UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		September 14, 2012
	)	

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INTERVENORS' REPLY IN OPPOSITION TO 'FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4 (SAMA ANALYSIS - SOURCE TERMS)'

Now come Beyond Nuclear, Citizens Environment Alliance of Southwestern Ontario (CEA), Don't Waste Michigan, and the Green Party of Ohio (collectively, "Intervenors"), by and through counsel, and reply in opposition to "FirstEnergy's Motion for Summary Disposition of Contention 4 (SAMA Analysis - Source Terms)."

A. Summary of Argument

After making significant modifications to its SAMA analysis in recent months (despite sweeping denials at the contention admissibility stage of any defect), FENOC still argues for the fundamental reliability of its now-bandaged assessment. But the improved SAMA, evidenced by the 117-page sheaf of documents FENOC deposited with the Commission on July 16, 2012, contains no reference to the recently-identified shield building cracking phenomena which Intervenors have been actively litigating, nor do the revised SAMA candidates and calculations address the technical predictions by NRC engineers that massive portions of the reinforced-concrete shield building are at risk of collapse or other failure mode in the event of even a mild seismic event or small load. The SAMA also omits to refer to apparent corrosion of the inner steel containment vessel, which was a mere 1.5 inches thick when brand new and does not have a lot of thickness to lose. Any corrosion is significant. The July 16 update of SAMA literally does not contain the words "shield building," "crack,' or "cracking." As of this writing, even the NRC Staff's requirement that FENOC perform a none-too-comprehensive investigation of the whole shield building for cracking will not be fulfilled until at least December 2012.[1]

To meet its burden on summary disposition, it is incumbent on FENOC, as the movant, to eliminate any real doubt as to the existence of any genuine issue of material fact. *Poller v. Columbia Broad. Sys. Inc.*, 368 U.S. 464 (1962);[2] *Sartor v. Ark. Natural Gas Corp.*, 321 U.S. 620, 627 (1954); *La. Power & Light Co.* (Waterford Steam Electric Station, Unit 3), LBP-81-48, 14 NRC 877, 883 (1981). *See also Entergy Nuclear Vermont Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) ("Summary disposition may be granted only if the truth is clear") (citing *Poller*, 368 U.S. at 467). Pretending that there is no new issue respecting shield building cracking, and reactor containment shell corrosion, does not dispel the existence of genuine fact issues, but bolsters Intervenors' argument that they are present. Because the "truth" isn't "clear" here, summary disposition must be denied so that this matter can proceed to hearing.

B. The Credibility Of The Davis-Besse SAMA Is Already Suspect In Light Of Recently-Admitted Errors

[1]Even the NRC staff referred, in a public meeting at Oak Harbor, Ohio High School on August 9, 2012, to the fact that the "plan was to have a plan" for investigation of the cracking in the future.

[2]Because the Commission's summary disposition rules borrow extensively from Rule 56 of the Federal Rules of Civil Procedure, it has long been held that federal court decisions interpreting and applying like provisions of Rule 56 are appropriate precedent for the Commission's rules. *Safety Light Corp.* (Bloomsburg Site Decommissioning and License Renewal Denials), LBP-95-9, 41 NRC 412, 449 n.167 (1995); *Duke Cogema Stone & Webster* (Savannah River Mixed Oxide Fuel Fabrication Facility), LBP-05-4, 61 NRC 71, 79 (2005).

At the March 1, 2011 hearing convened by the ASLB to hear arguments respecting admissibility of contentions, counsel for FENOC chastised Intervenors to the ASLB respecting their severe accident mitigation analysis (SAMA) contention (proposed Contention 1), remonstrating that the Board need not "get there" - *i.e.*, that the ASLB need not examine Intervenors' factual arguments because they were not legitimate, "site-specific" SAMA criticisms. But FENOC's SAMA assumptions, although modified by its July 16, 2012 *mea culpa* letter and attachments (Attachment 5 to FENOC's Motion for Summary Disposition), which acknowledge five notable miscalculations and erroneous or baseless assumptions[3] changed in the original SAMA, still do not pass muster under the National Environmental Policy Act ("NEPA"). In the original SAMA analysis of Davis-Besse, the wind direction data was input backwards (off by 180 degrees); real property valuation figures critical to estimation of economic losses were understated by a factor of 3; the land area potentially affected by a major radiological accident was grossly understated because of a calculation error; the escalation of decontamination costs was not performed according to the guidance of the Nuclear Energy Institute so as to incorporate the consumer price index; and mischaracterization of core inventory isotopic "mass" as "activity" in the Modular Accident Analysis Program (MAAP) code runs produced inaccurate conclusions instead of meaningful plant-specific values for the mass of the relevant fission product elements.

Following upon these errors in preparation of the original SAMA, it is consistent of FENOC to have failed to update the SAMA by detailing and discussing the looming problem of

[3]July 16, 2012, L-12-244, 10CFR54, FENOC's John C. Dominy, Director, Site Maintenance, FENOC, to NRC Document Control Desk, Attachment 1, Description of Errors Identified in the Davis-Besse Nuclear Power Station Unit No. 1 (Davis-Besse), License Renewal Application, Environment Report, Attachment E, Severe Accident Mitigation Alternatives Analyses, Page 1 of 1 [4/117 on .pdf counter].

concrete cracking in the shield building as well as corrosion of the steel containment vessel containing the Davis-Besse nuclear reactor.

C. Facts Which Create Genuine Issues Of Material Fact And Militate Against Summary Disposition

The shield building and the metal containment structure within it have structural problems. The shield building has experienced what appears to be a likely-underestimated epidemic of serious cracking, and the steel containment within it is corroding.

NRC staff engineers have identified and predicted considerable damage to the shield building. In an "Email from P. Hernandez, NRR to E. Sanchez-Santiago, RIII on Questions about Davis Besse Shield Building Report from DORL" dated November 4, 2011,[4] Pete Hernandez, assistant to the Lead PM [Project Manager] for Davis-Besse, responds to "C-CSS-099.20.054," a "calculation [of] the structural integrity of the SB [shield building]…considering the presence of an interfacial/circumferential crack between the SB structural concrete shell (*i.e.*, the 30'' thick reinforced concrete SB) and each architectural flute shoulder (16 flute shoulders in total), as described in Attachment B." He states:

> This description makes me think that they are looking at a single crack going in a circle. From what I understood the crack is pervasive along the entire surface, spidering in all directions, similar to a pane of tempered glass breaking. The description in Attachment B addresses only the crack at the opening and assumes that the crack is right along the rebar line. The core bores have shown that the cracks are at different depths so this doesn't seem to capture the current situation. Throughout the calculation, the word Crack, singular, is used. They also mention that the extent of the crack is only 10'-12.' This seems to greatly downplay the issue.
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> *****   *****   *****   *****
>
> At this point core bores of only the shoulders have been taken. So the only crack

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[4] http://pbadupws.nrc.gov/docs/ML1220/ML12200A192.pdf

widths we are aware of are those in the shoulders, which are not being addressed. **How can an analysis be done on the structurally credited concrete if no data from that area, in the form of core bores, has been taken? Shouldn't the structural integrity of the shoulders be calculated as well?**

> This seems to say that they are just doing calculations for the new concrete that is and ignores the rest of the building altogether. Is that right?
>
> **This says to me, that they are ignoring the shoulders, if they are ignoring all that concrete, it seems to be the opposite of conservative for evaluating the mechanical loads.**
>
> C-CSS-099.20.055 - Objective or Purpose: The purpose of this calculation is to demonstrate that during a seismic event, with the development of the crack in the architectural flute shoulder, the capacity of the rebar(s) can still provide adequate anchorage thus prevent cracked concrete piece from falling, and therefore Seismic II/I condition can be maintained.

(Emphasis added).

Hernandez further states:

> **I think the greater concern is will the SB stay standing and not whether or not the decorative concrete will fall off.** Because the licensee has not performed core bores to see if there is cracking in the credited concrete, **do they have a basis to say that the structural concrete will maintain a Seismic II/I condition?**
>
> This use of singular terminology also discounts this calculation because it seems that they are looking at only 1 crack and 1 shoulder or 1 flute. **Because cracks have been found through multiple core bores, shouldn't the appropriate calculations account for the combined effects of cracks in all the shoulders** and not just one by opening and not just individually?
>
> From what I understand, **IR mapping is only an indicator, but must be validated by core bores**. Does basing all the calculations on a length of a 12 foot crack discount the calculations altogether, because **we have indications of cracks at distances greater than 12 feet.** This also seems to assume that there is only 1 crack and not *many as the core bores seem to prove*. Isn't **IR mapping only useful at a limited depth too**, so that using it to evaluate a 48" thick piece of concrete is not realistic?

(Emphasis added).

Hernandez' concerns are shared by NRC staff engineer Abdul Sheikh, who in an "Email from A. Sheikh, NRR to E. Sanchez Santiago, RIII on Questions for the Conference Call,"[5]

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[5]http://pbadupws.nrc.gov/docs/ML1220/ML12200A213.pdf

states:

If this assumption is correct only 3-4 inches of the concrete on the inside face can be used in the structural analysis. In the response to the questions, the applicant stated that, 'Since we assume that outside reinforcement is to be treated ineffective in carrying any additional stress beyond 12.4 ksi, under accident thermal loads that may cause stresses in excess of what the rebar can carry (assumed 12.4 ksi), **the reinforcement is assumed to detach itself from the outer section of the shell.' These statements seems *(sic)* to be contradictory. In addition, I am concerned that the concrete will fail in this region due to bending in this region even under small loads.**

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Since we assume that outside reinforcement is to be treated ineffective in carrying any additional stress beyond 12.4 ksi, under accident thermal loads that may cause stresses in excess of what the rebar can carry (assumed 12.4 ksi), **the reinforcement is assumed to detach itself from the outer section of the shell.**

Additionally, Mr. Sheikh states:

5. The licensee justification for ignoring the dead (DL) and normal thermal (To) in calculation of rebars splice does not appear to be justified. The stresses due to dead load and thermal loads will be locked in the rebars and cannot be ignored.

6. The licensee considers the allowable stress in the rebar to be 60 ksi and ignores a phi factor (0.9) in his evaluation for lap splice. In addition, the licensee has not accounted for any additional uncertainty due the conditions.

7. I am not aware of any pull tests carried out with a crack in the plane of the rebar. Can the licensee provide any documentation for this statement.

8. The licensee is using numerous assumptions in his summary report and calculations that are not described in the UFSAR and ACI 318-63, and still calls it a design basis calculation. Can the licensee provide justification for this approach.

FENOC's contractor, Performance Improvement International ("PII"), documented that there are 14" deep cracks in the shield building walls, *see* PII's report, "Revised Root Cause Assessment: Davis-Besse Shield Building Laminar Cracking, Vol. 1," ML12138A037, pp. 92-93.[6] The presence of such cracking raises obvious questions about the ability of the shield

[6]"7. The exact depth of penetration used as input to the FE model varies. In "1D" areas, it is 4" or less. In "2D" areas, it is 14" or less. An inch one way or the other would shift the crack location about an inch - but a rigorous sensitivity study was not performed since we are not modeling growth rate."

building to fulfill its radiologically-critical function of containing radiation in the event of a release to the annulus. Abdul Sheikh, NRC engineer, found during his investigation of the cracking that "[The Davis-Besse] shield building has not been designed for containment accident pressure and temperature." NRC FOIA response, Appendix B, Document B/44, 12/13/11, Email from M. Galloway, NRR to A. Sheikh, NRR et al., RE: Davis-Besse Shield Building, 1 page.[7]

FENOC admits in its February 28, 2012 Root Cause Analysis that examination of the entire shield building at Davis-Besse has not taken place and planned inspections, which will not be comprehensive, will not be completed until December 1, 2012.[8] The RAI AMP foresees scant planned testing to be done during infrequent inspections over the coming decades, in the form of a mere handful of core bores. This is the NRC's "plan to have a plan" for FENOC inspections, which is inadequate from its inception.

Davis-Besse has water problems inside the shield building which seem to be contributing to corrosion of the steel containment. In RAI responses dated May 24, 2011 (ML 11151A90), the NRC staff had noted a "history of ground water infiltration into the annular space between the concrete shield building and steel containment":

> During a 2011 AMP audit, NRC staff also reviewed documentation that: [I]ndicated the presence of standing water in the annulus sand pocket region. The standing water appears to be a recurring issue of ground water leakage and areas of corrosion were observed on the containment vessel. In addition, during the audit the staff reviewed photographs that indicate peeling of clear coat on the containment vessel

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[7]http://www.beyondnuclear.org/storage/B%2044.pdf

[8]From FirstEnergy's April 5, 2012 "Reply to Requests for Additional Information" (ML12097A216) ("RAI AMP") at 11/29 of .pdf. "FENOC is developing a comprehensive engineering plan to re-establish the design and licensing basis conformance of the Shield Building. The plan is scheduled to be completed and issued by December 1, 2012. ***The plan will include a detailed structural analysis of the Shield Building and consider applicable effects."***

> annulus area, and degradation of the moisture barrier, concrete grout, and sealant in the annulus area that were installed in 2002-2003.

Id. at 47/280 of .pdf. FENOC has been required to commit to periodic inspections of the containment vessel for possible thinning of its wall. *See* ML11294A349. At p. 21/60 of this "Davis-Besse Commitment List,"[9] FirstEnergy commits to monitor water draining through the shield building for acidity and damage to reinforcing bars in the spent fuel pool and pits. *Id.* at 21/60 of .pdf. Before 2023, FENOC must check the concrete for compressive strength and degradation of reinforcing bars, apparently from refueling canal leakage. *Id.* at 22-23/60.

At p. 24/60 of this "Davis-Besse Commitment List," FENOC promises to:

> Compare the ultrasonic test (UT) thickness readings to minimum ASME Code vessel thickness requirements and to the results obtained during previous UT examinations of the Containment Vessel. Determine the need for maintenance or repair of the Containment Vessel based on the results and evaluation of the examinations.

At 25/60 of .pdf, FENOC is to "Perform visual inspection of 100 percent of the accessible areas of the wetted outer surface of the Containment Vessel in the sand pocket region." At 26/60 of the .pdf, FirstEnergy pledges to perform and evaluate core bores of the ECCS Pump Room No. 1 wall and the Room 109 ceiling, to examine core samples "for signs of corrosion or chemical effects of boric acid on the concrete or reinforcing bars" by 2014. There appears to be cracking on the underside of the spent fuel pool. At 27/60 of the .pdf, FENOC agrees to:

> Address the potential for borated water degradation of the steel containment vessel through the following actions:
>
> Access the inside surface of the embedded steel containment. A core bore will be completed by the end of 2014 (Phase 1). If necessary, a second core bore will be completed by the end of 2020 (Phase 2). If there is evidence of the presence of borated water in contact with the steel containment vessel, conduct non-destructive testing (NDT) to determine what effect, if any, the borated water has had on the steel containment

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[9] http://pbadupws.nrc.gov/docs/ML1129/ML11294A349.pdf

> vessel. Based on the results of NDT, perform a study to determine the effect through the period of extended operation of any identified loss of thickness in the steel containment due to exposure to borated water.

According to Abdul Sheikh, boric acid deposits had been observed over a large surface area of the Containment Incore Instrumentation Tunnel walls and the under-vessel area that are indicative of refueling canal leakage, the possible damage to structures has not been adequately addressed and FENOC has not clearly explained how the leakage will be reduced. (FOIA response document) Email, "5 Davis-Besse OIs [Open Items], January 10, 2012, 5/7 of .pdf.[10]

The MAAP code used for the license renewal at Davis-Besse has not been shown by Applicant to have been benchmarked against Davis-Besse's, nor any other nuclear power plant's, persistent, identified shield building cracking nor containment vessel corrosion thinning.

There are two uses of the NUREG-1465 source term. 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. "Joint Declaration of Kevin O'Kula and Grant Teagarden in Support of FirstEnergy's Motion for Summary Disposition of Intevenors' Contention 4 (SAMA Analysis Source Terms)" (O'Kula/Teagarden Decl.") (Attachment 2 to FENOC's MSD) ¶ 38. The second purpose for a reactor accident source term is to simulate a release of radioactive material to the environment (*i.e.*, outside containment) following a hypothetical reactor accident. This second source term is input to models of

[10]http://pbadupws.nrc.gov/docs/ML1201/ML12019A378.pdf

radionuclide environmental transport and dispersion and accident consequences that are used for Level 3 PRAs and SAMA cost-benefit analyses, which are best-estimate analyses. The use of the MAAP-based source term associated with releases to the environment for the Davis-Besse PRA and its SAMA analysis is a crucial element of Level 3 PRA and SAMA cost-benefit analyses. O'Kula/Teagarden Decl. ¶ 39.

The NUREG-1465 source term describes the amounts and types of radioactive material that would enter the containment, while MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident. O'Kula/Teagarden Decl. ¶ 43.

Plant-specific source terms developed for SAMA analysis must consider a spectrum of probabilistically-significant accident scenarios to have any meaning from a risk quantification perspective. O'Kula/Teagarden Decl. ¶ 44. The methodology used to develop source terms for a SAMA analysis must account for plant-unique conditions, plant design, support system dependencies, plant maintenance and operating procedures, operator training, and the interdependencies among these factors that can influence the core damage frequency (CDF) estimate for a specific plant. *Id.* ¶ 49.

Despite the evidence in this case of plant-unique, actual, negative conditions afflicting the shield building and its containment which cannot be wished or theorized away, FENOC's experts insist that "for Davis-Besse, approximately 90% of the core damage sequences involve accidents in which the containment *retains its structural integrity* (*i.e.,* radiological release is limited to containment leakage, as modeled in RC 9.1 and 9.2), and the remaining 10% would be the result of early containment failure and other events (*e.g.*, containment by-pass events, specifically steam generator tube rupture and interfacing system loss of coolant accidents)" (emphasis added). MSD p. 26. NRC staff engineer Sheikh contradicts this bias with his determination that up to 90% of the outer concrete of the shield building could collapse under mild stresses. Such a diametric contradiction necessitates trial on the merits.

FENOC omitted to include within its SAMA analyses any information about the Davis-Besse shield building cracking or the corroding steel barrier shell contained within it. There is zero analysis of the changed physical properties of those facilities, nor any discussion of the implications of those changed physical properties on the capacity of the shield building or the steel containment structure to contain radioactive materials in the event of an accident. FENOC makes the optimistic, self-serving assumption in its SAMA analyses and MAAP assumptions that the shield building, as well as steel containment vessel, are as good as new. This is not the case. The July 16, 2012 SAMA update (Attachment 5 to FENOC's MSD) literally does not contain the words "shield building, "crack," or "cracking," as revealed by word searches.

In FENOC's SAMA analysis, depicted in the Environmental Report ("ER") which is part of the license renewal application, only two (2) accident scenarios assume any negative involvement of the shield building or its components. At ER p. E-81, Table E.3-5, "Representative MAAP Accident Sequence" appears this scenario:

> Based on containment bypass sequence - guillotine rupture of the 12-inch diameter decay heat removal return line with failure of two valves in series. Primary system coolant is discharged to mechanical penetration room #2 which communicates with the shield building annulus (wire mesh doors). Following the pipe rupture, the room blowout panels fail allowing a release to the Auxiliary Building and environment. ECCS injection fails.

And at ER p. E-82, Table E.3-5, "Representative MAAP Accident Sequence" is this hypothetical mishap:

> Based on the Level 1 sequence TINYNINN; a SBO and loss of AFW at time zero. RCs (Release Categories) 6.1 and 6.2 assume direct impingement of entrained core debris on the containment free standing steel shell to obtain sidewall failure even with a coolable debris bed geometry. Sidewall failure 2 minutes after vessel failure results in early containment failure. Sidewall failures communicate with the shield building annulus and auxiliary building #4 mechanical penetration room. Release of fission products to the environment occurs following blow out panel failures; no annulus or auxiliary building decontamination factors are credited.
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> RC's 6.3 and 6.4 include uncoolable debris beds; the debris is assumed to pool in the lower compartment against the outer concrete curb. Late containment failure occurs when sufficient concrete is eroded.
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> RC's 6.1 and 6.3 include fission product scrubbing via containment spray and CACs.

At ER p. E-97, Table E.3-20, the "Frequency Vectors" of the above-mentioned Release Categories 6.1 through 6.4 are almost nonexistent, despite the cracking:

6.1	4.4E-10	0.00%
6.2	3.3E-11	0.00%
6.3	4.5E-09	0.04%
6.4	3.1E-08	0.31%

At ER p. E-179, "Qualitative Screening of SAMA Candidates," the question of improving seismic restraints in plant components remains unchanged from 2010 when the ER first appeared, down to the present:

SR-01 Increase seismic ruggedness of plant components. Criterion D	Very Low Benefit	The Seismic Qualifications Utility Group (SQUG) previously identified the need for additional seismic restraints in the plant. These restraints have already been added.

The Advisory Committee on Reactor Safeguards (ACRS) had serious concerns about the seismic qualifications of the Davis-Besse shield building, pre-operations, in the mid-1970s. This was revealed in FOIA Response Number 1, Appendix B, Document B/1, using a 0.15g maximum ground acceleration, instead of 0.20g.[11]

D. Argument

The cracking investigation of the shield building, which Intervenors detailed extensively in many filings since January 2012, arose in October 2011 and in NEPA terms, has generated significant new information which remains unaccounted for in the SAMA. Although an agency does not need to formally supplement an EIS whenever new information about a project comes to light, it must be reasonable in addressing new information, and consider its environmental significance and likely accuracy. *Warm Springs Dam Task Force v. Gribble*, 621 F.2d 1017, 1025 (9th Cir. 1980). NEPA imposes continuing obligations on the NRC, even after supposed completion of an environmental analysis. An agency that receives new and significant information casting doubt upon a previous environmental analysis must re-evaluate the prior analysis. *Marsh v. Oregon Natural Resources Council*, 490 U.S. 360, 374 (1989). This requirement is codified in NRC regulations at 10 C.F.R. §51.92(a). The NRC's license renewal application regulations also contain this obligation. 10 C.F.R. §51.53(c)(3)(iv) (ER must contain "any new and significant information regarding the environmental impacts of license renewal of which the applicant is aware"). This obligation extends to new and significant information even when such information pertains to a Category 1 issue. *See Duke Energy Corp.* (McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2), CLI-02-14, 55 NRC 278, 290 (2002). And 10 C.F.R. §51.45 requires that an environmental report shall discuss [(b)(1)] "[t]he impact of the proposed action on the environment" and that "[i]mpacts shall be discussed in

[11]"Licensing Basis Ground Motion Concern," released pursuant to January 2012 FOIA request of Intervenors, accessible online at http://www.beyondnuclear.org/storage/B%201.pdf

proportion to their significance. . . ." Moreover, §51.45(e) further requires candor from the applicant: "The information submitted pursuant to paragraphs (b) through (d) of this section should not be confined to information supporting the proposed action *but should also include adverse information*" (emphasis added).

The principal factor an agency should consider in exercising its discretion whether to supplement an existing EIS because of new information is the extent to which the new information presents a picture of the likely environmental consequences associated with the proposed action not envisioned by the original EIS. The issue is whether the subsequent information raises new concerns of sufficient gravity such that another, formal in-depth look at the environmental consequences of the proposed action is necessary. When the new information provides a seriously different picture of the environmental landscape such that another NEPA "hard look" is necessary, supplementation must take place. *State of Wis. v. Weinberger*, 745 F.2d 412, 418 (7th Cir. 1984)

FirstEnergy has not lived up to these standards and expectations. The Motion for Summary Disposition consists largely of hollow platitudes about SAMA contents and source term usages with which Intervenors don't much necessarily disagree, but which have arguably been ignored or breached as a consequence of FENOC's stunning failure to incorporate serious analysis about the damaged shield building, as well as the corroded steel containment vessel and their susceptibility into the SAMA candidate mix.

For example, FENOC maintains (MSD p. 2) that NUREG-1465 source terms represent radionuclides released into the containment atmosphere from a core-melt accident and are not the environmental source term used in a SAMA analysis. While that is generally true, to frame

Davis-Besse site-specifically requires one to assume that the shield building is severely cracked (as a fixed, starting and recurring assumption), and that whatever radionuclides enter the annulus between the corroded steel containment vessel and SB in the event of accident are likely to be accompanied by the very accident conditions (pressures and temperatures) which NRC's Abdul Sheikh has stated the shield building was not designed to withstand in the first place.[12] Given that the shield building is not brand new, but rather severely cracked, the potential for failure under even small loads (such as accident condition temperatures and pressures) cannot be ignored. The shield building could fail, releasing any and all gasesous, volatile, and even a large fraction of the solid particulate radioactivity directly into the environment, with no "sweeping and filtering" prior to discharge.

FENOC maintains (MSD p. 24) that the level of detail, and technical acceptability of its risk-informed analyses [PRAs] are to "'be based on the *as-built and as-operated and maintained plant*,' and reflect operating experience at the plant" (emphasis supplied) (citing Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 2, at 7 (May 2011) (Attachment 32 to FENOC's MSD). But the term "as-operated and maintained" at Davis-Besse must account for a uniquely-cracked shield building nearly unable to sustain its own weight. FENOC's argument holds neither water nor does it demonstrate containment of radioactivity.

Similarly, the mechanisms in place inside the shield building to "sweep and filter" fission products will have little useful effect if 90%, or a sizeable portion of the outer surface of the

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[12]NRC FOIA response, Appendix B, Document B/44, 12/13/11, Email from M.Galloway, NRR to A. Sheikh, NRR et al., RE: Davis-Besse Shield Building, http://pbadupws.nrc.gov/docs/ML1220 /ML12200A213.pdf

shield building were to collapse. Those devices and the maintenance of a negatively-pressured atmosphere within the shield building cannot be assumed to function correctly, or at least usefully, if shield building failure occurs because of the un- and under-investigated cracking. While at other nuclear plants it might be appropriate to not use NUREG-1465 source term release fractions as a "worst-case" analysis, the Davis-Besse shield building defects and potential for failure identified by Intervenors leave little to no margin for error. Analysis must begin at a much more conservative point than that used by FENOC. The evidence Intervenors have articulated in opposition to summary disposition thus establishes a "credible potential material deficiency in the [SAMA] analysis," and Intervenors should be accorded a hearing. *Entergy Nuclear Generation Co.* (Pilgrim Nuclear Power Station), CLI-12-15, 75 NRC __, slip op. at 13 (June 7, 2012) (absent a "credible potential material deficiency in the [SAMA] analysis, there is no … demonstration of a material issue for hearing").

Intervenors generally agree with FENOC (MSD p. 14) that "NEPA does not dictate adherence to a particular analytic protocol or even use of the 'best scientific methodology'" and that an applicant is permitted to select its own methodology, "provided that methodology is reasonable." But that reasonableness determination must be made in light of whether the new information at issue is "significant," "relevant to environmental concerns," and has a "bearing on the proposed action." 40 C.F.R. § 1502.9. Intervenors submit that the information about cracking is significant, indisputably is relevant to environmental concerns because of the potential for uncontrolled radiological release to the outside environment, and as a result, has an undeniable bearing on the proposed license extension for Davis-Besse. FENOC incorrectly assumes, for purposes of SAMA computations, the presence of an intact containment structure: "This is consistent with the requirement in Part 100 that, for licensing purposes, an accidental fission product release resulting from 'substantial meltdown' of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated *assuming that the containment remains intact* but leaks at its maximum allowable leak rate." (Emphasis added). MSD p. 23.

The statement of FENOC's which reflects the most magical thinking is that "The failure to credit the containment's presence as well as engineered safety features for mitigating and delaying releases leads to a worst-case source term scenario without any technically supported weighting by likelihood of occurrence." MSD p. 27. Under some circumstances, NEPA makes clear that "worst-case" scenarios such as not crediting the containment's presence at all, and treating containment failure sequences and containment intact sequences equivalently, are not "reasonable or appropriate" SAMA candidates. MSD p. 17. But in the site-specific case of Davis-Besse, it is reasonable and appropriate to take into account Davis-Besse's shoddy construction, the inconsistent safety culture at the plant for the last 35 years, and FENOC's incomplete understanding of the cracking phenomenon at this point, from which to infer that breach of containment from the shield building's condition is an obligatory factor that must be weighted in the SAMA analysis. After all, as FENOC says (MSD p. 25), "The methodology used to develop source terms for a SAMA analysis must account for plant-unique conditions, plant design, support system dependencies, plant maintenance and operating procedures, operator training, and the interdependencies among these factors that can influence the plant-specific CDF."

When the NEPA requirement that there be "worst-case" discussion of difficult-to-quantify risks was revoked in the 1980s, it was replaced with 40 C.F.R. §1502.22(b). That regulation stresses that where there is "incomplete or unavailable information," the EIS must include:

> (1) A statement that such information is incomplete or unavailable; (2) a statement of the relevance of the incomplete or unavailable information to evaluating reasonably foreseeable significant adverse impacts on the human environment; (3) a summary of existing credible scientific evidence which is relevant to evaluating the reasonably foreseeable significant adverse impacts on the human environment, and (4) the agency's evaluation of such impacts based upon theoretical approaches or research methods generally accepted in the scientific community. . . .

Where the missing information is "essential to a reasoned choice among alternatives," §1502.22 requires agencies to explicitly "acknowledge and discuss any flaws" in studies relied on in an EIS. *Cabinet Res. Group v. United States Fish & Wildlife Serv.*, 465 F. Supp. 2d 1067, 1099-1100 (D. Mont. 2006) (court set aside Forest Service's final EIS because it failed to address gaps in a key study it relied on in assessing a motorized access plan's impact on grizzly bears).

Respecting FENOC's assertion that the containment's presence and engineered safety features for mitigating and delaying releases must be credited while a worst-case source term scenario should be avoided, the fact of the matter is that the shield building at Davis-Besse may have failed and will collapse under minor stress. As a matter of law, "reasonably foreseeable significant adverse impacts on the human environment . . . include impacts which have catastrophic consequences, even if their probability of occurrence is low, provided that the analysis of the impacts is supported by credible scientific evidence, is not based on pure conjecture, and is within the rule of reason." 40 C.F.R. §1502.22(b). Unfortunately for FENOC, one "worst case" scenario for Davis-Besse is also the base case, is entirely foreseeable and is a proper SAMA candidate which might anticipate the four-fold increase in radiation releases to the environment, the difference between MAAP and NUREG-1465 source terms.[13]

It may well be that with the case of Davis-Besse, the outer frontiers of SAMA analysis have been reached. Intervenors are aware of their role, *arguendo*, to interject considerations of approximate relative cost and benefit of the SAMA candidate. *Entergy Nuclear Operations, Inc*. (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43, 102 n.308 (2008), which they do here. The SAMA candidates proposed by Intervenors rely on the basic (and factually-informed) conclusion that the shield building is severely cracked and could likely shed 90% of its concrete layers in the event of a minor seismological event, thus risking a breach failure to the outside environment; that is, the shield building and the corroded containment shell within it are compromised, not as a theoretical assumption, but as fact. Intervenors submit that unless this fact is included in the MAAP model, it will be difficult to conclude that the resulting SAMA is properly cost-beneficial and warrants serious consideration.

With respect to this sweeping, but realistic, factual conclusion that the shield building is probably compromised and incapable of serving its radiation containment functions, the SAMA candidate accidents and the physical project improvements necessary for mitigation, while seemingly audacious, are in fact merely obvious. Mitigation alternatives to remedy the shield building and containment shell failures include at least two options: (1) complete replacement, or (2) construction of an entirely new shield building over the existing one. Neither is likely to be

[13]*See* Lehner *et al.*, “Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants,” Final Letter Report, Brookhaven National Laboratory, Upton, NY, December 23, 2002, p. 17. ADAMS Accession Number ML031700011 (NUREG-1150 release fractions for the important radionuclides are about a factor of 4 higher than the ones used in the Duke PRA. The Duke results were obtained using the Modular Accident Analysis Package (MAAP) (See, for example, FENOC ER. Page 4.20-1 and E-17) code, while the NUREG-1150 results were obtained with the Source Term Code Package (NRC’s state-of-the-art methodology for source term analysis at the time of NUREG-1150) and MELCOR).

perceived as cost-beneficial. This refusal to fully and candidly assess the condition of the shield building and containment shell conveniently keeps FENOC from having to address the CRAC-2 damage predictions for Davis-Besse in the SAMA analysis. In "CRAC-2, Calculation of Reactor Accident Consequences," U.S. Nuclear Power Plants, Sandia National Laboratory, 1982, Sandia Laboratory predicted, using 1970 census data, that estimated the number of cancer deaths at Davis-Besse in a severe accident to be 10,000; early fatalities 1,400; and early injuries 73,000.[14] Peak fatalities were estimated by CRAC to occur within 20 miles of Davis-Besse; and peak injuries to occur within 65 miles of Davis-Besse from a core melt and catastrophic radioactivity release. CRAC-2's property damage prediction is $84 billion (expressed in 1982 dollar figures). Applying the Westegg inflation calculator[15] to that number, it becomes $187.24 billion in 2010 dollars. As reported by the Associated Press in June 2011,[16] populations have soared around U.S. nuclear power plants over the past 42 years, including around Davis-Besse. Thus, casualty figures would be even worse now than predicted by CRAC-2 in the early 1980s, based on 1970 U.S. Census data.

The reality here is, a severe accident at Davis-Besse arising from the degraded condition of the under-investigated shield building could cause tens to hundreds of billions of dollars of personal and property damage downwind and downstream (up the food chain, down the generations). The economics of the Davis-Besse SAMA which would emerge from serious consideration using the MAAP program of major radiological containment failure would identify

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[14]The chart from the CRAC-2 study, showing reactor-by-reactor consequences, can be found at www.beyondnuclear.org/storage/CRAC%202%20chart.pdf

[15]http://www.westegg.com/inflation/

[16]http://www.ap.org/company/awards/part-iii-aging-nukes

expenditure of a billion or so dollars on a new shield building to prevent perhaps hundreds of billions of dollars' worth of offsite damage - liability from which FENOC is legally buffered for the most part by the Price-Anderson Act. While it is not covered by the MAAP computer codes, the liability-shifting effects of federal law allow FirstEnergy to limit its "cost-benefit" concerns primarily to considerations of its own economic profitability and viability. Comparatively small expenditures to mitigate worse problems can be attractive to a nuclear operator up to a point, but there is no margin in confessing the need for replacing large, expensive components that undermine the nuclear power business model.

NRC SAMA analyses are not a substitute for, and do not represent, the NRC NEPA analysis of potential impacts of severe accidents. *Entergy Nuclear Generation Co. and Entergy Nuclear Operations, Inc.* (Pilgrim Nuclear Power Station), CLI-10-11, 71 NRC __ (Mar. 26, 2010) (slip op. at 37). The Generic Environmental Impact Statement for License Renewal "provides a generic evaluation of severe accident impacts and the technical basis for the NRC's conclusion that 'the probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts from severe accidents are small for all plants.'" *Id.* at 38. The NRC's generic assessment of the environmental impacts of severe accidents for all existing plants during the license renewal term is bounding. *Id*. Contrastingly, SAMA analysis is site-specific mitigation analysis for which NEPA does not demand a "fully developed plan" or a "detailed examination of specific measures which will be employed" to mitigate adverse environmental effects. *Id.* (citing *Duke Energy Corp*. (McGuire Nuclear Station, Units 1 & 2; Catawba Nuclear Station, Units 1 & 2), CLI-03-17, 58 NRC 419, 431 and *Robertson v. Methow Valley Citizens Council*, 490 U.S. 332, 353 (1989)). Accordingly, "unless it looks genuinely plausible that inclusion of an additional factor or use of other assumptions or models may change the cost-benefit conclusions for the SAMA candidates evaluated, no purpose would be served to further refine the SAMA analysis, whose goal is only to determine what safety enhancements are cost-effective to implement." CLI-10-11, 71 NRC __ (Mar. 26, 2010) (slip op. at 39). The question identified by Intervenors in the unique circumstance of the likely-failed Davis-Besse shield building is, cost-effective to whom?

Intervenors urge that "genuine plausibility" attaches to the very unique, very site-specific facts of Davis-Besse's shield building cracking problems, combined with its inner steel containment vessel corrosion problems, and that the integrity NEPA requires in severe accident mitigation analysis is at stake here. NEPA explicitly requires an agency undertaking an EIS to "insure the professional integrity, including scientific integrity, of the discussions and analyses in environmental impact statements." 40 C.F.R. §1502.24. It "is the agency, not an environmental plaintiff, that has a 'continuing duty to gather and evaluate new information relevant to the environmental impacts of its actions,' even after release of an [EA or EIS]." *Friends of the Clearwater v. Dombeck*, 222 F.3d 552, 559 (9th Cir. 2000) (quoting *Warm Springs Dam Task Force v. Gribble*, 621 F.2d at 1023); *See also Te-Moak Tribe v. Interior*, 608 F.3d 592, 605-606 (9th Cir. 2010); *Davis v. Coleman*, 521 F.2d 661, 671 (9th Cir. 1975) ("[C]ompliance with NEPA is a primary duty of every federal agency; fulfillment of this vital responsibility should not depend on the vigilance and limited resources of environmental plaintiffs"). NEPA requires the agency to develop alternatives that will "mitigate the adverse environmental consequences of a proposed project." *Robertson v. Methow Valley Citizens Council*, 490 U.S. 332, 351 (1989).

In the end, NEPA requires the exercise of reason, and reasonable accuracy in determining the inputs, if not the outcomes, of SAMA analysis, aimed at proving that the requisite "hard look" was taken by the agency. Here, if it is difficult to quantify for SAMA purposes the likely effects of shield building cracking and containment corrosion, then the alternative protocol mentioned above must be followed. "[W]hen the *nature* of the effect is reasonably foreseeable but its *extent* is not," the agency's EIS must follow §1502.22 *Mid States Coal. for Progress v. Surface Transp. Bd.*, 345 F. 3d 520, 549-50 (8th Cir. 2003). In other words, if FENOC's resistance focuses on an inability to quantify the implications of the shield building problems, then detailed qualitative disclosures remain mandatory.[17]

Given the responsibility placed indirectly by NEPA upon FENOC, which as applicant must prepare an accurate Environmental Report, and the NRC Staff, as the direct NEPA lead agency, summary disposition must be denied FENOC on the SAMA contention. A hearing must be held to determine whether the NRC and FENOC should be required to supplement and correct the MAAP analytical tools to accurately reflect that effectively zero weight is ascribed to the shield building for mitigation purposes in the event of a severe accident. That is all in fulfillment of the duty that the NRC ensure that there is a rational connection between the facts found and the choices made. *Burlington Truck Lines v. United States*, 371 U.S. 156, 158 (1972) (agency must consider "relevant factors" and articulate "a rational connection between the facts found and the choices made"). Until the site-specific factors of the cracked shield building and cor-

[17]Per 40 C.F.R. §1502.22(b): "(1) A statement that such information is incomplete or unavailable; (2) a statement of the relevance of the incomplete or unavailable information to evaluating reasonably foreseeable significant adverse impacts on the human environment; (3) a summary of existing credible scientific evidence which is relevant to evaluating the reasonably foreseeable significant adverse impacts on the human environment, and (4) the agency's evaluation of such impacts based upon theoretical approaches or research methods generally accepted in the scientific community. . . ."

roded containment are recognized as mandatory assumptions, the integrity of NEPA will be compromised.

Intervenors well recognize that NRC "adjudicatory hearings are not EIS editing sessions," *Private Fuel Storage* (Independent Spent Fuel Storage Installation), CLI-02-25, 56 NRC 340, 352 (2002), and that SAMA analysis is not a safety review performed under the Atomic Energy Act. But it is especially important for the NRC to disclose the SAMA costs and benefits in the SEIS, especially those such as complete replacement of the shield building which will be controversial and likely not implemented by FENOC (since NEPA requires only that there be disclosure of cost-benefit measures in the SAMA analysis, not their implementation).[18] It is also important for there to be factual faithfulness in deciding the SAMA data inputs, since the NRC's understanding of severe accident risks has changed in light of the Fukushima accident. See Jaczko dissent from *Southern Nuclear Operating Co.* (Vogtle Electric Generating Plant, Units 3 and 4) CLI-12-02, slip op. at 3-4.

CONCLUSION

Intervenors have presented a genuine issue of material fact, *i.e.*, that the SAMA for Davis-Besse may not be considered complete or must be supplemented and recalculated to recognize the fact of proven, and suspected, degradation of the shield building concrete, and corrosion of the containment vessel which shelters the reactor.

It falls to FENOC, as movant, to eliminate any real doubt as to the existence of any genuine issue of material fact. *Poller v. Columbia Broad. Sys. Inc.*, 368 U.S. 464 (1962).

[18] *Winter v. Nat. Res. Def. Council*, 129 S.Ct. 365, 376 (2008)(stating that "NEPA imposes only procedural requirements" and does not mandate any particular result).

"Summary disposition may be granted only if the truth is clear." *Entergy Nuclear Vermont Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (citing *Poller*, 368 U.S. at 467).

Intervenors have cast serious doubt on the consistency, comprehensiveness, and scientific completeness of the current Davis-Besse license renewal SAMA. It is not compliant with the regulations and expections imposed by the National Environmental Policy Act. Summary disposition must be denied here, however, because, per *Entergy Nuclear Vermont Yankee, L.L.C.*, the truth is not "clear," nor is it to be found within FENOC's moving papers.

The Commission recently found that:

> With respect to a SAMA analysis in particular, unless a contention, submitted with adequate factual, documentary, or expert support, raises a potentially significant deficiency in the SAMA analysis - that is, a deficiency that could credibly render the SAMA analysis altogether unreasonable under NEPA standards - a SAMA-related dispute will not be material to the licensing decision, and is not appropriate for litigation in an NRC proceeding.

Entergy Nuclear Generation Company and Entergy Nuclear Operations, Inc. (Pilgrim Nuclear Power Station), CLI-12-01, __ NRC ___ , slip op. at 25 (February 9, 2012). Here, Intervenors have demonstrated with factual and documentary support a "potentially significant deficiency" in the SAMA analysis for Davis-Besse. The failure of FENOC and ultimately, the NRC Staff, to identify, discuss and analyze within the SAMA analyses the structural inadequacies of the shield building and metal containment vessel "could credibly render the SAMA analysis altogether unreasonable."

Accordingly, summary disposition must be denied FirstEnergy on Contention 4.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		September 14, 2012
	)	

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INTERVENORS' STATEMENT OF MATERIAL FACTS IN OPPOSITION TO 'FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4'

1. Evaluation of source terms for a SAMA analysis requires a detailed analytical model that includes a multitude of physical process sub-models that account for, among other things, the timing and performance of both passive and active plant safety features and human (*i.e.*, operator) actions affecting accident progression and containment conditions. Any radionuclide releases outside of containment are sequentially modeled from their release from the reactor core through any release path from the containment (through partial containment failure or bypass conditions), and into the environment. Source terms depend on how rapidly the accident progresses, the path by which the radionuclides escape from the reactor into containment, the path through containment (or possibly bypassing containment altogether), and the effectiveness of both passive and active safety features, especially pools and sprays, that are intended to mitigate releases. "Joint Declaration of Kevin O'Kula and Grant Teagarden in Support of FirstEnergy's Motion for Summary Disposition of Intevenors' Contention 4 (SAMA Analysis Source Terms)" (O'Kula/Teagarden Decl.") (Attachment 2 to FENOC's MSD) ¶ 27.

2. The MAAP code as applied in this case has not been shown by Applicant to have been benchmarked against any nuclear plant's persistent, identified, widespread shield or containment building concrete cracking phenomenon.

3. There are two uses of the NUREG-1465 source term. 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. O'Kula/Teagarden Decl. ¶ 38. 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 is input to models of radionuclide environmental transport and dispersion and accident consequences that, among other purposes, are used for Level 3 PRAs and SAMA cost-benefit analyses, which are best-estimate analyses. The use of the MAAP-based source term associated with releases to the environment for the Davis-Besse PRA and its SAMA analysis supports this latter purpose; *i.e.*, it is a crucial element of Level 3 PRA and SAMA cost-benefit analyses. O'Kula/Teagarden Decl. ¶ 39.

4. The NUREG-1465 source term describes the amounts and types of radioactive material that would enter the containment, while MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident. O'Kula/ Teagarden Decl. ¶ 43.

5. MAAP is supposed to model and credit engineered safety features within the containment structure as fission product removal mechanisms. O'Kula/Teagarden Decl. ¶ 44.

6. Plant-specific source terms developed for SAMA analysis must consider a spectrum of probabilistically-significant accident scenarios to have any meaning from a risk quantification perspective. O'Kula/Teagarden Decl. ¶ 44.

7. The methodology used to develop source terms for a SAMA analysis must account for plant-unique conditions, plant design, support system dependencies, plant maintenance and operating procedures, operator training, and the interdependencies among these factors that can influence the core damage frequency (CDF) estimate for a specific plant. O'Kula/Teagarden Decl. ¶ 49.

8. For Davis-Besse, approximately 90% of the core damage sequences involve accidents in which the containment retains its structural integrity (*i.e.*, radiological release is limited to containment leakage, as modeled in RC 9.1 and 9.2), and the remaining 10% would be the result of early containment failure and other events (*e.g.*, containment bypass events, specifically steam generator tube rupture and interfacing system loss of coolant accidents). O'Kula/Teagarden Decl. ¶ 54.

9. FENOC omitted to include within its SAMA analyses any information specific to the Davis-Besse shield building or the corroding steel barrier shell contained within it, any analysis of the changed physical properties of those facilities, and any identification of the implications of those changed physical properties on the capacity of the shield building or the steel containment structure to contain radioactive materials in the event of an accident.

10. In its SAMA calculations, FirstEnergy assumes that the containment structure at Davis-Besse remains intact: "This is consistent with the requirement in Part 100 that, for

licensing purposes, an accidental fission product release resulting from 'substantial meltdown' of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated *assuming that the containment remains intact* but leaks at its maximum allowable leak rate." (Emphasis added). MSD p. 23.

11. The July 16, 2012 SAMA update (Attachment 5 to FENOC's MSD) literally does not contain the words "shield building," "crack," or "cracking."

12. In FENOC's SAMA analysis, depicted in the license renewal application Environmental Report ("ER"), there are two (2) accident scenarios which assume any negative involvement of the shield building or its components. At ER p. E-81, Table E.3-5, "Representative MAAP Accident Sequence" appears this scenario:

> Based on containment bypass sequence - guillotine rupture of the 12-inch diameter decay heat removal return line with failure of two valves in series. Primary system coolant is discharged to mechanical penetration room #2 which communicates with the shield building annulus (wire mesh doors). Following the pipe rupture, the room blowout panels fail allowing a release to the Auxiliary Building and environment. ECCS injection fails.

And at ER p. E-82, Table E.3-5, "Representative MAAP Accident Sequence" appears this scenario:

> Based on the Level 1 sequence TINYNINN; a SBO and loss of AFW at time zero. RCs (Release Categories) 6.1 and 6.2 assume direct impingement of entrained core debris on the containment free standing steel shell to obtain sidewall failure even with a coolable debris bed geometry. Sidewall failure 2 minutes after vessel failure results in early containment failure. Sidewall failures communicate with the shield building annulus and auxiliary building #4 mechanical penetration room. Release of fission products to the environment occurs following blow out panel failures; no annulus or auxiliary building decontamination factors are credited.
>
> RC's 6.3 and 6.4 include uncoolable debris beds; the debris is assumed to pool in the lower compartment against the outer concrete curb. Late containment failure occurs when sufficient concrete is eroded.
>
> RC's 6.1 and 6.3 include fission product scrubbing via containment spray and CACs.

13. At ER p. E-97, Table E.3-20, the "Frequency Vector" of Release Categories 6.1 through 6.4 are as follows:

6.1	4.4E-10	0.00%
6.2	3.3E-11	0.00%
6.3	4.5E-09	0.04%
6.4	3.1E-08	0.31%

14. At ER p. E-179, "Qualitative Screening of SAMA Candidates," it says (a year before discovery of the shield building cracking):

SR-01 Increase seismic ruggedness of plant components. Criterion D	Very Low Benefit	The Seismic Qualifications Utility Group (SQUG) previously identified the need for additional seismic restraints in the plant. These restraints have already been added.

Since the ER is dated 2010, it does not anticipate nor account for the shield building cracking.

15. NRC staff engineers have identified and predicted considerable damage to the shield building based upon their measurements, observations and various evidence. In an "Email from P. Hernandez, NRR to E. Sanchez-Santiago, RIII on Questions about Davis Besse Shield Building Report from DORL" dated November 4, 2011,[19] Pete Hernandez, assistant to the Lead PM [Project Manager] for Davis-Besse, responds to "C-CSS-099.20.054," a "calculation [of] the structural integrity of the SB [shield building]…considering the presence of an interfacial/ circumferential crack between the SB structural concrete shell (*i.e.*, the 30'' thick reinforced concrete SB) and each architectural flute shoulder (16 flute shoulders in total), as described in Attachment B." He states:

> This description makes me think that they are looking at a single crack going in a circle. From what I understood the crack is pervasive along the entire surface, spidering in all directions, similar to a pane of tempered glass breaking. The description in Attachment B addresses only the crack at the opening and assumes that the crack is right along the rebar line. The core bores have shown that the cracks are at different depths so this doesn't seem to capture the current situation. Throughout the calculation, the word Crack, singular, is used. They also mention that the extent of the crack is only 10'-12.' This seems to greatly downplay the issue.

Mr. Hernandez continues:

> At this point core bores of only the shoulders have been taken. So the only crack widths we are aware of are those in the shoulders, which are not being addressed. **How can an analysis be done on the structurally credited concrete if no data from that area, in the form of core bores, has been taken? Shouldn't the structural integrity of the shoulders be calculated as well?**
>
> This seems to say that they are just doing calculations for the new concrete that is and ignores the rest of the building altogether. Is that right?

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[19]http://pbadupws.nrc.gov/docs/ML1220/ML12200A192.pdf

> **This says to me, that they are ignoring the shoulders, if they are ignoring all that concrete, it seems to be the opposite of conservative for evaluating the mechanical loads.**
>
> C-CSS-099.20.055 - Objective or Purpose: The purpose of this calculation is to demonstrate that during a seismic event, with the development of the crack in the architectural flute shoulder, the capacity of the rebar(s) can still provide adequate anchorage thus prevent cracked concrete piece from falling, and therefore Seismic II/I condition can be maintained."

(Emphasis added).

Hernandez responds to this explanation as follows:

> **I think the greater concern is will the SB stay standing and not whether or not the decorative concrete will fall off.** Because the licensee has not performed core bores to see if there is cracking in the credited concrete, **do they have a basis to say that the structural concrete will maintain a Seismic II/I condition?**
>
> This use of singular terminology also discounts this calculation because it seems that they are looking at only 1 crack and 1 shoulder or 1 flute. **Because cracks have been found through multiple core bores, shouldn't the appropriate calculations account for the combined effects of cracks in all the shoulders** and not just one by opening and not just individually?
>
> From what I understand, **IR mapping is only an indicator, but must be validated by core bores**. Does basing all the calculations on a length of a 12 foot crack discount the calculations altogether, because **we have indications of cracks at distances greater than 12 feet.** This also seems to assume that there is only 1 crack and not *many as the core bores seem to prove*. Isn't **IR mapping only useful at a limited depth too**, so that using it to evaluate a 48" thick piece of concrete is not realistic?

(Emphasis added).

Mr. Hernandez' concerns are echoed by NRC staff engineer Abdul Sheikh, who in an "Email from A. Sheikh, NRR to E. Sanchez Santiago, RIII on Questions for the Conference Call,",[20] states:

> **If this assumption is correct only 3-4 inches of the concrete on the inside face can be used in the structural analysis.** In the response to the questions, the applicant stated that, 'Since we assume that outside reinforcement is to be treated ineffective in carrying any additional stress beyond 12.4 ksi, under accident thermal loads that may cause stresses in excess of what the rebar can carry (assumed 12.4 ksi), **the reinforcement is assumed to detach itself from the outer section of the shell.' These state-**

[20]http://pbadupws.nrc.gov/docs/ML1220/ML12200A213.pdf

ments seems *(sic)* to be contradictory. In addition, I am concerned that the concrete will fail in this region due to bending in this region even under small loads."

***** ***** ***** *****

Since we assume that outside reinforcement is to be treated ineffective in carrying any additional stress beyond 12.4 ksi, under accident thermal loads that may cause stresses in excess of what the rebar can carry (assumed 12.4 ksi), **the reinforcement is assumed to detach itself from the outer section of the shell.'**

Additionally, Mr. Sheikh states:

5. "The licensee justification for ignoring the dead (DL) and normal thermal (To) in calculation of rebars splice does not appear to be justified. The stresses due to dead load and thermal loads will be locked in the rebars and cannot be ignored."

6. The licensee considers the allowable stress in the rebar to be 60 ksi and ignores a phi factor (0.9) in his evaluation for lap splice. In addition, the licensee has not accounted for any additional uncertainty due the conditions.

7. I am not aware of any pull tests carried out with a crack in the plane of the rebar. Can the licensee provide any documentation for this statement.

8. The licensee is using numerous assumptions in his summary report and calculations that are not described in the UFSAR and ACI 318-63, and still calls it a design basis calculation. Can the licensee provide justification for this approach.

16. Abdul Sheikh, NRC engineer, found during his investigation of the cracking that "[The Davis-Besse] shield building has not been designed for containment accident pressure and temperature." NRC FOIA response, Appendix B, Document B/44, 12/13/11, Email from M. Galloway, NRR to A. Sheikh, NRR et al., RE: Davis-Besse Shield Building, 1 page.[21]

17. FENOC's contractor, Performance Improvement International, found that there are 14" deep cracks in the shield building walls. Performance Improvement International. PII's report, "Revised Root Cause Assessment: Davis-Besse Shield Building Laminar Cracking, Vol. 1," ML12138A037, pp. 92-93.[22] The presence of such cracking raises obvious questions about the ability of those locations on the shield building to fulfill their critical function of containing radiation in the event of a release to the annulus.

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[21] http://www.beyondnuclear.org/storage/B%2044.pdf

[22] "7. The exact depth of penetration used as input to the FE model varies. In "1D" areas, it is 4" or less. In "2D" areas, it is 14" or less. An inch one way or the other would shift the crack location about an inch - but a rigorous sensitivity study was not performed since we are not modeling growth rate."

18. FENOC admits in its February 28, 2012 Root Cause Analysis that examination of the entire shield building at Davis-Besse has not taken place and will not be completed until December 1, 2012:

> FENOC is developing a comprehensive engineering plan to re-establish the design and licensing basis conformance of the Shield Building. The plan is scheduled to be completed and issued by December 1, 2012. ***The plan will include a detailed structural analysis of the Shield Building and consider applicable effects.***

RAI AMP at 11/29 of .pdf. (Emphasis supplied). And not only does FENOC not have the intention of conducting a thorough investigation of the entire shield building, the RAI AMP foresees scant planned testing to be done during infrequent inspections over the coming decades, as, for example, a mere handful of core bores.

19. Davis-Besse has water problems inside the shield building. In RAI responses dated May 24, 2011 (ML 11151A90), the NRC staff had noted a "history of ground water infiltration into the annular space between the concrete shield building and steel containment:"

> During a 2011 AMP audit, NRC staff also reviewed documentation that: [I]ndicated the presence of standing water in the annulus sand pocket region. The standing water appears to be a recurring issue of ground water leakage and areas of corrosion were observed on the containment vessel. In addition, during the audit the staff reviewed photographs that indicate peeling of clear coat on the containment vessel annulus area, and degradation of the moisture barrier, concrete grout, and sealant in the annulus area that were installed in 2002-2003.

Id. at 47/280 of .pdf.

20. FENOC states in the February RCA that:

> The failure modes for the laminar cracking of the shield building concrete wall were primarily design related from about 40 years ago under a quality assurance program outside the control of FENOC. Therefore, the condition does not currently exist in other applicable programs /processes, equipment / systems, organizations, environments, and individuals.

21. FENOC has been required to commit to periodic inspections of the containment vessel for possible thinning of its wall. *See* ML11294A349. At p. 24/60 of this "Davis-Besse Commitment List,"[23] it states:

> Compare the ultrasonic test (UT) thickness readings to minimum ASME Code

[23] http://pbadupws.nrc.gov/docs/ML1129/ML11294A349.pdf

vessel thickness requirements and to the results obtained during previous UT examinations of the Containment Vessel. Determine the need for maintenance or repair of the Containment Vessel based on the results and evaluation of the examinations.

At 25/60 of .pdf:

Perform visual inspection of 100 percent of the accessible areas of the wetted outer surface of the Containment Vessel in the sand pocket region.

At 27/60 of .pdf:

Address the potential for borated water degradation of the steel containment vessel through the following actions:

Access the inside surface of the embedded steel containment. A core bore will be completed by the end of 2014 (Phase 1). If necessary, a second core bore will be completed by the end of 2020 (Phase 2). If there is evidence of the presence of borated water in contact with the steel containment vessel, conduct non-destructive testing (NDT) to determine what effect, if any, the borated water has had on the steel containment vessel. Based on the results of NDT, perform a study to determine the effect through the period of extended operation of any identified loss of thickness in the steel containment due to exposure to borated water.

22. Pursuant to 10 C.F.R. § 2.337(f), Intervenors request that ASLB take official notice of the facts alleged in their filings in this case respecting proposed Contention 5 on the cracking phenomena afflicting the Davis-Besse shield building, as well as the documents cited by Intervenors as the sources of those facts. *Cf. Yankee Atomic Elec. Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235 (1996).

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		September 14, 2012
.	)	

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CERTIFICATE OF SERVICE OF 'INTERVENORS' REPLY IN OPPOSITION TO FENOC MOTION FOR SUMMARY DISPOSITION OF CONTENTION NO. 4'

We hereby certify that a copy of the "INTERVENORS' REPLY IN OPPOSITION TO 'FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4 (SAMA ANALYSIS SOURCE TERMS)' and its accompanying "INTERVENORS' STATEMENT OF MATERIAL FACTS IN OPPOSITION TO 'FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4'" were sent by us to the following persons via electronic deposit filing with the Commission's EIE system on the 14th day of September, 2012:

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