ZIRCONIUM FIRES IN POOLS OF SPENT NUCLEAR FUEL: HIGH-PROBABILITY SCENARIOS AND PHENOMENA

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Version I

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I. EXECUTIVE SUMMARY

An extreme solar storm (geomagnetic disturbance) could cause over 300 extra high voltage transformers to fail, "leading to probable power system collapse[s] in the Northeast, Mid-Atlantic, and Pacific Northwest," which could last months or longer, "affecting a population in excess of 130 million." Such a solar storm could occur as frequently as once in 153 years to once in 500 years, according to the U.S. Nuclear Regulatory Commission ("NRC"), and initiate "a series of events potentially leading to core damage at multiple nuclear sites."[1]

In the event of prolonged electrical grid failures, neither the NRC nor any other government agency has a strategy for implementing measures that would effectively prevent multiple concurrent reactor core meltdowns and spent fuel pool ("SFP") fires, which would cause catastrophic releases of radiation. (This report focuses on SFP accidents; reactor core meltdown phenomena are primarily discussed when their consequences, such as the production of explosive hydrogen gas, could affect the progression of SFP accidents.)

If large-scale power outages were to last months or longer, multiple nuclear power plants ("NPP") would lose their supply of offsite alternating current ("ac") power, which is necessary for daily operation and *preventing* severe accidents. Multiple loss-of-offsite power ("LOOP") events—especially in the event of prolonged electrical grid failures—could lead to a number of station-blackouts ("SBO"); a SBO is a complete loss of both grid-supplied and backup onsite ac power. The Fukushima Dai-ichi accident was a SBO accident that caused three reactor core meltdowns.

Many of the safety systems that are required for cooling the reactor core and SFP in a SBO—removing decay heat: the heat generated by the radioactive decay of the nuclear fuel's fission products—need ac power to operate.

In a LOOP event, a NPP's emergency diesel generators ("EDG") are intended to "supply power [promptly and] continuously to the equipment needed to maintain the

[1] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," Docket No. PRM–50–96, NRC–2011–0069, Federal Register, Vol. 77, No. 243, December 18, 2012, pp. 74788-74798.

plant in a safe condition" for an extended time period, "with refueling every 7 days."[2] The NRC has stated that, in a LOOP event, EDGs should be able to maintain a NPP in a safe condition for a *mission time* of "typically around 30 days."[3] Most U.S. NPPs are required to have an a 7-day capacity of fuel oil for EDGs onsite; many NPPs have additional fuel oil onsite and arrangements to receive prompt deliveries of fuel oil.[4] However, there could be problems with transporting and maintaining a fuel supply, amidst varying degrees of social disruption, in the event of large-scale, long-term power outages.

It is worrisome that the frequency of extreme solar storms, causing the large-scale, long-term power outages that could lead to at least one SFP fire, is estimated to be *as high as once in 100 years*.[5] The U.S. is particularly vulnerable to SFP fires, because its SFPs are *densely-packed* with spent fuel assemblies. (Low-density storage would help prevent SFP fires.) For example, in August 2013, Indian Point Unit 3's SFP—located less than 25 miles north of New York City—contained 1199 fuel assemblies, approximately 89 percent of storage capacity.[6]

Indian Point's owner, Entergy, touts the safety of Indian Point Unit 3's SFP, explaining that it is "constructed with concrete walls 4 to 6 feet wide and with a half-inch stainless steel inner liner" and that it is "nearly 100% underground, so [it is] protected on all sides by rock and gravel."[7] However, if there were a SFP fire at Unit 3, *thousands of kilograms of explosive hydrogen gas* could be generated by the oxidation (burning) of the tens of thousands kilograms of zirconium—the cladding material of the fuel rods—in

[2] NRC, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," Regulatory Guide 1.9, March 2007, Revision 4, p. 2.
[3] NRC Inspection Manual, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing," May 2008, (ADAMS Accession No. ML080420064), p. 3.
[4] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74796.
[5] John Kappenman, "Geomagnetic Storms and Their Impacts on the U.S. Power Grid," Meta-R-319, January 2010, pp. 3-14, 3-22, 3-26, 3-27.
[6] NRC, "Summary of August 26, 2013, Meeting with Entergy Nuclear Operations, Inc. and Netco on Indian Point Unit 2 Spent Fuel Pool Management," September 24, 2013, (ADAMS Accession No. ML13256A086), p. 1.
[7] Entergy, "Safe, Secure, Vital: Indian Point Energy Center," website, "Spent Fuel," (located at http://www.safesecurevital.com/safe-secure-vital/spent-fuel.html: last visited on October 12, 2013).

storage. It is almost inevitable that hydrogen gas would detonate, breaching the barriers that are supposed to protect the public; releases of radiation could far exceed the quantity released by the Chernobyl Unit 4 accident. More land could be contaminated than the area encompassing the Chernobyl Exclusion Zone, with higher concentrations of radioactive cesium-137. The number of premature deaths from cancer and economic damages would perhaps be incalculable. (This report does not attempt to estimate the extent of the radiological releases, health consequences, land contamination, and economic damages that would ensue from such a catastrophe.)

This report provides an overview and detailed discussions of SFP fire scenarios and phenomena. For example, there is a discussion of the chemical reaction of zirconium (of the fuel rods) and oxygen *in air*, which has *significantly higher reaction rates* than the zirconium-oxygen reaction does in either pure oxygen or steam.[8]

This report draws conclusions from data of severe accident experiments that is pertinent to SFP accidents. A 2001 NRC report, NUREG-1738, also draws conclusions from such data, stating that "it is useful to consider the range of available data including core degradation testing in steam environments, since it is likely that many SFP accidents may involve some initial period during which steam oxidation kinetics controls the initial oxidation, heatup, and release of fission products."[9] Data of reactor loss-of-coolant accident experiments can also be pertinent to SFP accidents. In both types of accidents, fuel rods would heat up, causing their internal-pressures to increase up to the points at which they ballooned and burst, impeding local cooling of the fuel assemblies.

This report argues that the prototypical initiating event that would lead to either one or multiple concurrent SFP fires is the event of large-scale, long-term power outages, because the frequency of such an event is relatively high. In such an event, SFP fires could commence at some point after the water in the pools heated up and boiled off, uncovering the fuel assemblies.

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[8] O. Coindreau, C. Duriez, S. Ederli, "Air Oxidation of Zircaloy-4 in the 600-1000°C Temperature Range: Modeling for ASTEC Code Application," Journal of Nuclear Materials 405, 2010, p. 208.

[9] NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February 2001 (ADAMS Accession No. ML010430066), Appendix 1 B, p. Al B-2.

A June 2013 NRC report on how earthquakes could affect boiling water reactor ("BWR") Mark I SFPs and a September 2013 NRC report, NUREG-2157, both claim that a severe earthquake is *the prototypical* initiating event that would lead to a SFP fire. The June 2013 report assigns such a severe earthquake a frequency of once in 60,000 years; the frequency of such an earthquake leading to a SFP fire is claimed to be far lower.[10] And NUREG-2157 states that "the frequency of fuel being uncovered…is between [once in 416,667 years and once in 1,724,138 years] depending upon the seismic hazard assessment."[11]

Clearly, the authors of the June 2013 NRC report and NUREG-2157 are incorrect that a catastrophic earthquake should be considered as the prototypical initiating event that would lead to a SFP fire. They overlooked new information about the potential affects of solar-induced geomagnetic disturbances and that (in 2012) the NRC assigned frequencies to the occurrence of large-scale, long-term power outages that are *two orders of magnitude greater* than the frequencies assigned to the type of severe earthquake that could lead to a SFP fire. Furthermore, phenomena of SFP boil-off scenarios (which large-scale, long-term power outages could cause) are different than those of SFP rapid-drain scenarios (which beyond-design-basis earthquakes could cause).

This report discusses deficiencies of the NRC computer safety model, MELCOR, which under-predicts the severity of SFP accidents. The NRC has recently performed a number of post-Fukushima computer simulations of SFP accidents with MELCOR. However, MELCOR *does not simulate* how nitrogen gas (in air) accelerates the oxidation (burning) and degradation of zirconium fuel cladding *in air*,[12] which would affect the progression and severity of SFP accidents, including radioactive releases, "most notabl[y] ruthenium."[13] MELCOR also *does not simulate* the generation of heat from the chemical

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[10] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," June 2013, (ADAMS Accession No. ML13133A132), pp. viii, x.

[11] NRC, "Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment," NUREG-2157, September 2013, (ADAMS Accession No. ML13224A106), p. 4-81.

[12] K. C. Wagner, R. O. Gauntt, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," SAND1A Letter Report, Revision 2, November 2006, (ADAMS Accession No. ML120970086), p. 12.

[13] J. Stuckert, M. Große, Z. Hózer, M. Steinbrück, Karlsruhe Institute of Technology, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," KIT-SR 7634, May 2013, p. 1.

reaction of zirconium and nitrogen; neglecting to model a heat source that would affect the progression and severity of SFP accidents is another serious flaw.

The NRC's conclusions from its Post-Fukushima MELCOR simulations are non-conservative *and misleading*, because their conclusions *underestimate* the probabilities of large radiological releases from SFP accidents. By overlooking the deficiencies of its Post-Fukushima MELCOR simulations, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[14]

[14] 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, (ADAMS Accession No. ML111861807), p. 3.

II. SPENT FUEL POOL FIRE SCENARIOS AND PHENOMENA

II.A. Large-Scale, Long-Term Power Outages Could Lead to Multiple Concurrent Reactor Meltdowns and Spent Fuel Pool Fires

An extreme solar storm (geomagnetic disturbance) could cause over 300 extra high voltage ("EHV") transformers[15] to fail, "leading to probable power system collapse[s] in the Northeast, Mid-Atlantic, and Pacific Northwest," which could last months or longer, "affecting a population in excess of 130 million."[16] Such a solar storm—with an intensity similar to that of the 1859 Carrington event[17]—could occur as frequently as once in 153 years to once in 500 years (2.0×10^{-3}/yr to 6.5×10^{-3}/yr), according to the NRC, and initiate "a series of events potentially leading to core damage at multiple nuclear sites."[18] (This is an international nuclear safety issue, not only pertinent to the U.S.)

(On March 14, 2011, Thomas Popik, submitted a petition for rulemaking, PRM-50-96,[19] on behalf of the Foundation for Resilient Societies, requesting regulations to help prevent SFP severe accidents—like zirconium[20] fires in racks of densely-packed spent fuel assemblies—in the event of prolonged outages of "North American commercial electric power grids…caused by extreme space weather, such as coronal mass ejections and associated geomagnetic disturbances."[21] In 2012, the NRC decided to consider the issues raised in PRM-50-96 in its rulemaking process.[22])

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[15] The NRC has explained that "[l]arge transformers are very expensive to replace and few spares are available. Manufacturing lead times for new equipment range from 12 months to more than 2 years." See NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," Docket No. PRM–50–96, NRC–2011–0069, Federal Register, Vol. 77, No. 243, December 18, 2012, p. 74794.

[16] *Id.*, pp. 74788-74798.

[17] The Carrington event in 1859 is the largest solar storm ever recorded.

[18] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74790.

[19] Thomas Popik, PRM-50-96, March 14, 2011 (ADAMS Accession No. ML110750145).

[20] For consistency, this report will use the term "zirconium" to refer to all the various types of zirconium alloys that comprise fuel cladding. Zircaloy, ZIRLO, and M5 are particular types of zirconium alloy fuel cladding. In a SFP accident, the oxidation behavior of the different fuel cladding materials, with various zirconium alloys, would be similar because of their shared zirconium content.

[21] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74788.

[22] *Id.*, pp. 74788-74798.

Additionally, either devices designed specifically to disrupt (or destroy) electronic equipment or the detonation of a nuclear device high above the earth's atmosphere could also produce an electromagnetic pulse with a magnitude that could cause large-scale, long-term power outages.[23] A June 2010 North American Electric Reliability Corporation ("NERC") and U.S. Department of Energy ("DOE") report states that such power outages could also be caused by pandemics, "coordinated cyber, physical, and blended attacks"[24] and that "[d]eliberate attacks (including acts of war, terrorism, and coordinated criminal activity) pose especially unique scenarios due to their inherent unpredictability and significant national security implications."[25]

Regarding nuclear electromagnetic pulse ("EMP") attacks, a May 21, 2013 Wall Street Journal op-ed, "How North Korea Could Cripple the U.S.: A Single Nuke Exploded above America Could Cause a National Blackout for Months," states:

> The Congressional Electromagnetic Pulse Commission, the Congressional Strategic Posture Commission and several other U.S. government studies have established that detonating a nuclear weapon high above any part of the U.S. mainland would generate a catastrophic electromagnetic pulse.
>
> An EMP attack would collapse the electric grid and other infrastructure that depends on it—communications, transportation, banking and finance, food and water—necessary to sustain modern civilization and the lives of 300 million Americans.
>
> EMP effects can be made more powerful and more catastrophic by using an Enhanced Radiation Warhead. This is a low-yield nuclear weapon designed not to create a devastating explosion, but to emit large amounts of radiation, including the gamma rays that generate the EMP effect that fries electronics.[26]

And discussing the North American power grid's vulnerabilities to large-scale, long-term power outages, the executive summary for "Electromagnetic Pulse: Effects on

[23] Metatech Corporation, "Electromagnetic Pulse: Effects on the U.S. Power Grid," Executive Summary, January 2010.
[24] NERC, DOE, "The High-Impact, Low-Frequency (HILF) Event Risk Effort," June 2010, pp. 3, 8.
[25] *Id*.
[26] R. James Woolsey, Peter Vincent Pry, "How North Korea Could Cripple the U.S.: A single nuke exploded above America could cause a national blackout for months," Wall Street Journal, May 21, 2013.

the U.S. Power Grid," a series of reports Metatech prepared for the Oak Ridge National Laboratory ("ORNL"), states:

> The nation's power grid is vulnerable to the effects of an electromagnetic pulse (EMP), a sudden burst of electromagnetic radiation resulting from a natural or man-made event. EMP events occur with little or no warning and can have catastrophic effects, including causing outages to major portions of the U.S. power grid possibly lasting for months or longer. … The cost of damage from the most extreme solar event has been estimated at \$1 to \$2 trillion with a recovery time of four to ten years,[27] while the average yearly cost of installing equipment to mitigate an EMP event is estimated at less than 20 cents per year for the average residential customer.[28]

The NRC has pointed out that a 2012 NERC report, "Effects of Geomagnetic Disturbances on the Bulk Power System,"[29] disagrees with conclusions of the Metatech report, stating that "[b]ased on an assumed frequency of a once-in-100-year geomagnetic event, the NERC report indicates that potential damage to EHV transformers of recent design is of a low probability, and thus challenges the assertions of the Metatech report that 300 large EHV transformers would be at risk of failure."[30] The 2012 NERC report states that "[t]he most likely consequence of a strong GMD [geomagnetic disturbances] and the accompanying GIC [geomagnetic induced currents] is the increase of reactive power consumption and the loss of voltage stability," "which could lead to…power system collapse."[31] The NERC report concludes that if the power system were to collapse from a loss of voltage stability that it could be restored in a time period of "hours to days."[32]

However, Lawrence J. Zanetti, a physicist in the Space Department of the Johns Hopkins University Applied Physics Laboratory, disagrees with conclusions of the 2012

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[27] National Academy of Sciences, "Severe Space Weather Events—Understanding Societal and Economic Impacts: A Workshop Report," 2008.
[28] Metatech Corporation, "Electromagnetic Pulse: Effects on the U.S. Power Grid," Executive Summary, January 2010.
[29] NERC, "2012 Special Reliability Assessment Impact Report: Effects of Geomagnetic Disturbances on the Bulk Power System," February 2012.
[30] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74795.
[31] NERC, "2012 Special Reliability Assessment Impact Report: Effects of Geomagnetic Disturbances on the Bulk Power System," pp. iii, iv.
[32] *Id.*, p. iv.

NERC report, stating that "[i]n this NERC report, the strong denial of the likelihood of a large number of multiple transformer failures is misleading and purveys a false sense of grid security."[33]

If large-scale power outages were to last months or longer, multiple nuclear power plants ("NPP") would lose their supply of offsite alternating current ("ac") power, which is necessary for daily operation and preventing severe accidents. Multiple loss-of-offsite power ("LOOP") events—especially in the event of prolonged electrical grid failures—could lead to a number of station-blackouts ("SBO"); a SBO is a complete loss of both grid-supplied and backup onsite ac power. The Fukushima Dai-ichi accident was a SBO accident that caused three reactor core meltdowns.

Many of the safety systems that are required for cooling the reactor core and SFP in a SBO—removing decay heat: the heat generated by the radioactive decay of the nuclear fuel's fission products—need ac power to operate.

In a LOOP event, a NPP's emergency diesel generators ("EDG") are intended to "supply power [promptly and] continuously to the equipment needed to maintain the plant in a safe condition" for an extended time period, "with refueling every 7 days."[34] The NRC has stated that, in a LOOP event, EDGs should be able to maintain a NPP in a safe condition for a *mission time* of "typically around 30 days."[35] Most U.S. NPPs are required to have an a 7-day capacity of fuel oil for EDGs onsite; many NPPs have additional fuel oil onsite and arrangements to receive prompt deliveries of fuel oil.[36] However, there could be problems with transporting and maintaining a fuel supply, amidst varying degrees of social disruption, in the event of large-scale, long-term power outages.

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[33] Zanetti, L. J., "Review of North American Electric Reliability Corporation (NERC) Interim Report: Effects of Geomagnetic Disturbances on the Bulk Power System—February 2012," Space Weather, Vol. 11, doi:10.1002/swe.20060, 2013, p. 335.
[34] NRC, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," Regulatory Guide 1.9, March 2007, Revision 4, p. 2.
[35] NRC Inspection Manual, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing," May 2008, (ADAMS Accession No. ML080420064), p. 3.
[36] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74796.

There could be cases in which EDGs would not be able to either operate promptly or continuously for months (or longer), causing a SBO. A 2011 report, "Fukushima Fallout," states "that there have been recurrent prolonged malfunctions of [EDGs] at nuclear power plants in the U.S" and that "[i]n the past eight years there have been at least 69 reports of EDG inoperability at 33 nuclear power plants. A total of 48 reactors were affected, including 19 failures lasting over two weeks and 6 that lasted longer than a month."[37] (EDG endurance and margin tests are typically performed every 18 to 24 months; a 24-hour test period is intended to ensure that an EDG would be able to meet its 30-day mission time. The NRC allows some NPP personnel, including Indian Point's, to perform the test for *an 8-hour test period*.[38])

In a SBO, EDGs are inoperable and "reactor cooling is *temporarily provided* by systems that do not rely on ac power, such as turbine-driven pumps that are driven by steam from the reactor. Batteries also are used to provide direct current (dc) power to control the turbine-driven pumps and to power instrumentation"[39] [emphasis added]. Backup batteries would become depleted in four hours—for some reactors, eight hours. Without a timely restoration of ac power, a SBO will lead to a reactor core meltdown at each affected NPP unit, as occurred at Fukushima Dai-ichi. And, if there were freshly discharged fuel assemblies in a spent fuel pool ("SFP"), its water could heat up and boil off in 49.3 hours or 125.0 hours (depending on whether there had been a 1/3 or full core discharge, five days prior);[40] pools densely-packed with fuel assemblies would be likely to incur SFP fires.

"Fukushima Fallout," also states that "[a] review of the NRC's Standard Technical Specifications for nuclear power plants[41] indicates that spent fuel pools at

[37] The Staff of Congressman Edward J. Markey, "Fukushima Fallout: Regulatory Loopholes at U.S. Nuclear Plants," May 12, 2011, pp. 9, 25.
[38] NRC Inspection Manual, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing," pp. 1, 3.
[39] NRC, "Modeling Potential Reactor Accident Consequences," NUREG/BR-0359, January 2012, (ADAMS Accession No: ML12026A470), p. 11.
[40] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4-25.
[41] "Fukushima Fallout: Regulatory Loopholes at U.S. Nuclear Plants" (p. 13, note 46) states "[s]ee for example 'Standard Technical Specifications General Electric Plants, BWR/4' and 'Standard Technical Specifications for Westinghouse Plants.' "

nuclear reactors whose cores do not contain nuclear fuel (for example, because they [are] in the process of being refueled) do NOT require the presence of operable secondary emergency generation capacity" [emphasis not added], explaining that "licensees often perform maintenance on their [EDGs] when the reactors are undergoing refueling outages."[42]

It has been documented that solar storms have damaged NPPs. For example, in March 1989, a geomagnetic storm caused a generator step-up ("GSU") transformer to fail at the Salem Nuclear Plant.[43] And, in April 1994, a "moderate intensity" geomagnetic storm caused a GSU transformer at Zion Nuclear plant to fail: "[t]he failure was so severe that the transformer tank, containing thousands of gallons of oil, ruptured and started a major fire in the yard at the plant, which eventually involved control circuits and other sensitive systems."[44] It has also been documented that high-altitude electromagnetic pulses produced by nuclear detonations *caused diesel generators to fail* in 1962, when the USSR detonated a few nuclear weapons at high altitudes—above an altitude of approximately 30 kilometers—in an experimental program.[45] (The NRC maintains that solar storms would not adversely affect EDGs, because they are normally not operating and that "any [geomagnetically-induced currents] that enter the plant's electrical system during EDG operation should not result in excessive overheating of the generator windings."[46])

The NRC does not require NPP owners to be prepared for large-scale, long-term power outages, and notes that "in the event of a widespread electrical transmission system blackout for an extended duration (beyond 7 days and up to several months), it may not be possible to transport…necessary offsite resources to the affected NPPs in a

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[42] The Staff of Congressman Edward J. Markey, "Fukushima Fallout: Regulatory Loopholes at U.S. Nuclear Plants," May 12, 2011, p.13. "Fukushima Fallout" (p. 13, note 47) states that the sources of this information are from "[p]rivate communications from an individual working inside an operating nuclear power plant obtained by Rep. Markey's office and discussions with nuclear safety experts."
[43] John Kappenman, "Geomagnetic Storms and Their Impacts on the U.S. Power Grid," Metatech Report Meta-R–319, January 2010, p. 2-29.
[44] *Id*., p. 2-33.
[45] Edward Savage *et al*., "The Early-Time (E1) High-Altitude Electromagnetic Pulse (HEMP) and Its Impact on the U.S. Power Grid," Metatech Report Meta-R-320, January 2010, pp. i, 2-1, 3-4.
[46] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74796.

timely manner. Thus, government assistance (local, state, or Federal) may be necessary to maintain the capability to safely shutdown nuclear plants and cool spent fuel pools in the affected areas. Prior planning is needed to efficiently and effectively use government resources to ensure protection of public health and safety."[47] In other words, in the event of prolonged electrical grid failures, neither the NRC nor any other government agency has a strategy for implementing measures that would effectively prevent multiple concurrent reactor core meltdowns and SFP fires, which would cause catastrophic releases of radiation.

It is worrisome that the frequency of extreme solar storms, causing the large-scale, long-term power outages that could lead to at least one SFP fire, is estimated to be *as high as once in 100 years* (1.0×10^{-2}/yr).[48] The U.S. is particularly vulnerable to SFP fires, because its SFPs are densely-packed with spent fuel assemblies. (Low-density storage would help prevent SFP fires.) For example, in August 2013, Indian Point Unit 3's SFP—located less than 25 miles north of New York City—contained 1199 fuel assemblies, approximately 89 percent of storage capacity.[49]

II.B. Large-Scale, Long-Term Power Outages Should Be Considered the Prototypical Initiating Event that Would Lead to Spent Fuel Pool Fires

Large-scale, long-term power outages, which lasted months or longer, should be considered as the prototypical initiating event that would lead to either one or multiple concurrent SFP fires, because the frequency of such an event is estimated to be relatively high. (In such an event there would also be multiple concurrent reactor core meltdowns.[50]) However, a June 2013 NRC report on how earthquakes could affect BWR

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[47] *Id*., p. 74797.
[48] John Kappenman, "Geomagnetic Storms and Their Impacts on the U.S. Power Grid," Meta-R-319, pp. 3-14, 3-22, 3-26, 3-27.
[49] NRC, "Summary of August 26, 2013, Meeting with Entergy Nuclear Operations, Inc. and Netco on Indian Point Unit 2 Spent Fuel Pool Management," September 24, 2013, (ADAMS Accession No. ML13256A086), p. 1.
[50] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74790.

Mark I SFPs[51] and a September 2013 NRC report, NUREG-2157,[52] both claim that a severe earthquake is *the prototypical* initiating event that would lead to a SFP fire. The June 2013 report assigns such a severe earthquake a frequency of once in 60,000 years (1.67×10^{-5}/yr); the frequency of such an earthquake leading to a SFP fire is claimed to be far lower.[53] And NUREG-2157 states that "the frequency of fuel being uncovered…is between 5.8×10^{-7}/yr and 2.4×10^{-6}/yr [between once in 416,667 years and once in 1,724,138 years] depending upon the seismic hazard assessment."[54]

Clearly, the authors of the June 2013 NRC report and NUREG-2157 are incorrect that a catastrophic earthquake should be considered as the prototypical initiating event that would lead to a SFP fire. They overlooked new information about the potential affects of solar-induced geomagnetic disturbances and that (in 2012) the NRC assigned frequencies to the occurrence of large-scale, long-term power outages that are *two orders of magnitude greater* than the frequencies assigned to the type of severe earthquake that could lead to a SFP fire.

Large-scale, long-term power outages would lead to SBO scenarios in which the water in the pools heated up and boiled off, uncovering the fuel assemblies; severe earthquakes would lead to different scenarios. NUREG-2157 states that in the event of a beyond-design-basis earthquake with a magnitude significantly larger than what a SFP could withstand that "water would rapidly drain out of the pool. Only a small amount of water would remain and the spent fuel would be uncovered and exposed to the air."[55]

Contrary to NUREG-2157, a May 2013 Pennsylvania State University ("PSU") report claims that it is *unlikely* in the event of a SFP loss-of-coolant accident ("LOCA") that all the water would rapidly drain, except a small amount, completely uncovering the

[51] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," June 2013, (ADAMS Accession No. ML13133A132).

[52] NRC, "Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment," NUREG-2157, September 2013, (ADAMS Accession No. ML13224A106).

[53] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," pp. viii, x.

[54] NRC, "Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment," NUREG-2157, p. 4-81.

[55] *Id.*, p. 4-81.

fuel assemblies and exposing them to air over their entire length.[56] The 2013 PSU report opines that *partial* SFP LOCAs—in which "the water level in the SFP drains below the top of the fuel bundle"[57]—would be more likely.

In a SFP LOCA, partial fuel assembly uncovery would be a greater threat to safety than complete uncovery of the fuel assemblies. Complete uncovery of the fuel assemblies (with the water level dropping far enough below the bottom of the SFP baseplates,[58] which have holes) would enable air to flow through the fuel assemblies, entering at the base and exiting at the top. This would help cool the fuel assemblies. There would not be the same advantage if there were partial uncovery of the fuel assemblies. If the water level remained above the baseplates, it would essentially block the flow of air through the fuel assemblies and "effectively reduce the heat transfer rates from the fuel, causing the fuel to heat up at a higher rate than if natural circulation [were] occurring."[59]

SBO boil-off accidents resemble partial SFP LOCAs in that in both accidents there would be times in which there was partial uncovery of the fuel assemblies; the water level would be above the baseplates, essentially blocking the flow of air through the fuel assemblies and impeding the transfer of heat away from the fuel. The poor heat transfer conditions of SBO boil-off accidents make it more probable that they would lead to SFP fires.

The June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs claims that in the event of a complete SFP LOCA, the fuel assemblies would *not* be air coolable for 10 percent of a two year operating cycle (the approximate time interval between the loading of each reactor core discharge into the SFP); that is, the fuel assemblies would not be air coolable for 73 days. However, the June 2013 NRC report

[56] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, May 2013, p. 3.
[57] *Id*., p. 1.
[58] "[T]he distance between the pool floor liner and the bottom of the rack baseplate is…on average…26 centimeters (cm) (10.25 in.), depending on adjustments made to the leveling pad during installation." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 76, Footnote 1.
[59] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. v.

states that in a partial BWR Mark I SFP LOCA, the airflow could be impeded if there were channeled fuel assemblies in the SFP, which would "*increase the time to coolability*"[60] [emphasis added].

Elsewhere, the June 2013 NRC report states that a partial BWR Mark I SFP LOCA is *assumed not* to be air coolable for an entire two year operating cycle (730 days).[61] In other words, partial BWR Mark I SFP LOCAs, in which "the rack baseplate is not cleared and airflow is impeded,"[62] are *assumed not* to be air coolable during a reactor's entire life of operation, in which reactor core discharges would be loaded into the SFP every two years. As stated, the baseplates also would not be cleared in SBO boil-off accidents—another reason such accidents could lead to SFP fires.

According to the Electric Power Research Institute ("EPRI"), in a boil-off accident, "the temperature on the operating floor of the SFP…would also, at best, be tolerable from the standpoint of temperature and humidity for only a short period even with the help of protective clothing;"[63] which "may require a special suit and a breathing apparatus."[64] EPRI also states:

> [T]he possibility of increased radiation levels on the refueling floor would be a key concern since some emergency actions could involve installing hoses or pipes on the refueling floor to refill the pool. From the safety perspective, the SFP water depth absorbs the gamma rays emitted by the decay heat from reactor fuel. The gamma rays are attenuated exponentially as a function of the water depth…
>
> [I]f the initial water depth is decreased by a factor of two [50 percent], the radiation intensity would increase by 300 to 1000 times [from typical values]. When the radiation level increases to this extent, only minimal time should be spent on the refueling floor. Furthermore, the radiation

[60] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," Appendix B, p. B-10.
[61] *Id*., Appendix D, p. D-13.
[62] *Id*.
[63] Electric Power Research Institute ("EPRI"), "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, October 2012, Appendix EE, p. EE-2.
[64] Robert E. Henry, Fauske & Associates, "Additions and Changes to TBR Volume 2: Physics of Accident Progression," NRC Briefing on SAMG TBR, November 7, 2012, (ADAMS Accession No: ML12318A080), p. 66.

level will increase exponentially with a further decrease in the SFP water level.[65]

In fact, personnel access "should be limited when the [SFP water] level is significantly reduced and prohibited when the water level is below one-half of the nominal value."[66]

In a SBO boil-off accident, if enough water boiled off, the water level would drop in the pool, uncovering the fuel assemblies; and if temperatures in the SFP were to increase to approximately 657°C (1214°F), the Boral plates of the fuel assembly storage racks would melt.[67] (Boraflex would melt at even lower temperatures and not be effective once heated above approximately 300°C (572°F).[68]) Boral and Boraflex are neutron-absorber materials that are placed in high-density storage racks to help prevent criticality accidents. (Fission—the splitting of atoms in the nuclear fuel—occurs in a criticality accident.)

If Boral were to melt in *BWR* high-density storage racks, neutrons would diffuse throughout the SFP; in scenarios in which water was injected back into the boiled-off SFP, fission could possibly commence. (BWR SFPs do not use borated water.[69]) If fission were to occur, local fuel and fuel-cladding temperatures would rapidly increase. Fission would also "cause an increase in decay products, which [would] have a delayed effect on temperature increase[s]."[70] (A June 2013 NRC document states that "if an [inadvertent criticality event] were severe enough to produce significant heat, the fuel will be harder to cool."[71]) Rapid increases in the fuel and fuel-cladding temperatures could lead to a SFP fire. And radiation releases, caused by a criticality accident in a SFP,

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[65] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, pp. EE-1, EE-2.
[66] *Id*., Appendix EE, p. EE-2.
[67] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, May 2013, pp. 1-2.
[68] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-9.
[69] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 30.
[70] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, pp. 1-2.
[71] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 29.

would impede (or possibly prevent for significant time periods) efforts to mitigate either a partial SFP LOCA or SBO boil-off accident, making it more probable that such accident scenarios would lead to SFP fires. (Criticality accidents are discussed in more detail in Section D.)

As the NRC observes, large-scale, long-term power outages, which lasted months or longer, could initiate "a series of events potentially leading to core damage at multiple nuclear sites."[72] Radiological releases resulting from core damage would contaminate the NPP site and impede efforts to mitigate the accident, especially if radioactive debris were propelled throughout the site by hydrogen explosions, as occurred in the Fukushima Dai-ichi accident.[73] After the Fukushima Dai-ichi site was contaminated, workers had to wear additional protective clothing and limit the time they spent, working to mitigate the accident.[74] Efforts to mitigate a SFP accident would also be impeded (or possibly entirely prevented for significant time periods) by the radiologically-contaminated environment. Hence, if large-scale, long-term power outages lead to core damage, it would be more probable that such outages would also lead to at least one SFP boil-off accident and fire.

II.C. Station Blackout Boil-Off Scenarios Could Lead to Spent Fuel Pool Fires

SFPs store fuel assemblies (essentially bundles of fuel rods, comprised of zirconium alloy cladding sheathing uranium dioxide (UO_2) fuel pellets) after they are discharged from the reactor core. (See Figure 1.) If there were a loss of SFP cooling, the water in the pool would be heated by the fuel assemblies' decay heat (heat generated by the radioactive decay of the fuel's fission products) until it reached the boiling point; then the water would boil away, uncovering the fuel assemblies.

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[72] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74790.

[73] Institute of Nuclear Power Operations ("INPO"), "Special Report on the Nuclear Accident at the Fukushima Dai-ichi Nuclear Power Station," INPO 11-005, November 2011, pp. 9, 12, 21, 24, 25, 32, 37, 79, 85, 86, 96.

[74] *Id*., p. 9.

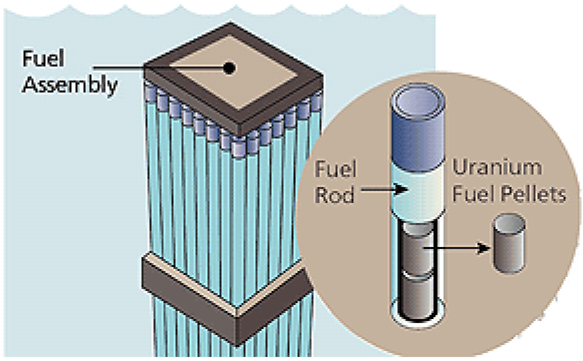

Figure 1. Fuel Assembly, Fuel Rod, and Uranium Dioxide Fuel Pellets[75]

SFPs have various depths; PWR and BWR SFPs typically have depths in a range from 38.0 feet to 40.0 feet.[76] And spent fuel assembles typically have heights of approximately 13 feet 4 inches,[77] so there is typically less than 27 feet of water above the top of the fuel assembles in SFPs. (In BWR Mark I and II designs, SFPs are typically located at the level of the operating floor, approximately 100 to 150 feet above ground level; and in PWR and BWR Mark III designs, SFPs are typically located at ground level.[78])

A number of factors would determine the "heat load" in the SFP, including how recently some of the fuel assemblies stored there had been discharged from the reactor core, because the amount of heat generated by decay heating progressively declines (nonetheless, decay heating remains a significant heat source for years). A 2011 IAEA report states that "[t]he heat load in spent fuel soon after irradiation is primarily due to the fission products [that is, primarily due to the decay heat generated by the fission products]. Much later in life it is due to the decaying actinides,"[79] predominantly

[75] NRC, Image from "Fact Sheet: Storage of Spent Nuclear Fuel."
[76] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4.5.
[77] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, October 2012, Appendix EE, p. EE-8.
[78] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, p. 4.6.
[79] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 18.

uranium, a small percentage of plutonium, and traces of other actinides. And the NRC states that the heat load in the SFP decreases rapidly over time from its peak value."[80]

The length of time prior to commencing movement of the fuel assemblies from the reactor core after shutdown to the SFP should also be considered in accessing SFP accidents. At some NPPs, the time before commencing fuel movement after shutdown has been reduced; for example, in 2002, PSEG Nuclear, the owner of Salem Nuclear Generating Station, Units 1 and 2, requested that the time prior to commencing fuel movement after shutdown at Salem be *reduced from 168 hours to 100 hours*.[81]

Fuel assemblies that had a higher burnup[82]—whose enriched-uranium fuel had been converted into energy to a greater extent in the reactor core—would produce greater quantities of decay heat; "[i]n general, the higher the burnup, the higher the heat load generated and the more heat rejection [cooling] capability is required."[83]

In the U.S. and elsewhere, the trend is to increase the burnup of fuel assemblies; this is done by extending the length of time the assemblies spend producing energy in the reactor core and/or by increasing the power levels of NPPs. Some power uprates—"[t]he process of increasing the licensed power level at a commercial [NPP]"[84]—in the U.S. have been substantial; in 2006, the NRC approved a 20 percent power uprate for Vermont Yankee Nuclear Power Station.

(In 1999, burnup levels for spent PWR fuel and spent BWR fuel were approximately 45 gigawatt-days thermal per metric ton[85] of enriched uranium ("GW·d/t U") and 37 GW·d/t U, respectively; in the U.S., by 2021, "[b]urnup levels for

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[80] NRC, "Review Standard for Extended Power Uprates," RS-001, Revision 0, December 2003, Attachment 2 to Matrix 5, (located at http://www.nrc.gov/reactors/operating/licensing/power-uprates/rs-001-rev-0-dec2003.pdf), p. 2.
[81] D.F. Garchow, PSEG Nuclear, "Request for Changes to Technical Specifications for Refueling Operations: Fuel Decay Time Prior to Commencing Core Alterations or Movement of Irradiated Fuel at Salem Nuclear Generating Station, Units 1 and 2," June 28, 2002, (ADAMS Accession No. ML021920053), p. 1.
[82] Burnup is the thermal energy produced per unit mass of enriched-uranium in the fuel.
[83] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 18.
[84] NRC, "Review Standard for Extended Power Uprates," RS-001, Revision 0, December 2003, Background.
[85] 1000 kilograms.

spent PWR fuel are anticipated to rise to ~55 GW·d/t U [and] burnup levels for spent BWR fuel will likely increase to over 40 GW·d/t U."[86])

Of course, the heat load in the SFP would also be determined by the quantity of fuel assemblies stored in the SFP. With high-density storage there are greater heat loads. If there were a loss of SFP cooling, the heat load in the SFP would affect how long it took for the water to reach the boiling point and boil away, uncovering the fuel assemblies. As an accident progressed, *local* heat up rates would be affected by fuel rack loading patterns—how the most recently discharged fuel assemblies were arranged with ones discharged over a year previously. In the SFP, fuel assemblies might be arranged within checkerboard configurations; there may be one more recently discharged fuel assembly out of every four cells; the other three could contain older fuel assemblies.

In *certain* boil-off scenarios, the water in a "typical" SFP[87] that had been loaded five days prior with a 1/3 core discharge, would heat from 51.7°C (125°F) to 100°C (212°F) in 11.2 hours and boil dry in 125.0 hours; and a typical SFP that had been loaded five days prior with a full core discharge, would heat from 66°C (150°F) to 100°C (212°F) in 3.1 hours and boil dry in 49.3 hours. (All of these values are for a SFP already stocked with 20 years of accumulated core discharges.)[88]

Regarding one-third and full core discharges (offloads), a 2005 National Research Council report, "Safety and Security of Commercial Spent Nuclear Fuel Storage," states:

> After a power reactor is shut down, its nuclear fuel continues to produce heat from radioactive decay… Although only one-third of the fuel in the reactor core is replaced during each refueling cycle, operators commonly offload the entire core (especially at PWR[s] [pressurized water reactor]) into the pool during refueling to facilitate loading of fresh fuel or for inspection or repair of the reactor vessel and internals. Heat generation in the pool is at its highest point just after the full core has been offloaded.[89]

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[86] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 9.

[87] Generic analyses of SFPs are limited. See J.H. Jo, P.F. Rose, S.D. Unwin, V.L. Sailor, Brookhaven National Laboratory, "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," NUREG/CR-5281, March 1989, (ADAMS Accession No. ML071690022), p. 5.

[88] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, p. 4-25.

[89] National Research Council, Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage, "Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report," 2005, p. 40.

The same report provides information on industry practices, stating:

> A 1996 survey by the Nuclear Regulatory Commission (USNRC, 1996[90]) found that the majority of commercial power reactors routinely offload their entire core to the spent fuel pool during refueling outages. The practice is more common among PWRs than BWRs, which tend to offload only that fuel that is to be replaced, but some BWRs do offload the full core. In response to a committee inquiry, an Energy Resources International staff member confirmed that this is still the case today [in 2005].[91]

In a SBO boil-off accident, if there were partial uncovery of the fuel assemblies, the water level would be above the baseplates, essentially blocking the flow of air through the fuel assemblies and impeding the transfer of heat away from the fuel; the poor heat transfer conditions would cause "the fuel to heat up at a higher rate than if natural circulation [were] occurring."[92]

A March 1979 report, NUREG/CR-0649, observes that in partial-uncovery scenarios, the "heat transfer advantages…gained by converting decay heat to boiling energy would be minimal," because the (boiling) water level would be "far" below the elevations of the fuel assemblies that had the maximum fuel-cladding temperatures and heatup rates.[93] There would also be partial cooling of the fuel cladding from the steam that was generated by the boiling water;[94] saturated steam would enhance the natural convection of heat away from the fuel assemblies because it has a high heat capacity.[95] However, in partial-uncovery scenarios, the heat-transfer benefits of steam would be minimal compared to how the blockage of air flow through the fuel assemblies would

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[90] NRC, "Refueling Practice Survey: Final Report," 1996 (ADAMS Accession No. ML003705757).

[91] National Research Council, "Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report," p. 40, Note 4.

[92] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. v.

[93] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 40.

[94] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 118.

[95] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 40.

impede the transfer of heat away from the fuel rods; in terms of heat transfer, partial uncovery is considered "the worst case scenario."[96]

(In a complete SFP LOCA, complete uncovery of the fuel assemblies (with the water level dropping far enough below the bottom of the SFP baseplates,[97] which have holes) would enable air to flow through the fuel assemblies, entering at the base and exiting at the top. NUREG/CR-0649, states that, in complete SFP LOCA scenarios**,** "the baseplate hole size can exert a marked effect on the heatup of the spent fuel, since a small baseplate hole tends to constrict the flow at the inlet to the fuel assembly. …if the temperature of self-sustaining clad oxidation is not attained, the peak clad temperature tends to reach a steady-state maximum value that remains essentially invariant with time. If a sufficiently high temperature is achieved, however, the clad oxidation reaction can become self-sustaining, leading to a temperature divergence that results in local clad melting. The temperature at which clad oxidation becomes self-sustaining is a function of the storage configuration, but tends to occur around 900°C."[98])

A June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs claims that in the event of a complete SFP LOCA, the fuel assemblies would *not* be air coolable for 10 percent of a two year operating cycle (the approximate time interval between the loading of each reactor core discharge into the SFP); that is, the fuel assemblies would not be air coolable for 73 days. However, the same June 2013 NRC report states that a partial BWR Mark I SFP LOCA "with channeled fuel could impede airflow and *increase the time to coolability*"[99] [emphasis added]. (The same poor heat transfer conditions would occur in a SBO boil-off accident, if there were partial uncovery of the fuel assemblies.) Elsewhere, the June 2013 NRC report states that a partial BWR Mark I SFP LOCA is *assumed not* to be air coolable for an entire two year operating

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[96] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. v.

[97] "[T]he distance between the pool floor liner and the bottom of the rack baseplate is…on average…26 centimeters (cm) (10.25 in.), depending on adjustments made to the leveling pad during installation." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 76, Footnote 1.

[98] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649,SAND77-1371, March 1979, p. 47.

[99] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," Appendix B, p. B-10.

cycle (730 days).[100] In other words, partial BWR Mark I SFP LOCAs, in which "the rack baseplate is not cleared and airflow is impeded,"[101] are *assumed not* to be air coolable during a reactor's entire life of operation, in which reactor core discharges would be loaded into the SFP every two years. As stated, the baseplates also would not be cleared in SBO boil-off accidents.

A 2013 PSU report, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," states that "[t]he time required for a SFP to reach temperatures high enough [827°C (1520°F) [102]] to result in fuel overheating could range from several days to weeks."[103] The 2013 PSU report is referring to partial SFP LOCAs; however, the same applies to SBO boil-off scenarios.

For certain boil-off scenarios, after the fuel assemblies were partly uncovered, the upper exposed elevations of the cladding of the fuel rods would *initially* heat up *very slowly*, at local rates possibly lower than 0.01°C/sec (0.018°F/sec) (lower than 36°C/hour (64.8°F/hour).[104] (These values are based on results of computer simulations conducted with the NRC TRACE computer safety model. The TRACE simulations were conducted to help Sandia National Laboratories ("SNL") develop a full-scale boil-off experiment, simulating a rapid partial SFP LOCAs, in which "the water level in the SFP drains below the top of the fuel bundle."[105] In the TRACE simulations of *particular* SFP LOCAs, with particular accident parameters, the initial temperature of the fuel cladding is 27°C (80°F); in a slower SBO boil-off scenario the initial temperature of the fuel cladding would be approximately 100°C (212°F) at the water surface. In the TRACE simulations, the local fuel-cladding temperature heated up from 27°C (80°F) to 827°C (1520°F) in less than 25 hours.[106] (See Figure 2.)

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[100] *Id.*," Appendix D, p. D-13.
[101] *Id.*
[102] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. 2.
[103] *Id.*, p. 1.
[104] *Id.*, p. 19.
[105] *Id.*, p. 1.
[106] *Id.*, p. 19.

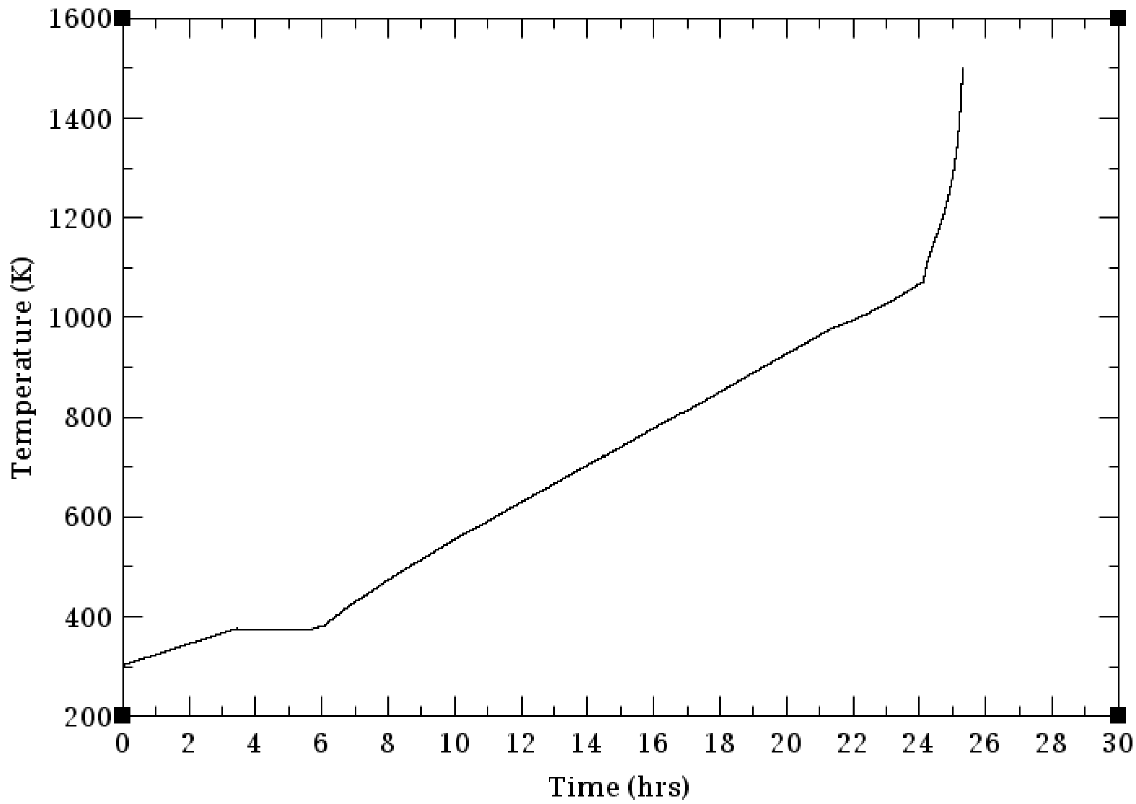

Figure 2. Local Cladding Temperature vs. Time in the TRACE Simulation[107]

According to EPRI, for certain PWR boil-off scenarios, after the fuel assemblies—with a decay heat of 42,334 watts per assembly—were partly uncovered, the upper exposed elevations of the cladding of the fuel rods would have local heatup rates of approximately 0.13°C/sec (0.23°F/sec).[108] This is still a relatively slow heatup rate; however, it is more than 10 times faster than that of the TRACE simulation example. If the local fuel-cladding heatup rate were 0.13°C/sec (0.23°F/sec), (without considering any additional heat that would be contributed by the zirconium-steam reaction) local fuel-cladding temperatures would heat from 100°C (212°F) to 827°C (1520°F) in approximately 1.6 hours.

(Below the water surface, the temperature of the fuel cladding *initially* would be higher than 100°C, because the boiling points of water are higher at greater pressures; for

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[107] Graph from "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE;" see Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. 19.
[108] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-10.

example, at 35 feet deep, the pressure would be approximately 30 pounds per square inch absolute ("psia") and the water temperature would exceed 121°C (250°F); the fuel cladding would be the same temperature at that depth.)

The temperature range of a SFP's water can vary during normal operation; for example, Palisades Nuclear Plant's SFP is maintained with water temperatures in a range between 75°F and 125°F.[109] Hence, in the event of a SFP accident, the initial water temperature would be at some value within a range of approximately 50 degrees Fahrenheit.

EPRI SFP accident guidance states that in a SBO boil-off accident, "should the water level decrease to 2/3 of the [spent fuel assembly] height, the overheating of the top of the fuel assemblies would approach 2000°F (1093°C)."[110] In a SBO boil-off accident, a SFP fire would possibly commence (*in a steam atmosphere*) if the fuel cladding reached local temperatures between approximately 1000°C (1832°F) and 1200°C (2192°F). *In air,* a SFP fire would most likely commence if the fuel cladding reached local temperatures between approximately 827°C (1520°F)[111] and 900°C (1652°F).[112]

II.C.1. Local Heavy Oxide and/or Crud Layers Would Partly Impede either the Local Steam or Air "Coolant" Flow through the Spent Fuel Assemblies in a SFP Accident

When high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 microns ("µm") thick (or greater);[113] there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 µm thick.[114]

[109] Entergy Nuclear Operations, "Commitments to Address Degraded Spent Fuel Pool Storage Rack Neutron Absorber: Palisades Nuclear Plant," August 27, 2008, (ADAMS Accession No. ML082410132), p. 1.
[110] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, October 2012, Appendix EE, p. EE-17.
[111] Zachary I. Franiewski *et al*., Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," pp. iv, 2, 3, 8, 13.
[112] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.
[113] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 30.
[114] *Id*., p. 29.

Local heavy oxide and/or crud layers would partly impede the local steam or air "coolant" flow through the spent fuel assemblies in a SFP boil-off accident or complete SFP LOCA, respectively, in at least the following aspects: 1) the amount of either steam or air "coolant" in the vicinity of the spent fuel cladding that had local heavy oxide and/or crud layers may be substantially less than if the cladding were clean; 2) the amount of either steam or air coolant flow past the vicinity of the spent fuel cladding that had local heavy crud and oxide layers may be substantially less than the flow past clean cladding; 3) if there were rapid oxidation, local growth of oxide layer thicknesses and increased degradation of the fuel cladding would further obstruct either the steam or air "coolant" flow.

Partly impeded local cooling, caused by local heavy oxide and/or crud layers, could cause local fuel-cladding temperatures to increase up the point at which zirconium would begin to rapidly chemically react with steam or air—at approximately 1000°C (1832°F) or 900°C (1652°F),[115] respectively. In a SFP accident, partly impeded local cooling, caused by local heavy oxide and/or crud layers, could decrease the time to the ignition of zirconium in either steam or air.

(It is doubtful that the NRC's computer safety model MELCOR simulates how local heavy oxide and/or crud layers would partly impede the local steam or air "coolant" flow through the spent fuel assemblies in a SFP boil-off accident or complete SFP LOCA, respectively.)

II.C.2. The Role of the Degraded Thermal Conductivity of High Burnup Fuel Rods in Spent Fuel Pool Boil-Off Accidents

As stated above, when high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 µm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 µm thick. (And medium to high burnup fuel cladding typically has a "hydrogen concentration in the

[115] Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

range of 100-1000 wppm [weight parts per million];" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[116])

High burnup fuel rods have thinner cladding, because a higher quantity of their zirconium content has oxidized during the operation of the reactor.[117] The thermal conductivity of oxide layers and crud layers is low—especially crud layers. (It is noteworthy that an EPRI report states that "[i]n some instances, BWR rods have been known to dislodge so much crud when moved around in [SFPs] that diminished pool clarity occurred."[118] Tenacious crud would not become dislodged from fuel rods in this fashion.)

And, as the burnup of fuel rods increases, there is an increase in their total *internal* thermal resistance. There is greater internal thermal resistance in high burnup fuel, because: 1) the thermal conductivity of the fuel pellets degrades, partly due to cracking and 2) there is an increased release of fission gas that degrades the thermal conductivity of the gap between the fuel pellet and the cladding.[119] (A 2011 IAEA report states that "[t]he fission gas released from the fuel pellets to the fuel cladding gap will increase *as much as ten-fold* for high burnup fuel over lower burnup fuel"[120] [emphasis added].) A 2012 NRC document states that "[t]he gap [thermal] resistance can…increase because of pellet densification (which increases the gap size) and/or degradation of the helium-gap gas conductivity by the addition of noble fission gases (xenon and krypton) released from the fuel pellets."[121]

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[116] K. Natesan, W.K. Soppet, Argonne National Laboratory, "Hydrogen Effects on Air Oxidation of Zirlo Alloy," NUREG/CR–6851, October 2004, (ADAMS Accession No. ML042870061), pp. iii, 3.
[117] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, pp. 29, 50.
[118] Electric Power Research Institute, "Technical Bases for Extended Dry Storage of Spent Nuclear Fuel," 1003416, December 2002, p. 3-8.
[119] NRC, "Letter to GE-Hitachi Nuclear Energy Americas (GEH) Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation," March 23, 2012, (ADAMS Accession No. ML120680571), Enclosure 2, "NRC Staff Assessment of General Electric-Hitachi Nuclear Energy and Global Nuclear Fuel—Americas Codes and Methods with Regard to Thermal Conductivity Degradation," March 23, 2012, (ADAMS Accession No. ML120750001), pp. 1-2.
[120] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 50.
[121] NRC, "Letter to GE-Hitachi Nuclear Energy Americas (GEH) Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation," Enclosure 2, "NRC Staff Assessment of General

(It is noteworthy that the fuel-cladding gap size does not necessarily increase in high burnup fuel; an October 2003 paper states that "[i]nner surface cladding oxidation and subsequent mechanical bonding between the fuel pellet and the cladding are well-known phenomena of high burnup and high duty fuels."[122])

During the operation of a reactor, the thermal resistance of crud and/or oxide layers on cladding increases the internal pressure of fuel rods. Regarding this phenomenon, a 2003 NRC document states:

> Clad[ding] oxidation can lead to significantly increased fuel rod internal pressures. … In addition to oxidation causing increases in rod internal pressures, crud deposition has a similar effect since crud is a poor conductor of heat. Keeping crud deposition to a minimum also reduces the impact on rod internal pressures.[123]

During the operation of a reactor, the fuel-cladding gap of high burnup fuel rods may reopen "when [the] internal pressure in the [fuel] rod exceeds [the] reactor coolant system pressure."[124] When the fuel-cladding gap reopens there is also thermal resistance caused by the extremely low thermal conductivity of the gases in the fuel-cladding gap. Regarding this phenomenon, the 2012 NRC document states:

> Should the gap [between the fuel pellet and the cladding] reopen, the increased thermal resistance will result in higher fuel pellet temperatures, resulting in higher fission gas release. The increased fission gas release will degrade gap conductivity while increasing the rod internal pressure, thus increasing pellet temperature and widening the gas gap further. The onset of gap reopening results in a runaway process of increasing gap opening until cladding failure.[125]

Electric-Hitachi Nuclear Energy and Global Nuclear Fuel—Americas Codes and Methods with Regard to Thermal Conductivity Degradation," pp. 1-2.

[122] Sven Van den Berghe *et al.*, "Observation of a Pellet-Cladding Bonding Layer in High Power Fuel," presented at "Advanced Fuel Pellet Materials and Designs for Water Cooled Reactors: Technical Committee Meeting," 20–24 October 2003, p. 307.

[123] NRC, "Safety Evaluation by the Office of Nuclear Regulation, Topical Report WCAP-15604-NP. REV. 1, 'Limited Scope High Burnup Lead Test Assemblies' Westinghouse Owners Group, Project No. 694," 2003, (ADAMS Accession No. ML070740225) (See Section A), p. 4.

[124] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998, (ADAMS Accession No. ML003730714), p. 1.

[125] NRC, "Letter to GE-Hitachi Nuclear Energy Americas (GEH) Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation," Enclosure 2, "NRC Staff Assessment of General Electric-Hitachi Nuclear Energy and Global Nuclear Fuel—Americas Codes and Methods with Regard to Thermal Conductivity Degradation," p. 8.

Regarding the heating of the fuel cladding *in a complete SFP LOCA*, a 1979 Sandia Laboratories report states that "[v]ariations in temperature from rod to rod in an assembly might occur as a result of variations in decay heat or *differences in the thickness of the oxide coating*, but these factors are difficult to predict and have not been accounted for"[126] [emphasis added]. And, discussing research and development priorities regarding the dry storage of spent fuel assemblies, which would also pertain to SFP accidents, a 2012 Pacific Northwest National Laboratory ("PNNL") report states that "[d]etermining actual clad emissivities[127] as a function of oxide and crud layer thicknesses under dry storage conditions is necessary to calculate actual temperature profiles…"[128] The PNNL report observes that this would be a "difficult and expensive task."[129]

In a boil-off accident, the thermal resistance of crud (corrosion products) and/or oxide layers on fuel-cladding would slightly decrease the radial heat losses of fuel assemblies to the external environment—slightly impeding the local cooling of the fuel assemblies. The thermal resistance of crud and/or oxide layers would primarily serve to decrease radial heat losses at the outer perimeters of the fuel assemblies; this effect would not be significant because the heat flux (rate of heat transfer from the fuel rods) would be relatively low. In fact, a 1979 Sandia Laboratories report states that "[a] calculation was made to determine whether a 100 micron [crud] Fe_2O_3 coating on the BWR fuel pins would affect the heatup of these pins during a pool drainage accident, and it was found that the overall effect on the fuel pin temperature was less than one degree."[130]

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[126] Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, SAND77-1371, March 1979, (ADAMS Accession No. ML120960637), p. 40.

[127] Emissivity is "[t]he ratio of the power per unit area radiated by a surface to that radiated by a black body at the same temperature. A black body therefore has an emissivity of 1 and a perfect reflector has an emissivity of 0." A black body is "[a] hypothetical body that absorbs all the radiation falling on it. … While a true black body is an imaginary concept, a small hole in the wall of an enclosure at uniform temperature is the nearest approach that can be made to it in practice." See Alan Isaacs *et al.*, "A Concise Dictionary of Physics," Oxford Reference, 1990, pp. 22, 88.

[128] Brady Hanson *et al.*, "Gap Analysis to Support Extended Storage of Used Nuclear Fuel." PNNL-20509, Rev. 0, Pacific Northwest National Laboratory, January 31, 2012, p. 88.

[129] *Id.*

[130] Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, SAND77-1371, March 1979, (ADAMS Accession No. ML120960637), p. 78.

It is doubtful that the Sandia Laboratories calculation used the lowest possible value that the thermal conductivity of crud layers can have; the morphology of crud plays more of a role than its chemical content does in determining the degree of its thermal resistance[131] (this was not necessarily known in 1979). It is also doubtful that a calculation done in 1979 or earlier would have accurately modeled (or attempted to model) the internal thermal resistance of spent fuel rods. (In fact, today, in 2013, the computer safety model the NRC uses for SFP accident analyses—MELCOR—does not model the gap between the fuel cladding and fuel pellets. MELCOR also replaces the thermo-physical properties of UO_2 fuel with properties of compacted magnesium oxide (MgO). In SFP-fire experiments conducted at SNL—used to benchmark MELCOR—zirconium cladding is packed with solid magnesium oxide filler.[132]) Furthermore, the burnups of spent fuel assemblies were far lower in 1979 than they are today. Nonetheless, the Sandia Laboratories calculation results are instructive: the overall effect of the degraded thermal conductivity of high burnup fuel rods in a SFP accident would be slight (unless such fuel rods were involved in a criticality accident[133]).

If high burnup fuel rods (or other spent fuel rods) were involved in a criticality accident as the water boiled away in the pool, any degraded thermal conductivity of such fuel rods would play a *significant role* in increasing local fuel and fuel-cladding temperatures, because the heat flux would be high.

II.D. Station Blackout Boil-Off Scenarios Could Lead to Criticality Accidents

II.D.1. Neutron-Absorber Materials and How the Potential for Criticality Accidents in SFPs Has Increased

Neutron-absorber materials are needed in the SFP storage racks that have densely packed fuel assemblies—high-density storage racks. Neutron-absorber materials are needed *to help prevent criticality accidents*; in fact, "new rack designs rely heavily on permanently

[131] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, (ADAMS Accession No. ML032940295), p. 240.
[132] Jeffrey Cardoni, Sandia National Laboratories, "MELCOR Model for an Experimental 17x17 Spent Fuel PWR Assembly," SAND2010-8249, November 2010, p. 4.
[133] Fission—the splitting of atoms in the nuclear fuel—occurs in a criticality accident.

installed neutron absorbers to maintain criticality requirements."[134] High-density storage racks also rely on the particular arrangements of discharged and fresh fuel assemblies that help to control reactivity in the SFP. In the SFP, fuel assemblies might be arranged within checkerboard configurations.

One of the reasons that criticality accidents could occur in SFP high-density storage racks is that the center-to-center distance between the spent fuel assemblies (the pitch) in such racks is close to that of fuel assemblies in the reactor core. For example, some BWR reactor cores have a fuel assembly pitch of 6.0 inches (in)[135] and some BWR SFPs have a spent fuel assembly pitch of 6.28 in.[136] Furthermore, some PWR reactor cores have a fuel assembly pitch of 8.587 in[137] and some PWR SFPs have a spent fuel assembly pitch of 9.0 in.[138]

A May 2010 NRC document states that "the dimensions of the SFPs cannot be changed so licensees are putting more fuel assemblies into the same volume." The May 2010 NRC document provides an example: "One plant went through several steps to go from a SFP originally licensed for a capacity of 600 fuel assemblies to its current licensed capacity of 3300."[139]

The May 2010 NRC document observes that over the years "fuel assemblies have become more reactive;" and states:

> Increased U235 enrichment is an example [of increased reactivity]. But there are other more subtle changes; *e.g.*, increased fuel pellet diameter; increased fuel pellet density; the BWR transition from fuel assemblies with 49 fuel rods to those with 91 fuel rods; increased use of fixed and integral burnable absorbers; and, changes to core operating parameters due

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134 NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," May 21, 2010, (ADAMS Accession No. ML101520463), p. 1.

135 OECD Nuclear Energy Agency (NEA), "Boiling Water Reactor Turbine Trip (TT) Benchmark," Volume I, "Final Specifications," NEA/NSC/DOC(2001)1, February 2001, p. 9.

136 K. C. Wagner, R. O. Gauntt, Sandia National Laboratories, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents And Extension of Reference Plant Analyses to Other Spent Fuel Pools," November 2006, (ADAMS Accession No. ML120970086), p. 57.

137 NRC, "Pressurized Water Reactor, B&W Technology, Crosstraining Course Manual," Chapter 2.1, "Core and Vessel Construction," Rev 10/2007, (ADAMS Accession No. ML11221A103), p. 2.1-14.

138 NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4-6.

139 NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," pp. 1, 2.

> to power uprates resulting in more reactive fuel assemblies to be stored in the SFP.[140]

Regarding historical limitations of critical experiments that did not include actinide and fission product nuclides that are important to determining reactivity in a SFP environment, the May 2010 NRC document states:

> [H]istorically the critical experiments used in the benchmarking do not include Actinide and Fission Product nuclides that are important to determining reactivity in a SFP environment. Rather than address the issue in the validation, SFP [license amendment requests] were silent on the issue. This is inconsistent with NRC guidance on performing these validations as described in NUREG/CR-6698[141] [published in January 2001]. Historically there were no publicly available experiments with Actinide and Fission Product nuclides. With the issuance of NUREG/CR-6979[142] [in September 2008], experiments with Actinides are available for benchmarking. However, *there are still limited experiments that contain Fission Products* that can be used in the validation[143] [emphasis added].

And regarding the fact that thinner fuel cladding usually results in a higher reactivity for spent fuel rods, the May 2010 NRC document states:

> As a fuel assembly is depleted in an operating reactor, it undergoes physical changes. Some of those changes have the potential to affect the SFP criticality analysis. For example, fuel rods experience irradiated rod growth. As the rods get longer, the cladding gets thinner. Thinner cladding usually results in a higher reactivity. As the amount of burnup credited in the analysis increases the more of an effect this could have on the criticality analysis.[144]

The May 2010 NRC document concludes that industry trends over the years "have resulted in reduced conservatism/margins to criticality, thus reducing or eliminating the ability to use engineering judgment when determining that there is reasonable assurance an inadvertent SFP criticality cannot occur. Additionally,

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[140] *Id*., p. 3.
[141] NRC, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, January 2001, (ADAMS Accession No. ML050250061).
[142] NRC, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," NUREG/CR-6979, September 2008, (ADAMS Accession No. ML082880452).
[143] NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," p. 5.
[144] *Id*.

uncertainties associated with SFP criticality analyses, due to lack of benchmark data for example, also decrease margins to criticality."[145]

II.D.1.a. Information about Newer BWR Fuel Assemblies

Regarding axial variations in BWR fuel assemblies of the loading of the U235 enrichment and gadolinium integrated burnable absorber material, an October 2000 Oak Ridge National Laboratory ("ORNL") report states:

> [N]ewer BWR fuel designs typically employ larger arrays (*e.g.*, 9 x 9 and 10 x 10) of smaller fuel rods with higher enrichments, increased gadolinium loading through higher concentrations and more gadolinium-bearing rods, and greater axial variation in enrichment and gadolinium loading. … The axial variations in enrichment and gadolinium loading naturally necessitate separate calculations for unique axial segments and are important to the criticality calculations.[146]

There are also axial variations in the number of fuel rods in BWR fuel assemblies.[147]

And regarding the effect of integrated burnable absorbers on the reactivity behavior of BWR fuel as a function of burnup, the October 2000 ORNL report states:

> [F]or BWR fuels (with integrated burnable absorbers) the reactivity increases as fuel burnup proceeds, reaches a maximum at a burnup where the absorber (gadolinium) is nearly depleted, and then decreases monotonically with burnup in a nearly linear fashion. The initial period of burnup (*i.e.*, before the gadolinium is depleted and the reactivity peaks) adds an additional complication to BWR depletion that is not present in the depletion of PWR fuels (without integrated burnable absorbers).[148]

(The October 2000 ORNL report states that "[f]or PWR fuels (without integrated burnable absorbers), the reactivity decreases monotonically with burnup in a nearly linear fashion."[149])

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[145] *Id.*, p. 3.
[146] J. C. Wagner, M. D. DeHart, and B. L. Broadhead, Oak Ridge National Laboratory ("ORNL"), "Investigation of Burnup Credit Modeling Issues Associated with BWR Fuel," ORNL/TM-1999/193, October 2000, pp. 62-63.
[147] NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," p. 1.
[148] J. C. Wagner, M. D. DeHart, and B. L. Broadhead, Oak Ridge National Laboratory ("ORNL"), "Investigation of Burnup Credit Modeling Issues Associated with BWR Fuel," ORNL/TM-1999/193, p. 13.
[149] *Id.*

II.D.2. Neutron-Absorber Materials Could Melt in a Station Blackout Boil-Off Accident

In a SBO boil-off accident, if enough water boiled off, the water level would drop in the pool and uncover the fuel assemblies. If fuel assemblies were uncovered, temperatures in the SFP could increase enough to cause neutron-absorber materials placed in high-density storage racks to melt. Boraflex and Boral are neutron-absorber materials. Boraflex vitrifies and melts at approximately 300°C (572°F) and 500°C (932°F), respectively; Boraflex would be ineffective once heated above approximately 300°C (572°F).[150] And Boral melts at approximately 657°C (1214°F).[151]

Regarding the melting of Boral in a SFP accident, EPRI SFP accident guidance states:

> With the aluminum cladding and the aluminum in the mixture, the BORAL would melt at temperatures of 1200°F (660°C). With this low melting temperature, the conservative evaluations for the intact response of those SFP configurations using this absorber are recommended to not use temperatures higher than 1100°F (593°C). If conditions are detected that would lead to conditions where the estimated fuel assembly/bundle temperature exceeds this value, the SFP should be assumed to have a degraded configuration *including the possible melting and downward relocation of the BORAL absorber plates to the bottom of the SFP*. Specifically, the reactivity of the SFP should be considered to be uncertain and somewhat increased from the nominal pool value[152] [emphasis added].

EPRI SFP accident guidance also notes that "[t]ypically the fuel assemblies/bundles are supported within stainless steel racks that form a square matrix" but that "[s]ome racks are also fabricated from aluminum."[153] And EPRI states that Boraflex could "begin to relocate due to softening or melting" above approximately 300°C (572°F).[154]

(It is noteworthy that in a SNL experiment simulating a complete SFP LOCA in which BWR fuel assemblies were heated in air, "[p]ost-mortem examination of the integral test assemblies revealed gross distortion of the pool rack and channel box,

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[150] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-9.

[151] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. 1.

[152] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," Appendix EE, pp. EE-8-EE-9.

[153] *Id.*, Appendix EE, p. EE-9.

[154] *Id.*

rubblization of the tubing bundle and accumulation of debris on the bottom tie plate that resulted in flow blockage. Flow blockage was also evident from *molten aluminum (originating from Boral plates built into the pool rack)* that *collected on and below the bottom tie plates*"[155] [emphasis added].)

And regarding what the SFP water levels would be when Boraflex and Boral began to lose their integrity, EPRI SFP accident guidance states:

> If the water level [in the SFP] were to decrease to approximately 0.85 (Boraflex) or 0.75 (Boral) of the fuel height, the fuel assembly/bundle outlet temperature would approach a level where the integrity of the neutron absorber shims would be in question and the geometric configuration of the structures in the SFP could begin to change.[156]

A 2001 NRC report, NUREG-1738, states that "[i]f the stored assemblies are separated by neutron absorber plates (*e.g.*, Boral or Boraflex), loss of these plates could result in a potential for criticality for BWR pools."[157] (BWR SFPs do not use borated water.[158] One of the reasons that criticality accidents could occur in BWR SFP high-density storage racks is that the center-to-center distance between the spent fuel assemblies (the pitch) in such racks is close to that of fuel assemblies in the reactor core.)

EPRI SFP accident guidance does not specify the extent that the neutron-absorber materials would relocate downward immediately after they began to melt in a SFP boil-off accident; however, it is not likely that neutron-absorber materials located below the water surface would be adversely affected by any downward relocation of melted materials. Hence, the neutron-absorber materials located below the water surface would remain intact and continue to prevent criticality accidents. But if the water level continued to drop, additional neutron-absorber materials, located at lower elevations, would melt. In this scenario, the SFP would be vulnerable to criticality accidents if water

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155 E. R. Lindgren, Sandia National Laboratory, "Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident," NUREG/CR-7143, March 2013 (ADAMS Accession No. ML13072A056), p. xx.

156 EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-17.

157 NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February 2001 (ADAMS Accession No. ML010430066), p. 3-26.

158 NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 30.

were injected back into the SFP to raise the water level and cover the fuel assemblies, again.

Regarding injecting water into a SFP with a reduced water level, EPRI SFP accident guidance recommends considering whether or not "the water level has been sufficiently low [such] that *the location of the absorber material* could have been jeopardized"[159] [emphasis added]. EPRI also recommends injecting *borated* water into the SFP if it is available.[160]

Discussing the progression of a partial *BWR* SFP LOCA, a 2013 PSU report observes that after the Boral in high-density storage racks melted, neutrons would diffuse throughout the SFP, and possibly cause fission to commence.[161] (Although not explicitly stated, the 2013 PSU report must be referring to scenarios in which water would be injected back into the drained SFP.) If fission were to occur, local fuel and fuel-cladding temperatures would rapidly increase. Fission would also "cause an increase in decay products, which [would] have a delayed effect on temperature increase[s]."[162] (A June 2013 NRC document states that "if an [inadvertent criticality event] were severe enough to produce significant heat, the fuel will be harder to cool."[163]) And radiation releases, caused by a criticality accident in a SFP, would impede (or possibly prevent for significant time periods) efforts to mitigate a SBO boil-off accident (or a partial SFP LOCA), making it more probable that such accident scenarios would lead to SFP fires. (SFPs would also be vulnerable to criticality accidents after Boraflex vitrified in high-density storage racks and became ineffective.)

EPRI SFP accident guidance states that "BWR analyses have indicated that spraying water into fresh, uncovered fuel bundles could result in a critical configuration. This could possibly also be the case for spent fuel where there has been sufficient decay

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[159] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-12.
[160] *Id.*, Appendix EE, p. EE-13.
[161] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, May 2013, p. 1.
[162] *Id.*, pp. 1-2.
[163] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 29.

of fission products that act as poisons."[164] *Perhaps* it would be valid to speculate that if there were one or more criticality accidents, severe enough to produce significant heat, a SFP fire would not always commence on a fuel assembly that was part of the group most recently discharged from the reactor core—the group producing the highest amount of decay heat in the pool.

(It is noteworthy that a June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs provides the results of a number of MELCOR computer safety model simulations of SFP LOCAs in which there was a *moderate leakage rate*.[165] In some simulations, SFP temperatures reached the point at which neutron-absorber materials would melt; spray cooling was employed,[166] however, the MELCOR simulations did not even consider the possibility of criticality accidents.[167])

II.D.3. The Potential for Criticality Accidents When Water is Sprayed onto Fuel Assemblies in Certain Spent Fuel Pools

Optimum-moderation conditions occur in water films and low-density water; optimum-moderation conditions could increase the potential for criticality accidents in a SBO boil-off accident if the pitch of PWR fuel assemblies were in a range between approximately 25 centimeters ("cm") (9.84 in) and 50 cm (19.68 in).[168] It would seem that the range of vulnerable pitches would be similar for BWR fuel assemblies; however, criticality analyses should be conducted on a case by case basis to make such a determination. Criticality analyses need to consider how optimum moderator conditions would affect BWR fuel assemblies with or without channel boxes. Plant specific criticality analyses should also be conducted for PWR fuel assemblies.

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[164] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-12.

[165] A moderate leakage rate is "[a] state with leakage from the bottom of the SFP, corresponding to through-wall concrete cracking at the bottom of the walls and tearing of the liner that propagates to an extent such that water leakage is controlled by the size of the cracks in the concrete." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 61.

[166] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," pp. 211-213.

[167] *Id.*, p. 20.

[168] G Caplin *et al.*, "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," Institut de Radioprotection et de Sûreté Nucléaire ("IRSN"), 2011, Information Sheet.

According to an April 1989 NRC report, NUREG-1353, BWR and PWR medium-density storage racks have fuel-assembly pitches of 22.86 cm (9.0 in) and 33.02 cm (13.0 in), respectively. And NUREG-1353 states that BWR and PWR low-density storage racks both have fuel-assembly pitches in a range from 50.8 cm (20.0 in) to 76.2 cm (30.0 in).[169] It is clear that NUREG-1353 provides values of typical fuel-assembly pitches; plant-specific values of fuel-assembly pitches would be likely to vary.

In the US, there are not many (if any) SFPs that have fuel assembles stored in low-density racks. Any plans to re-rack fuel assembles stored in high-density racks to either medium-density or low-density racks, need to consider requiring that neutron-absorber materials be placed into the new storage racks; criticality analyses should be conducted on a case by case basis to determine if neutron-absorber materials would be needed.

(It is noteworthy that "[a] PWR SFP would typically end up with high density and "moderate density" racks. … PWRs have a need to store fresh fuel assemblies in the SFP. To accommodate that need PWRs typically installed "moderate density" storage racks with smaller flux traps than the original and usually more neutron absorber than the high-density storage racks. With a fully intact neutron absorber, burnup requirements for the high-density storage racks can be fairly low and fresh fuel up to 5.00 w/o U235 could easily be accommodated in the moderate density storage racks.[170])

Optimum-moderation conditions in water films and low-density water do not increase the potential for criticality accidents in PWR SBO boil-off accidents if the pitch of the fuel assemblies is less than approximately 25 cm (9.84 in). It would seem that the dividing pitch-value of approximately 25 cm would also be true for BWR fuel assemblies; however, criticality analyses should be conducted on a case by case basis to make such a determination.

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[169] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4.6.

[170] NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," May 21, 2010, (ADAMS Accession No: ML101520463), pp. 1-2.

In an operating reactor core, a decrease of water density decreases the reactivity of the core,[171] because the pitch of the fuel assemblies is less than approximately 25 cm; in BWR cores, the fuel assembly pitch is approximately 6.0 inches. In fact, in the upper regions of BWR cores, steam voids, which decrease water density, reduce the core reactivity. An inherent safety feature of BWRs is that "a transient power increase will produce more steam voids, reducing reactivity, which reduces power and thus limits the excursion."[172]

A June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs recommends cooling fuel assemblies with a "spray flow" in partial SFP LOCA scenarios in which there would be "no natural circulation of air through the racks."[173] It is pertinent that in the Fukushima Dai-ichi accident "pumper trucks employing high booms spray[ed] water from a distance into the spent fuel pools." There was no other means available to the operators; hence, the Near-Term Task Force that the NRC established in response to the Fukushima Dai-ichi accident recommended that the NRC "[o]rder licensees to have an installed seismically qualified means to spray water into the spent fuel pools, including an easily accessible connection to supply the water (*e.g.*, using a portable pump or pumper truck) at grade outside the building."[174]

In the event of a partial SFP LOCA, it would be important to cool the fuel assemblies if a means were available. However, spraying water onto exposed fuel assemblies (especially unused fresh-fuel assemblies—more reactive than spent fuel), stored in racks that had a fuel-assembly pitch in a range between approximately 25 cm and 50 cm and did not have neutron-absorber materials, or whose neutron-absorber materials had previously melted, could cause a criticality accident. (In some partial SFP LOCA and boil-off scenarios, neutron-absorber materials would begin melting when the

171 G Caplin *et al.*, "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.

172 NRC, "BWR/4 Technology Manual (R-104B)," NRC Technical Training Center, Rev 0100, (ADAMS Accession No: ML022830867), p. 1.3-1.

173 NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," Appendix D, p. D-12.

174 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, (ADAMS Accession No. ML111861807), pp. 45, 46.

water level dropped 15 or 25 percent (depending on the materials) below the top of the fuel assemblies.)

Historically, neutron-absorber materials were not typically placed in either low-density open racks or medium-density racks with flux traps in order to help prevent criticality accidents.[175] (PWR SFPs are required to use water that is borated with at least 2000 parts per million ("ppm") of boron; BWR SFPs do not use borated water.[176])

A 2011 Institut de Radioprotection et de Sûreté Nucléaire ("IRSN") information sheet on preventing SFP criticality accidents, in the event that PWR fuel assemblies would be uncovered by the pool's water, discusses results of an investigation of the potential affects of low-density optimum-moderator water conditions. The CRISTAL computer code was used to simulate scenarios in which a SFP did *not* have neutron-absorber materials in its storage racks; the SFP contained 625 undamaged uranium-oxide PWR 17x17 assemblies in a height of 1.5-meters water.[177] The CRISTAL computer code simulated how different low-density optimum-moderator water conditions would affect the effective neutron multiplication factor (k_{eff})[178] of fuel assemblies that had different pitches in a range between approximately 25 cm and 50 cm.

The 2011 IRSN information sheet states that "injecting borated water to cool the [uncovered] assemblies [is] preferable when possible." The use of borated water would help prevent a criticality accident, because boron absorbs neutrons; however, the 2011 IRSN information sheet states that water borated with 2000 ppm of boron reduces yet does not completely eliminate the risk of criticality.[179] (It is pertinent that "BWR SFPs do not use borated water so the fact that the SFP may be refilled with unborated water is not a deviation from the norm."[180])

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[175] NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, p. 3-25.
[176] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 30.
[177] G Caplin *et al*., "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.
[178] The effective neutron multiplication factor (k_{eff}) is the estimated ratio of neutron production to neutron absorption and leakage.
[179] G Caplin *et al*., "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.
[180] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 30.

Experiments measuring the densities of water discharged from sprinklers and fire hoses have found that the maximum water density (0.004 g/cm^3) is well below the *optimum-moderation* densities of water (0.05 to 0.1 g/cm^3) that could cause fuel assemblies to have a criticality accident. However, "it has been observed[181] that a quantity of mist moderation judged to be safe might still be unacceptable due to water film formation on the fuel material. The film thickness is due to the viscosity of water."[182] (In a SBO boil-off scenario with heated uncovered fuel assemblies, there would be water film formation on fuel-cladding surfaces after the cladding was cooled by the sprayed water. EPRI SFP accident guidance states that "effective cooling could be initiated because the spray droplets wet the high temperature cladding surface and cause the formation of a sputtering water film that slowly drains over the high temperature cladding and quenches it."[183]) Simulations with a computer safety model—KENO V.a—demonstrated that super-criticality could occur if water films formed on fresh fuel assemblies in dry storage racks, "display[ing] this effect for fuel assemblies containing 256 rods, composed of UO_2 at 4.1 wt.% enrichment, in a 16 x 16 array. The assemblies were in 19 x 34 storage array."[184] (See Figure 3.)

[181] The source of this information is provided in the report: D. A. McCaughey and G. H. Bidinger, "Film Effects of Fire Sprinklers on Low-Enriched-Uranium Storage Systems," Transactions of the American Nuclear Society, Vol. 56, p. 329 (1988).
[182] E. D. Clayton, Pacific Northwest Laboratory, "Anomalies of Criticality: Revision 6," PNNL-19176, February 2010, pp. 76-77.
[183] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-11.
[184] E. D. Clayton, "Anomalies of Criticality: Revision 6," PNNL-19176, pp. 77, 80.

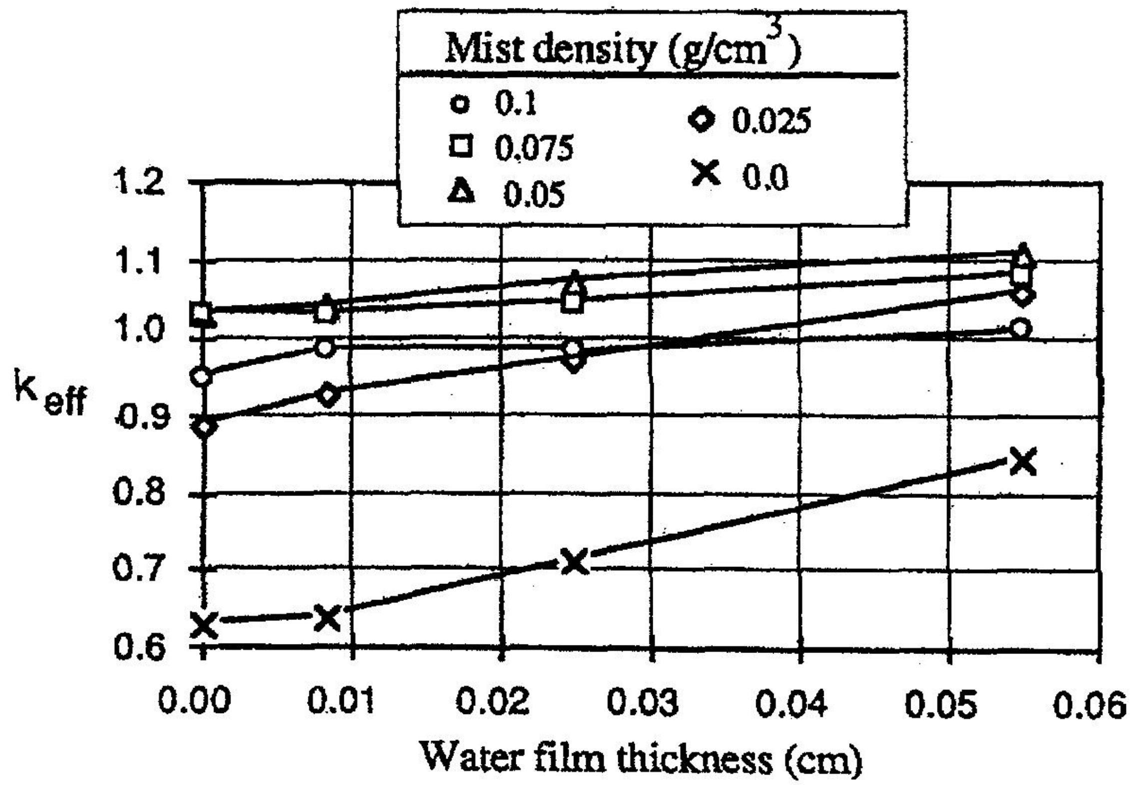

Figure 3. Film Effects of Water Sprinklers on Storage Array of 4.1%-enriched UO_2 Rods

In storage racks that had a fuel-assembly pitch in a range between approximately 25 cm and 50 cm and did not have neutron-absorber materials, or whose neutron-absorber materials had previously melted, the upper end of exposed *spent* fuel rods also could be susceptible to criticality if sprayed with water. In the case of PWR rods, "[t]he majority of [spent] PWR fuel assemblies have…significantly under-burned fuel at the ends (with burnup of 50 to 60% of the assembly average);" and the "under-burned [end] regions are dominant in terms of reactivity."[185] The 2011 IRSN information sheet states that there could be a risk of criticality in PWR rods with a burnup of 10 GWd/t *in their upper ends*[186] (their average burnup would be greater).

[185] J. C. Wagner, M. D. DeHart, Oak Ridge National Laboratory, "Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations, ORNL/TM-1999/246, March 2000, pp. 2, 5.

[186] G Caplin *et al*., "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.

II.D.4. The Potential for Criticality Accidents When Water Boils in Certain Spent Fuel Pools

If a SFP had a SBO boil-off accident, optimum-moderation conditions—caused by water films—could occur at locations where the boiling water's bubbly surface directly contacted fresh fuel assemblies.[187] The temperature at the surface of the fuel rods would be approximately 100°C; hence, water films in the bubbly surface would be in direct contact with the cladding. 100°C is below the temperature at which a boundary layer of vapor would form between water and a metal surface (the Leidenfrost phenomenon). Furthermore, the 2011 IRSN information sheet states that there could be a risk of criticality in a SFP boil-off accident from the water mist generated by boiling water;[188] the mist just above the boiling surface would be *extra-dense* water mist (with a density of approximately 0.05 g/cm^3) prone to optimum-moderation conditions.

Hence, after fuel assembles were uncovered, BWR SFPs would be susceptible to criticality accidents *in boiling water*—provided the fuel assemblies had a pitch in a range between approximately 25 cm and 50 cm and their storage racks did not have neutron-absorber materials. As stated, the 2011 IRSN information sheet states that water borated with 2000 ppm of boron reduces yet does not completely eliminate the risk of criticality;[189] therefore, if PWR fuel assembles were uncovered in boiling water, they would be also susceptible to criticality accidents.[190]

Therefore, any plans to re-rack fuel assembles stored in high-density racks to either medium-density or low-density racks, need to consider requiring that neutron-absorber materials be placed into the new storage racks; criticality analyses should be conducted on a case by case basis to determine if neutron-absorber materials would be needed. (In the US, there are not many (if any) SFPs that have fuel assembles stored in low-density racks.)

[187] This is pertinent to racks (without neutron-absorber materials) that had fresh fuel assemblies stored with a pitch in a range between approximately 25 cm and 50 cm.
[188] G Caplin *et al*., "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.
[189] *Id*.
[190] Provided the fuel assemblies had a pitch in a range between approximately 25 cm and 50 cm and their storage racks did not have neutron-absorber materials.

II.D.4.a. NRC Regulations: Criticality Accident Requirements

NRC regulations—10 C.F.R. § 50.68, Criticality Accident Requirements—require that safety analyses be conducted for scenarios in which fresh fuel assemblies, when housed in fresh fuel storage racks, *in a dry environment*, would be exposed to flooding, foam, or water mist, which fire fighting operations could cause. According to the NRC, "[f]oam or mist affects the neutron moderation in the [dry storage] array and can result in a peak in reactivity at low moderator density (called "optimum" moderation[191])."[192]

10 C.F.R. § 50.68, Criticality Accident Requirements, states:

> Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:
> …
> If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid [water is hydrogenous], the k-effective [the estimated ratio of neutron production to neutron absorption and leakage] corresponding to this optimum moderation must not exceed 0.98 [below 1.0 is subcritical], at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

10 C.F.R. § 50.68 requires that safety analyses be conducted for scenarios in which SFP "spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity" would be flooded with unborated (and with borated water) water. 10 C.F.R. § 50.68 also needs to require that safety analyses be conducted for SFP-accident scenarios in which fuel assemblies that had a pitch in a range between approximately 25 cm and 50 cm and were stored in racks that did not have neutron-absorber materials, or whose neutron-absorber materials had previously melted, would be exposed to either low-density, optimum-moderation firefighting foam or water mist, or water films.

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[191] J. M. Cano *et al*., "Supercriticality through Optimum Moderation in Nuclear Fuel Storage," Nuclear Technology, Vol. 48, p. 251 (1980).

[192] NRC, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1993, (ADAMS Accession No: ML072710248), Attachment 1, p. 4.

Perhaps the NRC would argue that in SFP-accident scenarios, it would *not* be a safety risk if such fuel assemblies[193] were exposed to either low-density, optimum-moderation firefighting foam or water mist, or water films.

A 2001 NRC report, NUREG-1738, states:

> The phenomenon of a peak in reactivity because of low-density (optimum) moderation (firefighting foam) *is not of concern* in spent fuel pools since the presence of relatively weak absorber materials, such as stainless steel plates or angle brackets, is sufficient to preclude neutronic coupling between assemblies. Therefore, personnel actions to refill a drained spent fuel pool containing *undeformed* fuel assemblies would not create the potential for a criticality[194] [emphasis added].

In their assessment that the presence of *relatively weak absorber materials*, such as stainless steel plates or angle brackets, would be sufficient to preclude neutronic coupling between assemblies, it is possible that the authors of NUREG-1738 were only considering spent fuel assemblies, overlooking the fact that fresh fuel assemblies—which are much more reactive—are also stored in SFPs. (In March 2011, the Fukushima Dai-ichi Unit 4 SFP was storing 204 fresh fuel assemblies (and 1331 spent assemblies).[195]) It is pertinent that fresh fuel storage racks, *in a dry environment*, also have stainless steel material—a relatively weak absorber material.

The authors of NUREG-1738 might have also overlooked the fact that the upper ends of *spent* fuel rods (perhaps PWR rods more than BWR rods) are significantly under-burned—"with burnup of 50 to 60% of the assembly average"—making those locations "dominant in terms of reactivity."[196] The authors of NUREG-1738 might have *non-conservatively* assumed that spent fuel assemblies have a uniform axial burnup distribution.

[193] Fuel assemblies that had a pitch in a range between approximately 25 cm and 50 cm and were stored in racks that did not have neutron-absorber materials, or whose neutron-absorber materials had previously melted.

[194] NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, Appendix 3, p. A3-2.

[195] Juan J. Carbajo, Oak Ridge National Laboratory, "MELCOR Model of the Spent Fuel Pool of Fukushima Dai-ichi Unit 4," 2012, p. 1.

[196] J. C. Wagner, M. D. DeHart, Oak Ridge National Laboratory, "Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations, ORNL/TM-1999/246, March 2000, pp. 2, 5.

Additionally, the authors of NUREG-1738 did not consider that "[s]ome [spent fuel] racks are also fabricated from aluminum,"[197] which like zirconium is quite transparent to neutrons.

The results of the CRISTAL computer code simulations discussed in the 2011 IRSN information sheet on preventing SFP criticality accidents, in the event that PWR fuel assemblies[198] would be uncovered by the pool's water, indicate that criticality accidents, caused by optimum-moderation low-density water or water films, could occur in SFPs.[199] This means that optimum-moderation conditions could cause criticality accidents in either low-density open racks, without neutron-absorber materials, or medium-density racks with non-borated flux traps—provided the stored fuel assemblies had a pitch in a range between approximately 25 cm and 50 cm.

II.D.5. Boraflex and Boral Degradation in Spent Fuel Racks

In a September 2013 NRC, Japan Lessons Learned Project ("JLLP") meeting, Rodney McCullum of the Nuclear Energy Institute ("NEI") stated: "We understand in the industry we can no longer rely on Boraflex. We're not relying on it anymore."[200] As stated above, Boraflex is a neutron absorber, intended to help prevent criticality accidents, that is located in spent fuel racks. Boraflex degrades; nonetheless, it is still used in a number of SFPs, including Indian Point Unit 2's.[201]

In May 2002, the high-density storage racks in Region 1-2 of the Indian Point Unit 2 SFP were "assumed to have sustained a 50 percent loss of Boraflex," due to degradation.[202] And, a NRC document, dated September 24, 2013, states that "[t]he

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[197] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-9.

[198] The PWR fuel assemblies could be either fresh fuel assemblies or assemblies with a burnup of 10 GWd/t *in their upper ends*; in both cases, the fuel assemblies would be stored in racks without neutron-absorber materials with a pitch in a range between approximately 25 cm and 50 cm.

[199] G Caplin *et al*., "Criticality Accident in Case of a Spent Fuel Pool Dry-Out," IRSN, 2011, Information Sheet.

[200] NRC, "Japan Lessons Learned Project Directorate Public Meeting," September 18, 2013, Transcript of Proceedings, (ADAMS Accession No: ML13277A215), p. 214.

[201] NRC, "Summary of August 26, 2013, Meeting with Entergy Nuclear Operations, Inc. and Netco on Indian Point Unit 2 Spent Fuel Pool Management," September 24, 2013, (ADAMS Accession No. ML13256A086), p. 1.

[202] NRC, Letter Dated May 29, 2002 to Michael R. Kansler, Entergy, "Indian Point Nuclear Generating Unit 2: Amendment Regarding Credit for Soluble Boron and Burnup in Spent Fuel

existing Unit 2 SFP criticality analysis of record, which takes credit for Boraflex inserts as neutron absorbers, was submitted by letter dated September 20, 2001."[203] Hence, even though there has doubtless been further degradation of Boraflex over the last dozen years, the Boraflex in a region of Indian Point Unit 2's SFP is still *assumed* to have sustained a 50 percent loss.

(Indian Point Unit 2 is a PWR; its SFP is required to use water that is borated with at least 2000 ppm of boron.)

Regarding Boraflex degradation and eroded subcriticality margins, a September 2012 NRC document states:

> Among neutron absorbing materials used in spent fuel pools, Boraflex degraded most severely. Boraflex is a neutron absorber material comprised of silicone polymer and boron carbide powder. When gamma-irradiated by spent nuclear fuel, Boraflex is prone to degradation and dissolution in the aqueous environment of the spent fuel pool. Consequently, *the subcriticality margins that existed when Boraflex was first installed have eroded*[204] [emphasis added].

As stated above, Boral is also a neutron absorber that is located in spent fuel racks; and like Boraflex, Boral degrades.

Regarding Boral degradation, NRC Information Notice 2009-26 states:

> Blisters and bulges of Boral cladding are material deformations that change the dimensions of the material. These blisters and bulges can be either water filled or gas filled (from the reaction of the SFP water and aluminum from the Boral), which may not be accounted for in the criticality analysis.[205]

And providing an example of the progression of Boral degradation, NRC Information Notice 2009-26 states:

> [T]he licensee at Beaver Valley stated that licensee inspections in 2007 of the Boral neutron absorber material coupons identified numerous blisters of the aluminum cladding, while only a few small blisters were identified

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Pit," Enclosure 2, "Safety Evaluation Regarding Indian Point Unit 2," (ADAMS Accession No: ML021230367), p. 3.

[203] NRC, "Summary of August 26, 2013, Meeting with Entergy Nuclear Operations, Inc. and Netco on Indian Point Unit 2 Spent Fuel Pool Management," p. 1.

[204] NRC, "Boraflex, RACKLIFE, and BADGER: Description and Uncertainties," September 2012, (ADAMS Accession No: ML12216A307), p. ii.

[205] NRC, "Information Notice 2009-26: Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool," October 28, 2009, (ADAMS Accession No: ML092440545), p. 4.

> in 2002. …blisters can displace water from the flux traps between storage cells and challenge dimensional assumptions used in the criticality analysis.[206]

The NRC's 2010 "On Site Spent Fuel Criticality Analyses NRR Action Plan," states that "virtually every permanently installed neutron absorber, for which a history can be established, has exhibited some degradation. Some have lost a significant portion of their neutron absorbing capability;" and states that "[t]he ability of licensees to control the material condition of any permanently installed neutron absorber that is credited for maintaining sub-criticality *is essential for the prevention of an inadvertent criticality event*" [emphasis added].[207] And NRC information notice 2009-26 states that "[t]he degradation mechanisms and *deformation rates* of any of the neutron-absorbing materials in the SFP are not well understood" [emphasis added].[208] (The degradation of neutron-absorber materials is especially worrisome for BWR SFPs, because they do not use borated water.[209])

It is possible that the degradation of neutron-absorber materials would increase the potential for a criticality accident occurring in the event a SFP boil-off accident. If overheated, it is probable that previously-degraded neutron-absorber materials would lose their effectiveness more quickly than non-degraded neutron-absorber materials.

II.E. Fuel Rods in a Spent Fuel Pool Would Balloon and Burst as It Boiled Dry, Impeding Local Cooling of the Fuel Assemblies

Heat produced by the radioactive decay heating of the fuel rods would cause the SFP's water to boil away; the fuel rods would become uncovered by water and heat up, increasing their local temperatures. When local fuel-cladding temperatures reached approximately 677°C (1250°F) the fuel rods would start to balloon and burst,[210]

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[206] *Id.*, p. 2.
[207] NRC, "On Site Spent Fuel Criticality Analyses NRR Action Plan," May 21, 2010, (ADAMS Accession No. ML101520463), pp. 1, 6.
[208] NRC, "Information Notice 2009-26: Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool," p. 4.
[209] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 30.
[210] The fuel rods would balloon and burst between approximately 677°C (1250°F) and 877°C (1610°F). See S. Güntay, J. Birchley, "MELCOR Further Development in the Area of Air

"releasing noble gases, such as xenon and krypton," into the environment.[211] This would occur because the fuel rods that are used in reactor cores are pre-pressurized: at higher temperatures, the internal gas pressure increases to points at which the fuel cladding balloons and bursts.

(Creep failure of the fuel cladding could occur from incurring stress for approximately 10 hours at cladding temperatures between approximately 565°C (1049°F) and 600°C (1112°F) or greater.[212] The NRC's NUREG-1738 states that "[w]hile failure of the cladding at these lower temperatures will lead to fission product release, such release is considerably smaller than that assumed for the cases where the temperature criterion is exceeded and significant fuel heatup and damage occurs."[213])

(It is noteworthy that the NRC computer safety model "MELCOR does not have a fuel cladding deformation and strain model. It uses a value of 900°C for widespread cladding failure."[214])

In a SFP boil-off accident, ballooning of the fuel cladding would most likely be in the form of sausage-type balloons, as occurred in the fuel-cleaning-tank accident at the Paks Nuclear Power Plant Unit 2 ("Paks-2"), in Hungary, in 2003.[215] In the Paks-2 accident, 30 fuel assemblies were severely damaged and their fuel rods ballooned—"long sausage balloons"[216] with "very long ballooned areas."[217] At a 2003 Advisory

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Ingress and Participation in OECDNEA SFP Project to Be Performed in the Time Frame 2009-2012," April 2009, p. 14.

[211] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, May 2013, p. 2.

[212] NRC, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, Appendix 1B, p. A1B-5.

[213] *Id*.

[214] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 26.

[215] In 2003, at the Paks Unit 2 plant in Hungry, there was a fuel cleaning tank accident in which 30 fuel assemblies incurred severe damage. In the Paks-2 accident, the fuel rods ballooned—"long sausage balloons"[215] with "very long ballooned areas."[215] See Advisory Committee on Reactor Safeguards Reactor Fuels Subcommittee, September 29, 2003, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2003/rf092903.pdf, pp. 212-225; see also IAEA, "OECD-IAEA Paks Fuel Project: Final Report," 2009, p. 12.

[216] Advisory Committee on Reactor Safeguards Reactor Fuels Subcommittee, September 29, 2003, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2003/rf092903.pdf, pp. 212- 225.

[217] IAEA, "OECD-IAEA Paks Fuel Project: Final Report," 2009, p. 12.

Committee on Reactor Safeguards ("ACRS") Reactor Fuels Subcommittee meeting, at least one participant thought that such long balloons would occur in reactor large-break loss-of-coolant accidents ("LOCA"). (*This is pertinent to the characteristics of the fuel-cladding ballooning that would occur in SFP accidents*, because, in both types of accidents, fuel rods would heat up to the point at which their internal-pressure increases caused them to balloon; in both types of accidents, the external pressure would be far less than the internal pressure of the fuel rods.)

In the ACRS meeting, Dr. Dana Powers (the lead author of "Cladding Swelling and Rupture Models for LOCA Analysis"[218]) stated: "If you're trying to persuade me that we'll never see long sausage balloons in reactor accidents, give up now while you're ahead;" and "where I run into trouble is saying x or y can never happen. Simply because you've never seen it in an experiment you've done with one foot sections [of fuel cladding]; that's where I have real trouble."[219]

(Experiments at Argonne Laboratories with segments of high burnup fuel rods—discussed in the same 2003 ACRS Subcommittee meeting—were conducted with 12 and 15 inch segments of fuel rods, with a "relatively uniform heating zone" *that was approximately five inches long*; hence, the ballooned locations of the fuel rods were not longer than five inches.[220])

In a SFP boil-off accident, it is highly probable that the ballooned sections of the fuel rods would be coplanar (at the same elevation); with coplanarity, there would also likely be some degree of local rod-to-rod contact. When local cladding temperatures reached the point at which the fuel rods ballooned, such temperatures would tend to be at approximately the same elevation. Additionally, in SFP boil-off accident, the fuel assemblies that were most recently loaded into the SFP (the hottest assemblies) would be the first ones to incur fuel-cladding ballooning.

In addition to the Paks fuel cleaning tank accident there is further evidence that there could be long sausage-like ballooned areas of the fuel cladding in a boil-off SFP

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[218] D. A. Powers, R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, April 1980, (ADAMS Accession No: ML053490337).
[219] Advisory Committee on Reactor Safeguards Reactor Fuels Subcommittee, September 29, 2003, pp. 217-218.
[220] *Id*. pp. 113, 181, and 195.

accident. (The experiments discussed in this paragraph are not SFP accident experiments; however, they apply to SFP accidents, because they are experiments in which fuel rod simulators were heated up to the point at which their internal pressure increases caused them to balloon.) For example: 1) the JAERI loss-of-accident tests had "axially extended contacts between rods (over more than 20 cm [7.9 in]) in [49-rod[221]] bundle configurations;"[222] 2) in the Materials Test 3 (MT-3), which had 12 full-length pre-pressurized fuel rods, "[t]he active strain [ballooned] region was spread over [a] ~2-[meter] (80-[in]) length" of the fuel rods[223] (this does not mean that there was a continuous ballooned length of about 80.0-in; however, it indicates that there was excessive ballooning); 3) an Oak Ridge National Laboratory ("ORNL") report states that for the CORA-16 experiment, there was *estimated* cladding strain (ballooning) on one of the fuel rods at the 550, 750, and 950 mm elevations, which indicates that the rod was estimated to have a ballooned length of at least 400 mm (15.75 in)[224] (the CORA experiments, which simulated meltdown accidents, were conducted with zirconium alloy multi-rod bundles that were two meters long);[225] 4) a second ORNL report states that for the CORA-33 experiment "the computed cladding strain [ballooning] was significant over 400 mm [15.75 in] of the rod length;"[226] and 5) the cladding balloons that occurred in the middle sections of the bundles from PWR FLECHT runs 2443 and 2544, which

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[221] European Commission: Nuclear Safety and the Environment, "Fuel Cladding Failure Criteria," September 1999, p. 88.

[222] Claude Grandjean, Institut de Radioprotection et de Sûreté Nucléaire (IRSN), "Coolability of Blocked Regions in a Rod Bundle after Ballooning under LOCA Conditions: Main Findings from a Review of Past Experimental Programmes," 2007.

[223] C. L. Wilson, G. M. Hesson, J. P. Pilger, L. L. King, F. E. Panisko, Pacific Northwest Laboratory, "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A," 1993, p. x.

[224] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," CONF-9105173-3-Extd.Abst., Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

[225] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, (ADAMS Accession No: ML042230460), p. 77.

[226] L. J. Ott, Siegfried Hagen, "Interpretation of the Results of the CORA-33 Dry Core Test," 1993.

had unintended internal gas pressure increases, [227] were substantially longer than a few inches.

Regarding assembly blockage in reactor LOCAs, resulting from newer zirconium fuel-cladding alloys like ZIRLO and M5, a 2004 OECD Nuclear Energy Agency report states that "[n]ew alloys have the tendency of being more ductile, which can increase ballooning size and thus increase blockage."[228] Furthermore, the same report states that "it can be anticipated, due to this better ductility that, for modern alloys, *the rod balloons will be bigger and the resulting flow blockage geometry at burst higher with more radial and axial extension* than for Zy4 [an older zirconium fuel-cladding alloy] rods when experiencing the same conditions at burst"[229] [emphasis added]. (As stated above, reactor LOCA fuel-cladding ballooning phenomena are pertinent to SFP accidents, because, in both types of accidents, fuel rods would heat up to the point at which their internal-pressure increases caused them to balloon; in both types of accidents, the external pressure would be far less than the internal pressure of the fuel rods.)

Interestingly, the 2004 OECD Nuclear Energy Agency report states that, *in a reactor LOCA*, "[t]here is a more uniform cladding temperature at high burn-up, which can lead to much larger cladding deformations and thus more pronounced flow blockage."[230] It is plausible that these same phenomena would occur in a SFP boil-off accident, because the fuel rods in the SFP would not have the pronounced *chopped-cosine axial heat flux distribution*[231] that the fuel rods have in operating reactor cores; the axial heat flux, albeit far less, would be far more evenly distributed in the fuel rods stored in the SFP.

The coplanar blockage of sausage-like fuel-cladding balloons (sections with a substantial axial extension), and any points of local rod-to-rod contact, would impede the

[227] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, (ADAMS Accession No: ML070780083), p. 3-95.
[228] OECD Nuclear Energy Agency, "Summary Record of the Experts meeting on the proposed OECD-IRSN STLOC Project," NEA/CSNI/R(2004)1, January 13, 2004, p. 5.
[229] *Id.*, p. 17.
[230] *Id.*, p. 5.
[231] The locations of the active length of the fuel rods are much hotter at the mid-elevation than at the upper and lower ends. The active length of a fuel rod is the length of the cladding containing the fuel pellets; it is approximately 12-feet long.

local cooling of the fuel assemblies; and local blockage-section surface temperatures could increase up the point at which the zirconium fuel-cladding began to rapidly chemically react with steam or air at approximately 1000°C (1832°F) or 900°C (1652°F),[232] respectively.

Ballooning and bursting would also cause the fuel-cladding to lose the protection of preexisting oxide layers, as clean surface locations opened up, facilitating exothermic (heat-generating) oxidation and hydriding of zirconium (both of these reactions are discussed below).

Additionally, local ballooning and bursting of zirconium fuel cladding at grid spacers would augment the cladding-to-grid contact. The NRC report, NUREG-2121, states that "[g]rid spacers may 'pin' rod ballooning… In bundle geometries, ballooning tends to occur such that all the balloons are coplanar, but ballooning is largely suppressed in the sections of fuel rods that cross a grid spacer."[233]

Regarding how fuel rod ballooning could *decrease* the time to the ignition of zirconium *in air* in a SFP accident, a 2009 paper about an OECD Nuclear Energy Agency SFP safety analysis project states:

> Fuel rod ballooning is an important phenomena expected to occur prior to ignition [of the zirconium fuel cladding in a SFP accident]. Rod ballooning has been shown to occur in the temperature range of 950 K to 1150 K [1250°F to 1610°F]. In the BWR 1×4 ignition test a peak clad temperature of 1050K [1430°F] was reached at 2.75 hrs and the rapid escalation to ignition began at 4.75 hrs at a peak clad temperature of 1200 K [1700°F]. Thus fuel rod ballooning is expected to occur during the crucial period prior to ignition *and could be expected to decrease the time to ignition by an hour or more*[234] [emphasis added].

It can be extrapolated that because fuel rod ballooning could decrease the time to the ignition of zirconium *in air* in a SFP accident, ballooning could also decrease the time to the ignition of zirconium *in steam* in a SFP accident.

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[232] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

[233] Patrick A.C. Raynaud, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," NUREG-2121, March 2012, (ADAMS Accession No: ML12090A018), p. 75.

[234] S. Güntay, J. Birchley, "MELCOR Further Development in the Area of Air Ingress and Participation in OECD NEA SFP Project to Be Performed in the Time Frame 2009-2012," April 2009, p. 14.

(It is noteworthy that the NRC claims that "rod ballooning has a low impact on the timing to breakaway oxidation."[235])

II.E.1. Fuel Fragmentation, Relocation, and Dispersal in a Spent Fuel Pool Boil-Off Accident

A March 2012 NRC report, NUREG-2121, states that "[f]uel fragmentation refers to any separation of the fuel pellet into more than one piece, regardless of when or why it occurred." In the reactor core, during typical operation, the uranium dioxide (UO_2) "fuel pellets develop many cracks because of thermal stresses."[236] A 2011 IAEA report explains that "[d]ue to thermal gradients, fuel pellets tend to fragment early in life,"[237] which can occur at fuel burnups "as low as a few megawatt days per metric ton uranium (MWd/MTU)."[238] It is likely that some degree of additional fuel fragmentation would occur in a reactor LOCA; a SFP accident would perhaps incur a lesser degree of additional fuel fragmentation than a reactor LOCA.

NUREG-2121 states that "[a]t higher values of burnup, fission gas production and migration is postulated to generate a 'rim' region in fuel pellets that is highly porous;" and that "[t]he size of fuel fragments is not uniform but tends to become smaller with increasing burnup."[239]

(It is noteworthy that a 2012 paper, "Oxidation Studies on Irradiated UO_2 Fuels," states that "[f]uel fragmentation would result in larger surface areas available for corrosion processes and radionuclide release."[240])

Defining fuel relocation, NUREG-2121 states that "fuel relocation can be described as any physical movement of fuel pellets or fuel fragments within the cladding. … *Radial* fuel relocation is the movement of the fuel outward toward the fuel cladding.

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[235] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 26.
[236] Patrick A.C. Raynaud, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," NUREG-2121, p. 3.
[237] IAEA, "Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management," No. NF-T-3.8, 2011, p. 37.
[238] Patrick A.C. Raynaud, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," NUREG-2121, p. 75.
[239] *Id.*, pp. 3, 75.
[240] D. Papaioannou *et al.*, "Oxidation Studies on Irradiated UO_2 Fuels" Top Fuel 2012 Transactions, European Nuclear Society, September 2-6, 2012, p. 5.

… *Axial* fuel relocation is the vertical movement of fuel fragments or particles within the cladding"[241] [emphasis not added].

Regarding *radial* fuel relocation, NUREG-2121 states that "fuel pellet cracking promotes an outward relocation of the pellet fragments that causes additional gap closure. This process is widely recognized in fuel performance analysis. It starts at beginning of life and quickly reaches equilibrium—by 5 GWd/MTU."[242] And regarding *axial* fuel relocation, NUREG-2121 states that "[u]nder normal operation, this process is usually limited by the fuel pellet immediately above or below the pellet in question." In experiments simulating reactor LOCAs, "voided regions of the cladding rod" and "additional fuel material [with]in the enlarged volume of the balloon region, or both" have been observed.[243]

Additionally, regarding the potential for the accumulation of relocated fuel fragments at the elevations of the spacer grids, NUREG-2121 states:

> Grid spacers may "pin" rod ballooning, potentially acting as choke points for fuel relocation. In bundle geometries, ballooning tends to occur such that all the balloons are coplanar, but ballooning is largely suppressed in the sections of fuel rods that cross a grid spacer.[244]

Regarding the fuel relocation which could occur in high burn-up fuel, in a reactor LOCA, a 2004 OECD Nuclear Energy Agency report states that "ANL [Argonne National Laboratory] tests have shown [the] potential for greater relocation at high burn-up due to increased fuel fragmentation. It is unknown if fuel-cladding bonding[245] delays relocation."[246] (This information is pertinent to the characteristics of the fuel-cladding ballooning and fuel relocation that could occur in SFP accidents, because, such

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[241] Patrick A.C. Raynaud, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," NUREG-2121, p. 3.
[242] *Id.*
[243] *Id.*
[244] *Id.*, p. 75.
[245] Regarding fuel-cladding bonding an October 2003 paper states: "Inner surface cladding oxidation and subsequent mechanical bonding between the fuel pellet and the cladding are well-known phenomena of high burnup and high duty fuels." See Sven Van den Berghe *et al.*, "Observation of a Pellet-Cladding Bonding Layer in High Power Fuel," presented at "Advanced Fuel Pellet Materials and Designs for Water Cooled Reactors: Technical Committee Meeting," 20–24 October 2003, p. 307.
[246] OECD Nuclear Energy Agency, "Summary Record of the Experts meeting on the proposed OECD-IRSN STLOC Project," NEA/CSNI/R(2004)1, January 13, 2004, p. 5.

phenomena could occur in both types of accidents.) The same report observes that larger fuel-cladding balloons—caused by "[n]ew alloys [that] have the tendency of being more ductile, which can increase ballooning size"—would be likely to facilitate a greater extent of fuel relocation and "the associated power generation increase."[247] (In a SFP accident any power generation increases caused by fuel relocation within fuel-cladding balloons would not be significant because the heat flux (rate of heat transfer from the fuel rods) would be relatively low.)

As stated above, the 2004 OECD Nuclear Energy Agency report states that "it can be anticipated, due to this better ductility that, for modern alloys, the rod balloons will be bigger and the resulting flow blockage geometry at burst higher with more radial and axial extension than for Zy4 [an older zirconium fuel-cladding alloy] rods when experiencing the same conditions at burst."[248]

The coplanar blockage of sausage-like fuel-cladding balloons (sections with a substantial axial extension) that had relocated fuel fragments, would impede the local cooling of the fuel assemblies; and local blockage-section surface temperatures could increase up the point at which the zirconium fuel-cladding began to rapidly chemically react with either steam or air at approximately 1000°C (1832°F) or 900°C (1652°F), respectively.

Defining fuel dispersal, NUREG-2121 states that "[f]uel dispersal is the ejection of fuel fragments or particles through a rupture or opening in the cladding."[249] Rapid reactor LOCA transient phenomena, such as rapid external depressurization, could enhance the dispersal fuel fragments from ruptured locations of the fuel cladding; external depressurization would not occur in SFP accidents.

Fuel dispersal during a reactor LOCA could occur with fuel that had a burnup lower than 62 GWd/MTU; previously it was believed that *significant* fuel dispersal during a reactor LOCA would not occur if the fuel had burnups lower than 62 GWd/MTU (average for the peak rod).[250] It seems probable that some degree of fuel

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[247] *Id.*
[248] *Id.*, p. 17.
[249] Patrick A.C. Raynaud, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," NUREG-2121, p. 3.
[250] *Id.*, p. 1.

dispersal would also occur in a SFP accident if the burnup were lower (or greater) than 62 GWd/MTU.) NUREG-2121 states that "[s]ome fuel dispersal has been observed in every case in which (1) rod rupture occurs, and (2) the fuel fragments are small enough to get through the rupture opening." And states that "[t]he amount of fuel that is dispersed can vary widely, from a puff of dust to large amounts of fragmented and pulverized fuel. Although evidence points to likely fuel dispersal in many tests, this phenomenon was not systematically investigated nor documented in the majority of test programs."[251]

II.F. Spent Fuel Pool Zirconium Fires in Steam and Air

Regarding the initiation and consequences of a SFP zirconium fire, a September 2013 NRC document, NUREG-2157, states:

> If cooling of the spent fuel were not reestablished, the fuel could heat up to temperatures on the order of 1,000°C (1,832°F). At this temperature, the spent fuel's zirconium cladding would begin to react with steam or air in a highly exothermic chemical reaction called a runaway zirconium oxidation reaction or autocatalytic ignition. This accident scenario is often referred to as a "spent fuel pool zirconium fire." Radioactive aerosols and vapors released from the damaged spent fuel could be carried throughout the spent fuel pool building and into the surrounding environment. This release could lead to exposures of the surrounding population and contamination of property (*e.g.*, land or structures) in the vicinity of the site.[252]

(Runaway zirconium oxidation would be more likely to commence in steam at local fuel-cladding temperatures between approximately 1000°C (1832°F) and 1200°C, (2192°F); and to commence in air at lower local fuel-cladding temperatures of 827°C (1520°F)[253] or 900°C (1652°F).[254]) (See Figure 4.)

[251] *Id.*, p. 75.

[252] NRC, "Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment," NUREG-2157, September 2013, (ADAMS Accession No. ML13224A106), Appendix F, p. F-2.

[253] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," pp. iv, 2, 3, 8, 13.

[254] Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

Figure 4. Zirconium Fuel Rod Simulators that Incurred Runaway Oxidation

II.F.1. In a Spent Fuel Pool Boil-Off Accident, a Zirconium Fire Could Ignite in Steam if Fuel-Cladding Temperatures Reached 1000°C (1832°F)

In a SBO boil-off accident, if the fuel assemblies were uncovered, the fuel cladding's zirconium content would initially chemically react with the steam produced by the boiling water in the SFP.[255] And if the water level in the SFP decreased to an elevation at approximately 66 percent of the height of the fuel assemblies, local fuel-cladding temperatures in the upper regions of the fuel assemblies would approach 2000°F (1093°C).[256] When local fuel cladding temperatures increased to approximately 1000°C (1832°F), the fuel cladding would incur significant additional heating from the exothermic (heat-generating) zirconium-steam reaction. The zirconium-steam reaction produces zirconium dioxide, hydrogen, and energy; the equation for the reaction is

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[255] Randall Gauntt *et al.*, Sandia National Laboratories "Fukushima Daiichi Accident Study: Status as of April 2012," SAND2012-6173, August 2012, p. 183.

[256] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-17.

written as Zr + $2H_2O \rightarrow ZrO_2 + 2H_2$ + energy. The energy (heat) generated by the reaction is approximately 6.45 megajoules per kilogram (kg) of Zr reacted.[257]

When zirconium reacts in steam it is possible for the reaction to become steam-starved, which occurs when hydrogen produced by the zirconium-steam reaction locally replaces steam (to varying degrees) at the surface of a fuel rod. This will mitigate oxidation rates or completely prevent oxidation.

The fuel-cladding outer surfaces of spent fuel assemblies are coated with varying thicknesses of zirconium dioxide layers (oxide layers). Oxide layers form on the fuel rods' cladding over the course of three or more years of operation in the reactor core, at elevated temperatures: typical BWR and PWR coolant temperatures are 540-550°F and 540-620°F, respectively.[258] There are also local crud (corrosion products) deposits on the outer surfaces of fuel cladding. Higher burnup fuel cladding typically has thicker oxide layers, and a higher hydrogen content. In a SFP accident the outer fuel-cladding oxide layer *can* function as a protective layer; the oxidation of zirconium at elevated temperatures *could* be "controlled by the diffusion of oxygen through the oxide [layer, with] the reaction rate [being] inversely proportional to the oxide thickness."[259] However, if the cladding temperature increases, the temperature may become the dominating factor that drives the zirconium-oxidation reaction, causing a rapid cladding-temperature escalation.[260] (In the PHEBUS B9R-2 test—conducted with a pre-oxidized test bundle—oxide layers did not prevent a rapid fuel cladding temperature escalation from commencing *in steam* at a relatively low temperature: 1027°C (1880°F); PHEBUS B9R-2 is discussed in Section F.2.a.)

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[257] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, (ADAMS Accession No. ML053490333), p. 8-2.
[258] IAEA, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels," IAEA-TECDOC-1470, October 2005, p. 7; and IAEA, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessels," IAEA-TECDOC-1120, October 1999, p. 5.
[259] S. Hagen, H. Malauschek, S. O. Peck, K.P. Wallenfels, "Temperature Escalation in PWR Fuel Rod simulator Bundles due to the Zircaloy-Steam Reaction: Test ESBU-1: Test Results Report," KfK-3508, December 1983, p. 4.
[260] *Id.*, p. 5.

(In air, nitrogen-related breakaway oxidation behavior would cause the *protective* oxide layer to degrade at approximately 800°C; and oxidation rates would begin accelerating.[261])

A SFP fire is *primarily a zirconium fire*: the *runaway* chemical reaction between zirconium and steam (or zirconium and air): *runaway zirconium oxidation*. Runaway zirconium oxidation causes *thermal runaway*, because zirconium oxidation is exothermic: the heat produced by the zirconium-steam reaction increases the local fuel-cladding temperature, which in turn increases the reaction rate, further increasing the local fuel-cladding temperature, and so on. Once runaway zirconium oxidation commences *in steam* (typically at local fuel-cladding temperatures between approximately 1000°C (1832°F) and 1200°C (2192°F), local fuel-cladding temperatures increase rapidly, leading to temperature increases of tens of degrees Fahrenheit per second. Hence, local fuel-cladding temperatures can escalate up to the point where zirconium melts—above 1816°C (3300°F)[262]—within a few minutes.

II.F.2. In a SFP Boil-Off Accident, a Zirconium Fire Might Not Ignite in Steam if Fuel-Cladding Temperatures Reached 1000°C (1832°F) or Greater

In SBO boil-off accident, it is possible that there would not be a temperature escalation, if local fuel-cladding temperatures increased to approximately 1000°C (1832°F), because the initial heatup rate of the fuel cladding would be very slow, as discussed in Section C.

After the fuel cladding were uncovered it would *initially* heat up *very slowly*, in some scenarios, at local rates lower than 0.01°C/sec (0.018°F/sec);[263] in other scenarios, local heatup rates would be approximately 0.13°C/sec (0.23°F/sec).[264]

Regarding the zirconium-steam reaction in the reactor core, a 1999 paper, "Current Knowledge on Core Degradation Phenomena, a Review," states that if the initial fuel-cladding temperature heat-up rate is 0.2°C/sec or lower, the heat-up rate will become

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[261] C. Duriez, T. Dupont, B. Schmet, F. Enoch, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," Journal of Nuclear Materials 380 (2008), pp. 30, 39, 40, 43, 44.

[262] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, (ADAMS Accession No: ML011800519), p. 3-1.

[263] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," p. 19.

[264] EPRI, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, Appendix EE, p. EE-10.

3.0°C/sec or lower if fuel-cladding temperatures reach 1200°C, because of the heat that would be contributed from the exothermic zirconium-steam reaction. The same paper also states that if the initial fuel-cladding temperature heat-up rate is 1.0°C/sec or greater, the heat-up rate will become 10.0°C/sec or greater if fuel-cladding temperatures reach 1200°C, because of the heat that would be contributed from the exothermic zirconium-steam reaction.[265]

An initial fuel-cladding temperature heat-up rate of 1°C/sec or greater means that there will be a thinner oxide thickness on the fuel cladding *for a particular temperature*; hence, oxidation rates become greater at fuel-cladding temperatures at which the exothermic zirconium-steam reaction contributes significant heat (6.45 megajoules per kg of Zr reacted).[266]

(It is noteworthy that if there were one or more criticality accidents in a SBO boil-off accident, after the fuel assemblies were uncovered, the heat generated from fission would cause rapid local fuel-cladding temperature increases.[267] Hence, it would be possible for initial heatup rates of the fuel cladding to be 1.0°C/sec or greater. If fuel-cladding temperatures that had initial heatup rates of 1.0°C/sec or greater were to increase to between approximately 1000°C (1832°F) and 1200°C (2192°F) *in a steam environment*, runaway zirconium oxidation would most likely commence.)

Regarding the fact that the CORA experiments conducted with lower heat-up rates did not have temperature escalations, a 1996 European Commission report states:

> The CORA experiments performed with lower heat-up rates demonstrated clearly that no temperature escalation took place. The chemical interaction energy evolved caused only an increased heat-up rate between [1200°C (2192°F)] and [1800°C (3272°F)] of about [1.0°C/sec (1.8°F/sec)]. The oxide layer which has formed on the cladding outer surface during heat-up delays the chemical interactions between Zircaloy and steam since the diffusion of oxygen through the ZrO_2 layer is the

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[265] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 205.

[266] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," NUREG/CP-0114, Vol. 2, 1990, (ADAMS Accession No. ML042250131), p. 7.

[267] Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," NucE431W S2013, May 2013, pp. 1-2.

reaction rate-determining step. The Zircaloy will be almost completely oxidized, or at least converted into α-Zr(O), before reaching the melting point of oxygen-poor (as-received) Zircaloy at about [1760°C (3200°F)[268]].[269]

The PHEBUS B9 test is an example of an experiment that did *not* have a rapid fuel-cladding temperature escalation that commenced at relatively low fuel-cladding temperatures, because *it had a low initial heatup rate*. In PHEBUS B9, conducted in December 1986, the initial fuel-cladding temperature heatup rate was 0.2°C/sec (0.36°F/sec); the test bundle heated up to 1547°C (2816°F) at a very slow rate, without a rapid fuel-cladding temperature escalation. At 1547°C (2816°F) a fuel-cladding temperature escalation of 5°C/sec commenced.[270]

The CORA-2 test is an example of an experiment that *had* a rapid fuel-cladding temperature escalation that commenced at relatively low fuel-cladding temperatures, because *it did not have a low initial heatup rate*. CORA-2 had an initial fuel-cladding temperature heatup rate of approximately 1.0°C/sec (1.8°F/sec). In CORA-2, a PWR-type test conducted with 25 fuel rods (16 heated and 9 unheated rods), an "uncontrolled temperature escalation started at about [1100°C (2012°F)]."[271] And the LOFT LP-FP-2 experiment is another example of an experiment that *had* a rapid fuel-cladding temperature escalation that commenced at relatively low fuel-cladding temperatures, because *it did not have a low initial heatup rate*. LOFT LP-FP-2, heated with "actual fission-product decay heating of the core,"[272] had an initial fuel-cladding temperature heatup rate of approximately 1.0°C/sec (1.8°F/sec).[273] In LOFT LP-FP-2, "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between

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[268] Zirconium melts at temperatures above 1816°C (3300°F). See NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, (ADAMS Accession No: ML011800519), p. 3-1.
[269] T.J. Haste *et al*., "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, 1996, p. 27.
[270] C. Gonnier *et al*., "PHEBUS Severe Fuel Damage Program Main Experimental Results and Instrumentation Behavior," Proceedings of the Seminar of the Phebus-FP (Fission Product) Project, Chateau Cadarache, St. Paul-Lez-Durance, France, June 5-7, 1991, p. 113.
[271] T.J. Haste *et al*., "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, pp. 15, 16.
[272] S. R. Kinnersly, *et al*., "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3.23.
[273] T. J. Haste *et al*., "Degraded Core Quench: A Status Report," August 1996, p. 13.

Zircaloy and water occurred at …1400 K [1127°C (2060°F)] on a guide tube." Hence, an analysis of LOFT LP-FP-2 "concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400 K (2060°F)."[274]

II.F.2.a. The PHEBUS B9R Test had a Low Initial Heatup Rate and a Rapid Fuel-Cladding Temperature Escalation at Relatively Low Temperatures

It needs to be clarified that even if there were a low initial heatup rate of the fuel cladding, it is still possible for a rapid fuel-cladding temperature escalation to commence at relatively low fuel-cladding temperatures. The PHEBUS B9R-2 test is an example of an experiment that *had* an unexpected rapid fuel-cladding temperature escalation that commenced at relatively low fuel-cladding temperatures, even though *it had a low initial heatup rate*.

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[275] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[276]

Discussing PHEBUS B9R-2, the 1996 European Commission report states:

> The B9R-2 test (second part of B9R) illustrates the oxidation in different cladding conditions representative of a pre-oxidized and fractured state. This state results from a first oxidation phase (first part name B9R-1, of the B9R test) terminated by a rapid cooling-down phase. During B9R-2, an unexpected strong escalation of the oxidation of the remaining Zr occurred when the bundle flow injection was switched from helium to steam while the maximum clad temperature was equal to 1300 K [1027°C

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274 J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, (ADAMS Accession No: ML062840091), pp. 30, 33.

275 G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 311.

276 T.J. Haste *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, p. 33.

> (1880°F)]. *The current oxidation model was not able to predict the strong heat-up rate observed* even taking into account the measured large clad deformation and the double-sided oxidation (final state of the cladding from macro-photographs).
>
> … *No mechanistic model is currently available to account for enhanced oxidation of pre-oxidized and cracked cladding*[277] [emphasis added].

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to SFP fires. The results of PHEBUS B9R-2 indicate that it is unpredictable as to whether or not rapid fuel-cladding temperature escalations would commence *in steam*, in a SFP accident, at relatively low fuel-cladding temperatures.

Spent fuel rods would also be "pre-oxidized": when high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 µm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 µm thick. And medium to high burnup fuel cladding typically has a "hydrogen concentration in the range of 100-1000 wppm [weight parts per million];" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[278]

According to an October 2000 OECD Nuclear Energy Agency report, the initial heatup rate in PHEBUS B9R-2 was less than 0.1°C/sec up to 727°C (1340°F) (during the pure helium phase of the experiment).[279] However, according to a graph with a plot of fuel-cladding temperature values at the 0.6 meter "hot level" of the PHEBUS B9R-2 test bundle, the initial heatup rate in PHEBUS B9R-2 was approximately 1.0°C/sec up to 727°C (1340°F); however, the heatup rate decreases to lower than 0.2°C/sec between

[277] *Id*., p. 126.
[278] K. Natesan, W.K. Soppet, Argonne National Laboratory, "Hydrogen Effects on Air Oxidation of Zirlo Alloy," NUREG/CR–6851, October 2004, (ADAMS Accession No: ML042870061), p. iii, 3.
[279] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix Update 1996-1999," NEA/CSNI/R(2000)21, October 2000, p. 97.

approximately 877°C (1610°F) and 1002°C (1835°F).[280] (See Figure 5.) As stated, the cladding-temperature escalation commenced at approximately 1027°C (1880°F).

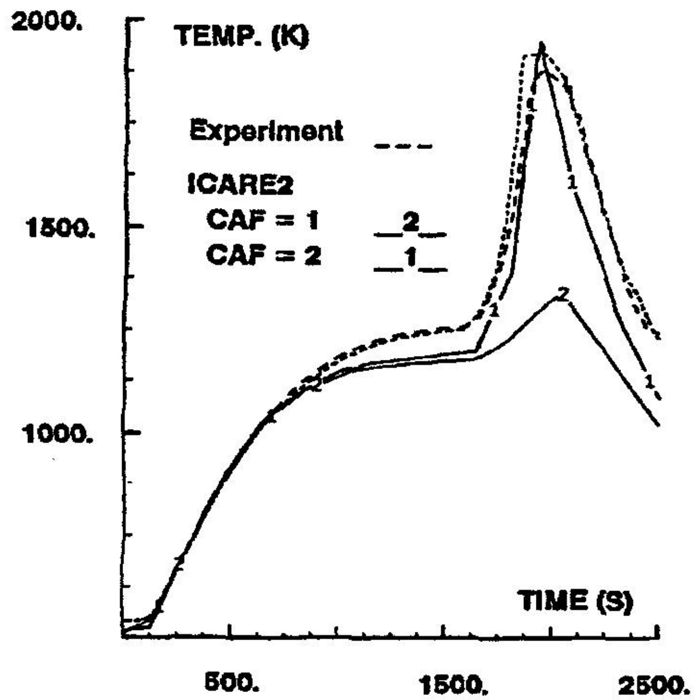

Figure 5. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[281]

(It is noteworthy that a September 2013 NRC document, NUREG-2157, states that if local fuel-cladding temperatures were to increase to approximately 1000°C (1832°F) in a SFP accident, a runaway zirconium oxidation reaction—*a SFP zirconium fire*—would commence *in steam*.[282] However, regarding zirconium alloy fuel-cladding behavior *in steam*, in a reactor LOCA, in October 2012, the NRC stated that "autocatalytic [zirconium oxidation] reactions have not occurred at temperatures less than

280 G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 312.

281 *Id.*

282 NRC, "Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment," NUREG-2157, Appendix F, p. F-2.

2200 degrees F;"[283] that is, runaway zirconium oxidation reactions have not commenced in experiments when fuel-cladding temperatures were lower than 1204.4°C (2200°F).

Hence, the NRC claims that runaway zirconium oxidation would commence at 1000°C (1832°F) *in steam*, in SFP accidents, which would have low initial heatup rates (except in certain criticality accident scenarios). Nonetheless, the NRC also claims that runaway zirconium oxidation would *not* commence below 1204.4°C (2200°F) *in steam*, in reactor LOCAs, which could have *high* initial heatup rates, exceeding 5.6°C/sec (10.0°F/sec).

Perhaps the NRC's statement regarding runaway zirconium oxidation *in steam*, in reactor LOCAs, is influenced by the fact that the NRC requires the maximum fuel-cladding temperature in a postulated reactor LOCA to not exceed 2200°F—10 C.F.R. § 50.46(b)(1) peak fuel-cladding temperature limit. If the NRC acknowledged that runaway zirconium oxidation *in steam* could commence in reactor LOCAs at fuel cladding temperatures below 2200°F, the NRC might realize that it needed to lower its Section 50.46 peak fuel-cladding temperature limit.[284])

II.F.3. In a Spent Fuel Pool Boil-Off Accident, a Zirconium Fire Would Most Likely Ignite in Air if Fuel-Cladding Temperatures Reached 900°C (1652°F) or Lower

In a SFP boil-off accident, after the fuel assemblies were uncovered, the fuel cladding's zirconium content would initially chemically react with the steam produced by the boiling water in the SFP. At some point, as more water boiled off and the water level decreased further (below the elevation at 66 percent of the height of the fuel assemblies), the fuel cladding would be exposed to local mixtures of steam and air. When zirconium is exposed to local mixtures of steam and air, the zirconium-oxygen reaction will

[283] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, (ADAMS Accession No: ML12265A277), p. 2.

[284] Full disclosure: in November 2009, the author of this report submitted a rulemaking petition (PRM-50-93) to the NRC, requesting that the NRC revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature, in a reactor LOCA, not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments. The author argued that data from multi-rod (assembly) severe fuel damage experiments (for example, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

dominate.[285] Then, as the water level dropped down even closer to the baseplates, the upper regions of the fuel assemblies would predominately be exposed to air. (After the fuel assemblies were uncovered there would be various local conditions; for example, there could be local steam starvation and local oxygen starvation.)

If there had been initial heatup rates that were very low (that is, if there had not been any criticality accidents that caused faster initial heatup rates) and a zirconium fire had not commenced *in steam*, a zirconium fire would most likely commence *in air*, provided water covered the baseplates at the lower end of the fuel assemblies. (If "water [is] above the base plate of the racks…the water at the bottom of the pool acts as a "plug," which prevents cooling of the assemblies by natural air circulation."[286])

II.F.4. Exothermic Reactions in Air: Zirconium Oxidation and Zirconium Nitriding

Runaway zirconium oxidation commences *in air* at lower local fuel-cladding temperatures—827°C (1520°F)[287] or 900°C (1652°F)[288]—than it does in steam; and the zirconium-oxygen reaction in air produces approximately twice as much energy (per kg of Zr reacted) as the zirconium-steam reaction. The zirconium-oxygen reaction in air produces zirconium dioxide and energy; the equation for the reaction is written as $Zr + O_2 \rightarrow ZrO_2$ + energy. The energy (heat) generated by the reaction is approximately 12.0 megajoules per kg of Zr reacted.[289]

And the zirconium-nitrogen reaction produces approximately 30 percent of the quantity of energy (per kg of Zr reacted) produced by the zirconium-oxygen reaction in air. The zirconium-nitrogen reaction produces zirconium nitride and energy; the equation

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285 C. Bals *et al.*, "Modelling of Accelerated Cladding Degradation in Air for Severe Accident Codes," The 3rd European Review Meeting on Severe Accident Research (ERMSAR-2008), Bulgaria, September 23-25, 2008, pp. 4, 5.

286 Randall Gauntt *et al.*, Sandia National Laboratories "Fukushima Daiichi Accident Study: Status as of April 2012," SAND2012-6173, August 2012, p. 183.

287 Zachary I. Franiewski *et al.*, Pennsylvania State University, "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," pp. iv, 2, 3, 8, 13.

288 Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

289 National Research Council, Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage, "Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report," 2005, p. 38.

for the reaction is written as $Zr + 1/2N_2 \rightarrow ZrN$ + energy. The energy (heat) generated by the reaction is approximately 3.76 megajoules per kg of Zr reacted.[290]

In *April 2000*, the ACRS told the NRC Staff that "nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample"[291] [emphasis added]. (A July 1987 NRC document, NUREG/CR-4982, states that the reaction of zirconium and nitrogen releases approximately 82,000 calories per mole of zirconium reacted.[292])

An August 2012 SNL report, "Fukushima Daiichi Accident Study" states that "[i]f *inadequate* cooling is provided, then the cladding will heat up and will rapidly oxidize (*i.e*., burn) and to a lesser extent, nitride (*i.e*., combine with nitrogen if no oxygen or steam are available). *Since the oxidation and nitride processes are exothermic*, the fuel rods could heat to melting conditions and structurally degrade"[293] [emphasis added].

II.F.5. Nitrogen Accelerates the Oxidation and Degradation of Zirconium Fuel-Cladding in Air

The nitrogen gas (in air) affects the oxidation of zirconium in air.[294] The presence of nitrogen accelerates the oxidation (burning) and degradation of zirconium fuel-cladding

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[290] V. L. Sailor *et al*., Brookhaven National Laboratory, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," NUREG/CR-4982, July 1987, p. 109.
[291] Dana A. Powers, Chairman of ACRS, Letter to Richard A. Meserve, Chairman of NRC, Regarding ACRS Recommendations for Improvements to the NRC Staff's "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," April 13, 2000, (ADAMS Accession No. ML003704532), pp. 3-4.
[292] V. L. Sailor *et al*., "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," NUREG/CR-4982, p. 109.
[293] Randall Gauntt *et al*., Sandia National Laboratories "Fukushima Daiichi Accident Study: Status as of April 2012," SAND2012-6173, August 2012, p. 183.
[294] K. C. Wagner, R. O. Gauntt, Sandia National Laboratories, Analysis and Modeling Division, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," SAND1A Letter Report, Revision 2, November 2006, (ADAMS Accession No. ML120970086), p. 12; and L. Fernandez-Moguel, J. Birchley, European MELCOR User's Group, "PSI air oxidation model in MELCOR: Part 2: Analysis of experiments and model assessment," Stockholm, May 2013, which states: "Neither MELCOR nor SCDAP [a severe accident computer safety model] are able to predict a nitride reaction."

in air,[295] which would affect the progression and severity of a SFP accident, including radioactive releases, "most notabl[y] ruthenium."[296] ("Ruthenium has a biological effectiveness equivalent to that of Iodine-131;"[297] Ruthenium-106 has half-life of 373.6 days.)

A 2010 Journal of Nuclear Materials paper observes that "[t]he complexity of air oxidation of Zircaloy arises out of the simultaneous oxidation and nitriding processes."[298] And a May 2013 report, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," discusses experimental data demonstrating that porous nitrides form inside oxide layers *under local or full oxygen-starvation conditions*.[299] (When zirconium reacts in air it is possible for the reaction to become oxygen-starved; however, if zirconium is locally oxygen-starved in air, nitrogen will react with it.) The porous, degraded condition of an oxide layer facilitates accelerated oxidation rates if additional oxygen becomes *locally* available; and any additional oxygen will react with the zirconium nitride (ZrN) within an existing oxide layer and form zirconium dioxide (ZrO_2) in a fast exothermic reaction.[300]

A 2008 Journal of Nuclear Materials paper, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," explains that "once initiated, the nitride-assisted degradation will be a self-sustaining process, because ZrN conversion into oxide leaves nitrogen trapped in the oxide scale and available for further nitriding, and because the oxide formed is undoubtedly non-protective. Where nitriding has initiated, the bright α-Zr(O) layer is thin, confirming the faster progression of the oxidation front there. The

[295] J. Stuckert, M. Große, Z. Hózer, M. Steinbrück, Karlsruhe Institute of Technology, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," KIT-SR 7634, May 2013, p. 1; and O. Coindreau, C. Duriez, S. Ederli, "Air Oxidation of Zircaloy-4 in the 600-1000°C Temperature Range: Modeling for ASTEC Code Application," Journal of Nuclear Materials 405, 2010, p. 208.
[296] J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 1.
[297] Dana A. Powers, Chairman of ACRS, Letter to Richard A. Meserve, Chairman of NRC, Regarding ACRS Recommendations for Improvements to the NRC Staff's "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," April 13, 2000, (ADAMS Accession No. ML003704532), p. 2.
[298] O. Coindreau, C. Duriez, S. Ederli, "Air Oxidation of Zircaloy-4 in the 600-1000°C Temperature Range: Modeling for ASTEC Code Application," p. 207.
[299] J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 10.
[300] Emilie Beuzet *et al*., "Modelling of Zry-4 Cladding Oxidation by Air Under Severe Accident Conditions using MAAP4 Code," International Conference Nuclear Energy for New Europe 2009, Slovenia, September 2009, p. 3.

self-sustainability of the nitriding-reoxidation sequence may also favor the lateral progressive propagation of the breakaway."[301]

Regarding nitrogen-induced breakaway oxidation, the 2008 Journal of Nuclear Materials paper explains that "[b]reakdown and loss of the dense scale protective effect occur and result in an accelerated degradation;" furthermore, the transition to nitrogen-induced breakaway oxidation occurs *earlier with pre-oxidized fuel cladding* than with fresh *non-oxidized* fuel cladding—"nitriding is favored by the 'corrosion' scale."[302]

It is clear that *in air*, in a SFP accident, a significant degree of zirconium oxidation would occur, because spent fuel rods would be "pre-oxidized." When high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 μm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick. And medium to high burnup fuel cladding typically has a "hydrogen concentration in the range of 100-1000 wppm [weight parts per million];" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[303]

Regarding the fact that air oxidation causes a fast progression of the oxidation front, the 2008 Journal of Nuclear Materials paper states:

> At 800°C and above, continuous acceleration is observed, as the consequence of a complex process involving nitride formation and re-oxidation, as well as dissolution of nitrogen in the zirconia anion sub-lattice. Important volume mismatches of the ZrO_2 and ZrN compounds, together with zirconia phase transformations lead to *growth of a highly cracked, porous, non-protective oxide. It results in fast progression of the oxidation front, as well as strong deformation of the cladding.* The barrier against fission product release provided by the fuel cladding is lost much earlier than during accident under steam atmosphere[304] [emphasis added].

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301 C. Duriez, T. Dupont, B. Schmet, F. Enoch, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," Journal of Nuclear Materials 380 (2008), p. 43.
302 *Id.*, p. 44.
303 K. Natesan, W.K. Soppet, Argonne National Laboratory, "Hydrogen Effects on Air Oxidation of Zirlo Alloy," NUREG/CR–6851, October 2004, (ADAMS Accession No: ML042870061), p. iii, 3.
304 C. Duriez, T. Dupont, B. Schmet, F. Enoch, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," Journal of Nuclear Materials 380 (2008), p. 44.

And regarding the fact that cladding degradation can be even much faster in oxygen starved situations (in air), the 2008 Journal of Nuclear Materials paper states:

> Kinetic data of this study have been obtained mainly in high air flow conditions. *In real accidental situations, where oxygen starved situations are likely to occur, cladding degradation can be even much faster than predictable from these high air flow data*, because of early initiation of the nitriding process, as shown by the few tests performed at the highest temperatures with insufficient air flow rate. All in all, more experimental investigations are required to address the various conditions that can be encountered in accidental situation.[305]

II.F.6. The Axial and Radial Propagation of a Spent Fuel Pool Fire

Regarding the axial propagation of the zirconium-*steam* reaction from its point of initiation, a 1990 Karlsruhe report, KfK 4378, states:

> [T]he temperature escalation starts at the hottest position in the bundle [of fuel rod simulators], at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward.[306]

And regarding axial and radial propagation of the zirconium-oxygen reaction (in steam and/or air), a September 2013 NRC document, NUREG-2157, states:

> Under certain conditions, the high temperature runaway zirconium oxidation reaction occurring in one part of the pool could also spread to other spent fuel in the pool. The proximity of fuel assemblies to one another, combined with the effects of [radiative] heat transfer when these assemblies are at very high temperatures, could allow the runaway oxidation reaction to spread from spent fuel with high decay heat to spent fuel with lower decay heat that would otherwise not have begun burning.[307]

As fuel rods heated up to melting temperatures, “the steel racks supporting the fuel assemblies will also heat due to convection and radiation from the fuel

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[305] *Id*., p. 44.

[306] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, “Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3),” Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[307] NRC, “Waste Confidence Generic Environmental Impact Statement: Draft Report for Comment,” NUREG-2157, p. F-2.

assemblies."[308] In the worst-case scenario, a SFP fire would propagate "throughout the entire spent fuel inventory in the pool"[309]

The zirconium-air reaction would propagate away from its point of initiation more rapidly than the propagation of the zirconium-steam reaction, because: 1) the heat produced by zirconium oxidation in air is greater than that in steam; 2) the nitrogen content in air would accelerate zirconium oxidation in air; and 3) heat would also be contributed by the exothermic zirconium-nitrogen reaction.

II.G. Other Chemical Reactions that Could Occur in a Spent Fuel Pool Fire

II.G.1. Zirconium Hydriding

It is widely known that hydrogen can detonate in air; hydrogen can also chemically react with zirconium. The reaction between hydrogen and zirconium *is exothermic*. Zirconium hydriding "can occur with [a] hydrogen-rich atmosphere and at [a] moderate temperature… The exothermic reaction is able to lead to *severe temperature escalations* in the temperature range of [627°C (1160°F) to 1127°C (2060°F)]"[310] [emphasis added]. (This information is based on data from an experiment—PHEBUS SFD C3—that was conducted under conditions very different than those that would occur in a SFP accident. PHEBUS SFD C3 "was performed with…high pressure (3.5 MPa [508 psia]), pure steam-starved conditions (pure hydrogen coolant) and very low cladding oxidation."[311])

Hydriding can occur when there are steam-starved conditions; however, there can also be simultaneous oxidation and hydriding.[312] In a SFP accident, hydriding would primarily occur at locations of the spent fuel rods that had freshly exposed zirconium as a result of fuel rod ballooning and rupturing; oxide layers inhibit hydrogen uptake.

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[308] K. C. Wagner, R. O. Gauntt, Sandia National Laboratories, Analysis and Modeling Division, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," SAND1A Letter Report, Revision 2, p. 12.

[309] J.H. Jo, P.F. Rose, S.D. Unwin, V.L. Sailor, Brookhaven National Laboratory, "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," NUREG/CR-5281, March 1989, (ADAMS Accession No. ML071690022), p. 8.

[310] T.J. Haste *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, 1996, p. 36.

[311] *Id.*, p. 33.

[312] *Id.*, pp. 70-71.

II.G.2. The Boron Carbide Contained in the Spent Fuel Racks

The boron carbide (“B_4C”) contained in the Boral and Boraflex neutron-absorber materials that are placed in spent fuel racks would melt before it could oxidize in the racks. EPRI SFP accident guidance states that it should be assumed that Boral and Boraflex materials would melt and relocate downward to the bottom of the SFP.[313] However, if some of the boron carbide were to oxidize (in dry air) later in the accident, the heat released—50,155 kJ/kg per kg of B_4C reacted—would be approximately 7.7 times greater (per gram) than the heat released by the oxidation of zirconium in steam—approximately 6500 kJ per kg of Zr reacted.[314]

II.G.3. Chemical Interactions between Zirconium and Inconel at “Low Temperatures”

Data from experiments studying severe reactor accidents can pertain to SFP accidents—as the NRC report NUREG-1738 concludes[315]—including information about the eutectic chemical interactions between materials that would occur in both types of accidents. Such experiments have demonstrated that eutectic chemical interactions between Inconel and Zircaloy occur at temperatures as “low” as 1832°F; hence, analysts have concluded that “[g]rid spacers can have a significant impact on the progression of damage in a reactor core during a severe accident. …in a reactor core with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers.”[316] It is pertinent that in the CORA severe reactor accident experiments, simulating meltdowns, “[i]n all cases, the damage of the bundle was initiated due to Zircaloy/stainless steel *and Zircaloy/Inconel interactions*. Localized liquefaction of these components started around

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313 EPRI, “Severe Accident Management Guidance Technical Basis Report,” Volume 2: “The Physics of Accident Progression,” 1025295, Appendix EE, p. EE-9.

314 L. Belovsky, “Heat Release from B_4C Oxidation in Steam and Air,” paper from “Behaviour of LWR core materials under accident conditions,” IAEA-TECDOC-921, Proceedings of a Technical Committee meeting held in Dimitrovgrad, Russian Federation, 9-13 October 1995, p. 57.

315 NRC, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” NUREG-1738, February 2001 (ADAMS Accession No. ML010430066), Appendix 1 B, p. Al B-2.

316 L.J. Siefken, M.V. Olsen, “A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core,” Nuclear Engineering and Design 146, 1994, p. 427.

1200°C."[317] In the CORA-2 and -3 experiments, "the meltdown of the Inconel spacer [took] less than one minute…[and] an enhanced melting in the midsection…shift[ed] the axial hot spot to the bottom of the bundle."[318]

(Inconel is an alloy that has a higher percentage of nickel (Ni) than stainless steel; for example, Inconel 600 is composed of 76.0 percent nickel by weight, 15.0 percent chromium by weight, 8.0 percent iron (Fe) by weight, and small percentages of other elements. Stainless steel is a metal alloy with contents of chromium and iron greater than 11.5 percent by weight and 50 percent by weight, respectively; stainless steel also contains nickel, manganese, and small percentages of other elements.)

The ballooning of zirconium fuel cladding would augment its contact with the Inconel grid spacers. If local temperatures were to increase to approximately 1200°C, the cladding-to-grid contact would initiate the eutectic chemical reaction between zirconium and Inconel. Hence, one could reasonably speculate that in a SFP boil-off accident, if a fuel assembly had Inconel grid spacers, the first location it liquefied would be in the vicinity of an upper Inconel grid spacer.

II.G.4. Chemical Interactions Between Zircaloy and Stainless Steel at "Low Temperatures"

Discussing chemical interactions between Zircaloy and stainless steel (and comparing them to those between Zircaloy and Inconel), "Current Knowledge on Core Degradation Phenomena, a Review" states:

> In a first approach, the reaction behavior of Zircaloy with Inconel 718 is comparable to that with Type 316 stainless steel.[319] At temperatures <1100°C, Inconel attacks the Zircaloy faster than stainless steel; above 1100°C, the situation is the reverse. In both cases, the melting of a relatively large quantity of Zircaloy with limited melting of the adjacent stainless steel or Inconel takes place. During heat-up of the stainless

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[317] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 202.

[318] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 6.

[319] P. Hofmann, M. Markiewicz, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Inconel 718 at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 4729, 1994.

> steel/Zircaloy…reaction [system], a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C. This may be the reason [one of the locations] that melt progression in a fuel rod bundle initiates [is] at absorber rod cladding (stainless steel)/Zircaloy guide tube contact areas.[320]

And discussing the affects of zirconium oxide (ZrO_2) layers on the chemical interaction between Zircaloy and stainless steel, "Current Knowledge on Core Degradation Phenomena, a Review" states:

> Oxide layers on the Zircaloy cladding outside diameter delay the chemical interactions between Zircaloy and steel, but they cannot prevent them. The influence of oxide layers becomes less important at temperatures >1100°C, since the dissolution of the protecting ZrO_2 layers occurs rather fast and the stainless steel is then in contact with metallic Zircaloy or oxygen-stabilized α-Zr(O).[321]

II.G.5. Molten Core Concrete Interaction in Spent Fuel Pool Accidents

For the SFP, the term "molten-core-concrete interaction" ("MCCI") is a misnomer; MCCI refers to molten fuel assemblies chemically interacting with the SFP's concrete content, after being relocated to the bottom of the SFP. MCCI would commence after the SFP's stainless steel liner melted; then the molten fuel assemblies would chemically interact with concrete, generating hydrogen and carbon monoxide gases.[322]

Regarding the MCCI that could occur in a SFP accident, a June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs states:

> MCCI may occur in selected scenarios in which the fuel relocated to the bottom of the pool following the failure of the rack baseplate and its temperature exceeded the concrete ablation temperature [approximately 1227°C (2240°F)]. These cases involve large-scale debris relocation and large releases of volatile fission products. Even without MCCI, the fuel in debris form continues to release fission products resulting in very large releases of volatiles.[323]

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[320] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[321] *Id*.
[322] M. Kowalik *et al*., "Severe Accident Analyses for Shutdown Modes and Spent Fuel Pools to Support PSA level 2 Activities," Eurosafe 2013 Seminar 1, p. 5.
[323] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," June 2013, (ADAMS Accession No. ML13133A132), p 26.

II.H. In a SBO Boil-Off Accident or Partial Spent Fuel Pool LOCA, Explosive Hydrogen Gas Would Be Produced by Zirconium and Other Materials Chemically Reacting with Steam

As the NRC observes, large-scale, long-term power outages, which lasted months or longer, could initiate "a series of events potentially leading to core damage at multiple nuclear sites."[324] Radiological releases resulting from core damage would contaminate the NPP site and impede efforts to mitigate the accident, especially if radioactive debris were propelled throughout the site by hydrogen explosions, as occurred in the Fukushima Dai-ichi accident.[325] After the Fukushima Dai-ichi site was contaminated, workers had to wear additional protective clothing and limit the time they spent, working to mitigate the accident.[326] Efforts to mitigate a SFP accident would also be impeded (or possibly entirely prevented for significant time periods) by the radiologically-contaminated environment.

II.H.1. How Hydrogen Explosions Could Affect BWR Mark I and Mark II Spent Fuel Pool Accidents

In BWR Mark I and Mark II designs, SFPs are typically located at the level of the operating floor, approximately 100 to 150 feet above ground level,[327] in the reactor building (secondary containment). If either a BWR Mark I or Mark II reactor core melted down and the total amount of the zirconium in the core—approximately 76,000 kg—were to chemically react with steam, approximately 3360 kg of hydrogen would be generated.[328] In the event of a severe accident at either a BWR Mark I or BWR Mark II, the Fukushima Dai-ichi accident scenario of hydrogen leaking from over-pressurized primary containments and/or hardened vent systems should be considered as

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[324] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74790.
[325] INPO, "Special Report on the Nuclear Accident at the Fukushima Dai-ichi Nuclear Power Station," INPO 11-005, November 2011, pp. 9, 12, 21, 24, 25, 32, 37, 79, 85, 86, 96.
[326] *Id.*, p. 9.
[327] NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4.6.
[328] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011," p. 10.

likely to occur again. In the Fukushima Dai-ichi accident, BWR Mark I reactor buildings—essentially industrial buildings with design pressures of approximately 3.0 psig[329]—were compromised by hydrogen explosions. BWR Mark II reactor buildings also have low design pressures.

Hence, BWR Mark I and Mark II SFPs are vulnerable to the hydrogen explosions that can occur in reactor buildings. A June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs states that "[t]he occurrence of a hydrogen combustion event from a concurrent reactor accident has the potential to generate debris which could impair SFP natural circulation air or steam cooling (should the fuel in the SFP become uncovered) for conditions in which the fuel might otherwise be cooled by means of these passive cooling modes."[330] Furthermore, if either a BWR Mark I or Mark II SFP were compromised by a hydrogen explosion, it could cause large radiological releases.

If a BWR Mark I or Mark II reactor building were breached by a hydrogen explosion there would be more available oxygen to facilitate oxidation of the zirconium cladding of the fuel assemblies. A June 2013 NRC report on how earthquakes could affect BWR Mark I SFPs states that if there were a hydrogen explosion in the reactor building, "damage could breach structures that would retain radioactive material, along with allowing more oxygen into the building, potentially increasing the severity of the spent fuel fire."[331] The accelerated zirconium oxidation would contribute additional heat, causing a quicker fuel-cladding temperature escalation, releasing yet more heat, causing a more rapid axial and radial propagation of the SFP fire. This would cause increased radiological releases from the SFP.

If the fuel assemblies were uncovered in either a SBO boil-off accident or a partial SFP LOCA, explosive hydrogen gas would be generated by the reaction of steam with the zirconium cladding of fuel rods. If enough hydrogen were generated, it could

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[329] Sherrell R. Greene, Oak Ridge National Laboratory, "The Role of BWR Mark I Secondary Containments in Severe Accident Mitigation," Proceedings of the 14th Water Reactor Safety Information Meeting at the National Bureau of Standards, Gaithersburg, Maryland, October 27-31, 1986, Exhibit 6.

[330] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 25.

[331] *Id.*, p. ix.

detonate.[332] Computer analyses conducted at ORNL with the MELCOR computer safety model found that in a *hypothetical* scenario if Fukushima Dai-ichi Unit 4's SFP had boiled dry, a total of 1800 to 2050 kilograms ("kg") of hydrogen could have been generated. A 2012 ORNL paper states that "[i]n theory, it [would be] possible to generate up to 3.4 kg of hydrogen per assembly (from oxidation of [zirconium] in the fuel cladding and box), or *a total of 4,525 kg* from the hot 1331 assemblies stored in [Unit 4's SFP]. The hydrogen generated from oxidation of steel and B_4C [boron carbide] in the racks [would] be additional"[333] [emphasis added].

(It is noteworthy that in MELCOR BWR Mark I "SFP calculation[s], [hydrogen] ignition is assumed to occur in the reactor building when the hydrogen concentration exceeds 10 percent by volume. In addition, MELCOR checks to determine whether there is sufficient oxygen. The minimum oxygen mole fraction for ignition is 5 percent."[334]

MELCOR SFP calculations of hydrogen combustion do not consider that significant deflagrations[335] of hydrogen can occur when local hydrogen concentrations are lower than 10 percent by volume. For example, in the Three Mile Island Unit 2 ("TMI-2") accident, a hydrogen deflagration occurred when the hydrogen concentration was 8.1 volume percent;[336] the deflagration caused a rapid pressure increase of approximately 28 pounds per square inch ("psi") in the containment.[337] Of course, the volume of a PWR large day containment, such as TMI-2 had, is different than that of a BWR Mark I reactor building; however, it is clear that a significant hydrogen deflagration would compromise a BWR Mark I reactor building, which has a relatively low design pressure.)

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[332] Juan J. Carbajo, Oak Ridge National Laboratory, "MELCOR Model of the Spent Fuel Pool of Fukushima Dai-ichi Unit 4," 2012, p. 1.
[333] *Id*., pp. 1-2.
[334] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 103.
[335] A deflagration is a combustion wave traveling at a subsonic speed (less than the speed of sound) relative to the unburned gas.
[336] Kahtan N. Jabbour, NRC, letter regarding Turkey Point Units 3 and 4, Exemption from Hydrogen Control Requirements, December 12, 2001, Attachment 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Turkey Point Units 3 and 4," (ADAMS Accession No. ML013390647), p. 4.
[337] W. E. Lowry *et al*., Lawrence Livermore National Laboratory, "Final Results of the Hydrogen Igniter Experimental Program," NUREG/CR-2486, February 1982, p. 4.

II.H.2. How Hydrogen Explosions Could Affect PWR and BWR Mark III Spent Fuel Pool Accidents

PWR and BWR Mark III SFPs are typically located at ground level.[338] In the event of a severe *reactor* accident, PWR and BWR Mark III SFPs would not be as vulnerable to the potential consequences of explosive hydrogen gas—generated from oxidized zirconium and other core materials—as BWR Mark I and Mark II SFPs. However, if the fuel assemblies were uncovered in either a SBO boil-off accident or a partial SFP LOCA, PWR and BWR Mark III SFPs, would be vulnerable to the explosive hydrogen gas that would be generated by the reaction of steam with zirconium and other materials in the SFP.

II.H.3. Indian Point Energy Center's Spent Fuel Pools are Located Underground; However, They Are Vulnerable to Hydrogen Explosions

Indian Point Energy Center is located less than 25 miles north of New York City; more than 17 million people live within a 50-mile radius of Indian Point.[339] On August 26, 2013, Indian Point Unit 2's SFP, which has a storage capacity of 1374 fuel assemblies, contained 1104 fuel assemblies (80 percent of capacity); and Indian Point Unit 3's SFP, which has a storage capacity of 1345 fuel assemblies, contained 1199 fuel assemblies (89 percent of capacity).[340] (The fuel assemblies in a typical PWR core have approximately 26,000 kg of zirconium that, if completely oxidized, would generate a total of approximately 1150 kg of hydrogen.[341] The cores of pressurized-water reactors, like Indian Point's, typically contain between 150 and 200 fuel assemblies.[342])

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338 NRC, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'," NUREG-1353, April 1989, (ADAMS Accession No. ML082330232), p. 4.6.

339 Edwin S. Lyman, Union of Concerned Scientists, "Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack at the Indian Point Nuclear Plant," September 2004, p. 23.

340 NRC, "Summary of August 26, 2013, Meeting with Entergy Nuclear Operations, Inc. and Netco on Indian Point Unit 2 Spent Fuel Pool Management," September 24, 2013, (ADAMS Accession No. ML13256A086), p. 1.

341 IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011," p. 10.

342 NRC, "Pressurized Water Reactors," (available at: http://www.nrc.gov/reactors/pwrs.html : last visited on 10/15/13).

Indian Point's owner, Entergy, touts the safety of Indian Point Unit 2 and 3's SFPs, explaining that "[t]hey are constructed with concrete walls 4 to 6 feet wide and with a half-inch stainless steel inner liner" and that "the fuel pool for Indian Point 2 is completely underground and Indian Point 3['s] is nearly 100% underground, so they are protected on all sides by rock and gravel."[343] However, if there were a SFP fire at either unit (or at both), *thousands of kilograms of explosive hydrogen gas* could be generated by the oxidation (burning) of the tens of thousands kilograms of zirconium—the cladding material of the fuel rods—in storage. It is almost inevitable that hydrogen gas would detonate, breaching the barriers that are supposed to protect the public; releases of radiation could far exceed the quantity released by the Chernobyl Unit 4 accident. More land could be contaminated than the area encompassing the Chernobyl Exclusion Zone, with higher concentrations of radioactive cesium-137. The number of premature deaths from cancer and economic damages would perhaps be incalculable.

An October 2011 Natural Resources Defense Council ("NRDC") report, "Nuclear Accident at Indian Point: Consequences and Costs," with analyses of the potential radiological consequences of *one full reactor core melt* at Indian Point, would perhaps help provide insight regarding the magnitude of the damages and suffering that would ensue from a SFP fire at Indian Point.

The NRDC report states:

> An accident at Indian Point Unit 3 involving a full reactor core melt approaching the scale of Chernobyl could put people in New York City at risk for receiving a whole-body radiation dose greater than 25 rem, resulting in a 7 percent increase in risk of premature death from cancer for an average individual. An accident of this scale would require the administration of stable iodine throughout the New York City metropolitan area, and put thousands at risk for radiation sickness in and near the Hudson Valley. …
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> A release of radiation on the scale of Chernobyl's would make Manhattan too radioactively contaminated to live in if the city fell within the plume.[344]

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[343] Entergy, "Safe, Secure, Vital: Indian Point Energy Center," website, "Spent Fuel," (located at http://www.safesecurevital.com/safe-secure-vital/spent-fuel.html: last visited on October 12, 2013).

[344] Matthew McKinzie, NRDC, "Nuclear Accident at Indian Point: Consequences and Costs," October 17, 2011, Cover Sheet, p. 1.

The prospect of a SFP fire at Indian Point is especially worrisome, given that an event that could lead to such a disaster—large-scale, long-term power outages, which could plausibly last months or longer (caused by an extreme solar storm)—"is plausible with a frequency in the range of once in 153 to once in 500 years"[345] or 2.0×10^{-3} and 6.5×10^{-3} per year.

II.I. Deficiencies of the NRC MELCOR Computer Safety Model, Regarding the Zirconium-Oxygen and Zirconium-Nitrogen Reactions in Air

A number of the limitations of the NRC MELCOR computer safety model have already been discussed in this report; it is clear that MELCOR under-predicts the severity of spent fuel pool accidents. Furthermore, recent NRC post-Fukushima MELCOR simulations of BWR Mark I SFP accidents have not considered realistic potential SFP accident scenarios; for example, criticality accidents were not modeled.[346] In this section MELCOR deficiencies, regarding modeling the zirconium-oxygen and zirconium-nitrogen reactions in air, are discussed.

II.I.1. MELCOR Does Not Model the Exothermic Zirconium-Nitrogen Reaction

The NRC has recently performed a number of post-Fukushima computer simulations of SFP accidents with the Sandia National Laboratories ("SNL") MELCOR computer safety model. However, MELCOR *does not simulate* the generation of heat from the chemical reaction of zirconium and nitrogen; neglecting to model a heat source that would affect the progression and severity of SFP accidents is a serious flaw.

Regarding limitations of the NRC's MELCOR computer safety model, in 2006, a SNL report observed that *MELCOR does not model the nitriding of zirconium alloy fuel cladding*, stating that fuel cladding would "combine with nitrogen if no oxygen or steam

345 NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools: Proposed Rules," p. 74790.

346 NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 20.

are available" and that the nitriding process is exothermic (heat-generating).[347] And in August 2012 a different SNL report, "Fukushima Daiichi Accident Study" stated: "If *inadequate* cooling is provided, then the cladding will heat up and will rapidly oxidize (*i.e.*, burn) and to a lesser extent, nitride (*i.e.*, combine with nitrogen if no oxygen or steam are available). *Since the oxidation and nitride processes are exothermic*, the fuel rods could heat to melting conditions and structurally degrade"[348] [emphasis added].

In an *April 2000* letter from Dana A. Powers, Chairman of the Advisory Committee on Reactor Safeguards ("ACRS"), to Richard A. Meserve, Chairman of the NRC, the ACRS advised the NRC Staff that an NRC report on SFP accident risk "relied on relatively geriatric work" for its *analysis of the interaction of air with zirconium fuel cladding*, stating that "[m]uch more is known now about air interactions with cladding," including knowledge gained "from studies being performed as part of a cooperative international program (PHEBUS FP[349]) in which NRC is a partner." The ACRS told the NRC Staff that "[a]mong the findings of this work *is that nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding*. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample"[350] [emphasis added].

As early as 1987, a report that was prepared for the NRC, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," stated that zirconium nitriding in air is an exothermic reaction, "releasing approximately 82 kcal/mole"—approximately 3.76 megajoules per kg of Zr reacted,[351] which is approximately 30 percent of the

[347] K. C. Wagner, R. O. Gauntt, Sandia National Laboratories, Analysis and Modeling Division, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," SAND1A Letter Report, Revision 2, p. 12.

[348] Randall Gauntt *et al*., Sandia National Laboratories "Fukushima Daiichi Accident Study: Status as of April 2012," SAND2012-6173, August 2012, p. 183.

[349] PHEBUS FP is an experimental program that researched severe-accident reactor core damage.

[350] Dana A. Powers, Chairman of ACRS, Letter to Richard A. Meserve, Chairman of NRC, Regarding ACRS Recommendations for Improvements to the NRC Staff's "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," April 13, 2000, (ADAMS Accession No. ML003704532), pp. 3-4.

[351] V. L. Sailor *et al*., Brookhaven National Laboratory, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," NUREG/CR-4982, July 1987, p. 109.

quantity of energy (per kg of Zr reacted) produced by the zirconium-oxygen reaction in air. Unfortunately, more than 25 years later, the NRC's Post-Fukushima MELCOR simulations still do not model how the nitrogen content of air would affect the progression of a SFP accident.

II.I.2. MELCOR Does Not Model How Nitrogen Accelerates the Oxidation and Degradation of Zirconium Fuel-Cladding in Air

MELCOR also *does not simulate* how nitrogen gas (in air) affects the oxidation of zirconium in air.[352] This is a serious flaw because the presence of nitrogen accelerates the oxidation (burning) and degradation of zirconium fuel-cladding *in air*,[353] which would affect the progression and severity of a SFP accident, including radioactive releases, "most notabl[y] ruthenium."[354] ("Ruthenium has a biological effectiveness equivalent to that of Iodine-131;"[355] Ruthenium-106 has half-life of 373.6 days.) Hence, the NRC's MELCOR simulations of SFP accidents *under-predict* the severity of such accidents.

A 2010 Journal of Nuclear Materials paper observes that "[t]he complexity of air oxidation of Zircaloy arises out of the simultaneous oxidation and nitriding processes."[356] And a May 2013 report, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," discusses experimental data demonstrating that porous nitrides form inside

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352 K. C. Wagner, R. O. Gauntt, Sandia National Laboratories, Analysis and Modeling Division, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," SAND1A Letter Report, Revision 2, p. 12; and L. Fernandez-Moguel, J. Birchley, European MELCOR User's Group, "PSI air oxidation model in MELCOR: Part 2: Analysis of experiments and model assessment," Stockholm, May 2013, which states: "Neither MELCOR nor SCDAP [a severe accident computer safety model] are able to predict a nitride reaction."

353 J. Stuckert, M. Große, Z. Hózer, M. Steinbrück, Karlsruhe Institute of Technology, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," KIT-SR 7634, May 2013, p. 1; and O. Coindreau, C. Duriez, S. Ederli, "Air Oxidation of Zircaloy-4 in the 600-1000°C Temperature Range: Modeling for ASTEC Code Application," p. 208.

354 J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 1.

355 Dana A. Powers, Chairman of ACRS, Letter to Richard A. Meserve, Chairman of NRC, Regarding ACRS Recommendations for Improvements to the NRC Staff's "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," p. 2.

356 O. Coindreau, C. Duriez, S. Ederli, "Air Oxidation of Zircaloy-4 in the 600-1000°C Temperature Range: Modeling for ASTEC Code Application," p. 207.

oxide layers *under local or full oxygen-starvation conditions*.[357] (When zirconium reacts in air it is possible for the reaction to become oxygen-starved; however, if zirconium is locally oxygen-starved in air, nitrogen will react with it.) The porous, degraded condition of an oxide layer facilitates accelerated oxidation rates if additional oxygen becomes *locally* available; and any additional oxygen will react with the zirconium nitride (ZrN) within an existing oxide layer and form zirconium dioxide (ZrO_2) in a fast exothermic reaction.[358]

As quoted above, an *April 2000* ACRS letter states that "[m]uch more is known now about air interactions with cladding;"[359] however, *a 2008* Journal of Nuclear Materials paper, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," states:

> Oxidation of zirconium alloys at high temperature for severe accident analysis has been widely studied in steam, however, the existing data regarding air oxidation in the temperature range of interest are scarce. …the exact role of zirconium nitride on the cladding degradation process is poorly understood. It remains unclear to [what] extent the nitrogen effect is responsible for the kinetic acceleration of the oxidation process that has been observed by these authors.
>
> Further[more], it should be stressed that most of the existing data have been obtained with bare [non-oxidized] samples.[360]

Regarding nitrogen-induced breakaway oxidation, the 2008 Journal of Nuclear Materials paper explains that "[b]reakdown and loss of the dense scale protective effect occur and result in an accelerated degradation;" furthermore, the transition to nitrogen-induced breakaway oxidation occurs *earlier with pre-oxidized fuel cladding* than with fresh *non-oxidized* fuel cladding—"nitriding is favored by the 'corrosion' scale."[361]

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[357] J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 10.
[358] Emilie Beuzet *et al*., "Modelling of Zry-4 Cladding Oxidation by Air Under Severe Accident Conditions using MAAP4 Code," International Conference Nuclear Energy for New Europe 2009, Slovenia, September 2009, p. 3.
[359] Dana A. Powers, Chairman of ACRS, Letter to Richard A. Meserve, Chairman of NRC, Regarding ACRS Recommendations for Improvements to the NRC Staff's "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," p. 3.
[360] C. Duriez, T. Dupont, B. Schmet, F. Enoch, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," Journal of Nuclear Materials 380 (2008), p. 30.
[361] *Id*., p. 44.

It is clear that *in air*, in a SFP accident, a significant degree of zirconium oxidation would occur, because spent fuel rods would be "pre-oxidized." When high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 μm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick. And medium to high burnup fuel cladding typically has a "hydrogen concentration in the range of 100-1000 wppm [weight parts per million];" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[362]

Regarding limitations of air oxidation models, the May 2013 report, "Results of the QUENCH-16 Bundle Experiment on Air Ingress," states that "[t]he models for air oxidation do not yet cover the whole range of representative conditions. The main aims of new bundle tests should be the investigation of areas where data [are] mostly missing."[363] And, a 2009 paper, regarding needed development for MELCOR *in the area of air ingress*, states that "air oxidation cannot be reliably predicted (or even described conservatively) by any of the models used in the currently available codes. A new modeling approach and an appropriate database are therefore necessary."[364] Additionally, information about the French Mozart Program to study the zirconium-air reaction states that "[b]ibliographic reviews reveal wide scattering of the existing kinetic data concerning the oxidation of Zircaloy-4 by air in the temperature range concerned [600°C to 1200°C]. *For recent alloys, such as M5 and Zirlo, there is virtually no data published in the open literature*"[365] [emphasis added].

In a June 2013 document, the NRC explained that a new air oxidation kinetics model was added to MELCOR version 1.8.6 (2005) that is based on data from

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[362] K. Natesan, W.K. Soppet, Argonne National Laboratory, "Hydrogen Effects on Air Oxidation of Zirlo Alloy," NUREG/CR–6851, October 2004, (ADAMS Accession No. ML042870061), p. iii, 3.

[363] J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 1.

[364] S. Güntay, J. Birchley, "MELCOR Further Development in the Area of Air Ingress and Participation in OECDNEA SFP Project to Be Performed in the Time Frame 2009-2012," April 2009, p. 4.

[365] IRSN, website description of the Mozart Program; available at: http://www.irsn.fr/EN/Research/Research-organisation/Research-programmes/SOURCE-TERM/MOZART/Pages/The-MOZART-programme-on-the-PWR-fuel-cladding-oxidation-in-air-3238.aspx (last visited 10/22/13).

isothermal[366] air zirconium-oxidation experiments conducted at Argonne National Laboratory ("ANL"). The ANL data (published in 2004) demonstrated that "air oxidation can be observed at temperatures as low as 600 K [327°C (620°F)];" and that the breakaway phenomenon that occurs when zirconium is oxidized in air causes "a sharp increase" in reaction and heatup rates in the post-breakaway regime. Apparently, MELCOR version 1.8.6 "provide[s] a better prediction of the measured data, including a transition to accelerated post-breakaway oxidation kinetics."[367]

MELCOR version 1.8.6 may provide a "better prediction" of the measured air oxidation data, than older versions. However, the Paul Scherrer Institute ("PSI") recently assessed MELCOR 1.8.6's ability to predict fuel-cladding behavior in accidents involving air ingress into the reactor vessel—which is pertinent to MELCOR's ability to predict zirconium-air reaction rates in SFP accidents—and "concluded that development of MELCOR was needed *to capture the accelerated cladding oxidation that can take place under air ingress conditions* (characterized by transition from formation of a protective oxide film to non-protective 'breakaway' oxidation at a significantly higher rate)"[368] [emphasis added].

PSI has also explained:

> Although there was not, [in] the 1980's, any systematic treatment of air oxidation, correlations had been developed on the basis of limited data[369] and these had been adapted for use in MELCOR in [an] attempt to provide a conservative statement of the thermal response to an air ingress scenario. A feature of all these correlations was that the controlling processes were similar to those which govern steam oxidation. The US-NRC later

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[366] The tests ANL were *isothermal tests*, in which "a [zirconium alloy] specimen was held at constant temperature and the weight gain associated with oxidation as a function of time was measured." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 93.
[367] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," pp. 93-94.
[368] S. Güntay, J. Birchley, "MELCOR Further Development in the Area of Air Ingress and Participation in OECDNEA SFP Project to Be Performed in the Time Frame 2009-2012," April 2009, p. 2.
[369] A. Benjamin *et al*., "Spent Fuel Heatup following Loss of Water during Storage," NUREG/CR-0649, SAND77-1371, March 1979, (ADAMS Accession No. ML120960637); and V. Sailor *et al*.," Severe Accidents in Spent Fuel Ponds in Support f Generic Issue 82", NUREG/CR-4982, July 1987.

> commissioned experimental studies[370] [the ANL isothermal experiments] to obtain data with which to establish a credible physical basis for using the correlations. *More recent experiments*[371] *demonstrated that the processes that govern air oxidation are quite different from those which apply to steam oxidation*[372] [emphasis added].

Clearly, the NRC's conclusions from its Post-Fukushima MELCOR simulations are non-conservative *and misleading*, because their conclusions *underestimate* the probabilities of large radiological releases from SFP accidents. By overlooking the deficiencies of its Post-Fukushima MELCOR simulations, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[373]

II.I.3. The NRC's Recent Non-Conservative Post-Fukushima MELCOR Simulations

A recent NRC Post-Fukushima MELCOR (version 1.8.6 of the code[374]) simulation of *a particular* BWR Mark I SFP fire scenario ("Unsuccessful Deployment of Mitigation for Moderate Leak (OCP3) Scenario"[375]) found that in the central area of the SFP, "Radial Ring 1"—where the newly discharged, hottest, fuel assemblies were stored—the peak fuel-cladding temperature would reach approximately 1800 K (1527°C) (2780°F) at "Axial Level 4."[376] However, the same simulation also found that "[a]fter the peak

[370] K. Natesan, W.K. Soppet, Argonne National Lab (ANL), "Air Oxidation Kinetics for Zr-Based Alloys," NUREG/CR-6846, July 2004, (ADAMS Accession No. ML041900069).

[371] These recent experiments are discussed in the four following reports: 1) M. Steinbrueck, U. Stegmeier, T. Ziegler, "Prototypical Experiments on Air Oxidation of Zircaloy-4 at High Temperature," FZK 7257, January 2007; 2) G. Schanz *et al*., "Results of QUENCH-10 Experiment on Air Ingress," FZKA 7057, May 2006; 3) Ch. Duriez *et al*., "Separate effect Tests on Zirconium Cladding Degradation in Air Ingress Situations," Proceedings of 2nd ERMSAR Conference, Karlsruhe, Germany, 2007; and 4) A. Auvinen *et al*., "Progress on ruthenium release and transport under air ingress Conditions," Nuclear Engineering and Design, 238, 2008, pp. 3418–3428.

[372] S. Güntay, J. Birchley, "MELCOR Further Development in the Area of Air Ingress and Participation in OECDNEA SFP Project to Be Performed in the Time Frame 2009-2012," p. 4.

[373] 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, (ADAMS Accession No. ML111861807), p. 3.

[374] The SFP models in MELCOR versions 1.8.6 and 2.1 are functionally the same. See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," pp. 92-93.

[375] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 142.

[376] For MELCOR "[t]he core is nodalized into a number of axial levels and radial rings (each ring represents a collection of assemblies);" and "MELCOR core models were originally designed for

temperature [is reached] at [Axial] Level 4, the peak temperature in the zirconium fire front decreases with each successive [axial] level. Radial heat transfer[377] from the fuel racks to the SFP wall…, *the buildup of the oxide layer on the fuel, and the depletion of the oxygen in the reactor building…cause the clad temperature to decrease*. After 24 hours, the fuel temperatures in [Radial] Ring 1 are relatively stable"[378] [emphasis added]. (In this scenario there is a depletion of the oxygen in the reactor building, because the reactor building was *not* breached by a hydrogen explosion (a total of four reactor buildings were breached by hydrogen explosions in the Fukushima Dai-ichi accident[379]).

This recent NRC MELCOR simulation—in which there is a depletion of the oxygen in the reactor building—would have had *different results* if it had modeled: 1) how nitriding would degrade the fuel-cladding's "protective" oxide layer and accelerate the zirconium oxidation, which would contribute additional heat; 2) the nitriding of zirconium under oxygen-starvation conditions; and 3) the significant additional heat that would be contributed from the exothermic nitrogen-zirconium reaction.

In other recent NRC MELCOR simulations of BWR Mark I SFP accident/fire scenarios, the reactor buildings were breached by hydrogen explosions, so there was more available oxygen to facilitate zirconium oxidation. However, those simulations would have had *different results* if they had modeled: 1) how nitriding would degrade the fuel-cladding's "protective" oxide layer and accelerate the zirconium oxidation, which

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the reactor core. Because of the code flexibility, the same modeling approach can be used for the spent fuel pool (with the addition of the rack as a separate component)." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 95, and p. 95, Note 12.

[377] "MELCOR attempts to model a multidimensional geometry with a simplified two-surface radiation model." See NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," p. 110, Note 23.

[378] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report," pp. 142-143.

[379] In the Fukushima Dai-ichi accident, hydrogen detonated in and essentially destroyed the secondary containments of Units 1, 3, and 4, causing large releases of radiation. And the secondary containment of Unit 2 was breached: a hydrogen explosion that occurred in the Unit 1 reactor building "caused a blowout panel in the Unit 2 reactor building to open, which resulted in a loss of secondary containment integrity." See INPO, "Special Report on the Nuclear Accident at the Fukushima Dai-ichi Nuclear Power Station," INPO 11-005, November 2011, p. 24.

would contribute additional heat and 2) the significant additional heat that would be contributed from the exothermic nitrogen-zirconium reaction.[380]

In actual SFP fires, there would be quicker fuel-cladding temperature escalations, releasing more heat, and quicker axial and radial propagation of zirconium fires than MELCOR indicates.

II.I.4. Recent Sandia National Laboratory Spent Fuel Pool Accident Experiments Are Unrealistic because They Were Conducted with Clean Non-Oxidized Cladding

Recent Sandia National Laboratory ("SNL") SFP accident experiments are unrealistic because they have been conducted with clean non-oxidized bundles of zirconium fuel rod simulators;[381] the spent fuel assemblies stored in SFPs have oxide layers. When high burnup (and other) fuel rods are discharged from the reactor core and loaded into the SFP, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 μm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick. And medium to high burnup fuel cladding typically has a "hydrogen concentration in the range of 100-1000 wppm [weight parts per million];" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[382]

Regarding nitrogen-induced breakaway oxidation, the 2008 Journal of Nuclear Materials paper explains that "[b]reakdown and loss of the dense scale protective effect occur and result in an accelerated degradation;" furthermore, the transition to nitrogen-induced breakaway oxidation occurs *earlier with pre-oxidized fuel cladding* than with fresh *non-oxidized* fuel cladding—"nitriding is favored by the 'corrosion' scale."[383]

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[380] NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor: Draft Report."

[381] E. R. Lindgren, Sandia National Laboratory, "Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident," NUREG/CR-7143, March 2013, (ADAMS Accession No. ML13072A056).

[382] K. Natesan, W.K. Soppet, Argonne National Laboratory, "Hydrogen Effects on Air Oxidation of Zirlo Alloy," NUREG/CR–6851, October 2004, (ADAMS Accession No: ML042870061), p. iii, 3.

[383] C. Duriez, T. Dupont, B. Schmet, F. Enoch, "Zircaloy-4 and M5 High Temperature Oxidation and Nitriding in Air," Journal of Nuclear Materials 380 (2008), p. 44.

It is clear that *in air*, in a SFP accident, there would be a significant degree of zirconium oxidation, because the spent fuel rods in the pool would be "pre-oxidized." This phenomenon of nitrogen attacking pre-oxidized zirconium alloy cladding is not simulated in SNL's experiments. Hence, data from SNL's SFP accident experiments is inadequate for benchmarking MELCOR. Benchmarking a computer safety model with data gathered from unrealistic experiments undermines the NRC's philosophy of defense-in-depth, which requires the application of conservative models.[384]

II.J. Experimental Data Indicates that MELCOR Under-Predicts the Zirconium-Steam Reaction Rates that Would Occur in a Spent Fuel Pool Accident

II.J.1. Oxidation Models Are Not Able to Predict the Fuel-Cladding Temperature Escalation that Commenced at "Low Temperatures" in the PHEBUS B9R Test

As stated above, the PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[385] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[386]

Discussing PHEBUS B9R-2, the 1996 European Commission report states:

> The B9R-2 test (second part of B9R) illustrates the oxidation in different cladding conditions representative of a pre-oxidized and fractured state. This state results from a first oxidation phase (first part name B9R-1, of the B9R test) terminated by a rapid cooling-down phase. During B9R-2, an unexpected strong escalation of the oxidation of the remaining Zr occurred when the bundle flow injection was switched from helium to steam while the maximum clad temperature was equal to 1300 K [1027°C (1880°F)]. *The current oxidation model was not able to predict the strong heat-up rate observed* even taking into account the measured large clad

[384] 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, p. 3.

[385] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 311.

[386] T.J. Haste *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, 1996, p. 33.

> deformation and the double-sided oxidation (final state of the cladding from macro-photographs).
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> … *No mechanistic model is currently available to account for enhanced oxidation of pre-oxidized and cracked cladding*[387] [emphasis added].

Today, in 2013, oxidation models still cannot accurately predict the local fuel-cladding temperature escalation that commenced in PHEBUS B9R when local fuel-cladding temperatures were 1027°C (1880°F). The PHEBUS B9R results indicate that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Urbanic-Heidrick correlations, are inadequate for use in computer safety models like MELCOR.

II.J.2. "Low Temperature" Oxidation Rates Are Under-Predicted for the CORA-16 Experiment

When Oak Ridge National Laboratory ("ORNL") investigators compared the results of the CORA-16 experiment—a BWR core severe fuel damage test, simulating a meltdown, conducted with a multi-rod zirconium alloy bundle—with the predictions of computer safety models, they found that the zirconium-steam reaction rates that occurred in the experiment were under-predicted. The investigators concluded that the "application of the available Zircaloy oxidation kinetics models [zirconium-steam reaction correlations] causes the low-temperature [1652-2192°F] oxidation to be underpredicted."[388]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in the CORA-16 experiment.[389] However, it is *unsubstantiated* that cladding strain actually increased reaction rates.

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, the NRC has pointed out that

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387 *Id.*, p. 126.

388 L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," ORNL/FTR-3780, October 16, 1990, p. 3.

389 L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," CONF-9105173-3-Extd.Abst., Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

an NRC report, NUREG/CR-4412,[390] "explain[s] that under *certain* conditions ballooning and deformation of the cladding can increase the available surface area for oxidation, thus enhancing the apparent oxidation rate"[391] [emphasis not added].

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA or [severe accident] will permit [fuel] cladding deformation (ballooning and possibly rupture) to occur because the fuel rod internal pressure may be greater than the external (coolant) pressure. In this case, oxidation and deformation can occur simultaneously. This in turn may result in an apparent enhancement of oxidation rates because: 1) ballooning increases the surface area of the cladding and permits more oxide to form per unit volume of Zircaloy and 2) the deformation may crack the oxide and provide increased accessibility of the oxygen to the metal. However deformation generally occurs before oxidation rates become significant; *i.e.*, below [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[392]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[393] NUREG/CR-4412 also explains that "it is possible to make a very crude estimate of the expected average enhancement of oxidation kinetics by deformation;"[394] the report provides a graph of the "rather sparse"[395] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[396]

NUREG/CR-4412 has a brief description of the rather sparse data; in one case, two investigators (Furuta and Kawasaki), who heated specimens up to temperatures

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[390] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, (ADAMS Accession No: ML083400371).
[391] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, (ADAMS Accession No: ML112211930), p. 3.
[392] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 27.
[393] *Id.*, pp. 27, 30.
[394] *Id.*, p. 30.
[395] *Id.*
[396] *Id.*, p. 29.

between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[397]

In fact, NUREG/CR-4412 states that only one pair of investigators (Bradhurst and Heuer) conducted tests that encompassed the temperature range—1652°F to 2192°F—in which zirconium-steam reaction rates were under-predicted for CORA-16. Bradhurst and Heuer reported that "[m]aximum enhancements occurred at slower strain rates. … However, the overall weight gain or average oxide thickness in [the Zircaloy-2 specimens] was only minimally increased because of the localization effects of cracks in the oxide layer."[398] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[399] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a very brief period of time, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain *may* only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater.[400]

Furthermore, ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F temperature range for CORA-16.

There is also one phenomenon NRC did not consider in its 2011 analysis of CORA-16: "[t]he swelling of the [fuel] cladding…alters [the] pellet-to-cladding gap in a

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[397] *Id.*, p. 30.
[398] *Id.*
[399] F. J. Erbacher, S. Leistikow, "A Review of Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident," Kernforschungszentrum Karlsruhe, KfK 3973, September 1985, p. 6.
[400] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 30.

manner that provides less efficient energy transport from the fuel to the cladding,"[401] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

CORA-16 is an example of an experiment that had zirconium-steam reaction rates that were under-predicted in the "low temperature" range from 1652°F to 2192°F by computer safety models. The CORA-16 results indicate that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Urbanic-Heidrick correlations, are inadequate for use in computer safety models like MELCOR.

About the Author

On March 15, 2007, Mark Leyse submitted a 10 C.F.R. § 2.802 petition for rulemaking ("PRM"), PRM-50-84,[402] to the NRC. PRM-50-84 requested: 1) that NRC make new regulations to help ensure licensees' compliance with 10 C.F.R. § 50.46(b) emergency core cooling systems ("ECCS") acceptance criteria and 2) to amend Appendix K to Part 50—ECCS Evaluation Models I(A)(1), "The Initial Stored Energy in the Fuel."

In 2008, the NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[403] And in 2009, NRC published "Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of a proposed rulemaking, addressing four objectives: the fourth being the issues raised in PRM-50-84.[404] In 2012, the NRC Commissioners voted unanimously to approve a

[401] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, ((ADAMS Accession No: ML041950059), p. 43.
[402] Mark Leyse, PRM-50-84, March 15, 2007 (ADAMS Accession No. ML070871368).
[403] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.
[404] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that was partly based on the safety issues Leyse raised in PRM-50-84.[405]

Leyse also coauthored a paper, "Considering the Thermal Resistance of Crud in LOCA Analysis,"[406] which was presented at the American Nuclear Society's 2009 Winter Meeting.

[405] 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).

[406] 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, 2009.2