December 9, 2016

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE SECRETARY OF THE COMMISSION

In the Matter of:	:
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ENTERGY NUCLEAR FITZPATRICK, LLC & ENTERGY NUCLEAR OPERATIONS, INC.	:
(James A. FitzPatrick Nuclear Power Plant; Docket No. 50-333-EA)	:

DECLARATION OF MARK LEYSE TO SUPPORT BEYOND NUCLEAR AND ALLIANCE FOR A GREEN ECONOMY'S PETITION TO REQUEST A HEARING AND LEAVE TO INTERVENE ON ENTERGY'S REQUESTS FOR AN EXTENSION TO COMPLY WITH NRC ORDERS EA-12-049, EA-12-051 AND EA-13-109 REQUIREMENTS FOR THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT

I. Introduction

I, Mark Leyse, declare that the following statements are true and correct to the best of my knowledge. I am *sui juris*. I am over the age of 18 years old.

1. Beyond Nuclear and Alliance for a Green Economy (AGREE) (or "Petitioners") have contracted my services to supply technical analysis and comments in support of their request for a public hearing and leave to intervene in the matter of Entergy's James A. FitzPatrick Nuclear Power Plant (FitzPatrick) "Request for Extension to Comply" with U.S. Nuclear Regulatory Commission (NRC) Orders EA-12-049, EA-12-051 and EA-13-109.
2. I studied nuclear engineering at the University of Wisconsin at Madison from 1979 to 1980. I have a Bachelor of Arts in Fine Arts from the University of California at Berkeley, completed in 1985.

3. I have worked as a nuclear safety consultant since 2010. I have worked for New England Coalition on Nuclear Pollution, Riverkeeper, and Natural Resources Defense Council. As a nuclear safety consultant, I have written 10 C.F.R. § 2.206 enforcement action petitions, a 10 C.F.R. § 2.802 petition for rulemaking, and reports.

4. For Natural Resources Defense Council, I wrote a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-103,[1] requesting post-Fukushima Daiichi accident revisions to 10 C.F.R. § 50.44, "Combustible Gas Control for Nuclear Power Reactors." I also wrote three reports: 1) "Preventing Hydrogen Explosions In Severe Nuclear Accidents: Unresolved Safety Issues Involving Hydrogen Generation And Mitigation;"[2] 2) "Preventing Hydrogen Explosions at Indian Point Nuclear Plant: Fact versus Industry Spin;"[3] and 3) "Post-Fukushima Hardened Vents with High-Capacity Filters for BWR Mark Is and Mark IIs."[4]

5. On March 15, 2007, I submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-84,[5] to the NRC. PRM-50-84 was summarized briefly in American Nuclear Society's *Nuclear News*'s June 2007 issue[6] and commented on and deemed "a well-documented justification

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[1] Mark Leyse, PRM-50-103, October 14, 2011, (ADAMS Accession No. ML11301A094).

[2] Mark Leyse, Author, and Christopher Paine, Contributing Editor, "Preventing Hydrogen Explosions In Severe Nuclear Accidents: Unresolved Safety Issues Involving Hydrogen Generation And Mitigation," NRDC Report, R:14-02-B, March 2014. (available at: https://www.nrdc.org/sites/default/files/hydrogen-generation-safety-report.pdf : last visited on 08/28/16)

[3] Mark Leyse and Christopher Paine, "Preventing Hydrogen Explosions at Indian Point Nuclear Plant: Fact versus Industry Spin," NRDC IB: 13-01-F, February 2013. (available at: https://www.nrdc.org/sites/default/files/IndianPoint-hydrogen-explosions-IB.pdf : last visited on 08/28/16)

[4] Mark Leyse, "Post-Fukushima Hardened Vents with High-Capacity Filters for BWR Mark Is and Mark IIs," Report for NRDC, July 2012, (ADAMS Accession No. ML12254A865).

[5] Mark Leyse, PRM-50-84, March 15, 2007 (ADAMS Accession No. ML070871368).

[6] American Nuclear Society, *Nuclear News*, June 2007, p. 64.

for…recommended changes to the [NRC's] regulations"[7] by the Union of Concerned Scientists ("UCS").

6. PRM-50-84 requested that NRC make new regulations: 1) to require licensees to operate light water reactors under conditions that effectively limit the thickness of crud (corrosion products) and/or oxide layers on fuel cladding, in order to help ensure compliance with 10 C.F.R. § 50.46(b) emergency core cooling system ("ECCS") acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.
7. Additionally, PRM-50-84 requested that NRC amend Appendix K to Part 50—ECCS Evaluation Models I(A)(1), *The Initial Stored Energy in the Fuel*, to require that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated loss-of-coolant accident ("LOCA") be calculated by factoring in the role that the thermal resistance of crud and/or oxide layers on cladding plays in increasing the stored energy in the fuel. PRM-50-84 also requested that these same requirements apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations. (Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.)
8. In 2008, the NRC decided to consider the safety issues raised in PRM-50-84 in its rulemaking process.[8] And in 2009, the NRC published "Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of a proposed rulemaking,

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[7] David Lochbaum, Union of Concerned Scientists, "Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84)," July 31, 2007, (ADAMS Accession No. ML072130342), p. 2.
[8] NRC, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," Docket No. PRM–50–84; NRC–2007–0013, Federal Register, Vol. 73, No. 228, November 25, 2008, pp. 71564-71569.

addressing four objectives: the fourth being the issues raised in PRM-50-84.[9] In 2012, the NRC Commissioners voted unanimously to approve a proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that is partly based on the safety issues I raised in PRM-50-84.[10]

9. With Rui Hu and Professor Mujid S. Kazimi of the Massachusetts Institute of Technology, I coauthored a paper, "Considering the Thermal Resistance of Crud in LOCA Analysis," that was presented at the American Nuclear Society's 2009 Winter Meeting.[11]

10. On November 17, 2009, I submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-93.[12] PRM-50-93 requests that NRC make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments; and 2) to stipulate minimum allowable core reflood rates, in the event of a LOCA.

11. Additionally, PRM-50-93 requests that NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments. These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.

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[9] NRC, "Performance-Based Emergency Core Cooling System Acceptance Criteria," NRC–2008–0332, Federal Register, Vol. 74, No. 155, August 13, 2009, pp. 40765-40776.

[10] NRC, Commission Voting Record, Decision Item: SECY-12-0034, Proposed Rulemaking—10 CFR 50.46(c): Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42), January 7, 2013, (ADAMS Accession No. ML13008A368).

[11] Rui Hu, Mujid S. Kazimi, Mark Leyse, "Considering the Thermal Resistance of Crud in LOCA Analysis," American Nuclear Society, 2009 Winter Meeting, Washington, D.C., November 15-19, 2009.

[12] Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No. ML093290250).

12. PRM-50-93 was discussed briefly in the American Nuclear Society's March 2010 issue of *Nuclear News*.[13] PRM-50-93 was also commented on by UCS.

13. Regarding PRM-50-93, UCS states:

> In our opinion, [PRM-50-93] addresses a genuine safety problem. We believe the NRC should embark on a rulemaking process based on this petition. We are confident that this process would culminate in revised regulations—perhaps not precisely the ones proposed [in PRM-50-93] but ones that would adequately resolve the issues…meticulously identified [in PRM-50-93]—that would better ensure safety in event of a loss of coolant accident.[14]

14. On October 27, 2010, the NRC published in the Federal Register that it had determined that a 10 C.F.R. § 2.206 petition, dated June 7, 2010, I wrote and submitted on behalf of New England Coalition—requesting that the NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature of VYNPS—meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802.[15] The NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95.[16] PRM-50-95 was discussed briefly in the July 30, 2010 issue of Platts's *Inside NRC*.[17]

15. My expert opinions and comments in this declaration are based both on my professional experience and on my review of relevant aspects of Entergy's request for extension to comply with NRC Order EA-13-109—"Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions"—at FitzPatrick.

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[13] American Nuclear Society, *Nuclear News*, March 2010, p. 36.

[14] David Lochbaum, Union of Concerned Scientists, "Comments Submitted by the Union of Concerned Scientists on the Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-93)," April 27, 2010, (ADAMS Accession No. ML101180175), p. 1.

[15] Federal Register, Vol. 75, No. 207, Notice of consolidation of petitions for rulemaking and re-opening of comment period, October 27, 2010, pp. 66007-66008.

[16] Mark Leyse, PRM-50-95, June 7, 2010, (ADAMS Accession No. ML101610121).

[17] Suzanne McElligott, *Inside NRC*, July 30, 2010.

II. Background

Hydrogen Generation in a Boiling Water Reactor Mark I Severe Accident: Rates and Quantities

16. In a boiling water reactor (BWR) severe accident, hydrogen generation could occur at rates from 0.1 to 5.0 kilograms (kg) per second.[18] An OECD Nuclear Energy Agency report states, a "rapid initial [hydrogen]-source occurs in practically all severe accident scenarios because the large chemical heat release of the [zirconium]-steam reaction causes a fast self-accelerating temperature excursion during which initially large surfaces and masses of reaction partners are available."[19]

17. If an overheated BWR reactor core were re-flooded with water, up to 300.0 kg of hydrogen could be generated in 60 seconds.[20] In this scenario, according to one report, between 5.0 and 10.0 kg of hydrogen could be generated per second.[21] (It is noteworthy that in the pressurized water reactor (PWR) Three Mile Island Unit 2 (TMI-2) accident, re-flooding of the uncovered reactor core by the emergency core cooling system (ECCS) caused a spike in the hydrogen generation rates; it has been estimated that approximately 33 percent of all the hydrogen produced in the TMI-2 accident occurred during reflooding.[22])

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[18] E. Bachellerie *et al*., "Generic approach for designing and implementing a passive autocatalytic recombiner PAR-system in nuclear power plant containments," Nuclear Engineering and Design, Volume 221, Nos. 1-3, April 2003, p. 158.
[19] OECD Nuclear Energy Agency, "State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI/R(2000)7, August 2000, (ADAMS Accession No: ML031340619), p. 6.38.
[20] E. Bachellerie *et al*., "Generic approach for designing and implementing a passive autocatalytic recombiner PAR-system in nuclear power plant containments," Nuclear Engineering and Design, Volume 221, Nos. 1-3, April 2003, p. 158.
[21] J. Starflinger, "Assessment of In-Vessel Hydrogen Sources," in *Projekt Nukleare Sicherheitsforschung: Jahresbericht 1999*, (Karlsruhe: Forschungszentrum Karlsruhe, FZKA-6480, 2000).
[22] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," Report by an OECD NEA Group of Experts, October 2000, p. 13.

18. The total quantity of hydrogen that could be generated in a severe accident at a BWR Mark I is extremely large. Considering hydrogen generated only from the oxidation of zirconium: if the total amount of the zirconium in a typical BWR core, approximately 76,000 kg, were to chemically react with steam, this would generate approximately 3,360 kg of hydrogen. (For PWRs, by contrast: considering hydrogen generated only from the oxidation of zirconium: if the total amount of the zirconium in a typical PWR core, approximately 26,000 kg, were to chemically react with steam, this would generate approximately 1,150 kg of hydrogen.)[23]

19. Large BWR cores typically have about a 58-percent greater initial uranium mass than large PWR cores, and this larger mass is divided into approximately 45 percent more fuel rods than in a PWR. However, these differences alone do not account for the fact that BWR cores have almost three times the mass of zirconium in their cores than PWRs.[24] BWR cores have significantly more zirconium mainly because, unlike PWRs, BWR fuel assemblies have "channel boxes" surrounding the fuel rods. The mass of each BWR assembly channel box is greater than 100 kg.[25]

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[23] International Atomic Energy Agency (IAEA), "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 10.

[24] These estimates are based on that fact that large BWR cores and large PWR cores have up to approximately 800 and 200 fuel assemblies, respectively (see NRC, "Boiling Water reactors" (located at: http://www. nrc.gov/reactors/bwrs.html) and NRC, "Pressurized Water Reactors" (located at: http://www.nrc.gov/reactors/pwrs.html)). Recent designs of BWR and PWR fuel assemblies have up to approximately 190 kg and 480 kg of initial uranium mass per assembly, respectively (see NRC, "Certificate of Compliance No. 1014," Appendix B, "Approved Contents and Design Features for the Hi-Storm 100 Cask System," (available at ADAMS No: ML13351A189), pp. 2.39, 2.44). Hence, large BWR cores and large PWR cores are estimated to have a total of approximately 152,000 kg and 96,000 kg of initial uranium mass, respectively. Recent designs of BWR and PWR fuel assemblies have on the order of 96 and 264 fuel rods per assembly, respectively. Hence BWR and PWR cores can have up to approximately 76,800 and 52,800 fuel rods per core, respectively; so BWR cores can have approximately 45 percent more fuel rods. NRC, "Certificate of Compliance No. 1014," Appendix B, "Approved Contents and Design Features for the Hi-Storm 100 Cask System," (available at ADAMS No: ML13351A189), pp. 2.39, 2.44.

[25] Yasuo Hirose *et al*., "An Alternative Process to Immobilize Intermediate Wastes from LWR Fuel Reprocessing," WM'99 Conference, February 28-March 4, 1999.

20. The total quantity of hydrogen generated in a severe accident can vary widely: The Fukushima Daiichi accident, which resulted in the meltdowns of three BWR Mark Is, likely generated more than 3,000 kg of hydrogen per affected unit. In a severe accident, hydrogen would also (in addition to that generated by the zirconium-steam reaction) be generated within the reactor vessel from the oxidation of non-zirconium materials: metallic structures and boron carbide (in BWR cores).[26] In the PWR TMI-2 accident, the oxidation of steel accounted for approximately 10 percent to 15 percent of the total hydrogen generation.[27]

21. In sum, in a BWR severe accident, the in-core "hydrogen sources are the zirconium-steam reaction, the boron carbide-steam reaction, the uranium-steam reaction, the steel-steam reaction, the…zirconium-steam reaction [from the] reflood[ing] of an overheated core, [and] the relocation of zirconium-rich mixtures."[28]

22. Regarding the behavior of boron carbide in the event of the reflooding of an overheated reactor core, it is important to consider that "a limited number of reflooding experiments with fuel rod bundles and B_4C [boron carbide] control blades have shown that the oxidation of B_4C may be an important contributor to the production of hydrogen and other combustible gases during reflood. In addition, the methane generated by the oxidation of B_4C may react with the iodine

[26] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 6.
[27] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/ CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 15.
[28] E. Bachellerie et al., "Generic approach for designing and implementing a passive autocatalytic recombiner PAR-system in nuclear power plant containments," Nuclear Engineering and Design, Volume 221, Nos. 1-3, April 2003, p. 158.

released from the fuel to form organic iodines and thus influence the source term considerably."[29]

23. In a case in which the molten core penetrated the reactor vessel, hydrogen would be generated from the oxidation of metallic material (chromium, iron, and any remaining zirconium) during direct containment heating and also from interaction of the molten core with concrete (out of which containment floors are made).[30] A safety study for the PWRs at Indian Point discusses a case in which interaction of a molten core with a concrete containment floor would generate more than 2,721.5 kg of hydrogen.[31] There would also be large quantities of hydrogen generated if the same scenario were to occur in a severe accident at a BWR Mark I.
24. If a molten core interacted with concrete, carbon monoxide (which, like hydrogen, is a combustible gas) would also be generated. Depending on different accident scenarios, concrete types, and geometrical factors affecting the molten core-concrete interaction, the quantities of carbon monoxide generated could vary greatly; concentrations could differ by up to several volume percent in the containment.[32] (The volume percent of the carbon monoxide in the containment is the volume of the carbon monoxide in the containment divided by the volume of the containment multiplied by 100.)

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[29] IAEA, "Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants," Safety Reports Series No.56, 2008, p. 12.
[30] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 6.
[31] Power Authority of the State of New York, Consolidated Edison Company of New York, "Indian Point Probabilistic Safety Study," Vol. 8, 1982, (available at ADAMS No: ML102520201), p. 4.3-10.
[32] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 47.

NRC and Industry Computer Safety Models Under-Predict Severe Accident Hydrogen Generation Rates

25. A 2001 OECD Nuclear Energy Agency report advises that high hydrogen generation rates "must be taken into account in risk analysis and in the design of hydrogen mitigation systems." However, the same report notes that computer safety models used by regulators under-predicted the actual rates of hydrogen generation that occurred in two sets of experiments simulating severe accidents: the CORA tests and LOFT LP-FP-2.[33] (The CORA and LOFT LP-FP-2 experiments were conducted to investigate accidents that lead to a meltdown of the reactor core. LOFT LP-FP-2 was conducted with an actual nuclear reactor, 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR." LOFT LP-FP-2 was an actual core meltdown—the most realistic severe accident experiment conducted to date; it combined decay heating, severe fuel damage, and the quenching of zirconium fuel cladding with water.[34]) Computer safety models also failed to predict hydrogen generation in the initial QUENCH facility experiments.[35] This indicates that computer safety models also under-predict the hydrogen generation rates that would occur in severe accidents.[36]

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[33] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/ CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9.

[34] T.J. Haste *et al.*, Organisation for Economic Co-Operation and Development, "Degraded Core Quench: A Status report," August 1996, p. 13.

[35] L.J. Ott, Oak Ridge National Laboratory, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," 1997, p. 4.

[36] LOFT LP-FP-2 was conducted in the Loss-of-Fluid Test Facility at Idaho National Engineering Laboratory in July 1985. The CORA and QUENCH tests were conducted at Karlsruhe Institute of Technology in Germany in the 1980s and 1990s.

26. A 1997 Oak Ridge National Laboratory (ORNL) report states that hydrogen generation in severe accidents can be divided into two separate phases: 1) a phase that runs from when the fuel cladding is still intact through the initial melting of the fuel cladding, which accounts for about 25.0 percent of the total hydrogen produced; and 2) a phase after the initial melting of the fuel cladding, in which there is additional melting, relocation, and the formation of uranium-zirconium-oxygen blockages, which accounts for about 75.0 percent of the total hydrogen generated (as indicated in analyses of the BWR CORA-28 and -33 tests).[37]

27. According to the 1997 ORNL report, computer safety models predict hydrogen generation rates "reasonably well" for the first phase, in which the fuel cladding remains intact, but predict hydrogen generation rates for the second phase "much less robustly." The 1997 ORNL report stresses that it is obvious that computer safety models need to accurately predict hydrogen generation rates when the fuel cladding is no longer intact, especially because most of the hydrogen generation occurs in that phase.[38]

28. A 2011 International Atomic Energy Agency (IAEA) report states that computer safety models under-predict the rates of hydrogen generation that would occur during a reflooding of an overheated reactor core.[39] The report cautions that, in different scenarios, reflooding could cause hydrogen generation rates to vary to a large degree and that predictions need to consider the possible range of outcomes in order to help prepare for severe accident hydrogen risk. In the

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[37] L.J. Ott, Oak Ridge National Laboratory, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," 1997, p. 4.
[38] L.J. Ott, Oak Ridge National Laboratory, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," 1997, p. 4.
[39] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 14.

BWR CORA-17 test, which simulated the reflooding and quenching of an overheated core, approximately 90 percent of the hydrogen generation occurred during reflooding.[40]

29. And also on difficulties of modeling the reflooding of an overheated reactor core, a 2008 IAEA report on severe accident analysis states: "The reflooding of a hot core is well understood with respect to thermohydraulic behaviour. In contrast to it, there are still open questions with respect to the hydrogen production (*e.g.*, loss of protective oxide shell) and additional R&D is necessary. The more the core is degraded, the less reliable data are available with respect to the thermohydraulics. Of course, if the thermohydraulics during reflood of a degraded core is not sufficiently known, *one cannot expect a reliable simulation of the related hydrogen production*"[41] [emphasis added].

30. Unfortunately, recent reports do not explicitly state the extent that computer safety models under-predict hydrogen generation rates during the reflooding and quenching of an overheated core—*i.e.*, a percentage value of the under-prediction has not been provided. However, a paper from a 2008 European meeting states that "the total hydrogen mass produced under reflooding remains *highly underestimated* in CORA-13 and LOFT LP-FP-2 experiments" [emphasis added]. In fact, regarding recent computer simulations of LOFT LP-FP-2, conducted using the ASTEC severe accident computer safety model, the same paper states: "High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test

[40] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," Report by an OECD NEA Group of Experts, October 2000, p. 210.
[41] IAEA, "Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants," Safety Reports Series No.56, 2008, p. 66.

during reflood were not reproduced by ASTEC *due to [a] lack of adequate modeling*."[42] [emphasis added].

31. Despite these reports dating back to 1997, the NRC's 2011 Near-Term Task Force report on insights from the Fukushima Daiichi accident failed to mention, much less discuss, the fact that the NRC's computer safety models—such as the widely used MELCOR code developed by Sandia National Laboratories—under-predict the hydrogen generation rates that occur in severe accidents. By overlooking the deficiencies of computer safety models, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[43] When hydrogen generation rates are under-predicted, hydrogen mitigation systems are not likely to be designed so that they could handle the generation rates that would occur in actual severe accidents.

32. Evidence indicates that the licensee of FitzPatrick, Entergy, has **not** conducted adequate, conservative computer simulations of the hydrogen generation rates that would occur in the event of a severe accident. Like everyone else's computer simulations, Entergy's simulations under-predict hydrogen generation rates. For this reason, among others, Entergy cannot guarantee that FitzPatrick's current decades-old wetwell vent would be effective in managing explosive hydrogen gas in the event of a severe accident.

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[42] G. Bandini, "Progress of ASTEC Validation on Circuit Thermal-Hydraulics and Core Degradation," The 3rd European Review Meeting on Severe Accident Research (ERMSAR-2008) Nesseber, Bulgaria, 23-25 September 2008, pp. 12, 14.

[43] Charles Miller *et al.*, NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: the Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident," SECY-11-0093, July 12, 2011, (available at ADAMS No: ML111861807), p. 3.

Why Boiling Water Reactor Mark I Primary Containments have Been Backfitted with Hardened Vents

33. The NRC's 2011 Near-Term Task Force report on insights from the Fukushima Daiichi accident states that NRC reports from 1975[44] and 1990[45] both concluded that in the event of a severe accident, boiling water reactor (BWR) Mark I primary containments have "a relatively high containment failure probability," because BWR Mark I primary containments have smaller volumes when compared to PWR containments[46]—about one-eighth the volume of PWR large dry containments. (BWR Mark I primary containments have a volume of approximately $0.28 \times 10^6\ ft^3$; pressurized water reactor (PWR) large dry containments have a volume of approximately $2.2 \times 10^6\ ft^3$.[47])

34. A BWR Mark I primary containment is comprised of a drywell, shaped like an inverted light bulb, and a wetwell (also termed "torus"), shaped like a doughnut. The wetwell is half filled with water (typically about 790,000 gallons[48])—the suppression pool.

35. In a severe accident, the water pumped into the reactor core to cool the fuel rods would heat up and produce thousands of kilograms of steam, which would enter the primary containment. The water in the suppression pool is intended to condense the steam and help absorb the heat released by the accident to reduce the pressure in the primary containment. Without the

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[44] NRC, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," NUREG-75-014, WASH-1400, October 1975.

[45] NRC, "Severe Accident Risks: An Assessment or Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

[46] Charles Miller, *et al.*, NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," SECY-11-0093, July 12, 2011, (available at ADAMS No: ML111861807), p. 39.

[47] M. F. Hessheimer, *et al.*, Sandia National Laboratories, "Containment Integrity Research at Sandia National Laboratories: An Overview," NUREG/CR-6906, July 2006, (available at ADAMS No: ML062440075), p. 24.

[48] NRC, "NRC Information Notice 2006-01: Torus Cracking in a BWR Mark I Containment," January 12, 2006, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML053060311, Attachment 1, p. 1.

condensation of the steam in the suppression pool, the relatively small primary containments of a BWR Mark I (often termed the "pressure suppression containment") would fail from becoming over-pressurized.

36. In a BWR severe accident, hundreds of kilograms of non-condensable hydrogen gas would also be produced (up to over 3000 kg[49])—at rates as high as between 5.0 and 10.0 kg per second, if there were a reflooding of an overheated reactor core[50]—which would increase the internal pressure of the primary containment. If enough hydrogen were produced, the containment could fail from becoming over-pressurized. To help address this problem, in 1989, the NRC sent Generic Letter 89-16, "Installation of a Hardened Wetwell Vent" to all the owners of BWR Mark Is, *recommending*[51] that hardened vents be installed in BWR Mark Is.[52] Hardened wetwell vents are intended to depressurize and remove decay heat from BWR Mark I primary containments; and the water in the wetwell is intended to have the function of helping scrub the fission products (excluding noble gases) that had entered the containment.[53]

[49] International Atomic Energy Agency (IAEA), "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 10.
[50] J. Starflinger, "Assessment of In-Vessel Hydrogen Sources," in "Projekt Nukleare Sicherheitsforschung: Jahresbericht 1999," Forschungszentrum Karlsruhe, FZKA-6480, 2000.
[51] Generic Letter 89-16 states that "the Commission has directed the [NRC] staff to approve installation of a hardened vent under the provisions of 10 CFR 50.59 ["Changes, Tests, and Experiments"] for licensees, who on their own initiative, elect to incorporate this plant improvement;" see NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989, p. 1.
[52] NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989, p. 1.
[53] R. Jack Dallman, *et al*., "Filtered Venting Considerations in the United States," May 17-18, 1988, CSNI Specialists Meeting on Filtered Vented Containment Systems, Paris France, p. 5.

There is No Guarantee that FitzPatrick's Current Decades-Old *Un*reliable Wetwell Vent Would Prevent a Hydrogen Explosion in the Event of a Severe Accident

37. In a report on an April 2011 post-Fukushima accident inspection at FitzPatrick, when discussing plant operators' procedures for responding in the event of a station blackout, the NRC stated:

> The licensee identified an apparent beyond design and licensing basis vulnerability, in that current procedures *do not address hydrogen considerations during primary containment venting*. This issue was documented in CR-JAF-2011-01529. As an immediate corrective action, the licensee revised TSG-9 to provide *a caution for operators to consider the presence of hydrogen* [emphasis added].
>
> The inspectors identified a beyond design and licensing bases vulnerability, in that FitzPatrick's current licensing basis did not require the plant to have a primary containment torus air space hardened vent system as part of their Mark I containment improvement program. The current licensed configuration is a hard pipe from primary containment to the suction of the standby gas treatment system, which is located outside the reactor building in an adjacent building. The NRC has established an agency task force to conduct a near term evaluation of the need for agency actions, which includes containment venting, following the events in Japan.[54]

38. The defective, antiquated BWR Mark I design performed poorly in the Fukushima Daiichi accident. In the accident, three BWR Mark I reactors melted down, each generating hundreds of kilograms of explosive hydrogen gas. Hydrogen **leaked** from the primary containments into the reactor buildings, where it accumulated and detonated at different times, destroying three reactor buildings, which released large quantities of harmful radioactive material into the environment. A wide area was contaminated, prompting the evacuation of about 90,000 people. Fukushima Daiichi's decades-old hardened vents **did not** prevent hydrogen from entering BWR Mark I secondary containments and detonating.

[54] NRC, Inspection Report for James A. FitzPatrick Nuclear Power Plant, Docket No: 50-333, April 18 through April 29, 2011, Inspection Report 05000333/2011008 (available at ADAMS No: ML111330455), pp. 8-9.

39. In the event of a severe accident at FitzPatrick, hundreds of kilograms of explosive hydrogen gas would be generated. There is no guarantee that FitzPatrick's current decades-old wetwell vent would prevent the hydrogen from exploding and destroying the reactor building, releasing large quantities of harmful radioactive material into the environment, as occurred in the Fukushima Daiichi accident.

40. Given the vulnerabilities of BWR Mark I primary containments—their relatively small volumes and dependence on suppression pools, which do not mitigate hydrogen—it is essential that a hardened containment vent be designed so that it would be reliable in a wide range of different severe accident scenarios. However, there is no guarantee that FitzPatrick's current decades-old wetwell vent would be reliable in a wide range of different severe accident scenarios.

41. There is no guarantee that FitzPatrick's current decades-old wetwell vent would perform adequately in severe accident scenarios in which there were rapid containment-pressure increases. It is pertinent that a 1983 Sandia National Laboratories manual cautions that "it may be difficult to design vents that can handle the rapid transients involved [in a severe accident]."[55] The scenario in which there would be the reflooding of an overheated reactor core, with the possible generation of between 5.0 and 10.0 kg of hydrogen per second,[56] would be a rapid transient.

42. It is also important to consider that "[t]he rapid heating and cooling [of the reactor core], and the associated high hydrogen production rates [that occur during the reflooding of an overheated core], can also affect other processes in the reactor system. Fission product release rates can increase rapidly due to the release of fission products on the grain boundaries during

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[55] Allen L. Camp, *et al*., Sandia National Laboratories, "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, August 1983, p. 2-66.

[56] J. Starflinger, "Assessment of In-Vessel Hydrogen Sources," in *Projekt Nukleare Sicherheitsforschung: Jahresbericht 1999*, (Karlsruhe: Forschungszentrum Karlsruhe, FZKA-6480, 2000).

quench, *and the pressure in the system can also be increased because of the additional steam and hydrogen produced during quenching*"[57] [emphasis added].

43. FitzPatrick's decades-old "current licensed configuration is a hard pipe from primary containment to the suction of the standby gas treatment system, which is located outside the reactor building in an adjacent building."[58] It hardly inspires confidence in FitzPatrick's "current licensed configuration" that in a post-Fukushima accident inspection "[t]he licensee identified an apparent beyond design and licensing basis vulnerability, in that current procedures do not address hydrogen considerations during primary containment venting. This issue was documented in CR-JAF-2011-01529. As an immediate corrective action, the licensee revised TSG-9 to provide a caution for operators to consider the presence of hydrogen."[59]

44. The licensee of FitzPatrick, Entergy, has **not** demonstrated that FitzPatrick's current decades-old wetwell vent (or "current licensed configuration") would perform adequately in severe accident scenarios in which there were rapid containment-pressure increases, rapid transients, or circumstances in which there would be the generation of between 5.0 and 10.0 kg of hydrogen per second.

45. The licensee of FitzPatrick, Entergy, has **not** demonstrated that FitzPatrick's current decades-old wetwell vent would be able to perform adequately in *certain* severe accident scenarios in which there were rapid containment-pressure increases. It is pertinent that a 1993 OECD Nuclear Energy Agency paper, "Non-Condensable Gases in Boiling Water Reactors," discusses

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[57] IAEA, "Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants," Safety Reports Series No.56, 2008, p. 8.

[58] NRC, Inspection Report for James A. FitzPatrick Nuclear Power Plant, Docket No: 50-333, April 18 through April 29, 2011, Inspection Report 05000333/2011008 (available at ADAMS No: ML111330455), pp. 8-9.

[59] NRC, Inspection Report for James A. FitzPatrick Nuclear Power Plant, Docket No: 50-333, April 18 through April 29, 2011, Inspection Report 05000333/2011008 (available at ADAMS No: ML111330455), p. 8.

severe accident scenarios in which there would be a rapid accumulation of steam in the drywell and non-condensable gas accumulation (nitrogen[60] and hydrogen) in the wetwell; in such scenarios, the primary containment's pressure could *rapidly* increase "up to the venting and failure levels."[61] "Non-Condensable Gases in Boiling Water Reactors" states that for a 3300 megawatt thermal BWR Mark I, in scenarios in which hydrogen would be produced from a zirconium-steam reaction of 40 percent, 70 percent, and 100 percent of all the zirconium in the reactor core,[62] if the total quantity of non-condensable gases (including nitrogen) were to accumulate in the wetwell, the primary containment's pressure would increase up to 107 pounds per square inch (psi), 161 psi, and 215 psi, respectively.[63]

<u>Computer Safety Models have Limitations in Predicting the Hydrogen Distribution and Steam Condensation that Would Occur in the Containment and Elsewhere in Different Severe Accident Scenarios</u>

46. In a September 2011 meeting of the Advisory Committee on Reactor Safeguards, Dana Powers, senior scientist at Sandia National Laboratories, expressed concern over the fact that hydrogen detonations occurred in the Fukushima Daiichi accident and stated that in experiments, "detonations are…extraordinarily hard to get."[64] Consequently, computer safety models (codes) derived from these experiments have limitations in predicting the hydrogen distribution and

[60] Nitrogen is used to inert BWR Mark I and Mark II primary containments.
[61] T. Okkonen, Nuclear Energy Agency OECD, "Non-Condensable Gases in Boiling Water Reactors," NEA/CSNI/R(94)7, May 1993, pp. 4-5.
[62] Equivalent to the quantity of hydrogen that would be produced from a zirconium-steam reaction of 72 percent, 126 percent, and 180 percent, respectively, of the active fuel cladding length.
[63] T. Okkonen, "Non-Condensable Gases in Boiling Water Reactors," p. 6.
[64] Advisory Committee on Reactor Safeguards, 586th Meeting, September 8, 2011, (ADAMS Accession No. ML11256A117), p. 95.

steam condensation that would occur in the containment and elsewhere in different severe accident scenarios.

47. A number of hydrogen combustion experiments have been conducted at Sandia National Laboratories; for example, such experiments were conducted in the 1980s at the FLAME facility—a rectangular channel 100 feet long, 8 feet high, and 6 feet wide.[65] Most experiments investigating the lower hydrogen concentration limits at which deflagration-to-detonation transition occurs have been conducted in detonation tubes; such tubes have been 39 to 70 feet long and about 11 to 17 inches in diameter.[66] Such experiments do not necessarily replicate the conditions that would occur in a BWR Mark I accident.

48. A 2007 OECD Nuclear Energy Agency report states, "Further work in code development…and code user training, supported by suitable complex experiments, is necessary to achieve more accurate predictive capabilities for containment thermal hydraulics and atmospheric gas/steam distribution. As a result of the code assessment, the modeling of the following three phenomena appeared to be the major issues: *condensation, gas density stratification, and jet injection*" [emphasis added].[67] This quote pertains to containments but it is pertinent to other areas of a nuclear plant where hydrogen could enter in the event of a severe accident.

49. Computer safety models also have limitations in predicting the phenomenon of hydrogen deflagrations transitioning into detonations; as well as the maximum pressure loads the containment (or other structures) would incur from detonations, in different scenarios. The

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65 M.P. Sherman *et al*., Sandia National Laboratories, "FLAME Facility: The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale," NUREG/CR-5275, (ADAMS Accession No. ML071700076), abstract.

66 OECD Nuclear Energy Agency, "State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI/R(2000)7, August 2000, (ADAMS Accession No. ML031340619), p. 3.5.

67 OECD Nuclear Energy Agency, "International Standard Problem ISP-47 on Containment Thermal Hydraulics: Final Report," NEA/CSNI/R(2007)10, September 2007, p. 7.

Fukushima Daiichi accident demonstrated that the NRC needs to conduct more realistic hydrogen combustion experiments—perhaps in facilities on the same scale as actual reactor containments, at elevated temperatures and with the large quantities of hydrogen that are produced in severe accidents.

50. Entergy referred to a decades-old safety evaluation report on FitzPatrick's hardened wetwell vent when it requested an extension for complying with NRC Order EA-13-109. In its request, Entergy stated:

> During the requested period of extension to comply with Phase 1 of the Order, the existing [FitzPatrick] containment vent system used to address GL 89-16,[68] as documented in the NRC Safety Evaluation (ML13015A634[69]), will continue to provide defense-in-depth measures and enhanced plant capability to mitigate the consequences of a beyond-design-basis external event and to prevent severe accident conditions in accordance with existing Emergency Operating Procedures.[70]

51. The safety evaluation report Entergy referred to is from September 1992. A 2007 OECD Nuclear Energy Agency report states that for simulating severe accidents, modeling of steam condensation, gas density stratification, and jet injection needs to be improved.[71] Clearly, the decades-old safety evaluation report did not realistically evaluate what would actually occur in the event of a **real** severe accident at FitzPatrick. Furthermore, the Fukushima Daiichi accident provided empirical evidence that there is **no guarantee** that decades-old hardened vents like

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[68] NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989.

[69] NRC, "Safety Evaluation Report: Hardened Wetwell Vent Capability at the James A. FitzPatrick Nuclear Power Plant," (ADAMS Accession No. ML13015A634), September 28, 1992.

[70] Entergy, "Request for Extension to Comply with NRC Order EA-13-109 at James A. FitzPatrick Nuclear Power Plant: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," License No. DPR-059, JAFP-16-0148, September 8, 2016, Attachment, p. 2.

[71] OECD Nuclear Energy Agency, "International Standard Problem ISP-47 on Containment Thermal Hydraulics: Final Report," NEA/CSNI/R(2007)10, September 2007, p. 7.

FitzPatrick's would be able to prevent hydrogen from entering BWR Mark I secondary containments and detonating, in the event of a severe accident.

Entergy's Request for Extension to Comply with NRC Order EA-13-109 at FitzPatrick Should Be Denied

52. Entergy's request for extension to comply with NRC Order EA-13-109—"Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions"—at FitzPatrick should be denied. Entergy has **not** demonstrated that FitzPatrick's current decades-old wetwell vent would be able to prevent a hydrogen explosion that could destroy the reactor building, in the event of a severe accident.
53. The defective, antiquated BWR Mark I design performed poorly in the Fukushima Daiichi accident. In the accident, three BWR Mark I reactors melted down, each generating hundreds of kilograms of explosive hydrogen gas. Hydrogen **leaked** from the primary containments into the reactor buildings, where it accumulated and detonated at different times, destroying three reactor buildings, which released large quantities of harmful radioactive material into the environment. Fukushima Daiichi's decades-old hardened vents **did not** prevent hydrogen from entering BWR Mark I secondary containments and detonating.
54. Entergy has **not** demonstrated that FitzPatrick's current decades-old wetwell vent would be able to prevent a hydrogen explosion that could destroy the reactor building, in the event of a severe accident. There is no reason to believe that a severe accident at FitzPatrick would turn out any differently than it did at in the Fukushima Daiichi accident, in which hydrogen **leaked** from primary containments into reactor buildings and detonated.

Respectfully submitted on behalf of

Beyond Nuclear and AGREE,

/s/

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