PRM-50-93 & 50-95
(75FR66007)

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From: Mark Leyse
To: RulemakingComments Resource; PDR Resource; Inverso, Tara; Bladey, Cindy; Clifford, Paul; Dudley, Richard; Doyle, Daniel
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Subject: NRC-2009-0554 (Eighth)
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Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Leyse's, Petitioner's, eighth response, dated April 12, 2014, to the NRC's notice of solicitation of public comments on PRM-50-93 and PRM-50-95, NRC-2009-0554, published in the Federal Register on October 27, 2010.

Sincerely,

Mark Leyse

April 12, 2014

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

Attention: Rulemakings and Adjudications Staff

Comments on

Nuclear Regulatory Commission's Draft Interim Reviews of

Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554

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Mark Edward Leyse's Comments on Nuclear Regulatory Commission's Draft Interim Reviews of Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554

In these comments, Mark Edward Leyse ("Petitioner") comments on the U.S. Nuclear Regulatory Commission's ("NRC") Draft Interim Reviews ("DIR") of two petitions for rulemaking: PRM-50-93[1] and PRM-50-95[2] ("PRM-50-93/95"). Petitioner highlights some of the pertinent information, submitted by Petitioner in PRM-50-93/95 and in public comments on PRM-50-93/95, which NRC did not consider in its DIRs. Problems with NRC's TRACE simulations of FLECHT run 9573 are also discussed.

I. NRC has Overlooked Specific Data Cited by Petitioner from Experiments in which Runaway Oxidation Commenced at Temperatures Lower than the 10 C.F.R. § 50.46(b)(1) 2200°F Peak Fuel-Cladding Temperature Limit

> *The heat evolved from the zircaloy-[steam] reaction at temperatures above 2000°F is significant and produces an autocatalytic effect.*[3]— General Electric, 1959

Regarding the 2200°F 10 C.F.R. § 50.46(b)(1) peak fuel-cladding temperature ("PCT") limit, in NRC's October 2012 DIR of PRM-50-93/95, NRC concludes:

> [A]utocatalytic reactions have not occurred at temperatures less than 2200 degrees F. Accordingly, the 2200 degree F regulatory limit is sufficient provided the correlations used to determine the metal-water reaction rate below 2200 degrees F are suitably conservative such that excessive reaction rates do not occur below that value.[4]

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[1] Mark Leyse, PRM-50-93, November 17, 2009, available at: NRC's ADAMS Documents, Accession Number: ML093290250.

[2] Mark Leyse, PRM-50-95, June 7, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101610121. (PRM-50-95 was originally a 10 C.F.R. § 2.206 enforcement action petition that Petitioner wrote on behalf of New England Coalition ("NEC"), dated June 7, 2010. In October 2010, NRC published a notice in the Federal Register stating that it had determined the NEC petition met the requirements for a petition for rulemaking under 10 C.F.R. § 2.802.)

[3] J. I. Owens, R. W. Lockhart, D.R. Iltis, K. Hikido, General Electric Company, "Metal-Water Reactions: VIII. Preliminary Consideration of the Effects of a Zircaloy-Water Reaction during a Loss-of-Coolant Accident in a Nuclear Reactor," GEAP-3279, September 30, 1959, p. 34.

[4] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12265A277, p. 2.

In PRM-50-93/95 and in comments on PRM-50-93/95, Petitioner submitted information stating that runaway (autocatalytic) zirconium-steam reactions ("runaway oxidation") *have* commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit. For example, PRM-50-93 (pages 46-47) quotes an OECD Nuclear Energy Agency report, which states that runaway oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater.[5] The NRC's October 2012 DIR of PRM-50-93/95 fails to respond to or even acknowledge the existence of this information.

In its October 2012 DIR of PRM-50-93/95, NRC neither acknowledges nor discusses the fact that Dr. Robert E. Henry, in presentation slides from "TMI-2: A Textbook in Severe Accident Management," postulated that in the Three Mile Island Unit 2 ("TMI-2") accident, the heat produced by the exothermic zirconium-steam reaction caused thermal runaway to commence in the reactor core when fuel-cladding temperatures reached approximately 1000°C (1832°F).[6] Dr. Henry's postulation is discussed in Petitioner's comments on PRM-50-93/95, dated November 23, 2010, (pages 11-14).[7]

Interestingly, a March 2002 NRC document, "Perspectives on Reactor Safety," states that in a postulated station blackout scenario at Grand Gulf, runaway zirconium oxidation would commence at 1832°F.[8] (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

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[5] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," Executive Summary, February 2000, p. 9. (Regarding the statement that runaway (autocatalytic) oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater, "Degraded Core Quench: Summary of Progress 1996-1999" explicitly states that "[a] notable feature of the [QUENCH] experiments was the occurrence of temperature excursions starting in the unheated region at the top of the shroud, from temperatures of 750-800°C, which is more than 300 K lower than excursion temperatures associated with runaway oxidation by steam.")

[6] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007, seven of these presentation slides are in attachment 2 of the transcript from "10 C.F.R. 2.206 Petition Review Board Re: Vermont Yankee Nuclear Power Station", July 26, 2010, available at: NRC's ADAMS Documents, Accession Number: ML102140405, Attachment 2.

[7] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.

[8] NRC, "Perspectives on Reactor Safety," NUREG/CR-6042, Rev. 2, March 2002, available at: NRC's ADAMS Documents, Accession Number: ML021080117, pp. 3.7-4, 3.7-5, 3.7-29.

Furthermore, in NRC's own September 2011 DIR of PRM-50-93/95, NRC presented data demonstrating that runaway oxidation commenced in the LOFT LP-FP-2 experiment when fuel-cladding temperatures were lower than 2200°F. (In PRM-50-93 (pages 27, 33, 41, 42), Petitioner quoted a Pacific Northwest Laboratory paper, which states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec [18°F/sec], signal[s] the onset of an autocatalytic oxidation reaction."[9] This is for cases in which there would be relatively low initial heatup rates—for example, 1.0 K/sec (1.8°F/sec)—followed by substantially higher heatup rates, caused by the contribution of heat generated by the exothermic zirconium-steam reaction.) In NRC's September 2011 DIR of PRM-50-93/95, NRC presented data stating that in LOFT LP-FP-2, when local temperatures reached 1477 K (2199.2°F), just under the regulatory limit, the heatup rates at two fuel-cladding locations (TE-5C07-042 and TE-5D13-042) were already 10.3 K/sec (18.5°F/sec) and 11.9 K/sec (21.4°F/sec), respectively.[10]

Hence, NRC's October 2012 DIR of PRM-50-93/95 overlooks data that NRC itself provided in September 2011 demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F PCT limit. Clearly, NRC needs to correct, and explore the safety implications of its erroneous conclusion that runaway oxidation has not commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

It is noteworthy that a report regarding best-estimate predictions for LOFT LP-FP-2 states that runaway oxidation would commence if fuel-cladding temperatures were to start increasing at a rate of 3.0 K/sec (5.4°F/sec);[11] this is for cases in which there would be relatively low initial heatup rates. (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

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[9] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, available at: NRC's ADAMS Documents, Accession Number: ML042230126, p. 282.

[10] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," September 2011, available at: NRC's ADAMS Documents, Accession Number: ML112650009, p. 4.

[11] S. Guntay, M. Carboneau, Y. Anoda, "Best Estimate Prediction for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3803, June 1985, available at: NRC's ADAMS Documents, Accession Number: ML071940361, p. 38.

NRC's September 2011 DIR of PRM-50-93/95 failed to report that in LOFT LP-FP-2, at one location, due to the rapid Zircaloy-steam reaction on a Zircaloy guide tube, the temperature increased from 1400 K to 1800 K (2060.6°F to 2780.6°F) in 21 seconds.[12] The September 2011 DIR of PRM-50-93/95 also failed to note the heatup rate at the Zircaloy guide tube location (TE-5H08-027) when temperatures reached 1477 K (2199.2°F)—most likely the heatup rate exceeded 10 K/sec. At that location (TE-5H08-027), the *average* heatup rate was 19 K/sec (approximately 34.3°F/sec) from 1400 K to 1800 K (2060.6°F to 2780.6°F) over a period of 21 seconds.

The NRC's September 2011 DIR of PRM-50-93/95, states that a report, "Quick Look Report on OECD LOFT Experiment LP-FP-2," concluded that "rapid oxidation of zircaloy started at approximately 1480 seconds" and that "thermocouples [temperature measuring devices] at the 42-inch elevation confirms this, as the[ir measurements] exceed[ed] 1477 K (2200°F) by 1460 seconds."[13] NRC is incorrect: the report actually states that "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between zircaloy and water occurred at about 1430 [seconds] and 1400 K [2060°F];"[14] furthermore, the report states that *recorded* temperatures on a Zircaloy guide tube reached 1800 K (2780.6°F) at 1451 seconds and that *recorded* temperatures on fuel cladding reached 1800 K (2780.6°F) at 1475 seconds.[15]

The "Quick Look Report"" also states:

> The first recorded (and qualified) rapid temperature rise caused by the exothermic reaction between the steam and the zircaloy is at about 1430 s[econds] on guide tube thermocouple TE-5H08-027. (Thermocouple TE-5EIl-027 was judged to have failed at 1311 s[econds], but the mode of failure suggests that temperatures reached 1800 K (2780°F) at some location in the core by 1381 s[econds].) The rapid temperature rise began from approximately 1400 K (2060°F).[16]

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[12] Adams, J. P., *et al*., "Quick Look Report on OECD LOFT Experiment LP-FP-2," OECD LOFT-T-3804, September 1985, available at: NRC's ADAMS Documents, Accession Number: ML071940358, pp. 30, E-4, E-8.

[13] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," p. 4.

[14] Adams, J. P., *et al*., "Quick Look Report on OECD LOFT Experiment LP-FP-2," p. 30.

[15] *Id*., p. E-8.

[16] *Id*., p. E-4.

In PRM-50-93 (page 39), Petitioner quoted a report that stated that "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation."[17] And Petitioner, in PRM-50-93 (page 40), quoted the same report, which stated that "[i]t can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400°K (2060°F)."[18] NRC overlooked the fact that the very same sentence is on page 30 of the report it referenced: "Quick Look Report on OECD LOFT Experiment LP-FP-2.")

LOFT LP-FP-2 combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water;[19] and "[t]he [LOFT LP-FP-2] experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[20]

(See Appendix A for information about the BWR FLECHT Zr2K test and Thermal Hydraulic 1 test 130: design basis accident experiments in which runaway oxidation (most likely) commenced and almost commenced, respectively, at fuel-cladding temperatures that were lower than the 2200°F PCT limit. Although neither mentioned in PRM-50-93/95 nor in comments on PRM-50-93/95, the PHEBUS B9R-2 test is also discussed.)

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[17] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, available at: NRC's ADAMS Documents, Accession Number: ML062840091, p. 30.

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

[19] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.

[20] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 3. 23.

I.A. NRC Overlooked an Experiment in which Runaway Oxidation either Commenced at a Temperature Lower than the 2200°F PCT Limit or at a Temperature Not High Enough above 2200°F to Provide a Necessary Margin of Safety

NRC's October 2012 DIR of PRM-50-93/95 falsely claims that Petitioner omitted "some important information from the "Compendium of ECCS Research for Realistic LOCA Analysis," [which] discusses conservatism in the regulatory criteria, and provides some justification."[21]

The October 2012 DIR of PRM-50-93/95 quotes the "important information" from "Compendium of ECCS Research for Realistic LOCA Analysis":

> The MT-6B test conducted in June 1984 showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant. In the FLHT-test, completed in March 1985, 12 ruptured zircaloy clad rods were subjected to an autocatalytic temperature excursion. From the measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500 – 2600°F (1371 – 1427°C) temperature region.[22]

The first sentence from the quote above, regarding the MT-6B test (Materials Test 6B) was already quoted in PRM-50-93 (pages 31, 35). And PRM-50-93 discussed the MT-6B test (pages 30-31, 35). One of the things that PRM-50-93 points out is that three publications report *different* peak fuel-cladding temperature values for the MT-6B test: the PCT was reported variously as 2060°F (1400 K),[23] 2200°F (1477 K),[24] and 2336°F (1553 K).[25]

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[21] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 2.

[22] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 2; the source of this quote is NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, available at: NRC's ADAMS Documents, Accession Number: ML053490333, p. 8-2.

[23] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, p. viii.

[24] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

[25] G. M. Hesson, *et al.,* Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," 1993, p. 2.

The second and third sentences from the quote above, regarding the FLHT-test (actually the FLHT-1 test: Full-Length High-Temperature Severe Fuel Damage Test 1) were also already quoted in PRM-50-93 (page 37). And PRM-50-93 discusses the FLHT-1 test (pages 31-38); and Appendix E of PRM-50-93 has graphs depicting cladding temperature values for the maximum temperature region of the FLHT-1 test fuel assembly; the FLHT-1 test is also discussed in Petitioner's comments on PRM-50-93/95, dated December 27, 2010, (pages 31-36).[26] PRM-50-93 already highlighted that it is highly likely that in the FLHT-1 test, runaway oxidation commenced at cladding temperatures of approximately 1520°K (2277°F) or lower. Even if it were determined that runaway oxidation commenced at 77°F above NRC's 2200°F PCT limit, this would indicate that the 2200°F PCT limit is non-conservative, because the limit would not provide a necessary margin of safety in the event of a loss-of-coolant accident ("LOCA").

In PRM-50-93 (pages 34-35), Petitioner explains why he believes that in the FLHT-1 test, the cladding temperature excursion began at a temperature of approximately 1520°K (2277°F) or lower.

In PRM-50-93 (page 34), a quote is provided that describes the procedure the conductors of the FLHT-1 test followed. Regarding the test procedure, "Full-Length High-Temperature Severe Fuel Damage Test 1" states:

> When the temperature reached about 1475°K (2200°F), the bundle coolant flow [rate] was again increased to stop the temperature ramp. This led to a stabilized condition. The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly until the test director requested that the reactor power be reduced to zero power.[27]

PRM-50-93 argues (pages 34-35) that it is obvious from the description in the quote above and from the cladding-temperature plots provided in Appendix E of PRM-50-93 that when cladding temperatures reached approximately 1475°K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was *not* achieved. (The slopes of the lines of the cladding-temperature value plots of the FLHT-1 test

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[26] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

[27] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.6.

become nearly vertical, after the cladding-temperature values reach approximately 1520°K (2277°F), indicating that only a short time period passed before temperatures increased to approximately 2275°K (3636°F).) In fact, cladding temperatures continued to increase. This is clearly stated in the quote above, which states that increased "flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly…"[28]

Clearly, the conductors of the FLHT-1 test *could not terminate* the cladding-temperature increase after peak cladding temperatures reached approximately 1475°K (2200°F); they increased the coolant flow rates yet still could not prevent the runaway zirconium-steam reaction from commencing. Peak cladding temperatures increased from approximately 1520°K (2277°F) or lower to approximately 2275°K (3636°F), within approximately 85 seconds.[29]

It is unfortunate that NRC overlooked the information provided in PRM-50-93 on the FLHT-1 test and did not review the FLHT-1 test.

II. NRC Has Not Considered the Problems with the Metallurgical Data from the Four Zircaloy PWR-FLECHT Experiments

Regarding the metallurgical data from the four Zircaloy PWR-FLECHT experiments, in NRC's October 2012 DIR of PRM-50-93/95, NRC states:

> Furthermore, while PRM-50-93 takes issue and disagrees with parts of the NRC's evaluation of petition PRM-50-76, it fails to consider that in the NRC evaluation there were calculations of oxygen uptake and ZrO_2 thickness for the four FLECHT Zircaloy experiments (Cadek *et al*., 1971). The calculations showed Cathcart-Pawel to be best-estimate and Baker-Just to be conservative.[30]

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[28] *Id.*
[29] *Id.*, pp. v, 4.6.
[30] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 6.

When NRC performed its technical safety analysis of PRM-50-76,[31] NRC was evidently unaware of the serious problems with the metallurgical data that Westinghouse took and analyzed from the four FLECHT Zircaloy experiments.

In NRC's October 2012 DIR of PRM-50-93/95, NRC overlooked *new information*—not discussed in PRM-50-76—that Petitioner provided in PRM-50-93 (pages 49-50) and in comments on PRM-50-93/95, dated November 23, 2010 (pages 45-47),[32] dated March 15, 2010 (pages 32-34),[33] dated April 7, 2011 (pages 7-9),[34] which indicates Westinghouse's metallurgical data from Zircaloy PWR FLECHT run 9573 is invalid. And in comments on PRM-50-93/95, dated July 30, 2011 (page 18),[35] Petitioner provided new information indicating that the metallurgical data from Zircaloy PWR FLECHT run 8874 is also invalid; see Section II.A.

Appendixes A and B of PRM-50-93 have photographs of the sections of the test bundles from FLECHT runs 9573 and 8874 that incurred runaway oxidation, respectively.

Furthermore, although neither discussed in PRM-50-93 nor in comments on PRM-50-93/95, there are also significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from Zircaloy PWR FLECHT runs 2443 and 2544; see Section II.B.

II.A. NRC Overlooked Problems with the Metallurgical Data from FLECHT Runs 8874 and 9573

In PRM-50-93 and in comments on PRM-50-93/95, Petitioner *emphasized* that there are significant problems with Westinghouse's examinations of the metallographic cross-

[31] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," April 29, 2004, available at: NRC's ADAMS Documents, Accession Number: ML041210109.
[32] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[33] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[34] Mark Leyse, Comments on PRM-50-93/95, April 7, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111020046.
[35] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

sections that were taken from test rods from Zircaloy PWR FLECHT run 9573, because Westinghouse did not obtain metallurgical data from the locations of the rods from run 9573 that incurred runaway oxidation.[36] Then, in comments on PRM-50-93/95, Petitioner stated that Zircaloy PWR FLECHT run 8874 had also incurred runaway oxidation and that Westinghouse did not obtain metallurgical data from the locations of the rods from run 8874 that incurred runaway oxidation. It is probable that the locations of the test bundles from runs 8874 and 9573 that Westinghouse did examine were steam starved: the examined locations had limited oxidation because they had been exposed to a limited amount of steam.

It is reasonable to assume that—as in CORA-2, in which local steam starvation conditions are postulated to have occurred[37]—in FLECHT runs 8874 and 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, Westinghouse's application of the Baker-Just zirconium-steam correlation (used in computer safety models) to the oxide layers on the test bundles from FLECHT runs 8874 and 9573 were to locations that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, that is not a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models.

Subsequently, NRC applied the Baker-Just and Cathcart-Pawel correlations to the metallurgical data from the four FLECHT Zircaloy experiments:[38] unfortunately, NRC did not apply the Baker-Just and Cathcart-Pawel correlations to metallurgical data from the locations of FLECHT runs 8874 and 9573 that incurred runaway oxidation. Hence,

[36] Runaway oxidation was not expected to occur in any of Westinghouse's PWR FLECHT tests. "PWR FLECHT Final Report" does not mention that the bundles from PWR FLECHT runs 8874 and 9573 incurred runaway oxidation.

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

[38] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, available at: NRC's ADAMS Documents, Accession Number: ML050250359, pp. 21-22.

NRC's analyses are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

It is unfortunate that NRC has overlooked the information Petitioner provided which indicates that Westinghouse's metallurgical data from FLECHT runs 8874 and 9573 is invalid.

(See Appendixes B and C for photographs of the sections of the test bundles from FLECHT runs 9573 and 8874 that incurred runaway oxidation, respectively.)

II.B. Problems with the Metallurgical Data from FLECHT Runs 2443 and 2544

Although neither discussed in PRM-50-93/95 nor in comments on PRM-50-93/95, there are also significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from Zircaloy PWR FLECHT runs 2443 and 2544.

A Westinghouse report states that two of the PWR FLECHT experiments—runs 2443 and 2544—with Zircaloy test bundles had unintended internal gas pressure increases, at the middle sections of the bundles, which caused the Zircaloy cladding to balloon and move away from the heat source of the internally heated rods and from the location of the thermocouples.[39] The actual temperatures of the Zircaloy cladding of the test bundles at the middle section were lower than the temperatures Westinghouse recorded. Therefore, the quantity of oxidation which occurred at the middle sections of the test bundles from FLECHT runs 2443 and 2544, occurred at lower temperatures than Westinghouse claimed.

Westinghouse would have accurately measured the thickness of each oxide layer; however, Westinghouse concluded that the thicknesses of the oxide layers from the middle sections of the test bundles from FLECHT runs 2443 and 2544 had been produced at higher temperatures than they were actually produced at. Hence, the metallurgical data was erroneously associated with cladding temperatures that were too high. Clearly, Westinghouse's metallurgical data from FLECHT runs 2443 and 2544 is not valid for

[39] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, available at: NRC's ADAMS Documents, Accession Number: ML070780083, p. 3-95.

performing a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models. NRC's subsequent analyses—which used data from FLECHT runs 2443 and 2544—are also not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

(Interestingly, in Westinghouse's comparison of eight metallurgical samples from run 2443, taken from two feet above and below the midplane location, *all* of the measured oxide thicknesses *exceeded* the predicted oxide thicknesses.[40])

III. NRC's TRACE Simulations of FLECHT Run 9573 Are Invalid because They Did Not Simulate the Section of the Test Bundle that Incurred Runaway Oxidation

In NRC's October 2012 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573 that it performed.[41] NRC provides results of its TRACE simulations for the 2, 4, 6, 8, and 10-foot elevations of the FLECHT run 9573 test bundle, which were the elevations where thermocouples were located on the bundle.[42]

Unfortunately, in FLECHT run 9573 there were no thermocouples located at the section of the test bundle which incurred runaway oxidation—around the 7 ft elevation. (There was a steam probe thermocouple located at the 7-foot elevation.[43]) Hence, NRC's TRACE simulations of FLECHT run 9573 did not include the section of the test bundle that incurred runaway oxidation.

As already stated in PRM-50-93 (pages 59, 60), Westinghouse reported, regarding the FLECHT run 9573 bundle, that a "[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[44] (See Figure 1.) And, as previously stated in PRM-50-93 (page 60), Westinghouse reported that "[t]he remainder of the [FLECHT run 9573] bundle was in

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[40] In all eight cases measured oxide thicknesses were less than 0.1 x 10^{-3} inches thick; however, all the predicted thicknesses were zero inches. See F. D. Kingsbury, J. F. Mellor, A. P. Suda, Westinghouse Electric Corporation, Appendix B, "Materials Evaluation," of "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. B-9.

[41] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," pp. 7-8.

[42] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 2-10.

[43] *Id.*, p. 2-13.

[44] *Id.*, p. 3-97.

excellent condition." [45] (Appendix A of PRM-50-93 has photographs of the "locally severe damage zone," which incurred runaway oxidation, of the test bundle from FLECHT run 9573.)

Figure 1. Section of the Test Bundle from PWR FLECHT Run 9573 that Incurred Runaway Oxidation

As stated in Section II.A, it is reasonable to assume that—as in CORA-2, in which local steam starvation conditions are postulated to have occurred[46]—in FLECHT run 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, NRC's TRACE simulations of FLECHT run 9573, using the Baker-Just and Cathcart-Pawel correlations, encompassed locations—the 2, 4, 6, 8, and 10-foot elevations of the test bundle—that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Petitioner contends on the basis of this evidence that NRC's TRACE

[45] *Id.*

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

simulations are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

(See Appendix B for photographs of the section of the test bundle from FLECHT run 9573 that incurred runaway oxidation.)

III.A. NRC's TRACE Simulations of FLECHT Run 9573 Did Not Include Data Taken from the Seven-Foot Elevation of the Test Bundle

The highest predicted temperature in NRC's TRACE simulations of FLECHT run 9573 was 1598.4 K (2417.7°F) at the 6-foot elevation, *at 18 seconds* after flooding commenced: predicted by the TRACE simulation using the Baker-Just correlation.[47] As stated in PRM-50-93 (pages 10-11, 59, 63), Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573.[48] And, as stated in PRM-50-93 (pages 59-60, 60-61), Westinghouse reported that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[49] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F higher than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F higher than the highest temperature predicted using the Cathcart-Pawel correlation.

In NRC's October 2012 DIR of PRM-50-93/95, NRC states that "it should be noted that over the first 18 seconds of FLECHT run 9573, the heatup rate was below the 15 K/sec that is considered in the petition to be an indication of an "autocatalytic reaction" rate.[50] In fact, as stated in Section I, PRM-50-93 quotes a paper stating that "a rapid [cladding] temperature escalation, [greater than] *10 K/sec* [18°F/sec], signal[s] the

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[47] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.
[48] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[49] *Id*.
[50] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.

onset of an autocatalytic oxidation reaction"[51] [emphasis added]. (This is for cases in which there would be relatively low initial heatup rates—for example, 1.0 K/sec (1.8°F/sec)—followed by substantially higher heatup rates, caused by the contribution of heat generated by the exothermic zirconium-steam reaction.) The NRC staff response misrepresents a statement made in the petition.

Regarding the heatup rates, NRC states:

> At the elevations where cladding oxidation was significant ([4, 6, and 8 feet]), both the Cathcart-Pawel and the Baker-Just correlations resulted in an over-prediction of the measured heatup rate. Heatup rates with the Baker-Just correlation were greater than those obtained with the Cathcart-Pawel correlation, and were significantly greater than the heatup rates observed in the experimental data. At the peak power elevation ([6 feet]), the heatup rate using the Baker-Just correlation exceeded the experimental value by 41 percent.[52]

As already stated in PRM-50-93 (pages 66-67), Westinghouse reported, regarding the FLECHT run 9573 test bundle that "[t]he steam probe thermocouple located one foot above midplane [at the 7-foot elevation] in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding."[53] (Appendix I of PRM-50-93 is a Westinghouse memorandum, dated December 14, 1970, reporting that the steam heatup rate exceeded 300°F/sec, at the 7-foot elevation.)

Hence, there is yet another reason why NRC's TRACE simulations FLECHT run 9573 were not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models. NRC's TRACE simulations did not include data taken from the 7-foot elevation of the FLECHT run 9573 test bundle, where a steam probe thermocouple measured steam temperature heatup rates that exceeded 300°F/sec. Surely, at the 7-foot elevation, at 18 seconds after flooding

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[51] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

[52] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.

[53] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.

commenced, there were local cladding temperature heatup rates that exceeded 16.1 K/sec (29°F/sec): the maximum heatup rate predicted by NRC's TRACE simulation using the Baker-Just correlation.[54]

It is unfortunate that NRC has overlooked the *new information* on FLECHT run 9573—not discussed in PRM-50-76—that Petitioner provided in PRM-50-93 and in comments on PRM-50-93/95.

(See Appendix D for information about experiments in which zirconium-steam reaction rates occurred that are under-predicted by computer safety models.)

III.B. Results of NRC's TRACE Simulations of FLECHT Run 9573 Were Not Compared to the Highest Cladding Temperatures and Heatup Rates

There are serious problems with the fact that NRC compared the results of its TRACE simulations of FLECHT run 9573 to the *average* value of different thermocouple measurements—data taken from the FLECHT run 9573 test bundle at the 2, 4, 6, 8, and 10-foot elevations, at 18 seconds after flooding commenced. NRC compared its TRACE results regarding cladding temperatures to "the average of the available thermocouple measurements at a particular elevation;"[55] and compared its TRACE results regarding cladding temperature heatup rates to "the average of the available thermocouple measurements at each elevation."[56] The values of the averages of the cladding temperatures and heatup rates would be lower than the maximum values of the cladding temperatures and heatup rates at each elevation. Assessing the Baker-Just and Cathcart-Pawel correlations for use in computer safety models by comparing TRACE results with averaged thermocouple measurements is not a legitimate assessment.

Furthermore, in comments on PRM-50-93/95, dated April 12, 2010 (pages 26-27), Petitioner pointed out that in the PWR FLECHT tests—including run 9573—there were radiative heat losses from the test bundles to the bundle housing, which "constituted a

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[54] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.
[55] *Id*., p. 7.
[56] *Id*., p. 8.

700°F cold spot;"[57] therefore, especially, the peripheral rods of the FLECHT run 9573 bundle would have radiated heat to the surrounding bundle housing.

Regarding the fact that the FLECHT run 9573 test bundle's interior rods were hotter than the peripheral rods, NRC's October 2012 DIR of PRM-50-93/95 states:

> In FLECHT run 9573 there were three thermocouples that registered temperatures greater than 2200 degrees F at a time of 18 seconds. … These were thermocouples numbered 3D3, 2D2, and 4E3. Each of these three thermocouples was on the interior of the bundle and shielded from the housing by at least one row of heater rods. Because of the low thermal radiation view factor, the [bundle] housing is not expected to have had a large influence on local heat transfer coefficients on the interior of the bundle.[58]

Hence, NRC acknowledges that temperatures were hotter in the interior of the test bundle; nonetheless, NRC decided to compare its TRACE results to the average value of different thermocouple measurements—hotter interior temperatures averaged with the cooler temperatures of the bundle's peripheral rods.

(In a LOCA, the concern would be that the *maximum fuel element cladding temperature* did not exceed the 2200°F 10 C.F.R. § 50.46(b)(1) PCT limit: the PCT limit pertains to the "hot spot," not to the average of cladding temperatures at a particular elevation.)

IV. NRC Overlooked Information Pertaining to PWR FLECHT Run 9573 Heat Transfer Coefficients

Regarding Petitioner's comments on PRM-50-93/95 dated March 15, 2010 (pages 5-9),[59] concerning FLECHT run 9573 heat transfer coefficients, NRC's October 2012 DIR states:

> The comments discuss the negative heat transfer coefficients near the mid-plane elevation in FLECHT run 9573 and that, as pointed out in the data

[57] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, Union of Concerned Scientists, 1974, p. 5.31.

[58] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 5.

[59] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.

> report [WCAP-7665] (Cadek *et al.*, 1971),[60] this occurred at approximately the time when heater rods began to fail in the bundle and the cladding temperatures were 2200-2300 degrees F. The comments also noted that heat transfer coefficients in this test were lower than those in other FLECHT tests with Zircaloy cladding. The petitioner, however, failed to recognize or acknowledge that this aspect of FLECHT run 9573 was addressed in the NRC technical evaluation of PRM-50-76 where this anomaly was attributed to the data reduction process. (See page 7 of NRC, 2004.)[61]

In the passage above, NRC has made an incorrect statement and overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients. It needs to be clarified that, as previously and correctly stated in PRM-50-93 (pages 59-60, 60-61), WCAP-7665 reports that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F"[62]—*i.e.*, they were not caused by temperatures in the range of 2200 to 2300°F.

First, NRC incorrectly described the statement from its own technical evaluation of PRM-50-76. NRC's technical evaluation does not say that the "anomaly," regarding heat transfer coefficients, was *definitely* attributed to the data reduction process. NRC's technical evaluation states that "*[s]ome* of the anomaly [lower 'measured' heat transfer coefficients] *can probably be explained* due to a deficiency in the data reduction process" [emphasis added].[63]

(More importantly, NRC needs to acknowledge that additional information regarding FLECHT run 9573 was provided in PRM-50-93 and that NRC's technical evaluation of PRM-50-76 is seriously flawed. For example, NRC's technical evaluation of PRM-50-76 does not mention the fact that the FLECHT run 9573 test bundle incurred runaway oxidation—there is still no NRC analysis of the sections of the bundle that incurred runaway oxidation.)

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[60] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665.
[61] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 5.
[62] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[63] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 CFR Part 50 and Regulatory Guide 1.157," p. 7.

In fact, Westinghouse's 1971 report, WCAP-7665, states that "anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. *These may have been related to the high steam probe temperatures measured at the 7 ft elevation*" [emphasis added].[64] (The high steam probe temperatures "exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure." [65])

Second, NRC has overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients that Petitioner provided in PRM-50-93 (pages 9-11, 59-70) and comments on PRM-50-93/95, dated March 15, 2010 (pages 5-9),[66] dated November 23, 2010 (pages 29-34),[67] and dated December 27, 2010 (pages 15-21).[68] As stated, Westinghouse postulated that the negative heat transfer coefficients observed in FLECHT run 9573 "may have been related to the high steam probe temperatures measured at the 7 ft elevation."[69] In PRM-50-93 and comments on PRM-50-93/95, Petitioner argues that the high steam temperatures were in fact the cause of the negative heat transfer coefficients; the negative heat transfer coefficients were a result of heat transfer from the steam—measured at temperatures exceeding 2500°F—to the test bundle rods.

Regarding FLECHT run 9573, in October 2002, Westinghouse stated, "[t]he high fluid [steam] temperature was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[70] Hence, the heat generated by the zirconium-steam reaction is what heated the steam to temperatures exceeding 2500°F—a phenomenon that could occur in a large break LOCA.

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[64] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.
[65] *Id*., p. 3-97.
[66] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[67] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[68] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.
[69] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.
[70] H. A. Sepp, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, available at: NRC's ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

IV.A. NRC's Incorrect Claim that Its TRACE Simulations of FLECHT Run 9573 Demonstrate that Conservative Heat Transfer Models Can Be Developed from Data Obtained Primarily from Experiments Conducted with Stainless Steel Rods

In its October 2012 DIR of PRM-50-93/95, the NRC Staff claims:

> The TRACE simulations…demonstrate that it is possible to develop heat transfer models based on data obtained primarily from stainless steel rods and conservatively simulate FLECHT run 9573. When either the Cathcart-Pawel or Baker-Just correlations are used to determine the metal-water reaction rate, TRACE was found to conservatively predict the cladding temperatures at each elevation. … The staff concludes that there is nothing in the petition that [indicates] use of stainless steel clad rod data is inaccurate or insufficient for development of heat transfer models.[71]

As discussed in Section III, NRC's TRACE simulations of FLECHT run 9573 are invalid because they did not simulate the section of the test bundle that incurred runaway oxidation. The simulations of FLECHT run 9573 encompassed locations of the test bundle that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Furthermore, the simulations did not include data taken from the 7-foot elevation of the test bundle, where a steam probe thermocouple measured steam temperature heatup rates that exceeded 300°F/sec. There are also serious problems with the fact that NRC compared the results of its TRACE simulations of FLECHT run 9573 to the *average* value of different thermocouple measurements taken at each elevation and not to the maximum values of the cladding temperatures measured at each elevation (the 2, 4, 6, 8, and 10-foot elevations of the test bundle, at 18 seconds after flooding commenced).

Clearly, NRC's TRACE simulations are neither legitimate simulations of FLECHT run 9573 nor legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models. Hence, the TRACE simulations do *not* "demonstrate that it is possible to develop heat transfer models based on data obtained primarily from stainless steel rods."[72]

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[71] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 9.
[72] *Id*., p. 9.

As stated in Section IV, NRC has overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients that Petitioner provided in PRM-50-93 (pages 9-11, 59-70) and comments on PRM-50-93/95, dated March 15, 2010 (pages 5-9),[73] dated November 23, 2010 (pages 29-34),[74] and dated December 27, 2010 (pages 15-21).[75] The information Petitioner provided supports the claim that Appendix K to Part 50 Section I.D.5—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]"—is erroneously based on the assumption that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

V. NRC's Conclusions Regarding Reflood Rates Are Invalid because They Are Based on NRC's TRACE Simulations of FLECHT Run 9573, which Did Not Simulate the Section of the Test Bundle that Incurred Runaway Oxidation

In its March 2013 DIR of PRM-50-93/95, NRC's conclusions regarding reflood rates are based on NRC's TRACE simulations of FLECHT run 9573. As discussed in Section III, NRC's TRACE simulations of FLECHT run 9573 are invalid because they did not simulate the section of the test bundle that incurred runaway oxidation. In fact, NRC's TRACE simulations of FLECHT run 9573 encompassed locations of the test bundle that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

In its March 2013 DIR of PRM-50-93/95, NRC *incorrectly* concludes that its "TRACE simulation of Test 9573 showed reasonable agreement with available data, with TRACE exceeding the measured maximum cladding temperature 18 seconds into the

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[73] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[74] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[75] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

test."[76] As discussed in Section III.A, Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573[77] and that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[78] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F *higher* than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F *higher* than the highest temperature predicted using the Cathcart-Pawel correlation.

V.A. Comparisons of NRC's TRACE Simulations of FLECHT Run 9573 with Actual Experimental Data

In order to reach its conclusions regarding reflood rates for its DIR of PRM-50-93/95, NRC relies on invalid TRACE simulations of FLECHT run 9573. Different conclusions would be reached by objectively reviewing actual experimental data from tests *conducted with zirconium alloy bundles*. (Interestingly, the TRACE simulations of FLECHT run 9573 (the ones done in order to reach conclusions regarding reflood rates) seem to have only used the Cathcart-Pawel correlation; apparently, the Baker-Just correlation was not used in any of the simulations.[79])

[76] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," March 8, 2013, available at: NRC's ADAMS Documents, Accession Number: ML13067A261, p. 4.
[77] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[78] *Id*.
[79] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," pp. 4, 7.

V.A.1. TRACE Simulations of Reflood Cooling Compared to Actual Experimental Data

In its March 2013 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573 in which:

> In each case, the initial axial cladding temperature profile was scaled to that of Test 9573 to obtain the desired maximum cladding temperature at the start of each simulation. The reflood rate was assumed to be 1.1 inch/sec, consistent with Test 9573. At maximum initial cladding temperatures less than approximately 1200 degrees F (922 K), typical of those expected following the blowdown period of a LOCA, the peak cladding temperature[s] remain below 1800 degrees F (1255 K).[80]

In FLECHT run 9573 the *actual* PCT at the onset of reflood was 1970°F;[81] however, for the NRC TRACE simulations discussed in this section (V.A.1), FLECHT run 9573 was assigned PCTs at the onset of reflood that were less than approximately 1200°F. These TRACE simulations each resulted in FLECHT run 9573 having an overall PCT that was less than 1800°F. But there are problems with these TRACE simulations because there is data from *actual* thermal hydraulic LOCA experiments conducted with zirconium alloy bundles that indicates these simulations under-predict the overall PCT that FLECHT run 9573 would have had if its PCT at the onset of reflood had been 1200°F or lower. NRU Thermal Hydraulic 1 ("TH-1") test nos. 109 and 125 were conducted with reflood rates of 1.3 inches/second (in/sec) and 1.4 in/sec, respectively. TH-1 test no. 109 had a PCT at the onset of reflood of 1158°F and an overall PCT of 1881°F; and TH-1 test no. 125 had a PCT at the onset of reflood of 1138°F and an overall PCT of 1802°F.[82]

TH-1 test nos. 109 and 125 both had greater reflood rates than FLECHT run 9573. The greater reflood rates of TH-1 test nos. 109 and 125 would have had more of an effect on mitigating the overall PCT increases in those tests than the lower reflood rate of FLECHT run 9573 had on mitigating run 9573's overall PCT increase. (As discussed in

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[80] *Id*., p. 4.

[81] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.

[82] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, available at: NRC's ADAMS Documents, Accession Number: ML101960414, p. 13.

Section V.B.1, the flooding rate is the most influential parameter that affects the overall PCT in thermal hydraulic LOCA experiments.)

And the TH-1 tests had an average fuel rod power of 0.38 kW/ft;[83] the peak rod power of FLECHT run 9573 was 1.24 kW/ft.[84] The lower fuel rod power of the TH-1 tests would not have affected the overall PCT increases as much as the greater fuel rod power of FLECHT run 9573 affected run 9573's overall PCT increase. (Regarding low power runs of thermal hydraulic LOCA experiments, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report" states that "[the] temperature rises…are smaller for the low power [runs] since lower energy removal rates and temperature differences are needed to remove the generated energy."[85]) Nonetheless, TH-1 test nos. 109 and 125, which both had initial PCTs that were less than 1200°F, had overall PCTs that exceeded 1800°F. (NRC's TRACE simulations of FLECHT run 9573—conducted with assigned initial PCTs of less than 1200°F for run 9573—predicted that run 9573's overall PCT would remain below 1800°F.) Such actual experimental data is further evidence that NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

V.A.2. TRACE Simulations of Steam Cooling Compared to Actual Experimental Data

In its March 2013 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573; NRC states:

> Consider the TRACE model of the Zircaloy clad bundle that represented the bundle used in FLECHT Test 9573. Assuming an initial temperature profile with a maximum temperature of 1200 degrees F (922 K), a simulation was conducted with no liquid injection but with steam-only cooling of the bundle. [The] steam-only mass flow rate [was] 0.114 kg/s through the bundle. The peak cladding temperature obtained [was] 1325.7

[83] C. L. Mohr *et al.,* Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, available at: NRC's ADAMS Documents, Accession Number: ML083470834, p. 9-40.
[84] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.
[85] G. P. Lilly, H. C. Yeh, L. E. Hochreiter, N. Yamaguchi, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report," WCAP-8838, March 1977, available at: NRC's ADAMS Documents, Accession Number: ML070780090, p. 3-5.

> K (1927 degrees F). No liquid injection can be interpreted as a reflooding rate of 0.0 in/sec. Cooling was accomplished not by reflood of the bundle, but only by convective cooling to the steam. The cladding exceeded 1000 C (1832 degrees F), and thus metal-water reaction became a significant source of heat. Nevertheless, the peak cladding temperature remained below 2200 degrees F and an "autocatalytic" (runaway) oxidation did not occur.[86]

Again, there is data from *actual* thermal hydraulic LOCA experiments conducted with zirconium alloy bundles that indicates NRC's TRACE simulations under-predict the overall PCT that FLECHT run 9573 would have had if its PCT at the onset of reflood had been 1200°F and its reflood rate had been 0.0 in/sec, with a steam-only mass flow rate of 0.114 kilograms/second ("kg/sec") through the test bundle.

In FLECHT run 9573, a steam-only mass flow rate of 0.114 kg/sec would be approximately equal to a reflood rate of 0.68 in/sec, if the steam were condensed.[87] In NRC's TRACE simulation, the steam was assigned an inlet temperature of approximately 307°F.[88]

TH-1 test nos. 127 and 130 were conducted with reflood rates of 1.0 in/sec and 0.74 in/sec, respectively. TH-1 test no. 127 had a PCT at the onset of reflood of 966°F and an overall PCT of 1991°F; and TH-1 test no. 130 had a PCT at the onset of reflood of 998°F and an overall PCT of 2040°F.[89]

In TH-1 test no. 130, the reactor *actually* tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures increased

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[86] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," pp. 5-6.

[87] In the four PWR FLECHT facility tests with zirconium alloy (7 x 7) bundles, the bundle housing was square with internal dimensions of 4.200 inches (in) and there were 42 test rods with a diameter of 0.422 inch, six control rod thimbles a diameter of 0.545 inch, and one instrument tube with a diameter of 0.463 inch. See F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," pp. 2.1, 2.11, 3.8. In FLECHT run 9573, the cross-sectional flow area was 10.198 in^2, which is calculated by subtracting the total of $42\pi(.211\ in^2) + 6\pi(.2725\ in^2) + \pi(.2315\ in^2)$ from $(4.2\ in^2)$. A mass of 0.114 kg of water has a volume of 6.957 in^3. In FLECHT tests with 7 x 7 bundles, a volume of 6.957 in^3 of water—with a cross-sectional area of 10.198 in^2—would have had a height of 0.68 in.

[88] In NRC's TRACE simulation of steam-only cooling of FLECHT run 9573, the steam was saturated steam at a pressure of 0.42 MPa. See NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 6. Saturated steam at a pressure of 0.42 MPa (60.9 pounds per square inch) would have a temperature of approximately 307°F.

[89] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, pp. v, 13.

by 190°F, because of the heat generated by the zirconium-steam reaction (of course, there would have also been a slight amount of actual decay heat[90]) and the peak measured cladding temperature was 2040°F.[91] In TH-1 test no. 130, if the reactor had not shutdown when the PCT was approximately 1850°F, the overall PCT would have exceeded 2040°F. In fact, if the reactor had not shutdown when the PCT was approximately 1850°F it is possible that the combination of the simulated decay heat and heat generated by the zirconium-steam reaction would have caused the test bundle to incur runaway oxidation; in such a case, the PCT would have increased to greater than 3300°F.

(TH-1 test no. 130 is discussed on pages 24-25 of Petitioner's comments on PRM-50-93/95, dated December 27, 2010,[92] on page 5 of Petitioner's comments on PRM-50-93/95, dated July 27, 2011,[93] and on pages 9-11 of Petitioner's comments on PRM-50-93/95, dated July 30, 2011.[94])

TH-1 test nos. 127 and 130 both had greater coolant inlet rates (reflood rates of 1.0 in/sec and 0.74 in/sec, respectively) than the steam-only mass flow rate of 0.114 kg/sec (approximately equal to a reflood rate of 0.68 in/sec, if the steam were condensed) that was assigned to FLECHT run 9573 for NRC's TRACE simulations. The greater coolant inlet rates of TH-1 test nos. 127 and 130 would have had more of an effect on mitigating the overall PCT increases in those tests than the lower coolant inlet rate assigned to FLECHT run 9573 had on mitigating run 9573's overall PCT increase. And the TH-1 tests had an average fuel rod power of 0.38 kW/ft;[95] the peak rod power of

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[90] TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft. See C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.

[91] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.

[92] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

[93] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.

[94] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

[95] C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.

FLECHT run 9573 was 1.24 kW/ft.[96] The lower fuel rod power of the TH-1 tests would not have affected their overall PCT increases as much as the higher fuel rod power of FLECHT run 9573 affected its overall PCT increase. Furthermore, TH-1 test nos. 127 and 130 both had initial PCTs that were less than 1000°F; and 1200°F was the initial PCT assigned to FLECHT run 9573 for NRC's TRACE simulations. Nonetheless, TH-1 test nos. 127 and 130 had overall PCTs of 1991°F and 2040°F, respectively. (NRC's TRACE simulations of FLECHT run 9573 predicted that run 9573's overall PCT would be 1927°F.) Such actual experimental data is yet further evidence that NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

NRC has incorrectly concluded that "[t]he [TRACE] steam-only cooling calculation demonstrates that it is possible to cool a Zircaloy clad bundle without reflooding."[97] NRC should review actual experimental data and not rely on invalid TRACE simulations of FLECHT run 9573, which did not simulate the section of the test bundle that incurred runaway oxidation.

V.B. Information Pertaining to LOCA-Reflood Phenomena that NRC Overlooked

V.B.1. NRC Overlooked the Significant Role that Reflood Rates have in Determining the PCT in a LOCA

Regarding reflood LOCA hydraulics, in its March 2013 DIR of PRM-50-93/95, NRC states that "[b]ecause numerous parameters have an effect on reflood hydraulics, no single parameter completely controls the peak cladding temperature for a particular transient."[98] While NRC's assertion is correct as far as it goes, it does not go far enough. As previously stated in PRM-50-93 (page 13), regarding the significance that coolant flood rates played in the PWR FLECHT test program, the "PWR FLECHT Final Report" states, "[i]n general, the effect on heat transfer coefficient[s] of varying system parameters was clearly discernable, *with flooding rate being by far the most influential*

[96] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.
[97] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 6.
[98] *Id*., p. 2.

parameter investigated"[99] [emphasis added]. Hence, reflood rates would have a significant role in determining the PCT in a LOCA; and thus there needs to be a new regulation stipulating minimum allowable core reflood rates in the event of a LOCA, as requested in PRM-50-93.

V.B.2. NRC Overlooked the Role that the Heat Generated by the Exothermic Zirconium-Steam Reaction has in Increasing Fuel-Cladding Temperatures in a LOCA

Regarding fuel-cladding temperature increases of over 1000°F that were observed in NRU reflood tests conducted with Zircaloy fuel-cladding, in its March 2013 DIR of PRM-50-93/95, NRC states:

> Part of the basis for the petition's request for a limit on reflood rate, is the significant temperature increases observed in the NRU reflood tests. Starting from initial cladding temperatures less than 1000 degrees F, several NRU tests produced temperature increases of over 1000 degree F. The petition cites NRU test 127 and 130 as examples. The petition appears to imply that similar temperature increases would occur if the initial cladding temperatures had been 1200 degrees F or more. *This is not correct, however*[100] [emphasis added].

PRM-50-93/95 does in fact state that it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower would not, with high probability, prevent zirconium alloy fuel cladding with peak cladding temperatures of approximately 1200°F or greater at the onset of reflood, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. NRC claims that this is incorrect (NRC's argument is quoted below in this section (V.B.2)); however, NRC has overlooked the role that the heat generated by the exothermic zirconium-steam reaction has in increasing fuel-cladding temperatures in a LOCA.

As already discussed in section V.A.2, in TH-1 test no. 130 (conducted with zirconium alloy fuel cladding), the reactor shutdown when the PCT was approximately 1850°F and after the reactor shutdown, cladding temperatures increased by 190°F, because of the heat generated by the zirconium-steam reaction (of course, there would

[99] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 5-1.
[100] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

have also been a slight amount of actual decay heat[101]) and the peak measured cladding temperature was 2040°F.[102] If the reactor had not shutdown when the PCT was approximately 1850°F, the overall PCT would have exceeded 2040°F; and it is highly probable that the test bundle would have incurred runaway oxidation and that the PCT would have increased to greater than 3300°F.

NRC needs to consider that if TH-1 test no. 130 had been conducted with an initial PCT of 1200°F and the reactor did not shutdown when the PCT was approximately 1850°F, with high probability, the overall PCT would have exceeded 2200°F, because of the heat generated by the zirconium-steam reaction.

Regarding the results of LOCA tests *conducted with stainless steel bundles* in three experimental programs—PWR FLECHT SEASET,[103] PWR FLECHT Cosine,[104] and PWR FLECHT Skewed[105]—in its March 2013 DIR of PRM-50-93/95, NRC states:

> Thermal radiation becomes more important in transferring heat away from hot spots, and as rod temperatures increase the temperature difference between the cladding and the coolant increases. Figure 1…shows the effect of initial cladding temperature on temperature rise from tests in three experimental facilities. As the initial cladding temperature increases, the overall temperature rise *decreases*[106] [emphasis not added].

It is important to recognize that *only* thermal hydraulic LOCA experiments conducted with stainless steel bundles demonstrate the phenomenon of higher cladding temperature increases for tests with lower PCTs at the onset of reflood (in the entire

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[101] TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft. See C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.
[102] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.
[103] Lee, N., Wong, S., Yeh, H.C., and Hochreiter, L.E., "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Evaluation and Analysis Report," WCAP-9891, NUREG/CR-2256, February 1982, available at: NRC's ADAMS Documents, Accession Number: ML070740214.
[104] G. P. Lilly, H. C. Yeh, L. E. Hochreiter, N. Yamaguchi, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report," WCAP-8838.
[105] Lilly, G.P. *et al.*, "PWR FLECHT Skewed Profile Low Flooding Rate Test Series Evaluation Report," WCAP-9183, November 1977, available at: NRC's ADAMS Documents, Accession Number: ML070780095.
[106] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

design basis accident cladding temperature range, below 2200°F). And, of course, nuclear power plants use zirconium alloy fuel rod cladding—not stainless steel fuel rod cladding.

At lower temperatures thermal hydraulic LOCA experiments conducted with Zircaloy bundles also demonstrate the phenomenon of higher cladding temperature increases for tests with lower PCTs at the onset of reflood; however, the results of experiments conducted with Zircaloy bundles are *different* at higher temperatures. In the temperature range at which the oxidation of Zircaloy becomes significant, the heat generated by the zirconium-steam reaction causes higher cladding temperature increases, as PCTs at the onset of reflood increase.

This trend is seen in four Zircaloy tests—TH-1 test nos. 105, 107, 110, and 128—conducted with an average fuel rod power of 0.38 kW/ft;[107] the first three tests had a reflood rate of 1.9 in/sec; the fourth test had a reflood rate of 2.0 in/sec. TH-1 test no. 105 had a PCT at the onset of reflood of 907°F and an overall PCT of 1364°F (an increase of 457°F); TH-1 test no. 107 had a PCT at the onset of reflood of 1154°F and an overall PCT of 1578°F (an increase of 424°F); TH-1 test no. 110 (Zircaloy) had a PCT at the onset of reflood of 1314°F and an overall PCT of 1665°F (an increase of 351°F); and TH-1 test no. 128 (Zircaloy) had a PCT at the onset of reflood of 1604°F and an overall PCT of 1991°F (an increase of 387°F).[108]

TH-1 test nos. 105, 107, and 110, demonstrate the phenomenon of higher cladding temperature increases for tests that had lower PCTs at the onset of reflood (for thermal hydraulic experiments conducted with Zircaloy bundles *at lower temperatures*). However, in TH-1 test no. 128, with a PCT at the onset of reflood of 1604°F, the overall PCT increase is 36°F greater than the overall PCT increase in TH-1 test no. 110, with a PCT at the onset of reflood of 1314°F. The overall PCT increased more in TH-1 test no. 128—with a slightly higher reflood rate—because of the heat that was generated by the zirconium-steam reaction.

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[107] C. L. Mohr *et al.,* "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.

[108] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.

(Unfortunately, an extremely limited amount of tests have been conducted with zirconium alloy bundles, so there is not much experimental data available to discuss.) NRC is incorrect in its conclusion that "[a]s the initial cladding temperature increases, the overall temperature rise *decreases*"[109] [emphasis not added]. Incredibly, NRC has *only* considered data from thermal hydraulic LOCA experiments conducted with stainless steel bundles and *overlooked* data from experiments conducted with the industry-standard zirconium alloy bundles.

VI. Conclusion

NRC's October 2012 DIR of PRM-50-93/95 actually overlooks experimental data NRC itself provided in its September 2011 DIR demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F PCT limit.[110] Clearly, the NRC Staff needs to correct its erroneous conclusion that runaway oxidation has not commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

It is unfortunate that NRC has also overlooked the new information Petitioner provided which indicates that Westinghouse's metallurgical data from FLECHT run 9573 is invalid. There are significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from FLECHT run 9573, because Westinghouse did not obtain metallurgical data from the locations of the rods from run 9573 that incurred runaway oxidation.

Additionally, NRC's TRACE simulations of FLECHT run 9573 did not include the section of the test bundle that incurred runaway oxidation. In fact, NRC's TRACE simulations encompassed locations of the test bundle that were most likely steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, NRC's TRACE simulations are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in

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[109] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.
[110] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," p. 4.

computer safety models, and not legitimate verifications of NRC's conclusions regarding reflood rates.

The highest predicted temperatures in NRC's TRACE simulations of FLECHT run 9573 at 18 seconds after flooding commenced, using the Baker-Just correlation and Cathcart-Pawel correlation, were 2417.7°F and 2338.2°F, respectively.[111] Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573.[112] And Westinghouse reported that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[113] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F higher than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F higher than the highest temperature predicted using the Cathcart-Pawel correlation. Hence, NRC's TRACE simulations of FLECHT run 9573 indicate that the Baker-Just and Cathcart-Pawel correlations are not sufficiently conservative for use in computer safety models.

(See Appendix A for information about the BWR FLECHT Zr2K test and TH-1 test 130, design basis accident experiments in which runaway oxidation (most likely) commenced and almost commenced, respectively, at fuel-cladding temperatures that were lower than the 2200°F PCT limit. And see Appendix D for information about experiments in which zirconium-steam reaction rates occurred that are under-predicted by computer safety models.)

It is also important to recognize the limitations of thermal hydraulic LOCA experiments that were conducted with stainless steel bundles. Of course, nuclear power plants use zirconium alloy fuel-cladding—not stainless steel fuel-cladding.

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111 NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.
112 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
113 *Id*.

Respectfully submitted,

/s/

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Appendix A Experiments in which Runaway Oxidation (Most Likely) either Commenced or Almost Commenced at Fuel Cladding Temperatures Lower than the 2200°F PCT Limit

I. An Experiment in which Runaway Oxidation Most Likely Commenced at a Temperature Lower than the 2200°F PCT Limit: The BWR FLECHT Zr2K Test

NRC's October 2012 Draft Interim Review ("DIR") of PRM-50-93/95 concluded that "autocatalytic reactions have not occurred at temperatures less than [the 2200°F PCT limit];"[1] however, the NRC's DIR overlooked information Petitioner presented on the BWR FLECHT Zr2K test. (The BWR FLECHT Zr2K test is discussed on pages 35-45 of Petitioner's comments on PRM-50-93, dated March 15, 2010,[2] with information in Appendix F of the March 15, 2010 comments; and discussed on pages 39-49 of PRM-50-95, with information in Appendix G of PRM-50-95.)

In the Atomic Energy Commission's ("AEC") emergency core cooling systems ("ECCS") rulemaking hearing, conducted in the early 1970s, Dr. Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[3] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[4] conducted with a pressurized Zircaloy multi-rod bundle. Among other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [Zircaloy-steam] reactions"[5] and that the test "was

[1] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12265A277, p. 2.

[2] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.

[3] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists ("UCS").

[4] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-17; this paper cites UCS, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

[5] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-18.

saved only as a 'consequence of the extensive heater failures that occurred'."[6] In the hearing, Dr. Roger Griebe, the Aerojet Nuclear Company (Aerojet) project engineer who coordinated the BWR-FLECHT program, testified that "there is *no* convincing proof available from [Zr2K] test data to demonstrate that near-thermal runaway definitely did not exist" in the Zr2K test [emphasis not added].[7, 8]

(Petitioner would argue that actual thermal runaway—not *near* thermal runaway—occurred in the BWR FLECHT Zr2K test, because local test bundle cladding temperatures increased from lower than 2200°F to greater than 2900°F in approximately 40 seconds.[9])

General Electric ("GE") argued that the exothermic Zircaloy-steam reaction was insignificant in the thermal response of the Zircaloy heater rods and estimated that the energy from the exothermic Zircaloy-steam reaction was between 5 and 10% of the total energy input.[10] However, it is probable that GE was incorrect: in some of the BWR CORA experiments, conducted years later, in the 1980s, the Zircaloy-steam reaction contributed between 33 and 48% of the total energy input, once cladding temperatures reached approximately 2200°F.[11]

Thermocouple (a temperature measuring device) measurements taken during the Zr2K test, recorded that at between approximately 2100 and 2200°F, local cladding temperatures began to rapidly increase, leading to increases of tens of degrees Fahrenheit per second: in some intervals (approximately 20 seconds long), there were local

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[6] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-24; this paper cites UCS, "An Evaluation of Nuclear Reactor Safety," p. 5.63, as the source of this information.

[7] Official Transcript of the AEC's Emergency Core Cooling Systems Rulemaking Hearing, pp. 7138-7139.

[8] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, UCS, 1974, p. 5.11.

[9] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Figure 12, as the source of this information.

[10] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-18, A8-19.

[11] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, p. 5.

temperature increases of several hundred degrees Fahrenheit.[12] The thermocouples recorded that local cladding temperatures increased to greater than 2900°F.

GE argued that the thermocouple measurements of the rapid cladding-temperature increases taken in the Zr2K test were not valid, claiming "that the 'erratic thermocouple outputs[13] do not represent actual cladding temperatures, but are the result of equipment malfunctions'[14] associated with the Zr2K test."[15] In the rulemaking hearing, the AEC agreed with GE that the thermocouple measurements of the rapid cladding-temperature increases taken in the Zr2K test were not valid; the AEC stated that "[i]n [the Zr2K test], the maximum cladding temperature was approximately 2250°F."[16]

However, it is highly probable that GE and the AEC were incorrect: the thermocouple measurements taken in the Zr2K test resemble thermocouple measurements taken in BWR severe fuel damage experiments, in which there were rapid cladding-temperature increases that commenced below 2200°F, leading to increases of

[12] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-25, A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13l97, June 1971, Figures A-11, A-12, and J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Figure 12, as the sources of this information.

[13] A California Institute of Technology report which analyzed data from the Zr2K test, concluded that the observed thermocouple measurements were not erratic; see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-21, A8-23.

[14] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Appendix D, p. 107.

[15] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-24, A8-27.

[16] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1104-1105. This document is available at: NRC's ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999; the source of this information is Exhibit 1069, pp. 53-54, from the rulemaking hearing.

tens of degrees Fahrenheit per second. Local cladding temperatures in such experiments exceeded 2900°F.[17]

In the ECCS rulemaking hearing, Dr. Kendall and Ford contended in their direct testimony that "GE's interpretation of [the Zr2K test] is based on a…maximum cladding temperature curve that…constituted false reporting of the test data;" [18] and Dr. Griebe testified "that GE 'tremendously slanted' BWR-FLECHT data "towards the lower temperatures and towards the interpretation GE obviously presented in their report'."[19]

(In their final decision on the issues raised in the ECCS rulemaking hearing, the AEC commissioners observed that "[t]he conditions in [the BWR FLECHT Zr-2 test] were stated to be significantly more severe than the conditions reasonably expected to prevail during a postulated BWR LOCA, even for the 'hot' bundle."[20])

II. An Experiment that Most Likely Would have Incurred Runaway Oxidation if the Reactor had Not Shutdown When Maximum Fuel Cladding Temperatures Were Approximately 1850°F: Thermal Hydraulic 1 Test 130

In NRC's October 2012 DIR of PRM-50-93/95, NRC states that "[b]ecause of the initial high temperature in FLECHT run 9573, the conditions exceeded design basis LOCA conditions and were more typical of a severe accident test."[21] Indeed, FLECHT run 9573 had high initial cladding temperatures (the BWR FLECHT Zr-2 test also exceeded design basis LOCA conditions, as noted in Section I of Appendix A). However, a different PWR LOCA test (NRU Thermal Hydraulic 1 ("TH-1") test 130), which in some ways resembles FLECHT run 9573, did not have high initial cladding temperatures; TH-1 test no. 130 was also conducted with a relatively low power level.

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[17] L. Sepold *et al.*, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," FZKA 7447, pp. I, 1.
[18] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," pp. 5.12, 5.14.
[19] *Id*.
[20] Dixy Lee Ray *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," pp. 1104-1105; the source of this information is Exhibit 1148, p. P-15, from the rulemaking hearing.
[21] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.

(TH-1 test no. 130 is discussed on pages 24-25 of Petitioner's comments on PRM-50-93/95, dated December 27, 2010,[22] on page 5 of Petitioner's comments on PRM-50-93/95, dated July 27, 2011,[23] and on pages 9-11 of Petitioner's comments on PRM-50-93/95, dated July 30, 2011.[24])

In TH-1 test no. 130, there was a reflood rate of 0.74 in./sec.[25] At the onset of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[26] (TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft.[27])

In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures increased by 190°F, because of the heat generated from the zirconium-steam reaction (of course, there would have also been a slight amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[28]

It is clear that, in TH-1 test no. 130, if the reactor had not shutdown when the PCT was approximately 1850°F, that the overall PCT would have exceeded 2040°F. In fact, it is highly probable that the multi-rod bundle in the TH-1 test no. 130, would have incurred runaway oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

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[22] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.
[23] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.
[24] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.
[25] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, available at: NRC's ADAMS Documents, Accession Number: ML101960414, Abstract, p. v. The Abstract states that the lowest reflood rate in the TH-1 tests was 1.88 cm/ sec (0.74 in./sec); the Summary states that the lowest reflood rate in the TH-1 tests was 0.74 in./sec; page 13 states that the reflood rate of TH-1 test no. 130 was 0.7 in./sec: so the value of "0.7 in./sec," given on page 13, was rounded off from 0.74 in./sec.
[26] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.
[27] C. L. Mohr *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, available at: NRC's ADAMS Documents, Accession Number: ML083470834, p. 9-40.
[28] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.

III. In the PHEBUS B9R-2 Test, a Rapid Fuel-Cladding Temperature Escalation Commenced at Approximately 1880°F

(The information discussed in this section was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

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.[29] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[30]

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. … 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)].[31]

According to an October 2000 OECD Nuclear Energy Agency report, the initial heatup rate in PHEBUS B9R-2 was less than 0.1°C/sec up to 727°C (1340°F) (during the pure helium phase of the experiment).[32] However, according to a graph with a plot of fuel-cladding temperature values at the 0.6 meter "hot level" of the PHEBUS B9R-2 test bundle, the initial heatup rate in PHEBUS B9R-2 was approximately 1.0°C/sec up to 727°C (1340°F); however, the heatup rate decreases to lower than 0.2°C/sec between

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[29] 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.

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

[31] *Id.*, p. 126.

[32] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix Update 1996-1999," NEA/CSNI/R(2000)21, October 2000, p. 97.

approximately 877°C (1610°F) and 1002°C (1835°F).[33] (See Figure 1.) As stated, the cladding-temperature escalation commenced at approximately 1027°C (1880°F).

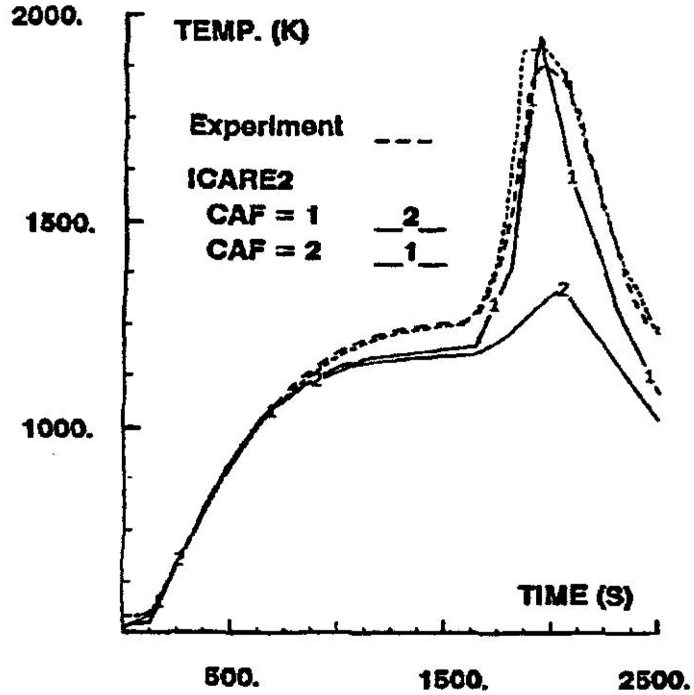
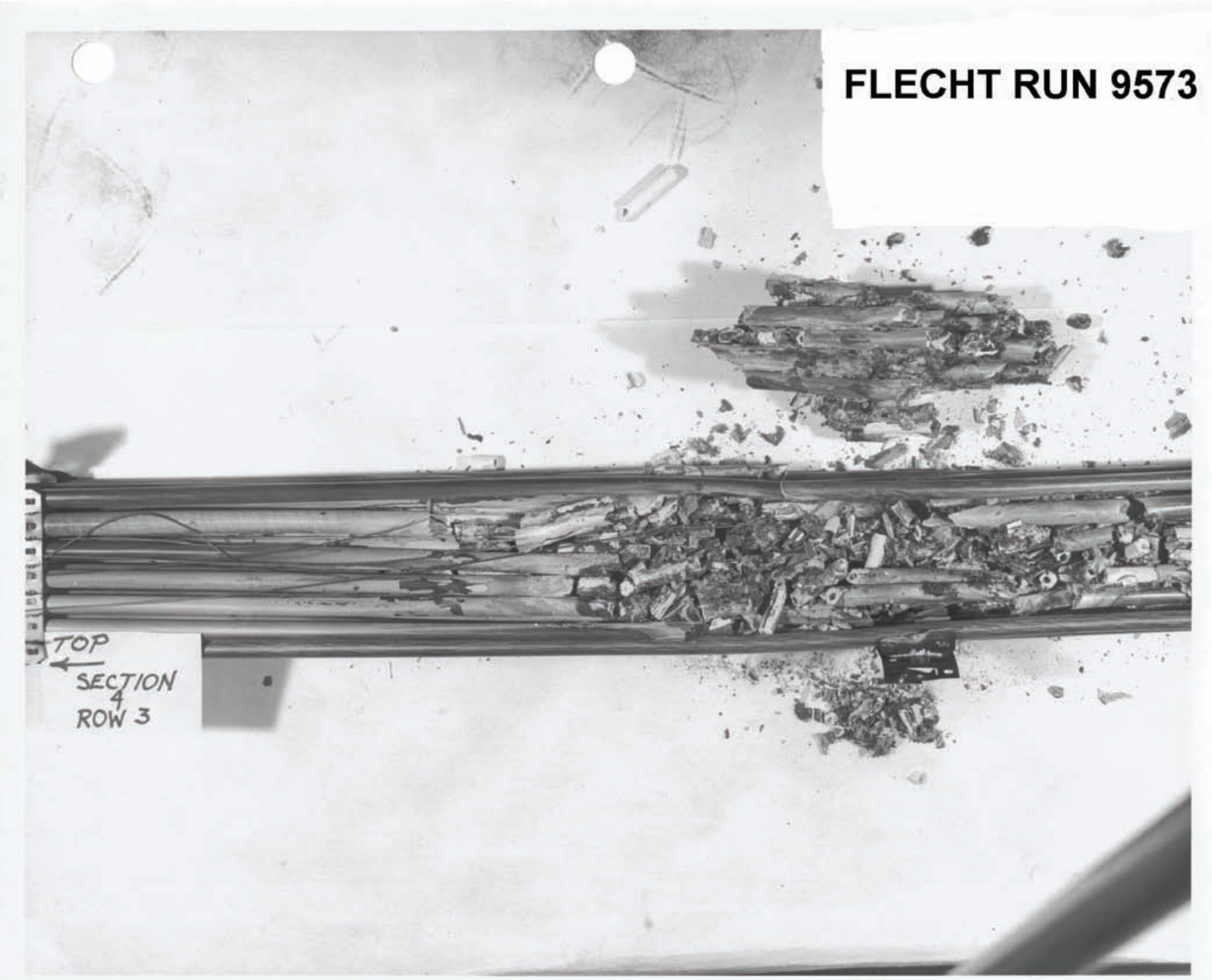


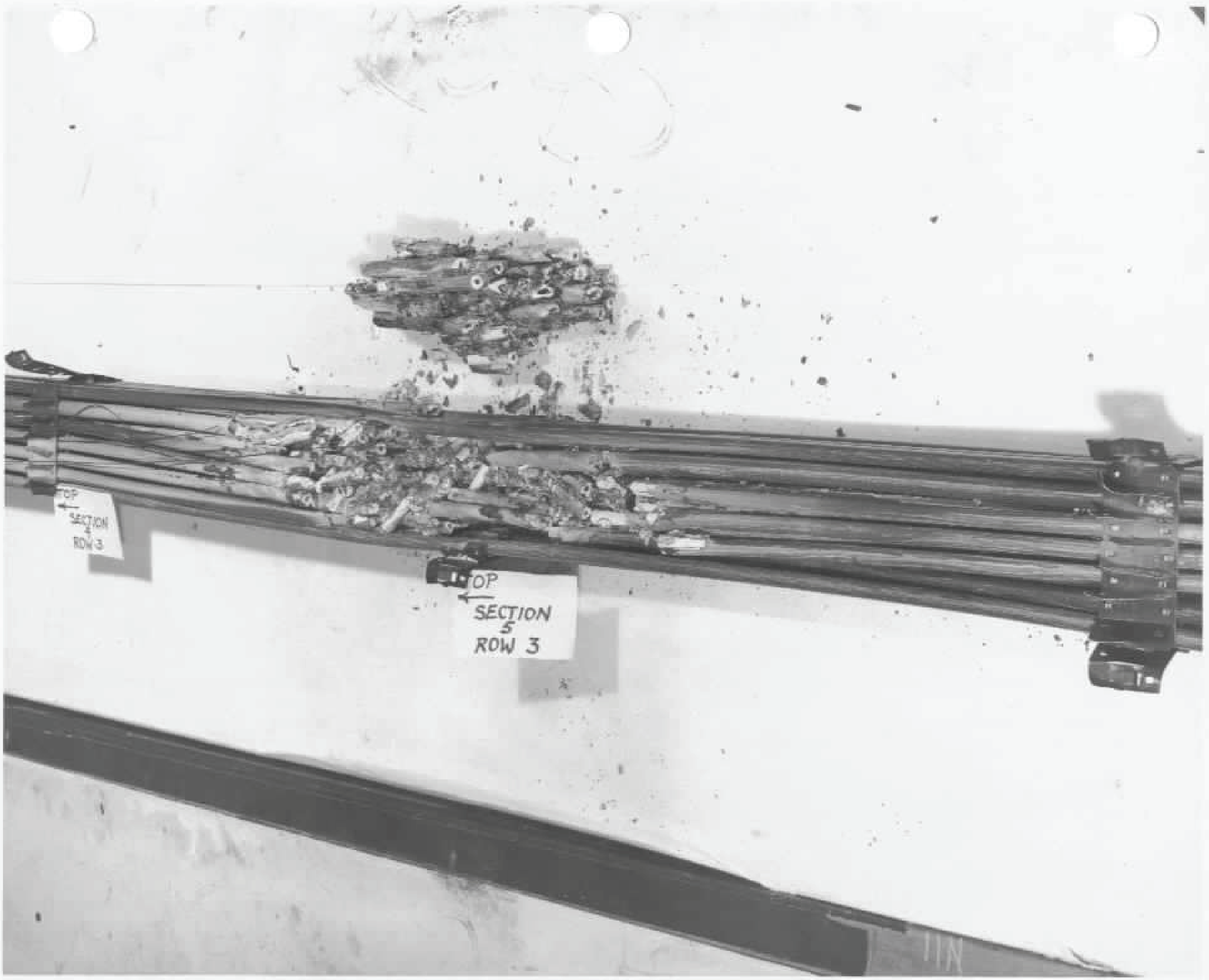
Figure 1. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[34]

[33] G. Hache, R. Gonzalez, B. Adroguer, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, p. 312.
[34] *Id*.

Appendix B Photographs of the Section of the Test Bundle from FLECHT Run 9573 that Incurred Runaway Oxidation

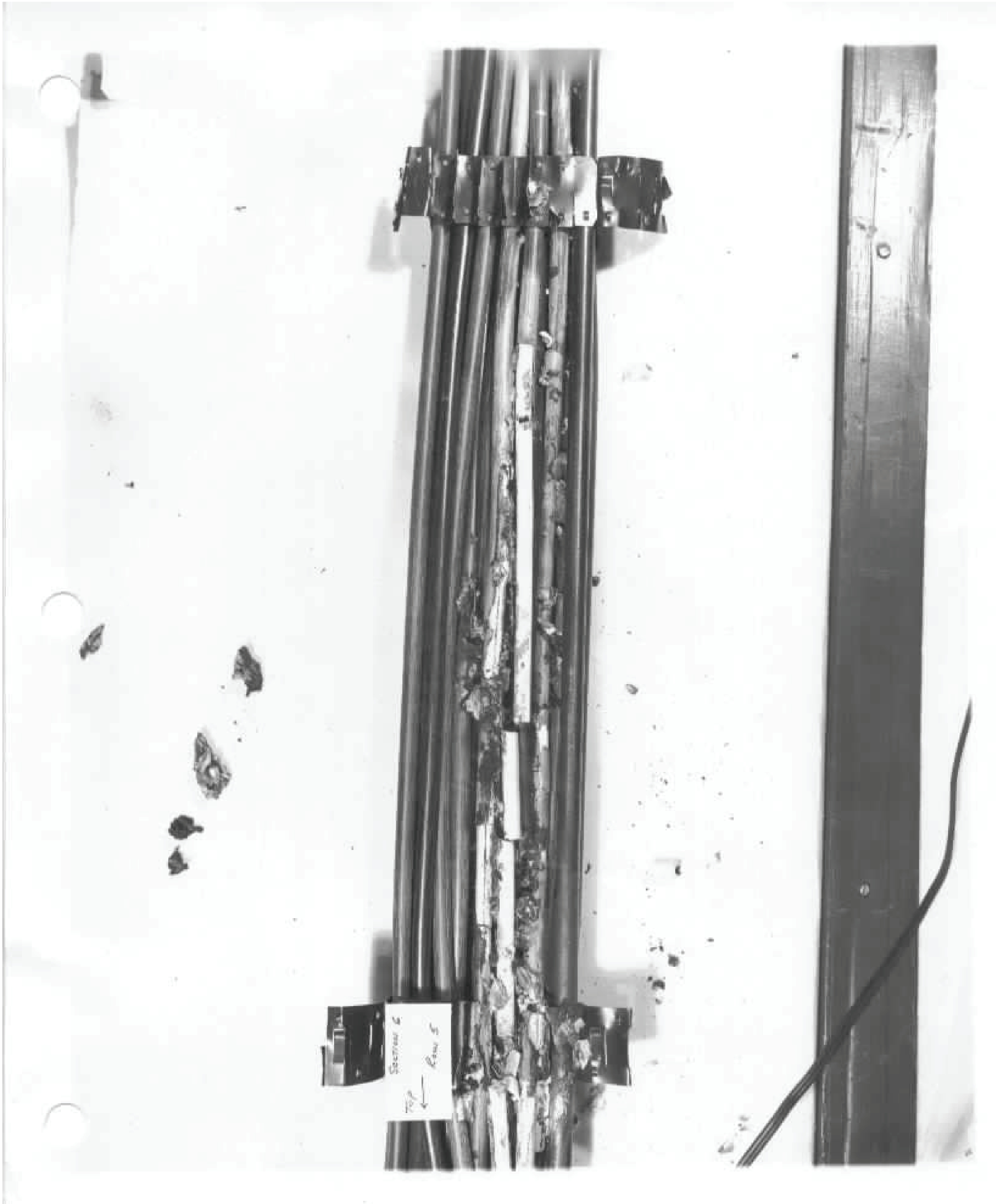
FLECHT RUN 9573





TOP
SECTION 4
ROW 3
TOP
SECTION 5
ROW 3

Appendix C Photograph of the Section of the Test Bundle from FLECHT Run 8874 that Incurred Runaway Oxidation


Section 6
Row 5
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Appendix D Experiments in which Zirconium-Steam Reaction Rates Occurred that Exceed the Rates Predicted by Computer Safety Models

I. Severe Accident Experiments in which Hydrogen Generation Rates Occurred that Exceed the Rates Predicted by Computer Safety Models

In Petitioner's comments on PRM-50-93/95 (page 5), dated April 7, 2011,[1] Petitioner quoted an OECD Nuclear Energy Agency report, published in 2001, which explicitly states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[2] PRM-50-93/95 argues that computer safety models using either the Baker-Just correlation or Cathcart-Pawel correlation—both among the available Zircaloy-steam oxidation correlations—under-predict the zirconium-steam reaction rates that would occur in loss-of-coolant accidents and severe accidents. However, NRC's draft interim reviews of PRM-50-93/95 on the CORA and LOFT LP-FP-2 experiments neither discuss nor mention Nuclear Energy Agency's statement, which pertains to the Baker-Just and Cathcart-Pawel correlations.

In fact, NRC's August 2011 Draft Interim Review ("DIR") of PRM-50-93/95, NRC concludes:

> The results of [the] CORA [experiments] do not suggest that the Cathcart-Pawel or Baker-Just correlations are non-conservative. The assertions made by the petition with regards to Cathcart-Pawel and Baker-Just are not substantiated by the CORA data. [3]

And NRC's September 2011 DIR of PRM-50-93/95, NRC concludes:

> A close examination of thermocouple data for LOFT LP-FP-2 found that the heatup rates below 2200ºF did not indicate presence of an exothermic "autocatalytic" reaction. The results of LOFT Test LP-FP-2 do not therefore suggest that the Cathcart-Pawel or Baker-Just correlations are

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[1] Mark Leyse, Comments on PRM-50-93/95, April 7, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111020046.

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

[3] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: NRC's ADAMS Documents, Accession Number: ML112211930, p. 3.

> non-conservative. The assertions made in PRM-50-93/95 with regards to Cathcart-Pawel and Baker-Just are not substantiated by the results of this LOFT test.[4]

(As discussed in Section I of Petitioner's letter with comments on NRC's DIRs of PRM-50-93/95, NRC has overlooked data that NRC provided in September 2011 demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F peak cladding temperature ("PCT") limit.)

It is unfortunate that NRC overlooked the Nuclear Energy Agency's statement that the available Zircaloy-steam oxidation correlations—which the Baker-Just and Cathcart-Pawel correlations are among—are not suitable for use in computer safety models to determine the increased hydrogen production in the CORA and LOFT LP-FP-2 experiments.

The Nuclear Energy Agency's statement pertains to the increased hydrogen production that would occur in severe accidents during a reflooding of an overheated reactor core.[5] A 1999 paper explains that "[n]o models are yet available to predict correctly the quenching processes in the CORA and LOFT LP-FP-2 tests. …the increased hydrogen production during quenching cannot be determined on the basis of the available Zircaloy/steam oxidation correlations."[6]

The Nuclear Energy Agency's statement does not pertain to the design basis accident temperature range. However, PRM-50-95—originally a 10 C.F.R. § 2.206 enforcement action petition, *which NRC decided to make into a petition for rulemaking*[7]—discusses boiling water reactor ("BWR") severe accident phenomena, in

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[4] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," September 2011, available at: NRC's ADAMS Documents, Accession Number: ML112650009, p. 5.

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

[6] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, Vol. 270, 1999, pp. 207-208.

[7] Mark Leyse, PRM-50-95, June 7, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101610121. (PRM-50-95 was originally a 10 C.F.R. § 2.206 enforcement action petition that Petitioner wrote on behalf of New England Coalition (NEC), dated June 7, 2010. In

addition to phenomena which would occur in the design basis accident temperature range: fuel cladding temperatures lower than the 2200°F PCT limit. Given that the Fukushima Dai-ichi accident occurred in March 2011 and that NRC has since performed simulations of BWR severe accidents with the MELCOR computer safety model, it would seem appropriate for NRC to acknowledge that MELCOR under-predicts the hydrogen generation rates that occur during a reflooding of an overheated reactor core.

II. Computer Safety Models Fail to Accurately Predict the Onset of the Fuel-Cladding Temperature Escalation that Commenced in the LOFT LP-FP-2 Experiment (in the Design Basis Accident Temperature Range)

As discussed in Section I of Petitioner's letter with comments on NRC's DIRs of PRM-50-93/95, the onset of the fuel-cladding temperature escalation commenced in the LOFT LP-FP-2 experiment when fuel-cladding temperatures were lower than the 2200°F PCT limit.

Computer safety models have failed to accurately predict the onset of the fuel-cladding temperature escalation that occurred in the LOFT LP-FP-2 experiment. Regarding a fairly recent computer safety model (ASTEC V1.3 code) simulation of the LOFT LP-FP-2 experiment, a 2010 paper, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17), but the onset of temperature escalation in the upper part of the CFM [center fuel module] was delayed.[8]

In "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17, the graph of the cladding-temperature values in the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment depicts that the onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature greater than 1700 K (2600°F); figure 17 also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature well

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October 2010, NRC published a notice in the Federal Register stating that it had determined that the NEC petition, met the requirements for a petition for rulemaking under 10 C.F.R. § 2.802.)

[8] G. Bandini *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," Progress in Nuclear Energy, 52, 2010, p. 155.

below 1500 K (2240°F).[9] Hence, the difference between the calculated and actual experimental value for the onset of the temperature escalation (at the 1.067 m elevation) is greater than 200 K (360°F)—a significant difference.

(It is noteworthy that, regarding the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment during reflood, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflooding were not reproduced by ASTEC due to lack of adequate modeling.[10])

III. An Experiment for which the Quantity of Hydrogen Produced by the Zircaloy-Steam Reaction at about 1800°F Is Under-Predicted by Computer Safety Models: The FRF-1 Experiment

The FRF-1 experiment—conducted in the TREAT facility[11]—was not a large-scale experiment yet Union of Concerned Scientists and the authors of a report on the FRF-1 experiment[12] claimed that, as of 1971, it simulated "the most realistic loss-of-coolant accident conditions of any experiment to date."[13]

(The FRF-1 experiment is discussed in Petitioner's comments on PRM-50-93/95, dated November 23, 2010 (pages 37-45),[14] and dated July 27, 2011 (pages 1-2);[15] and in Appendix A to Petitioner's comments on PRM-50-93/95, dated November 23, 2010, there is a graph depicting the maximum cladding temperatures which occurred in the FRF-1 experiment.)

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[9] *Id*.
[10] *Id*.
[11] The First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") was conducted in the Transient Reactor Test Facility ("TREAT").
[12] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971.
[13] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer-Verlag, New York, 2000, p. 43.
[14] Mark Leyse, Comments on PRM-50-93, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[15] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.

Data from the FRF-1 experiment indicates that computer safety models under predict the quantity of hydrogen produced by the Zircaloy-steam reaction. In the experiment, at fuel rod temperatures of about 1800°F, the Zircaloy-steam reaction generated 1.2 ± 0.6 liters of hydrogen. In the Indian Point Unit 2 ("IP-2") licensing hearing, Westinghouse Electric, which had performed experimental simulations of loss-of-coolant accidents, and conducted computer simulations of such accidents, testified that their computer safety models predicted that there would be no zirconium-steam reaction at 1800°F—that no hydrogen would be produced in a loss-of-coolant accident if local temperatures of the fuel rods were to reach 1800°F.[16]

In the IP-2 licensing hearing, Dr. Jack Roll of Westinghouse contended that data from the FRF-1 experiment was not reliable, because "the measurement of the extent of [zirconium-steam] reaction was in fact by an inferred route, and there were no direct measurements taken," that "[t]here was a large uncertainty in the measurement of total hydrogen evolution during the experiment," and that there was "an uncertainty in the temperatures of the fuel [rods] during the experiment."[17] Westinghouse concluded that it is not possible to know if the data from the FRF-1 experiment actually demonstrated that the extent of the zirconium-steam reaction was higher (or much higher) than would be predicted by computer safety models.

Unfortunately, there was not a means to confirm if Westinghouse's claims were correct or not, because the Atomic Energy Commission decided to discontinue funding for the TREAT facility loss-of-coolant accident experimental program.[18] The FRF-1 experiment could not be replicated; its results could not be confirmed.

[16] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, available at: NRC's ADAMS Documents, Accession Number: ML100350644, pp. 2152-2153.
[17] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, available at: NRC's ADAMS Documents, Accession Number: ML100350642, pp. 2297-2299.
[18] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.

IV. Problems with the Explanation for Why Low-Temperature Oxidation Rates Are Under-Predicted for the CORA-16 Experiment

As stated in PRM-50-95 (pages 12, 13, 26, 27) and in Petitioner's comments on PRM-50-93/95, March 15, 2010 (page 30),[19] dated April 12, 2010 (page 8),[20] dated November 24, 2010 (page 7),[21] dated July 30, 2011 (page 16),[22] and April 16, 2012 (pages 6, 7, 9, 11, 20),[23] when investigators compared the results of the CORA-16 experiment—a BWR 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."[24]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in the CORA-16 experiment.[25] (In Petitioner's comments on PRM-50-93/95, dated April 16, 2012 (pages 5-13),[26] Petitioner provided information indicating that it is *unlikely* that cladding strain increased the zirconium-steam reaction rates that occurred in the CORA-16 experiment; it is certainly *unsubstantiated* that cladding strain increased reaction rates.)

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[19] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.

[20] Mark Leyse, Comments on PRM-50-93/95, April 12, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101020564.

[21] Mark Leyse, Comments on PRM-50-93/95, November 24, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340248; NRC dates these comments November 23, 2010.

[22] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

[23] Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12109A084.

[24] 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.

[25] 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.

[26] Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12109A084.

In NRC's 2011 evaluation of the CORA-16 experiment, NRC stated that an ORNL paper, "In-Vessel Phenomena—CORA," noted that in CORA-16, "cladding strain could be a factor and that cladding strain and significant oxidation occurred simultaneously."[27] However, NRC erroneously observed that "In-Vessel Phenomena—CORA" "provided an analytical adjustment that improved the timing prediction with respect to the measured temperatures."[28]

In fact, the ORNL paper's authors employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16.[29] There are three graphs in the ORNL paper depicting cladding temperature plots from different cladding elevations (550 mm, 750 mm, and 950 mm) of "heated rod 5.3" in CORA-16:[30] each plot illustrates that cladding temperatures were greater in the experiment than computer safety models—using the available zirconium-steam reaction correlations—initially predicted (*with no enhancement*), indicating that zirconium-steam reaction rates were also under-predicted. Each graph also depicts predicted cladding temperature plots that were computer generated by using a simple *multiplier* to *enhance* the predicted zirconium-steam reaction rates (and the amount of heat the zirconium-steam reaction produced). By using the multiplier the predicted reaction rates were matched closer to the reaction rates that occurred in the experiment; hence, the multiplier also helped the predicted cladding temperatures match the cladding temperatures that occurred in the experiment.

NRC also erroneously stated that "In-Vessel Phenomena—CORA," did not report that computer safety models under-predicted zirconium-steam reaction rates in CORA-16:[31] a simple glance at the three graphs described above[32] reveals that the paper reported that reaction rates were under-predicted. And a second ORNL paper explicitly states that

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[27] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: NRC's ADAMS Documents, Accession Number: ML112211930, p. 3.
[28] *Id*.
[29] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[30] See Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[31] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[32] See Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.

the low-temperature (1652°F to 2192°F) oxidation that occurred in CORA-16 was under-predicted.[33] (Petitioner has quoted the second ORNL paper in a number of different comments on PRM-50-93/95 that Petitioner has sent to NRC.)

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, NRC pointed out that an NRC report, NUREG/CR-4412,[34] "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" [emphasis not added].[35]

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.[36]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[37] 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;"[38] the report provides a graph of the "rather sparse"[39] data. The graph indicates that the general trend

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[33] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.
[34] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, available at: NRC's ADAMS Documents, Accession Number: ML083400371.
[35] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[36] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 27.
[37] *Id*., pp. 27, 30.
[38] *Id*., p. 30.
[39] *Id*.

is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[40]

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]."[41]

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."[42] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[43] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a very brief period of time, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain *may* only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater[44] (in a LOCA environment in which local cladding temperatures would be increasing rapidly). 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

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[40] *Id*., p. 29.
[41] *Id*., p. 30.
[42] *Id*.
[43] 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.
[44] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 30.

actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F temperature range for CORA-16.

There is also one phenomenon NRC did not consider in its 2011 analysis of CORA-16: "[t]he swelling of the [fuel] cladding…alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,"[45] 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.

In NRC's 2011 evaluation of CORA-16, NRC concluded that the fact zirconium-steam reaction rates were under-predicted by computer safety models—using the available zirconium-steam reaction correlations—"is inadequate as a basis to revise regulations or invalidate the use of [the] Baker-Just and Cathcart-Pawel [correlations] for design basis calculations of oxidation."[46] (The Baker-Just and Cathcart-Pawel correlations are among the available zirconium-steam reaction correlations.) NRC's conclusion is unsubstantiated, as the information presented in this section indicates. When NRC chooses to invalidate experimental data, which is important for simulating accidents, with unsubstantiated postulations, NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative computer safety models.[47]

A plausible explanation for why zirconium-steam reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F by computer

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[45] 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, available at: NRC's ADAMS Documents, Accession Number: ML041950059, p. 43.

[46] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.

[47] Charles Miller, *et al*., NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," SECY-11-0093, July 12, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111861807, p. 3.

safety models would be that the currently used zirconium-steam reaction correlations are inadequate for use in computer safety models.

V. Oxidation Models Are Not Able to Predict the Fuel-Cladding Temperature Escalation that Commenced at Approximately 1880°F in the PHEBUS B9R-2 Test

(The information discussed in this section was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

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.[48] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[49]

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*[50] [emphasis added].

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to the cladding of high burnup fuel rods. The

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[48] 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.

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

[50] *Id*., p. 126.

PHEBUS B9R-2 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just and Cathcart-Pawel correlations, are inadequate for use in computer safety models.