MAAP4 to ensure that it is state-of-the-art and applicable for accident management evaluations. Affidavit of Kyle W. Ross Concerning the Motion for Summary Disposition of Contention 4 ("Ross' Affidavit") at ¶ 3 -8.

45. MAAP and its successor versions, including MAAP4, were developed in accordance 10 C.F.R. Part 50, Appendix B and International Organization for Standardization ("ISO") 9001 quality assurance requirements. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 33.

Response: Admitted to the extent that the MAAP4 Applications Guidance states that the code was developed and is maintained under Fauske & Associates, L.L.C., quality assurance ("QA") program, and the Applications Guidance states that the QA program is in compliance with 10 C.F.R. Part 50, Appendix B, and ISO 9001 quality assurance requirements. *See* Ross' Affidavit at ¶ 3 -8.

46. EPRI has identified the MAAP code (versions 4.0.5 and later) as a "consensus model" suitable for use in evaluation of PRA success criteria. EPRI Report 1013492, "Probabilistic Risk Assessment Compendium of Candidate Consensus Models" at 2-3 (2006) (Attach. 23); Joint Decl. ¶ 33.

Response: Admitted.

47. MAAP4 has been benchmarked against numerous severe accident studies and the Three Mile Island Unit-2 ("TMI-2") core melt accident. The benchmarking of MAAP is documented in Section 7 (MAAP Benchmarks) and Appendix F (Summaries of MAAP Benchmarks) of EPRI's MAAP4 Applications Guidance and also in the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations report "Recent Developments in Level 2 PSA and Severe Accident Management." Committee on the Safety of Nuclear Installations, Nuclear Energy Agency, Organization for Economic Co-operation and Development, NEA/CSNI/R(2007)16, at 36 (Nov. 2007) (Attach. 24); Joint Decl. ¶ 34.

Response: Admitted.

48. EPRI licenses MAAP4 to a wide array of entities, such as utilities, vendors, and research organizations, including universities. The majority of MAAP4 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. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 32.

Response: Admitted.

49. In general, a computer code in itself is not validated by the NRC, but its use for specific applications may be found acceptable for estimating certain phenomena within certain defined regimes. For example, a computer code may be used to predict the coupled thermal hydraulic fission product transport response of reactor systems to severe accident events. If inputs and assumptions are appropriate for the computer model, and sources of uncertainty are understood, then the results of that code may be accepted by a reviewer or regulator for purposes of the application. Joint Decl. ¶ 30; 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) (Attach. 22).

Response: Admitted to the extent that in general, a computer code in itself is not validated by the NRC, but its use for specific applications may be found acceptable for estimating certain phenomena within certain defined regimes. **Denied** to the extent that the statement suggests that computer codes cannot be validated by the NRC, when in fact some codes have been validated. For example, the NRC published NUREG-1795 in August 2007, "FAVOR Code Versions 2.4 and 3.1 Verification and Validation Summary Report." This NUREG describes a summary of verification and validation (V&V) of the probabilistic fracture mechanics models in the Fracture Analysis of Vessels-Oak Ridge (FAVOR) code [NUREG-1795 abstract]. *See* Ross' Affidavit at ¶ 3 - 8.

50. The MAAP code has been used by nuclear plant licensees and other entities to predict the responses of nuclear power plants during postulated severe accidents and is the most commonly used code in the U.S. for such purposes. Joint Decl. ¶ 35; Kenneth D. Kok, Ed., *Nuclear Engineering Handbook* at 539 (2009) (Attach. 25).

Response: Admitted.

51. The use of MAAP and its successor versions in IPEs and subsequent PRA applications, including those related to advanced reactor standard design certification applications, has been accepted by the NRC Staff. *See*, *e.g.*, NUREG-1503, "Final Safety Evaluation Report Related to Certification of the ABWR Reactor Design," Vol. 1 at 19-53 to 19-55 (July 1994) (Attach. 47); NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Vol. 1 at 19-61 (Sept. 2004) (Attach. 48); Joint Decl. ¶ 35.

Response: Admitted.

52. Numerous NRC license renewal applicants have used the MAAP code to support NRC-approved SAMA analyses, including 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) (Attach. 26); 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) (Attach. 27); Joint Decl. ¶ 36.

Response: Admitted.

B. Differences in MAAP4-Generated and NUREG-1465 Source Terms (Basis 2)

53. The reactor accident source term generally serves two purposes in the U.S. nuclear regulatory process. The first purpose is for licensing, safety analysis, and regulatory compliance, particularly in meeting 10 C.F.R. Part 100 siting requirements. For this purpose, a source term representing the release of radioactive materials into the reactor containment is used to assess the adequacy of reactor containments and engineered safety systems, as well as the environmental qualification of equipment inside the containment that must function following a design-basis accident. This source term also is used to show that dose criteria at the exclusion area boundary are met by assuming the maximum allowable design leak rate from the

containment. NUREG-1465 source terms are applicable for the first purpose described above. 10 C.F.R. § 50.34(a)(1)(ii)(D) & 10 C.F.R. § 100.11; Joint Decl. ¶ 38.

Response: Admitted.

54. The second purpose for which a reactor accident source term is developed is to simulate a release of radioactive material to the environment (*i.e.*, outside containment) following a hypothetical reactor accident. This second source term may be used as input to radionuclide dispersal and accident consequence models (*e.g.*, MACCS2) that are used for Level 3 PRA and SAMA evaluations. Joint Decl. ¶ 39.

Response: Admitted.

55. NUREG-1465 states that it was developed "to define a revised accident source term for regulatory application for future LWRs" and "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, Accident Source Terms for Light-Water Nuclear Power Plants, at vii, 3 (Feb. 1995) (Attach. 8); Joint Decl. ¶ 41.

Response: Admitted.

56. NUREG-1465 provides generic, default source terms, whereas PRA and SAMA analyses are intended to be best-estimate engineering evaluations that seek to maximize the use of plant-specific data. Joint Decl. ¶ 45.

Response: Admitted.

57. NUREG-1465 assumes a "release resulting from 'substantial meltdown' of the core into the containment . . . [and assumes] that the containment remains intact but leaks at its maximum allowable leak rate." NUREG-1465 at vii, 1, 3 (Attach. 8); Joint Decl. ¶ 42.

Response: Admitted to the extent that NUREG-1465 assumes a substantial release into containment for the purpose of reactor siting criteria. Ross' Affidavit at ¶ 9 - 12

58. NUREG-1465 discusses in-containment fission product removal mechanisms, such as engineered safety features ("ESFs"), but does not provide numerical estimates of source terms that account for the effects of such mechanisms (*e.g.*, containment sprays, aerosol deposition). NUREG-1465 directs the reader to use appropriate methodologies in crediting fission product removal or reduction within containment. NUREG-1465 at 17-21 (Attach. 8); Joint Decl. ¶ 44.

Response: Admitted.

59. The NUREG-1465 source term solely represents radionuclides released into the containment. It does not specify the source term released from containment into the environment following a severe accident, and it does not take into account the reductions of the source term that would occur in those circumstances. Joint Decl. ¶¶ 42-43.

Response: Admitted.

60. The MAAP code produces results that are different from, and generally smaller than, the release fractions specified in NUREG-1465, because MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident.

MAAP models and credits fission product removal mechanisms such as containment ESFs (*e.g.*, containment air coolers, containment spray) and natural depletion processes (*e.g.*, aerosol deposition and containment holdup). Joint Decl. ¶¶ 43-44.

Response: Admitted.

C. Differences in MAAP4-Generated Source Terms and Source Terms Discussed in Other Historical Studies Cited by Intervenors (Basis 3)

61. Contention 4 cites two documents containing historical comparisons between release fractions developed using earlier versions of the MAAP code and release fractions developed using other codes. The first document is a 1987 draft of the NUREG-1150 severe accident risk study that, in examining accident risk at Zion Nuclear Station, stated that “the MAAP estimates for environmental release fractions were significantly smaller” than those obtained with the Source Term Code Package [footnote omitted] (“STCP”) computer code (the primary code used in the NUREG-1150 study). Petition at 114 (citing Office of Nuclear Regulatory Research, “Draft for Comment, Reactor Risk Reference Document,” NUREG-1150, Vol. 1, at 5-14 (Feb. 1987) (Attach. 9)). This statement does not appear in the final December 1990 version of NUREG-1150 (Attach. 10). Joint Decl. ¶¶ 58, 59.

Response: Admitted.

62. The second is a 2002 Brookhaven National Laboratory (“BNL”) report reviewing combustible gas control availability at ice condenser and Mark III containment plants. The BNL study 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). 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”) (Attach. 34); Joint Decl. ¶ 58.

Response: Admitted.

63. 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.” BNL report at 17 (Attach. 34); Joint Decl. ¶ 58.

Response: Admitted.

64. Severe accident source term estimates depend on many plant-specific design features, operational practices, and the technical accuracy provided by computer code models used for source term quantification Joint Decl. ¶ 59.

Response: Admitted.

65. The NUREG-1150 study (issued as a final report in 1990) was completed over 20 years ago and involved an assessment of the risks from severe accidents at five commercial nuclear power plants in the United States. Davis-Besse was not one of those five plants. Joint Decl. ¶ 59; NUREG-1150 (Attach. 10).

Response: Admitted.

66. The IDCOR (MAAP) to NUREG-1150 (STCP) comparison of Zion results cited by Intervenors was only one of four sets of plant results compared in the February 1987 draft of NUREG-1150. 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. The report and its underlying technical analyses were substantially modified in two rounds of review before the report's final publication in December 1990. One of the changes included deleting the specific discussion comparing the MAAP and STCP results for Zion, such that the comparison cited by Intervenors in Contention 4 was not incorporated into the final December 1990 version of NUREG-1150. Joint Decl. ¶ 59; Draft NUREG-1150, Vol. 1 at 5-14 (Attach. 9); NUREG-1150, Vol. 1 (Attach. 10).

Response: Admitted. In addition, the draft NUREG-1150 also contained MAAP comparisons for Sequoyah, Peach Bottom and Grand Gulf. Draft NUREG-1150, Vol. 1 (FirstEnergy's Attach. 9), at 5-17, 5-19, 5-23, 5-27. Ross' Affidavit at ¶ 19.

67. The final NUREG-1150 report states that 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 (Attach. 10); Joint Decl. ¶ 60. 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 estimated source term. NUREG-1150, Vol. 3, App. D at D-17 (Attach. 10); Joint Decl. ¶ 60.

Response: Admitted

68. The BNL report's 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). The comparison of MAAP-based source terms with those estimated over ten years earlier with STCP (a simpler code) and an earlier version of MELCOR—and for different plants—is expected to show differences. Joint Decl. ¶ 63.

Response: Admitted.

69. Also, the BNL report 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. *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,'" Attach. 1, 3 (Oct. 8, 2002) (Attach. 38); Joint Decl. ¶ 63.

Response: Admitted.

70. Since the issuance of NUREG-1150, better understanding of heat transfer and removal from the reactor pressure vessel during severe accident sequences, improved insights on iodine, cesium, and other fission product groups' chemistry from contemporary research, and modeling improvements suggest that the early containment failure releases would be smaller than previously estimated. Joint Decl. ¶ 64.

Response: Admitted. For example, Sandia recently published the State-of-the-Art Reactor Consequence Analyses (SOARCA) project (NUREG/CR 7110 Volumes 1 and 2) which was a project commissioned by the NRC to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents. Ross' Affidavit at ¶ 20 - 21.

71. In its 2002 Supplemental Environmental Impact Statement for Catawba license renewal, the NRC Staff compared similar sequences between the two studies—NUREG-1150 and Revision 2b of the Catawba PRA, [footnote omitted] which included the plant's IPE models—and concluded there was "reasonable agreement" for the closest corresponding release scenarios. NUREG-1437, Supp. 9, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Catawba Nuclear Station, Units 1 and 2 – Final Report at 5-9 to 5-10 (Dec. 2002) (Attach. 37); Joint Decl. ¶ 62.

Response: Admitted.

72. The state of the art for source term analysis has significantly improved since the NUREG-1150 study was performed in the 1980s. For example, in 2006, the NRC initiated the State-of-the-Art Reactor Consequence Analyses ("SOARCA") project 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 such as WASH-1400, the 1982 Siting Study, and NUREG-1150. 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. Joint Decl. ¶ 65; NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, Draft Report for Comment" (Jan. 2012) ("Draft NUREG-1935") (Attach. 39).

Response: Admitted.

73. 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), and the NUREG-1150 (1990) study. The Staff analyzed one plant unit at each site. Joint Decl. ¶ 66; Draft NUREG-1935 (Attach. 39).

Response: Admitted.

74. 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. Joint Decl. ¶ 68; Draft NUREG-1935 (Attach. 39).

Response: Admitted.

75. In January 2012, the NRC published the results of its SOARCA assessment, including plant-specific reports for Peach Bottom and Surry. Among the findings, the NRC found that, in addition to delayed radiological releases, the magnitude of the radionuclide release, especially with respect to the key radioisotopic (iodine and cesium) groups, is much smaller

than estimated in prior studies. Joint Decl. ¶ 69; Draft NUREG-1935 (Attach. 39); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 1: Peach Bottom Integrated Analysis" (Jan. 2012) (Attach. 40); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 2: Surry Integrated Analysis" (Jan. 2012) (Attach. 42); NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences" (Jan. 2012) (Attach. 43).

Response: Admitted.

76. Both MAAP and MELCOR were used soon after of 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 in March 2011. 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) (Attach. 45). Sandia applied MELCOR in modeling the Station Blackout sequence for the NRC in support of the Japanese Government. Joint Decl. ¶ 72.

Response: Admitted to the extent that MAAP and MELCOR were used after the March 2011 Fukushima Dai-ichi nuclear power plant accident in Japan. Tokyo Electric Power Company has used MAAP to inform its understanding of the accident progression in Units 1-3 during the earthquake and subsequent tsunami event in March 2011 and Sandia applied MELCOR in modeling Fukushima-tailored station blackout sequences for the NRC and the Department of Energy in support of the Japanese Government.

Respectfully submitted,

Signed (electronically) by

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