Rulemaking1CEm Resource

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Attachments: Mark Leyse's Comments on the NRC's "Regulatory Improvements for Decommissioning Power Reactors;" Proposed Rules; NRC-2015-0070.pdf

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From: Mark Leyse [mailto:markleyse@gmail.com]
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Subject: [External_Sender] NRC-2015-0070

Dear Annette L. Vietti-Cook:

In the attached comments, dated March 18, 2016, I respond to the U.S. Nuclear Regulatory Commission's notice of solicitation of public comments on "Regulatory Improvements for Decommissioning Power Reactors;" Proposed Rules; NRC-2015-0070, published in the Federal Register on November 19, 2015.

In the attached comments, I provide evidence that the NRC's MELCOR analyses of spent fuel pool (SFP) accidents and fires are not based on sound science. MELCOR analyses under-predict the severity and radiological releases of SFP fires. The NRC cannot plan for the portions of decommissioning in which there would be irradiated fuel in the SFP until it conducts analyses of SFP accidents and fires that are based on sound science.

In accordance with its philosophy of defense-in-depth, which requires the application of conservative models, the NRC must conduct safety analyses of SFP accidents and fires that are based on sound science.

On the basis of its non-conservative MELCOR analyses, the NRC has already decided to not require licensees to expedite the transfer of spent fuel assemblies from SFPs to dry cask storage. The NRC must disregard the results of its flawed MELCOR analyses. Many questions pertinent to the storage of spent fuel assemblies—

including emergency-preparedness questions for decommissioning—cannot be answered until safety analyses of SFP accidents and fires are based on sound science.

Sincerely,

Mark Leyse

March 18, 2016

Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

Mark Leyse's Comments on the NRC's "Regulatory Improvements for Decommissioning Power Reactors;" Proposed Rules; NRC-2015-0070

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March 18, 2016

Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

Subject: Response to the U.S. Nuclear Regulatory Commission's ("NRC") notice of solicitation of public comments on "Regulatory Improvements for Decommissioning Power Reactors;" Proposed Rules; NRC-2015-0070.

Mark Leyse's Comments on the NRC's "Regulatory Improvements for Decommissioning Power Reactors;" Proposed Rules; NRC-2015-0070

I, Mark Leyse ("Commenter"), am responding to the Federal Register notice that the NRC published on November 19, 2015 soliciting public comments on its proposed rules on improvements for decommissioning nuclear power reactors.

In these comments Commenter addresses safety issues pertinent to questions that the NRC asked in its solicitation of public comments on its proposed rules. In its Federal Register notice, among other things, the NRC asked "questions related to emergency preparedness requirements for decommissioning power reactor licensees."[1]

The NRC also stated that its "[r]ulemaking may involve a tiered approach for modifying [emergency preparedness] requirements based on several factors, including, but not limited to, the source term after cessation of power operations, removal of fuel from the reactor vessel, elapsed time after permanent defueling, and type of long-term onsite fuel storage."[2]

In its second series of emergency-preparedness questions—EP-2—the NRC asked the following three (out of a total of four) questions:

> a. What tiers and associated EP requirements would be appropriate to consider for this approach?
>
> b. What factors should be considered in establishing each tier?

[1] NRC, "10 CFR Parts 26, 50, 52, 73, and 140: "Regulatory Improvements for Decommissioning Power Reactors," Proposed Rules; NRC-2015-0070, Federal Register, Vol. 80, No. 223, November 19, 2015, p. 72362.
[2] *Id*., p. 72363.

c. What type of basis could be established to support each tier or factor?[3]

Commenter believes that David Lochbaum, director of the Union of Concerned Scientists' Nuclear Safety Project, has provided an excellent answer to Question a.

Regarding how the tiers of the decommissioning of nuclear power reactors should be defined, Mr. Lochbaum recommends that:

> The tiers during decommissioning are: (1) from cessation of reactor operation until removal of all irradiated fuel from the reactor vessel, (2) from removal of all irradiated fuel from the reactor vessel until removal of all irradiated fuel from the spent fuel pool, (3) removal of all irradiated fuel from the spent fuel pool until removal of all irradiated fuel from the site, and (4) from removal of all irradiated fuel from the site until termination of the license.[4]

Commenter's comments address safety issues pertinent to the first and second tiers, in which there would be irradiated fuel in the spent fuel pool ("SFP"). As, the NRC states in its Federal Register notice, during decommissioning, a SFP fire could lead to a significant radiological release.[5] In these comments Commenter provides evidence that the NRC's safety analyses of SFP accidents and fires are not based on sound science; the NRC's analyses under-predict the severity and radiological releases of SFP fires.

The NRC cannot plan for the first and second tiers of decommissioning until it conducts safety analyses of SFP accidents and fires that are based on sound science. Many questions pertinent to the storage of spent fuel assemblies—including emergency-preparedness questions for decommissioning—cannot be answered until realistic safety analyses are conducted.

I. Statement of Commenter's Interest

On March 15, 2007, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-84,[6] to the NRC. PRM-50-84 requested: 1) that the NRC make new regulations to help ensure licensees' compliance with 10 C.F.R. § 50.46(b) emergency core cooling

[3] *Id*.
[4] David Lochbaum, "Comments on the NRC's 'Regulatory Improvements for Decommissioning Power Reactors;' Proposed Rules; NRC-2015-0070," December 22, 2015, (ADAMS Accession No. ML16013A124).
[5] NRC, "10 CFR Parts 26, 50, 52, 73, and 140: "Regulatory Improvements for Decommissioning Power Reactors," Proposed Rules; NRC-2015-0070, p. 72359.
[6] Mark Edward Leyse, PRM-50-84, March 15, 2007 (ADAMS Accession No. ML070871368).

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 safety issues raised in PRM-50-84 in its rulemaking process.[7] And in 2009, the 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.[8] In 2012, the NRC Commissioners voted unanimously to approve a proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that was partly based on the safety issues Commenter raised in PRM-50-84.[9]

Commenter also coauthored a paper, "Considering the Thermal Resistance of Crud in LOCA Analysis," which was presented at the American Nuclear Society's 2009 Winter Meeting.[10]

On June 19, 2014, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-108,[11] to the NRC. Among other things, PRM-50-108 requests that the NRC enact new regulations requiring that computer simulations of postulated SFP accidents model how the affects of nitrogen would accelerate the progression of and increase the severity of SFP fires.

[7] NRC, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," Docket No. PRM–50–84; NRC–2007–0013, Federal Register, Vol. 73, No. 228, November 25, 2008, pp. 71564-71569.

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

[9] 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).

[10] 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.

[11] Mark Edward Leyse, PRM-50-108, June 19, 2014 (ADAMS Accession No. ML14195A388).

II. The Spent Fuel Pool Fire Scenario May Not Be "Highly Unlikely"—As the NRC Claims in Its Advance Notice of a Proposed Rulemaking

In its advance notice of a proposed rulemaking, regarding "Regulatory Improvements for Decommissioning Power Reactors," the NRC states:

> During reactor decommissioning, the principal radiological risks are associated with the storage of spent fuel onsite. Generally, a few months after the reactor has been permanently shut down, there are no possible design-basis events that could result in a radiological release exceeding the limits established by the U.S. Environmental Protection Agency's (EPA) early-phase Protective Action Guidelines of [one] roentgen equivalent man at the exclusion area boundary. The only accident that might lead to a significant radiological release at a decommissioning reactor is a zirconium fire. The zirconium fire scenario is a postulated, but *highly unlikely*, beyond-design-basis accident scenario that involves a major loss of water inventory from the spent fuel pool (SFP), resulting in a significant heat-up of the spent fuel, and culminating in substantial zirconium cladding oxidation and fuel damage. The analyses of spent fuel heat-up scenarios that might result in a zirconium fire are related to the decay heat of the irradiated fuel stored in the SFP. Therefore, the probability of a zirconium fire scenario continues to decrease as a function of the time that the decommissioning reactor has been permanently shut down[12] [emphasis added].

According to statements that the NRC published in the Federal Register in December 2012 announcing that PRM-50-96[13] had been accepted, the SFP fire scenario may not be "highly unlikely."

PRM-50-96 was submitted by Thomas Popik of The Foundation for Resilient Societies on March 14, 2011, three days after a tsunami and earthquake struck Japan, leading to three meltdowns at the Fukushima Daiichi nuclear plant. PRM-50-96 requested that new regulations be enacted to help prevent SFP fires in the event of long-term power blackouts.

PRM-50-96 addressed scenarios in which the North American power grids would experience long-term blackouts that would last months to years. PRM-50-96 cited and quoted reports stating that coronal mass ejections from the Sun could direct electrically-

[12] NRC, "10 CFR Parts 26, 50, 52, 73, and 140: "Regulatory Improvements for Decommissioning Power Reactors," Proposed Rules; NRC-2015-0070, p. 72359.

[13] Thomas Popik, The Foundation for Resilient Societies, PRM-50-96, March 14, 2011, (ADAMS Accession No. ML110750145).

charged particles to Earth, causing "a 1-in-100-year geomagnetic storm." Such an event could, in turn, cause more than 300 extra high voltage transformers[14] to overheat and incur permanent damage, leading to large-scale, long-term blackouts.[15]

There are other events that could also lead to power blackouts that would last months to years, including cyberattacks. It is noteworthy that in 2015 Ted Koppel released a bestselling book, *Lights Out: A Cyberattack, A Nation Unprepared, Surviving the Aftermath*, addressing the threat of cyberattacks collapsing the U.S. power grid for a period of months. Koppel interviewed General Lloyd Austin III, Commander of U.S. Central Command for *Lights Out*. Koppel asked the General if he thinks that a cyberattack could disable a large portion of the U.S. power grid. "It's not a question of if," the General replied, "it's a question of when someone will try that."[16]

If large-scale power outages were to last months or longer, multiple nuclear power plants 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 events—especially in the event of prolonged power grid failures—could lead to a number of station-blackouts; a station-blackout is a complete loss of both grid-supplied and backup onsite AC power. The Fukushima Dai-ichi accident was a station-blackout accident that led to three reactor core meltdowns.

The NRC seems to agree with PRM-50-96's argument that SFPs are vulnerable to zirconium fires in the event of blackouts that would last months to years. In December 2012, NRC staff members decided to consider safety issues raised in PRM-50-96 in its rulemaking process.[17]

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[14] 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.

[15] Oak Ridge National Laboratory, Executive summary of "Electromagnetic Pulse: Effects on the U.S. Power Grid," a collection of six technical reports written for ORNL by Metatech Corporation, January 2010, pp. i, ii.

[16] Ted Koppel, *Lights Out: A Cyberattack, A Nation Unprepared, Surviving the Aftermath*, (New York: Crown Publishers, 2015), p. 89.

[17] 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.

In its December 2012 Federal Register notice announcing that PRM-50-96 had been accepted, the NRC stated:

> The NRC's initial evaluation of available information indicates that the likelihood of an extreme solar storm (similar to the 1859 Carrington event[18]) is plausible with a frequency in the range of once in 153 to once in 500 years (2E–3 to 6.5E–3 per year). The probability of the petitioner's postulated catastrophic grid failure, given a Carrington-like event, is not known with certainty. However, based on the NRC's review of the existing data, the NRC believes that there is insufficient information for the NRC to conclude that the overall frequency of a series of events potentially leading to core damage at multiple nuclear sites is acceptably low such that no regulatory action is needed. Thus, the NRC concludes that the petitioner's scenario is sufficiently credible to require consideration of emergency planning and response capabilities under such circumstances. Accordingly, the NRC intends to further evaluate the petitioner's concerns in the NRC rulemaking process.[19]

The NRC's conclusion is that the frequency of a "catastrophic grid failure" from a Carrington-type event may be as high as once in 153 years. And the NRC also concludes that such a power failure, in turn, could initiate "a series of events potentially leading to core damage at multiple nuclear sites." But has the NRC ever conducted probabilistic risk assessments ("PRA") estimating the core damage frequency that could occur at "multiple nuclear sites" in the event of long-term catastrophic grid failures—blackouts that would last months to years? Or has the NRC conducted PRAs estimating the frequency of SFP fires that could occur at multiple nuclear sites in the event of long-term catastrophic grid failures?

An NRC Fact Sheet on PRAs explains: "One of the Nuclear Regulatory Commission's key responsibilities is to ensure the operation of nuclear power plants and other NRC-licensed facilities present no undue risk to public health and safety."[20] Given that the frequency of a catastrophic grid failure may be as high as once in 153 years, a PRA considering the frequency of core damage and SFP fires at multiple nuclear sites, in

[18] The Carrington event occurred in 1859; it is the largest solar storm to hit Earth ever recorded.
[19] NRC, "Long-Term Cooling and Unattended Water Makeup of Spent Fuel Pools," Proposed Rules, Docket No. PRM–50–96, NRC–2011–0069, p. 74790.
[20] NRC, "Fact Sheet: Probabilistic Risk Assessment," October 2007, (ADAMS Accession No. ML032200337).

the event of a catastrophic grid failure, may find that the operation of nuclear power plants and their *over-packed* SFPs presents an "undue risk to public health and safety."

PRM-50-96 identified one of the events—blackouts that would last months to years—that most threatens public health and safety. The NRC's Federal Register notice announcement on PRM-50-96's acceptance also warns about the threat of long-term-blackouts. However, the NRC has *failed* to properly address the fact that the safety of SFPs (as well as reactor cores) is threatened by long-term-blackouts.

III. The NRC's Computer Safety Model MELCOR, which Simulates Postulated Spent Fuel Pool Accidents and Fires

MELCOR is a computer safety model that the NRC uses to simulate postulated SFP accidents. The NRC has conducted MELCOR simulations of postulated SFP accidents and fires in the aftermath of the Fukushima Daiichi accident. A number of such MELCOR simulations are discussed in the September 2014 document, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" (hereinafter: NUREG-2161).[21]

III.A. Recent NRC MELCOR Simulations of Spent Fuel Pool Accidents and Fires Did Not Consider the Worst Case Scenario

Referring to NUREG-2161, in its advance notice of a proposed rulemaking, regarding "Regulatory Improvements for Decommissioning Power Reactors," the NRC states:

> The NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (ADAMS Accession No. ML14255A365), evaluated the potential benefits of strategies required in § 50.54(hh)(2). The NUREG-2161 found that successful implementation of mitigation strategies significantly reduces the likelihood of a release from the SFP in the event of a loss of cooling water. Additionally, NUREG-2161 found that the placement of spent fuel in a dispersed configuration in the SFP, such as the 1 x 4 pattern, would have a positive effect in promoting natural circulation, which enhances air coolability and

[21] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, September 2014, (ADAMS Accession No. ML14255A365).

> thereby reduces the likelihood of a release from a completely drained SFP.[22]

NUREG-2161—"the most detailed analysis to date of SFP consequences"[23]—is limited because it only considered that postulated SFP accidents and fires would be initiated by a large seismic event.[24] And, as the NRC's proposed rulemaking points out, NUREG-2161 only considered scenarios in which there would be "a completely drained SFP."[25]

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 fuel assemblies and exposing them to air over their entire length.[26] 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"[27]—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,[28] 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

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[22] NRC, "10 CFR Parts 26, 50, 52, 73, and 140: "Regulatory Improvements for Decommissioning Power Reactors," Proposed Rules; NRC-2015-0070, p. 72360.
[23] Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix B, p. B-10.
[24] *Id*., p. 6.
[25] *Id*., Appendix E, pp. E.26-E.27, E.32.
[26] 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.
[27] *Id*., p. 1.
[28] "[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 Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 77, Footnote 1.

from the fuel, causing the fuel to heat up at a higher rate than if natural circulation [were] occurring."[29]

Partial SFP LOCAs and SFP boil-off accidents resemble each other in that in both accidents there would be long times in which there was partial uncovery of the fuel assemblies. In both types of accidents, 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 SFP LOCAs (partial) and boil-off accidents make it more probable that those types of accidents would lead to SFP fires.

NUREG-2161 claims that in the event of a *complete* SFP LOCA, the spent fuel assemblies would *not* be air coolable for the first 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 the first 73 days out of every two years. However, NUREG-2161 states: "A *partial draindown event* with [BWR] channeled fuel could impede airflow and *increase the time to coolability*"[30] [emphasis added].

Elsewhere, NUREG-2161 states that a *partial* LOCA of a BWR Mark I SFP is assumed *not* to be air coolable for an entire two year operating cycle (730 days).[31] In other words, *partial* LOCAs of BWR Mark I SFPs, in which "the rack baseplate is not cleared and airflow is impeded,"[32] are *assumed not* to be air coolable during a reactor's entire life of operation. Over the course of its entire life of operation, a reactor's discharges of fuel assemblies are loaded into the SFP every two years. And, according to NUREG-2161, in the decommissioning process, for two years after reactor operation has terminated, a BWR Mark I SFP is assumed *not* to be air coolable in the event of a *partial* LOCA. As stated above, the baseplates also would not be cleared in SFP boil-off accidents—another reason such accidents could lead to SFP fires.

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[29] 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.
[30] Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix B, p. B-10.
[31] *Id*., Appendix D, p. D-13.
[32] *Id*.

IV. MELCOR Under-Predicts the Severity of Spent Fuel Pool Fires

The U.S. is particularly vulnerable to SFP fires, because its SFPs are *densely-packed* with spent fuel assemblies. For example, one SFP at a U.S. plant was originally intended to store 600 fuel assemblies. Subsequently, the plant was permitted by the NRC to store up to 3300 assemblies in the same SFP.[33] Gordon R. Thompson, executive director of the Institute for Resource and Security Studies, David Lochbaum of the Union of Concerned Scientists, and others, have argued for years that storing spent fuel less densely would help improve public safety.[34] However, the U.S. nuclear industry prefers high-density storage because it is cheaper. The NRC allows high-density storage to persist.

In the high-density storage racks of SFPs, the center-to-center distance between the spent fuel assemblies (the "pitch") is similar to the pitch of fuel assemblies in the reactor core. For example, some BWR reactor cores have a fuel assembly pitch of 6.0 inches[35] and some BWR SFPs have a spent fuel assembly pitch of 6.28 inches.[36] Additionally, some PWR reactor cores have a fuel assembly pitch of 8.587 inches[37] and some PWR SFPs have a spent fuel assembly pitch of 9.0 inches.[38]

The NRC once considered requiring licensees to expedite the transfer of spent fuel assemblies from SFPs to dry cask storage. That would have made SFPs far less

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

[34] Here are examples of comments of Mr. Thompson and Mr. Lochbaum have made arguing that storing spent fuel less densely would help improve public safety. See Gordon R. Thompson, Institute for Resource and Security Studies, "Comments on the US Nuclear Regulatory Commission's Draft Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor," August 1, 2013, (ADAMS Accession No. ML13225A397); see also David Lochbaum, Union of Concerned Scientists, "Comment On: NRC-2013-0136-0002 Draft Reports; Availability: Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," July 18, 2013, (ADAMS Accession No. ML13210A139).

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

[36] 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.

[37] 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.

[38] 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.

vulnerable to SFP fires. However, in 2013, the NRC decided to not require licensees to expedite the transfer of spent fuel assemblies. As explained in the NRC document COMSECY-13-0030, the NRC used the results of its NUREG-2161 MELCOR computer simulations—comparing postulated SFP accidents for a reference plant's SFP with high-density storage to the same SFP with low-density storage[39]—to help justify its decision to not expedite the transfer of spent fuel assemblies. (MELCOR version 1.8.6—released to users in July 2005[40]—was used for the simulations.[41])

The NUREG-2161 MELCOR analyses are seriously flawed because they under-predict the severity of SFP fires. For example, as discussed in detail in Sections IV.B.1 and IV.B.2, MELCOR *does not simulate* how nitrogen gas (in air) accelerates the oxidation (burning) and degradation of zirconium fuel cladding in *air*.[42] Nitrogen would accelerate the progression of and increase the severity of SFP accidents, including increasing their radiological releases. MELCOR also *does not simulate* the generation of heat from the chemical reaction of zirconium and nitrogen.[43] Neglecting to model a heat source that would also affect the progression and severity of SFP accidents is a second serious flaw.

The NRC's conclusions from its MELCOR analyses are non-conservative and *misleading*, because the NRC's conclusions *underestimate* the severity of SFP fires as well as the probabilities of large radiological releases from SFP fires. By overlooking the

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[39] NRC, COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," November 12, 2013, (ADAMS Accession No. ML13273A601), p. 3.

[40] Sandia National Laboratories ("SNL") developed the MELCOR computer model. A SNL website about MELCOR states that MELCOR version 1.8.6 was released to users in July 2005. See SNL, "MELCOR: A computer code for analyzing severe accidents in nuclear plants and the design basis accidents for advanced power plant applications," available at https://melcor.sandia.gov/about.html (last visited February 10, 2016).

[41] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, pp. 95-96. It is noteworthy that the SFP models in MELCOR versions 1.8.6 and 2.1 are functionally the same.

[42] 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.

[43] 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," p. 12.

deficiencies of its MELCOR simulations, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[44]

On the basis of its non-conservative MELCOR analyses, the NRC decided to not require licensees to expedite the transfer of spent fuel assemblies from SFPs to dry cask storage. The NRC is neglecting its duty to protect the public from potential SFP fires and their radiological releases.

IV.A. Deficiencies of MELCOR, Regarding the Air Cooling of Spent Fuel Assemblies in Storage Pools

According to a 2014 *Annals of Nuclear Energy* paper, severe accident codes, including MELCOR, use thermal hydraulic models that are not necessarily appropriate for SFPs. Regarding SFP modeling limitations, the paper states:

> The phenomena of natural convection and boiling in the fuel building. In fact, the conclusions on the coolability of [fuel assemblies] can be very different, in function of the calculations. Some studies show, for a loss of water transient (conducting to fast dewatering and air ingress in the [fuel assemblies]), that air flow is sufficient to remove the power, for other studies this conclusion depends on the air flow that could actually flow in the [fuel assemblies]. (Remark: Most of these calculations seem to *use thermal hydraulic parameters/models which seem not appropriate for SFP geometries*. Therefore, the gas flow is *strongly overestimated* and *non-conservative*. OECD SFP experiment showed ignition in a simulated 3 year old spent fuel element in air)[45] [emphasis added].

As stated in Section III.A, NUREG-2161 claims that in the event of a *complete* SFP LOCA, the spent fuel assemblies would be air coolable for 90 percent of a two year operating cycle. That is, the fuel assemblies would be air coolable for the last 657 days out of every two years (730 days); the fuel assemblies would *not* be air coolable for the first 73 days out of every two years.[46] NUREG-2161 also states that a *partial* SFP LOCA of a BWR Mark I SFP is assumed *not* to be air coolable for an entire two year operating

[44] 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.
[45] J. Fleurot *et al.*, "Synthesis of spent fuel pool accident assessments using severe accident codes," *Annals of Nuclear Energy*, 74, 2014, p. 70.
[46] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix B, p. B-10.

cycle (730 days).[47] However, differing from NUREG-2161, the 2014 *Annals of Nuclear Energy* paper states that at least one "experiment showed ignition in a simulated 3 year old spent fuel element in air."[48]

Experimental data demonstrates that claims of NUREG-2161 regarding the air coolability of spent fuel assemblies are false. And the fact that the simulated gas flow in MELCOR analyses of SFP accidents is *strongly overestimated* means that NUREG-2161's conclusions regarding the air coolability of spent fuel assemblies are extremely *non-conservative*.

IV.A.1. Fuel Rods in a Spent Fuel Pool Would Balloon and Burst as It Boiled Dry, Impeding Local Cooling of the Fuel Assemblies

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."[49] This is an additional flaw of MELCOR, regarding its modeling the air coolability of spent fuel assemblies. This is yet another reason to doubt NUREG-2161's claim that in the event of a *complete* SFP LOCA, the spent fuel assemblies would be air coolable for 90 percent of a two year operating cycle. That is, the fuel assemblies would be air coolable for the last 657 days out of every two years (730 days); the fuel assemblies would *not* be air coolable for the first 73 days out of every two years.[50]

In a SFP LOCA, the fuel rods would become uncovered by water and heat up from heat produced by decay heating, increasing their local temperatures. And, in a SFP boil-off accident, heat produced by decay heating would cause the SFP's water to boil away; the fuel rods would become uncovered by water and heat up, increasing their local temperatures. In both types of accidents, when local fuel-cladding temperatures reached

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47 Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix D, p. D-13.

48 J. Fleurot *et al.*, "Synthesis of spent fuel pool accident assessments using severe accident codes," p. 70.

49 Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 26.

50 Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix B, p. B-10.

approximately 677°C (1250°F) the fuel rods would start to balloon and burst,[51] "releasing noble gases, such as xenon and krypton," into the environment.[52] 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.

In a SFP LOCA or 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.[53] In the Paks-2 accident, 30 fuel assemblies were severely damaged and their fuel rods ballooned—"long sausage balloons"[54] with "very long ballooned areas."[55] At a 2003 Advisory 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"[56]) stated: "If you're trying to persuade me that

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

[52] 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.

[53] 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"[53] with "very long ballooned areas."[53] 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.

[54] 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.

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

[56] D. A. Powers, R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, April 1980, (ADAMS Accession No: ML053490337).

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."[57]

(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.[58])

In a SFP LOCA or 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 a SFP LOCA or 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 SFP LOCA or boil-off 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[59]] bundle configurations;"[60] 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]

[57] Advisory Committee on Reactor Safeguards Reactor Fuels Subcommittee, September 29, 2003, pp. 217-218.
[58] *Id*. pp. 113, 181, and 195.
[59] European Commission: Nuclear Safety and the Environment, "Fuel Cladding Failure Criteria: Final Report," EUR 19256 EN, September 1999, p. 88.
[60] 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.

~2-[meter] (80-[in]) length" of the fuel rods[61] (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)[62] (the CORA experiments, which simulated meltdown accidents, were conducted with zirconium alloy multi-rod bundles that were two meters long);[63] 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;"[64] and 5) the cladding balloons that occurred in the middle sections of the bundles from PWR FLECHT runs 2443 and 2544, which had unintended internal gas pressure increases, [65] 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."[66] 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

[61] 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.
[62] 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.
[63] 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.
[64] L. J. Ott, Siegfried Hagen, "Interpretation of the Results of the CORA-33 Dry Core Test," 1993.
[65] 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.
[66] 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.

experiencing the same conditions at burst"[67] [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."[68] It is plausible that these same phenomena would occur in a SFP LOCA or boil-off accident, because the fuel rods in the SFP would not have the pronounced *chopped-cosine axial heat flux distribution*[69] 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 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),[70] respectively.

Ballooning and bursting would also cause the fuel-cladding to lose the protection of preexisting oxide layers, as clean surface locations opened up, in *steam*, facilitating exothermic (heat-generating) oxidation and hydriding of zirconium.

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

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[67] *Id*., p. 17.
[68] *Id*., p. 5.
[69] 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.
[70] Allan S. Benjamin *et al*., Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

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."[71]

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*[72] [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.

(It is noteworthy that the NRC claims that in a SFP accident, "rod ballooning has a low impact on the timing to breakaway oxidation."[73])

IV.A.2. 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 Spent Fuel Pool Accident

It is doubtful that 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 LOCA or boil-off accident. If in fact this affect of local heavy oxide and/or crud layers is not simulated, it is an additional flaw of MELCOR, regarding its ability to model the air coolability of spent fuel assemblies. This is yet another reason to doubt NUREG-2161's claim that in the event of a *complete* SFP LOCA, the spent fuel

[71] 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.
[72] 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.
[73] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 26.

assemblies would be air coolable for 90 percent of a two year operating cycle. That is, the fuel assemblies would be air coolable for the last 657 days out of every two years (730 days); the fuel assemblies would *not* be air coolable for the first 73 days out of every two years.[74]

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);[75] there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick.[76]

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 LOCA or boil-off accident, in at least the following ways: 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),[77] 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.

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[74] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, Appendix B, p. B-10.
[75] 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.
[76] *Id.*, p. 29.
[77] Allan S. Benjamin *et al.*, Sandia Laboratories, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979, p. 47.

IV.B. Deficiencies of MELCOR, Regarding the Zirconium-Oxygen and Zirconium-Nitrogen Reactions in Air

IV.B.1. MELCOR Does Not Model the Exothermic Zirconium-Nitrogen Reaction

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 MELCOR, in 2006, a Sandia National Laboratory ("SNL") report observed that MELCOR *does not* model the nitriding of zirconium fuel cladding, stating that fuel cladding would "combine with nitrogen if no oxygen or steam are available" and that the nitriding process is exothermic (heat-generating).[78] 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"[79] [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 told 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*. The ACRS stated 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[80]) in which the NRC is a partner." And 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

[78] 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.

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

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

to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample"[81] [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,[82] which is approximately 30 percent of the 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.

IV.B.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.[83] This is a serious flaw because the presence of nitrogen accelerates the oxidation (burning) and degradation of zirconium fuel-cladding in *air*,[84] which would affect the progression and severity of a SFP accident, including radioactive releases, "most notabl[y] ruthenium."[85] ("Ruthenium has a biological effectiveness equivalent to

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[81] 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.

[82] 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.

[83] 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."

[84] 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.

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

that of Iodine-131;"[86] 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."[87] 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*.[88] (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.[89]

As quoted above, an *April 2000* ACRS letter states that "[m]uch more is known now about air interactions with [zirconium fuel] cladding."[90] 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.

86 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.

87 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.

88 J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 10.

89 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.

90 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.

> Further[more], it should be stressed that most of the existing data have been obtained with bare [non-oxidized] samples.[91]

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."[92]

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;" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[93]

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."[94] 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."[95]

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[91] 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.
[92] *Id*., p. 44.
[93] 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.
[94] J. Stuckert *et al*., "Results of the QUENCH-16 Bundle Experiment on Air Ingress," p. 1.
[95] 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.

Additionally, information on the Institut de Radioprotection et de Sûreté Nucléaire's website about the French Mozart Program for studying the zirconium-air reaction states: "Bibliographic 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*"[96] [emphasis added].

In NUREG-2161, 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 *isothermal*[97] 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."[98]

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

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[96] 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 3/17/16).

[97] The Argonne National Laboratory tests 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 Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 96.

[98] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, pp. 96-97.

protective oxide film to non-protective ‘breakaway’ oxidation at a significantly higher rate)”[99] [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[100] 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 commissioned experimental studies[101] [the ANL isothermal experiments] to obtain data with which to establish a credible physical basis for using the correlations. *More recent experiments*[102] *demonstrated that the processes that govern air oxidation are quite different from those which apply to steam oxidation*[103] [emphasis added].

Clearly, the NRC’s conclusions from its NUREG-2161, 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 MELCOR simulations, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[104]

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[99] 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. 2.

[100] 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. 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.

[101] 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).

[102] 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.

[103] 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.

[104] 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.

IV.B.3. The NRC's Recent Non-Conservative Post-Fukushima MELCOR Simulations

A recent NUREG-2161, Post-Fukushima MELCOR (version 1.8.6 of the code[105]) simulation of *a particular* BWR Mark I SFP fire scenario ("Unsuccessful Deployment of Mitigation for Moderate Leak (OCP3) Scenario"[106]) 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."[107] However, the same simulation also found that "[a]fter the peak temperature [is reached] at [Axial] Level 4, the peak temperature in the zirconium fire front decreases with each successive [axial] level. Radial heat transfer[108] 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"[109] [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[110]).

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105 The SFP models in MELCOR versions 1.8.6 and 2.1 are functionally the same. See Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, pp. 95-96.

106 Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 145.

107 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 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 Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 98, and p. 98, Note 12.

108 "MELCOR attempts to model a multidimensional geometry with a simplified two-surface radiation model." See Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 113, Note 23.

109 Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, pp. 145-146.

110 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

This NUREG-2161 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 NUREG-2161 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.[111] 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 would contribute additional heat and 2) the significant additional heat that would be contributed from the exothermic nitrogen-zirconium reaction.

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.

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

Recent SNL SFP accident experiments are unrealistic because they have been conducted with clean non-oxidized bundles of zirconium fuel rod simulators;[112] 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

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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.

[111] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161.

[112] 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).

to 100 μm thick. And medium to high burnup fuel cladding typically has a "hydrogen concentration in the range of 100-1000 wppm;" "[z]irconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen…"[113]

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."[114]

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 cladding is not simulated in SNL's experiments. Hence, data from SNL's SFP accident experiments is inappropriate 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.[115]

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

IV.C.1. Oxidation Models Are Not Able to Predict the Fuel-Cladding Temperature Escalation that Commenced at "Low Temperatures" in the PHEBUS B9R-2 Test

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.[116] A 1996 European

[113] 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.
[114] 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.
[115] 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.
[116] 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.

Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region (see Figure 1); the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[117]

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 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*[118] [emphasis added].

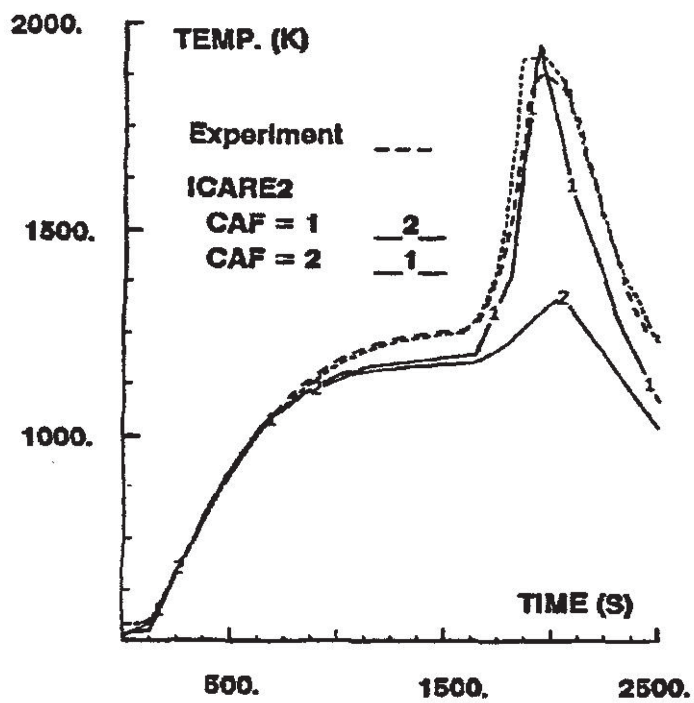

Figure 1. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[119]

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[117] T.J. Haste *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, 1996, p. 33.
[118] *Id.*, p. 126.

Today, in 2016, oxidation models still cannot accurately predict the local fuel-cladding temperature escalation that commenced in PHEBUS B9R-2 in *steam* when local fuel-cladding temperatures were 1027°C (1880°F). The PHEBUS B9R-2 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.

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to SFP fires. 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…"[120]

IV.C.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."[121]

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[119] 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.

[120] 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.

[121] 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.

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16.[122] 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 an NRC report, NUREG/CR-4412,[123] "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"[124] [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.[125]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[126] 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;"[127] the report

[122] 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.
[123] 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).
[124] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, (ADAMS Accession No: ML112211930), p. 3.
[125] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 27.
[126] *Id*., pp. 27, 30.
[127] *Id*., p. 30.

provides a graph of the "rather sparse"[128] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[129]

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 between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[130]

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." [131] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[132] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a 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.[133]

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

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[128] *Id*.
[129] *Id*., p. 29.
[130] *Id*., p. 30.
[131] *Id*.
[132] 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.
[133] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 30.

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 the 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 manner that provides less efficient energy transport from the fuel to the cladding,"[134] 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.

IV.C.3. "Low Temperature" Oxidation Rates Are Under-Predicted for FLECHT Run 9573

Westinghouse's "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" (hereinafter: "WCAP-7665") states that, "[t]he objective of the PWR FLECHT…test program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[135] The FLECHT tests were conducted with bundles of heater rods sheathed in zirconium alloy (Zircaloy) cladding.

[134] 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.

[135] 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. 1.1.

Runaway oxidation was not expected to occur in any of the tests; however, the FLECHT Run 9573 test bundle incurred runaway oxidation (see Figure 2).

Figure 2. Section of the FLECHT Run 9573 Test Bundle that Incurred Runaway Oxidation

The FLECHT Run 9573 test bundle incurred runaway oxidation around its seven foot elevation. WCAP-7665 states: "Post-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation. The heater rod failures were apparently caused by localized temperatures in excess of 2500°F." WCAP-7665 also states: "During the test, heater element failures started at 18.2 seconds... At the time of the initial failures, midplane [at the 6 foot elevation] clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure)."[136]

The NRC conducted TRACE code computer simulations of FLECHT Run 9573 and found that TRACE *under-predicted* temperatures that were reported by Westinghouse at the 7 ft elevation of the test bundle. On November 24, 2015, Aby

[136] *Id*., p. 3.97.

Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, sent Commenter an e-mail regarding the NRC's TRACE computer simulation of FLECHT Run 9573. In his e-mail, Mr. Mohseni disclosed findings of "the completed simulation [for] the cladding and steam temperatures at the 7-ft elevation (at 18 seconds)."[137]

TRACE *under-predicted* cladding and steam temperatures at the 7-foot elevation of the FLECHT Run 9573 test bundle. TRACE is supposed to *over-predict* temperatures in order to ensure an adequate margin of safety. The Baker-Just and Cathcart-Pawel zirconium-steam reaction correlations were used for the TRACE simulations. The TRACE simulations need to be considered as evidence that the NRC and nuclear industry's computer safety models under-predict the zirconium-steam reaction rates that would occur in the event of a SFP accident.

IV.C.3.a. FLECHT Run 9573—a Comparison between the TRACE Predictions and the Results Westinghouse Reported

According to Mr. Mohseni's e-mail, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *cladding* temperatures of 1526 K (2287°F) and 1561 K (2350°F), respectively. And, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *steam* temperatures of 1370 K (2006°F) and 1397 K (2055°F), respectively. Those are predicted cladding and steam temperatures for the FLECHT Run 9573 test bundle at the 7-ft elevation, at 18 seconds.[138]

Westinghouse reported that at 18.2 seconds, heater rod failures occurred around the 7 foot elevation when *cladding* temperatures were in excess of 1644 K (2500°F). (Who knows how high the cladding temperatures actually were; they could have been hundreds of degrees Fahrenheit higher than 1644 K (2500°F).)

And Westinghouse reported that at 16.0 seconds, a steam probe at the 7 foot elevation recorded *steam* temperatures that exceeded 1644 K (2500°F). And a Westinghouse memorandum stated that after 12 seconds, the steam-probe thermocouple

[137] Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, e-mail to Mark Leyse, regarding the NRC's TRACE computer simulation of the FLECHT Run 9573 test bundle, November 24, 2015, (ADAMS Accession No: ML15341A160).
[138] *Id*.

recorded "an extremely rapid rate of temperature rise (over 300°F/sec)."[139] (Who knows how high the steam temperatures actually were at 18 seconds; they were likely hundreds of degrees Fahrenheit higher than 1644 K (2500°F).)

Taking the time difference of 0.2 seconds (between 18 and 18.2 seconds) into account, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *cladding* temperatures that were at least 200°F and 140°F lower, respectively, than the temperatures Westinghouse reported. That is *non-conservative*.

When TRACE used the Cathcart-Pawel and Baker-Just correlations, at 18 seconds it predicted *steam* temperatures that were about 500°F and 450°F lower, respectively, than the temperatures Westinghouse measured at 16 seconds. Westinghouse also reported that after 12 seconds, steam temperatures were increasing at a rate greater than 300°F/sec. So steam temperatures were even greater at 18 seconds than they were at 16 seconds. Hence, the TRACE predictions for steam temperatures are *non-conservative*.

The FLECHT Run 9573 results indicate that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are inadequate for use in computer safety models like MELCOR.

V. MELCOR Does Not Simulate Criticality Accidents in Certain Spent Fuel Pool Accident Scenarios

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 installed neutron absorbers to maintain criticality requirements."[140] (Discharged and fresh fuel assemblies are also placed into particular arrangements in high-density storage racks in order to help control reactivity in the SFP. In a SFP, fuel assemblies might be arranged within checkerboard configurations.)

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[139] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970. This Memorandum is available at Appendix I of PRM-50-93. See Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No: ML093290250), Appendix I.

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

In the event of a power outage that would last months to years, a nuclear power plant could have a station-blackout—a complete loss of AC power. In the event of a station-blackout, the SFP could boil off enough water to partly or completely expose the fuel assemblies to air. The fuel assemblies could also be exposed to air for a significant period of time in the event of a SFP LOCA. 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).[141] And Boral melts at approximately 657°C (1214°F).[142]

As stated above in Sections IV.A, IV.B, and IV.C, the NUREG-2161 MELCOR analyses that the NRC conducted to help justify its decision to not expedite the transfer of spent fuel assemblies from SFPs to dry cask storage are seriously flawed. The NRC's MELCOR analyses are additionally flawed because they did not model criticality accidents.[143]

NUREG-2161 provides the results of a number of MELCOR analyses of SFP loss-of-coolant accidents in which there was a *moderate leakage rate*.[144] In some of the NRC's simulations, SFP temperatures reached the point at which neutron-absorber materials would melt. Then spray cooling was simulated in the same MELCOR analyses. The SFP was, at least partly, refilled with water.[145] However, the MELCOR analyses did

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[141] Electric Power Research Institute, "Severe Accident Management Guidance Technical Basis Report," Volume 2: "The Physics of Accident Progression," 1025295, October 2012, Appendix EE, p. EE-9.

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

[143] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 20.

[144] 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 Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 61.

[145] Andrew Barto *et al.*, NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, pp. 216-218.

not even consider the possibility of criticality accidents.[146] In the real world, there could be criticality accidents in such SFP accident scenarios.

The MELCOR analyses of NUREG-2161 did not model criticality accidents; however, the very same NRC report warns about criticality accidents.

Regarding criticality accidents, which NUREG-2161 terms "inadvertent criticality events," the report states:

> Inadvertent criticality events (ICEs) may be possible for specific combinations of conditions (*e.g.*, *during reflood of a drained pool* for a region of the pool storing higher reactivity fuel assemblies where the boron poison in the rack panels has been significantly displaced as a result of the earthquake). If such an event affected a region of the pool (as opposed to only a portion of a particular assembly), and if it occurred at a point in the accident where the fuel was only partially covered, the event could have an important impact on onsite dose rates. Further, if an ICE were severe enough to produce significant heat, the fuel will be harder to cool and short-lived radionuclides will be produced[147] [emphasis added].

Hence, the NRC believes that criticality accidents, or inadvertent criticality events, could play a significant role in a SFP accident in cases in which neutron-absorber materials would become ineffective. If a critically accident were to occur, local fuel and fuel-cladding temperatures would rapidly increase. A critically accident would also "cause an increase in decay products, which [would] have a delayed effect on temperature increase[s]."[148] Increased onsite dose rates, caused by a criticality accident, would impede (or possibly prevent for significant time periods) efforts to mitigate a SFP accident.

NUREG-2161 cautions that "[t]he possibility of a criticality event cannot be summarily dismissed."[149] Furthermore, the NRC has a regulation pertaining to preventing criticality accidents in the event that a SFP would be partly or completely refilled with either unborated or borated water—10 C.F.R. § 50.68 Criticality Accident

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[146] *Id*., p. 20.
[147] *Id*., p. 29.
[148] Zachary I. Franiewski *et al*., "Spent Fuel Pool Analysis of a BWR-4 Fuel Bundle Under Loss of Coolant Conditions Using TRACE," pp. 1-2.
[149] Andrew Barto *et al*., NRC, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, p. 30.

Requirements. (Borated water would absorb neutrons and help prevent criticality accidents.)

Regarding refilling a SFP with unborated water, 10 C.F.R. § 50.68(b)(4) states:

> If no credit for soluble boron is taken, the k-effective [the estimated ratio of neutron production to neutron absorption and leakage] of the spent fuel storage racks loaded with fuel of the maximum fuel assembly *reactivity must not exceed 0.95* [below 1.0 is subcritical], at a 95 percent probability, 95 percent confidence level, if flooded with unborated water [emphasis added].

Nonetheless, the NRC has overlooked how SFP criticality "events" could make a SFP accident far worse: the NUREG-2161 MELCOR analyses that the NRC conducted to help justify its decision to not expedite the transfer of spent fuel assemblies did not consider criticality accidents.

(It is noteworthy that the NRC's recent (November 2015) proposed requirements for 10 C.F.R. § 50.155(b)(1) do not have provisions for preventing criticality accidents in cases in which water would be sprayed into a SFP (the refilling of a drained SFP).[150])

In the event of a SFP accident, in some scenarios, a SFP could be partly or completely refilled with unborated water. In the Fukushima Dai-ichi accident, SFPs were refilled (at least to some degree) with seawater;[151] and reactors cores were injected with both unborated and borated seawater.[152]

VI. Conclusion

In these comments Commenter has provided evidence that the NRC's MELCOR analyses of SFP accidents and fires are not based on sound science. MELCOR analyses under-predict the severity and radiological releases of SFP fires. The NRC cannot plan for the first and second tiers of decommissioning—in which there would be irradiated fuel in the

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[150] NRC, "10 CFR Parts 50 and 52: Mitigation of Beyond-Design-Basis Events," Proposed Rules, Docket Nos. PRM–50–97 and PRM–50–98; NRC–2011–0189 and NRC-2014-0240, Federal Register, Vol. 80, No. 219, November 13, 2015.

[151] Tokyo Electric Power Company March 25, 2011 Press Release, "Plant Status of Fukushima Daiichi Nuclear Power Station (as of 10:30 PM Mar 25th)," available at http://www.tepco.co.jp/en/press/corp-com/release/11032515-e.html (last visited February 11, 2016).

[152] Institute of Nuclear Power Operations, "Special Report on the Nuclear Accident at the Fukushima Dai-ichi Nuclear Power Station," INPO 11-005, November 2011, pp. 9-10, 21, 28, 79, 80.

SFP—until it conducts analyses of SFP accidents and fires that are based on sound science.

In accordance with its philosophy of defense-in-depth, which requires the application of conservative models,[153] the NRC must conduct safety analyses of SFP accidents and fires that are based on sound science.

On the basis of its non-conservative MELCOR analyses, the NRC has already decided to not require licensees to expedite the transfer of spent fuel assemblies from SFPs to dry cask storage. The NRC must disregard the results of its flawed MELCOR analyses. Many questions pertinent to the storage of spent fuel assemblies—including emergency-preparedness questions for decommissioning—cannot be answered until safety analyses of SFP accidents and fires are based on sound science.

Respectfully submitted,

/s/

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[153] 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.