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July 30, 2011

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

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 AND PRM-50-95; NRC-2009-0554

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TABLE OF CONTENTS

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I. Statement of Petitioner's Interest

On November 17, 2009, Mark Edward Leyse, Petitioner (in these comments "Petitioner" means Petitioner for PRM-50-93 and sole author of PRM-50-95), submitted a petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the Nuclear Regulatory Commission ("NRC") make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments;[1] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[2, 3]

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models, Sources of Heat during the LOCA, Metal-Water Reaction Rate*, to require that

[1] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

[2] It can be extrapolated from experimental data from Thermal-Hydraulic Experiment 1, conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, that, in the event a large break ("LB") LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater and an average fuel rod power of approximately 0.37 kW/ft or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LB LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

[3] It is noteworthy that in 1975, Fred C. Finlayson stated, "[r]ecommendations are made for improvements in criteria conservatism, especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates);" see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, Abstract, p. iii.

the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[5]

On June 7, 2010, Petitioner, submitted an enforcement action 10 C.F.R. § 2.206 petition on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

On October 27, 2010, NRC published in the Federal Register a notice stating that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner submitted on behalf of NEC, meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[6]

When Petitioner wrote the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner did not foresee that NRC would docket it as PRM-50-95. PRM-50-95 was written and framed as a 10 C.F.R. § 2.206 petition, not as a 10 C.F.R. § 2.802 petition; however, it is laudable that NRC is reviewing the issues Petitioner raised in PRM-50-95.

II. Supplementary Information to PRM-50-93 and PRM-50-95

In these comments on PRM-50-93 and PRM-50-95, Petitioner discusses data that indicates that analyses using the Baker-Just and Cathcart-Pawel correlations under-

[4] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would commence in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA.

[5] Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.

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

predict the amount of heat that Zircaloy oxidation generated in Thermal-Hydraulic Experiment 1 ("TH-1") test no. 128, a thermal hydraulic experiment simulating LOCA conditions, conducted with a full-length Zircaloy 32-rod UO_2 fuel bundle.[7]

Analyses using the Baker-Just and Cathcart-Pawel correlations would also most likely under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 130. In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[8] So the peak cladding temperature increased by 190°F after the reactor shutdown, because of the heat generated from the metal-water reaction.

It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown.

The data from TH-1 test no. 128 (and most likely also TH-1 test no. 130) is another piece of evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

Perhaps it could be argued that there may have been a problem in TH-1 test no. 128 (although no problems were reported in TH-1 test no. 128[9]) and that therefore it is not certain that analyses using the Baker-Just and Cathcart-Pawel correlations under-

[7] NRC, "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-76)," Attachment 1, Federal Register Notice, June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 18-19 (hereinafter "Denial of PRM-50-76," Attachment 1).

[8] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119, p. 13 (hereinafter "Prototypic Thermal-Hydraulic Experiment").

[9] Sometimes thermal hydraulic experiments simulating LOCA conditions reach different overall PCTs, even when they are conducted with similar test parameters. However, "Prototypic Thermal-Hydraulic Experiment" does not report that there were any problems with TH-1 test no. 128 and the data from the TH-1 tests is fairly consistent. See C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882.

predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128. However, any such claims *cannot* be substantiated without conducting more tests with the same parameters as TH-1 test no. 128, to find out if the overall PCT would in fact be more than 100°F lower than it was in TH-1 test no. 128, as analyses using the Baker-Just and Cathcart-Pawel correlations would predict.

Furthermore, in the interest of conservatism and to uphold NRC's congressional mandate to protect the lives, property, and environment of the people of the United States of America, NRC needs to consider the data from TH-1 test no. 128 (and most likely also TH-1 test no. 130), as evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

It is clear that NRC needs to conduct pressurized water reactor ("PWR") thermal hydraulic experiments, simulating LOCA conditions, with a realistic range of different reflood rates, with full-length zirconium alloy[10] multi-rod bundles (comprised of either fuel rods sheathing UO_2 fuel or realistic, pressurized[11] and non-pressurized fuel rod simulators), to investigate the behavior of such bundles when reaching local cladding temperatures of up to 2200°F or higher.

The conductors of such experiments would be able to measure Zircaloy oxidation rates.

A. Analyses Using the Baker-Just Correlation Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in Thermal-Hydraulic Experiment 1 Test No. 128

In this section Petitioner compares data from the TH-1 tests—conducted at National Research Universal ("NRU") at Chalk River, Ontario, Canada to evaluate the

[10] Zircaloy fuel cladding is a particular type of zirconium alloy fuel cladding. ZIRLO and M5 fuel cladding materials are also zirconium alloys; however, they are different zirconium alloys than Zircaloy.

[11] It is noteworthy that "Prototypic Thermal-Hydraulic Experiment" states that experiments with pressurized fuel rod simulators for materials deformation tests would "concentrate on evaluating not only ballooning and rupture but also the added effects on the thermal hydraulic behavior of flow blockage." See C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 2.

thermal-hydraulic behavior of a full-length Zircaloy 32-rod UO_2 fuel bundle during the heatup, reflood, and quench phases of a large-break LOCA[12]—with an example of a prediction (using the Baker-Just correlation) of the behavior of Zircaloy UO_2 fuel rods under LOCA conditions, which is discussed in "PWR FLECHT Final Report."[13] In this comparison, it is evident that analyses using the Baker-Just correlation under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128.

In TH-1 test no. 128, with a peak power of 0.55 kW/ft,[14] a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1604°F, the overall PCT was 1991°F (an increase of 387°F).[15] And in the "PWR FLECHT Final Report" example, the UO_2 Zircaloy fuel assembly, with a peak power of 1.24 kW/ft, a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1600°F, was predicted to have an overall PCT of approximately 1880°F (an increase of approximately 280°F).[16]

So with similar parameters (but with a lower fuel rod power) TH-1 test no. 128 had an overall PCT increase that was more than 100°F greater than the overall PCT increase predicted in the UO_2 Zircaloy fuel assembly example discussed in "PWR FLECHT Final Report." This indicates that analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions.

Data from TH-1 test no. 128 is another piece of evidence that indicates the Baker-Just correlation is not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

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[12] NRC, "Denial of PRM-50-76," Attachment 1, pp. 18-19.

[13] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 4-2, 4-3, 4-4 (hereinafter "PWR FLECHT Final Report").

[14] 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, located in ADAMS Public Legacy, Accession Number: 8104140024, pp. 6-13, 6-15 (hereinafter "Safety Analysis Report").

[15] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.

[16] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," WCAP-7665, pp. 4-2, 4-3, 4-4.

1. Analyses Using the Cathcart-Pawel Correlation would also Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in TH-1 Test No. 128

As discussed in section II.A., analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128. This means that analyses using the Cathcart-Pawel correlation would also under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128.

Discussing the Baker-Just and Cathcart-Pawel correlations in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K," the NRC states:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F...[17]

(It is noteworthy that data from TH-1 test no. 128 indicates that the Baker-Just correlation is *not* substantially conservative at 2200°F. In fact, data from TH-1 test no. 128 indicates that the Baker-Just correlation is non-conservative.)

So at the same temperatures, analyses using the Cathcart-Pawel correlation predict a lower heat generation rate than analyses using the Baker-Just correlation predict. Therefore, analyses using the Cathcart-Pawel correlation would under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions.

Data from TH-1 test no. 128 is another piece of evidence that indicates the Cathcart-Pawel correlation is not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

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[17] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 3; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.

2. It is Probable that in Addition to TH-1 Test No. 128, Analyses Using the Baker-Just and Cathcart-Pawel Correlations would Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in Other Tests in Thermal-Hydraulic Experiment 1

One of the guidelines for the TH-1 tests was that the fuel cladding temperatures would *not* exceed 1900°F[18]—300°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In three of the TH-1 tests the overall PCTs exceeded 1900°F, exceeding the PCTs predicted for the tests. The overall PCTs of TH-1 test nos. 127, 128, and 130 were 1991°F, 1991°F, and 2040°F, respectively. So it is probable that the Baker-Just and Cathcart-Pawel correlations would under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test nos. 127 and 130.

As discussed in section II., in TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[19]

a. TH-1 Test No. 130

In Atomic Energy Commission ("AEC") responses to questions submitted by Anthony Z. Roisman, pertaining to the IP-2 licensing hearing, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[20]

Indeed, computer codes using the Baker-Just correlation may calculate that the Zircaloy-steam reaction is negligible below 1900°F; however, experimental data from

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[18] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, p. 3-3.
[19] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[20] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

multi-rod thermal hydraulic experiments demonstrates that the Zircaloy-steam reaction is substantial below 1900°F.

Data from TH-1 test no. 130 demonstrates that the Zircaloy-steam reaction is *not* negligible below 1900°F.

In TH-1 test no. 130, there was a peak power of 0.55 kW/ft[21] and a reflood rate of 0.74 in./sec.[22] 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.[23]

In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[24] So because of the heat generated from the metal-water reaction, the peak cladding temperature increased by 190°F, after the reactor shutdown.

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 been greater than 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.

(TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average and peak power of TH-1 test no. 130 would have been 0.37 kW/ft[25] and 0.55 kW/ft,[26] respectively, in the pre-transient phase of the test.)

Of course, in the event of an actual LOCA, the energy from decay heating would not suddenly terminate if cladding temperatures were to reach approximately 1850°F.

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[21] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.
[22] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, 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.
[23] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[24] *Id.*
[25] *Id.*, p. 10.
[26] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.

The data of TH-1 test no. 130 indicates, in the event of a LOCA, at a PWR, with high probability, if peak cladding temperatures reached temperatures of approximately 1850°F, the Zircaloy cladding would begin to rapidly oxidize, and that—with the combination of heat generated by the metal-water reaction and decay heat—the oxidation would become autocatalytic and cladding temperatures would start increasing at a rate of tens of degrees Fahrenheit per second. Within a period of approximately 60 seconds peak cladding temperatures would increase to 3000°F or greater; the melting point of Zircaloy is approximately 3308°F.[27]

(Of course, as stated above, there would have been a small amount of actual decay heat in the bundle of TH-1 test no. 130, after the reactor shutdown; however, it would have been substantially lower than the amount of decay heat in a counterpart bundle, in the event of a LOCA.)

Analyses using the Baker-Just and Cathcart-Pawel correlations would most likely under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 130. It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown.

The data from TH-1 test no. 130 is most likely another piece of evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

[27] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

B. A More Detailed Discussion of the Example of a Prediction (Using the Baker-Just Correlation) of the Behavior of Zircaloy UO_2 Fuel Rods under LOCA Conditions

Regarding the example—discussed in section II.A. above—of a prediction (using the Baker-Just correlation) of the behavior of Zircaloy UO_2 fuel rods under LOCA conditions, "PWR FLECHT Final Report" states:

> Figure 4-1[28] shows a comparison of the temperature response of boron nitride-stainless steel (BN-SS), boron nitride-Zircaloy (BN-Zr), and uranium dioxide-Zircaloy (UO_2-Zr) rods for 6 and 2 in./sec. flooding rates. The curves were generated by a conduction code using heat transfer coefficients obtained from stainless steel PWR FLECHT tests.[29] The gap coefficients for the BN and UO_2 cases were 10,000 and 500 $Btu{\cdot}hr^{-1}{\cdot}ft^{-2}{\cdot}°F^{-1}$, respectively. Initial temperature distributions were assumed to be uniform in the BN cases, whereas a 59°F difference between peak pellet and initial clad temperature was used in the UO_2 cases. Metal-water reaction was predicted in the Zircaloy cases using the Baker-Just parabolic rate equation (reference 4).[30] The BN-SS curves are generally representative of the behavior of Group I and II PWR FLECHT heater rods and were found to be in good agreement with the measured temperature response for the same run conditions. The BN-Zr curves are representative of the behavior of Group III PWR FLECHT rods while the UO_2-Zr curves are representative of reactor fuel rod response, assuming the BN-SS heat transfer coefficients apply.[31]

Figure 4-1., "Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods,"[32] depicts temperature plots of the BN-SS, BN-Zr, and UO_2-Zr representative

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[28] See Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods.

[29] The heat transfer coefficients obtained from stainless steel PWR FLECHT tests are used in Appendix K to Part 50 ECCS evaluation calculations. Appendix K to Part 50—ECCS Evaluation Models (I)(D)(5), *Required and Acceptable Features of the Evaluation Models, Post-Blowdown Phenomena, Refill and Reflood Heat Transfer for Pressurized Water Reactors*, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the stainless steel] FLECHT results [reported in "PWR FLECHT Final Report"]."

[30] Baker, L., Just, L. C., "Studies of Metal-Water Reactions at High Temperatures. III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," Argonne National Laboratory, ANL-6548, May 1962, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050550198.

[31] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," WCAP-7665, pp. 4-2, 4-4.

[32] See Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods.

examples for the flooding rates of 6 in./sec. and 2 in./sec. Petitioner will only discuss the representative examples with the 2 in./sec. flooding rate.

For the BN-SS, BN-Zr, and UO_2-Zr representative examples, with the 2 in./sec. flooding rate, the maximum overall PCTs of all three representative examples are lower than 1900°F.

The UO_2-Zr representative example (with a temperature plot drawn with a solid line) has the middle overall PCT value of approximately 1880°F.

The BN-SS and BN-Zr representative examples have overall PCTs of approximately 1840°F and 1890°F, respectively.

Figure 4-1., "Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods" lists some of the parameters for the BN-SS, BN-Zr, and UO_2-Zr representative examples: initial PCT of 1600°F, flooding rate of 2 in./sec., pressure of 60 psia, peak power of 1.24 kW/ft, inlet coolant temperature of 150°F.

C. A Comparison between the TH-1 Tests and the PWR FLECHT Tests

Regarding the fact that the TH-1 tests can be compared to the PWR FLECHT tests, "Safety Analysis Report" states:

> The largest body of information bearing on fuel rewetting or quench is that of the Westinghouse FLECHT experimental series. Cadek (1972) and Rosal (1978) have written reports that describe the experiments and results and cover the same range of reflood rates as in the [TH-1, TH-2, and TH-3] tests proposed for NRU. ...
>
> The NRU [TH-1, TH-2, and TH-3] LOCA [tests] will be quite similar to that of the [PWR] FLECHT tests, with the following major exceptions:
>
> 1) NRU LOCA has nuclear-heated rods; FLECHT has electrically-heated rods.
>
> 2) NRU has Zircaloy-clad rods; FLECHT has stainless steel-clad rods.
>
> 3) NRU has peak-to-average axial power distribution of 1.51; FLECHT's peak-to-average axial power distribution is 1.66.
>
> 4) The NRU test has 32 rods. The tests have different rod surface-to-shroud surface ratios.

5) Pre-transient steam cooling in the NRU tests distorts the initial axial temperature distribution.[33]

"Safety Analysis Report" also points out another major difference between the NRU Th-1 tests and the PWR FLECHT tests: "At high cladding temperatures the steam will react with the Zircaloy cladding as given by: $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ [+ ΔH_R]."[34] In which the heat of reaction, ΔH_R, is 559kJ (143 kcal) per mole Zr.[35]

However, the main point of "Safety Analysis Report" is that the TH-1 tests can be compared to the PWR FLECHT tests.

And discussing the goals of the TH-1 tests and other NRU experiments, "Prototypic Thermal-Hydraulic Experiment" states:

> The data [from the NRU experiments] will be used to assess various calculational models for reactor safety analyses and conclusions derived from the large series of electrically heated tests and smaller scale in-pile tests being conducted elsewhere. The test data provide information for evaluating cooling degraded cores as a result of either an accident or an off-normal operating transient. ...
>
> The results of the program will be used to provide data for model calibration or to help define the primary heat transfer mechanisms for new analytical models. The geometry, mass flux, heat capacity, and materials are all prototypic, which eliminates much of the uncertainty of prior test results from other programs. Major concerns of other programs, such as length of fuel bundle or type of heating, [electrical instead of nuclear], should be answered by these test results.[36]

So one of the goals of the TH-1 tests was to use the test data to "assess...conclusions derived from the large series of electrically heated tests."[37]

1. A Comparison of the Results of TH-1 Test No. 107 and PWR FLECHT Run 3724: Tests with Lower PCTs at the Onset of Reflood

It is informative to compare the results of TH-1 test no. 107 and PWR FLECHT run 3724.

[33] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 9-31, 9-32.
[34] *Id.*, p. 9-40.
[35] *Id.*, p. 9-41.
[36] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, pp. 2-3.
[37] *Id.*, p. 2.

Parameters of the two tests:

1) TH-1 test no. 107 (Zircaloy) had a peak power of 0.55 kW/ft,[38] reflood rate of 1.9 in./sec., PCT at the onset of reflood of 1154°F and an overall PCT of 1578°F (an increase of 424°F);[39]

2) PWR FLECHT run 3724 (stainless steel) had a peak power of 1.24 kW/ft, reflood rate of 1.9 in./sec., a PCT at the onset of reflood of 1187°F, and an overall PCT of 1614°F (an increase of 427°F).[40]

TH-1 test no. 107 and PWR FLECHT run 3724, with lower PCTs at the onset of reflood of 1154°F and 1187°F, respectively, had cladding temperature increases of 424°F and 427°F, respectively. So at temperatures where the oxidation of Zircaloy does not produce much heat, the results of TH-1 test no. 107 and PWR FLECHT run 3724 are similar. This indicates that the results of the TH-1 tests can be compared with the results of PWR FLECHT tests.

D. When NRC Denied PRM-50-76, it Overlooked Data which Indicates that Analyses Using the Baker-Just and Cathcart-Pawel Correlations Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in TH-1 Test No. 128

In 2005, NRC denied PRM-50-76,[41] which addressed the fact that the Baker-Just and Cathcart-Pawel correlations are deficient because they were not developed to consider how heat transfer would affect Zircaloy-steam reaction kinetics in the event of a LOCA.[42]

In 2005, regarding the fact that data from isothermal tests are used for the development of Zircaloy-steam oxidation correlations, NRC stated:

> For the development of oxidation correlations, limited by oxygen diffusion into the metal, well-characterized isothermal tests are more important than the complex thermal hydraulics suggested by [Robert H. Leyse]. [Robert H. Leyse's] suggested use of complex thermal-hydraulic conditions would be counter-productive in reaction kinetics tests because temperature

[38] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.
[39] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[40] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-5.
[41] NRC, "Denial of PRM-50-76," Attachment 1.
[42] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009.

> control is required to develop a consistent set of data for correlation development. Isothermal tests allow this needed temperature control. *It is more appropriate to apply the developed correlations to more prototypic transients (including complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations are indeed limiting.* This is what was done by Westinghouse in WCAP-7665, by Cathcart and Pawel in NUREG-17 and by the NRC in its technical safety analysis of PRM-50-76[43] [emphasis added].

So "Denial of PRM-50-76," Attachment 1 states that the Baker-Just and Cathcart-Pawel correlations were used in analyses of prototypic transients (including those with complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations were limiting.

First, as pointed out in Petitioner's comments on PRM-50-93 and PRM-50-95, dated April 7, 2011, NRC overlooked the fact that it was reported in 2001, in an OECD Nuclear Energy Agency report, 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."[44]

Second, NRC overlooked the fact that ORNL reports from 1990 and 1991, discussing the CORA-16 experiment, explicitly state that "[c]ladding oxidation was not accurately predicted by available correlations"[45] and that "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted."[46]

The fact that analyses using the available Zircaloy-steam oxidation correlations under predict the oxidation rates that occur in large-scale integral severe fuel damage

[43] NRC, "Denial of PRM-50-76," Attachment 1, pp. 21-22.

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

[45] 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.

[46] 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.

experiments indicates that the available Zircaloy-steam oxidation correlations are also inadequate for use in ECCS evaluation models predicting the oxidation rates that would occur in the event of a LOCA.

Third, as discussed in sections II., II.A., and II.A.1. above, NRC overlooked the fact that analyses using the Baker-Just and Cathcart-Pawel correlations under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions. TH-1 test no. 128 (with a lower fuel rod power but with otherwise similar parameters) had an overall PCT increase that was more than 100°F greater than the overall predicted PCT increase of the UO_2 Zircaloy fuel assembly example, discussed in "PWR FLECHT Final Report."

Data from TH-1 test no. 128 is another piece of evidence that indicates the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

(It is noteworthy that NRC's technical safety analysis of PRM-50-76 lists "Prototypic Thermal-Hydraulic Experiment," which reports on the data of the TH-1 tests, as reference 17.[47]

And noteworthy that NRC's technical safety analysis of PRM-50-76 states:

> NRC has continued to study complex thermal hydraulic effects on ECCS heat transfer processes during accident conditions related to LOCAs[48] consistent with Commission direction. The NRC funded more than 50 Zircaloy clad bundle reflood experiments at the NRU reactor [the program the TH-1 tests were part of].[49, 50]

It is also noteworthy that, in NRC's "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-

[47] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041210109, p. 12.

[48] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333.

[49] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882 and C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208.

[50] 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, p. 10.

76)," NRC stated that it was "reviewing...data from [the early '80s, from the program the TH-1 tests were part of,] to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[51])

Fourth, there is no metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation, because Westinghouse did not obtain such data. So neither Westinghouse nor NRC applied the Baker-Just correlation to metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation; furthermore, NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation.[52]

Fifth, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments, in which local steam starvation conditions are postulated to have occurred[53]—during PWR FLECHT runs 8874 and 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

So Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to oxide layers on the bundles from PWR FLECHT runs 8874 and 9573 were to locations that most likely were steam starved: those are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in ECCS evaluation calculations.

III. CONCLUSION

It is unfortunate that NRC has ignored data from TH-1 test no. 128, a multi-rod bundle thermal hydraulic experiment, that indicates that the Baker-Just and Cathcart-

[51] NRC, "Denial of PRM-50-76," Attachment 1, p. 19.
[52] Westinghouse obtained a total of 13, 2, 15, and 3 metallurgical samples from PWR FLECHT Zircaloy runs 2443, 2544, 8874, and 9573, respectively. See F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," pp. B-2, B-3.
[53] 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.

Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA.

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 would help improve public and plant-worker safety.

Respectfully submitted,

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Dated: July 30, 2011

Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods[1]

[1] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 4-3.

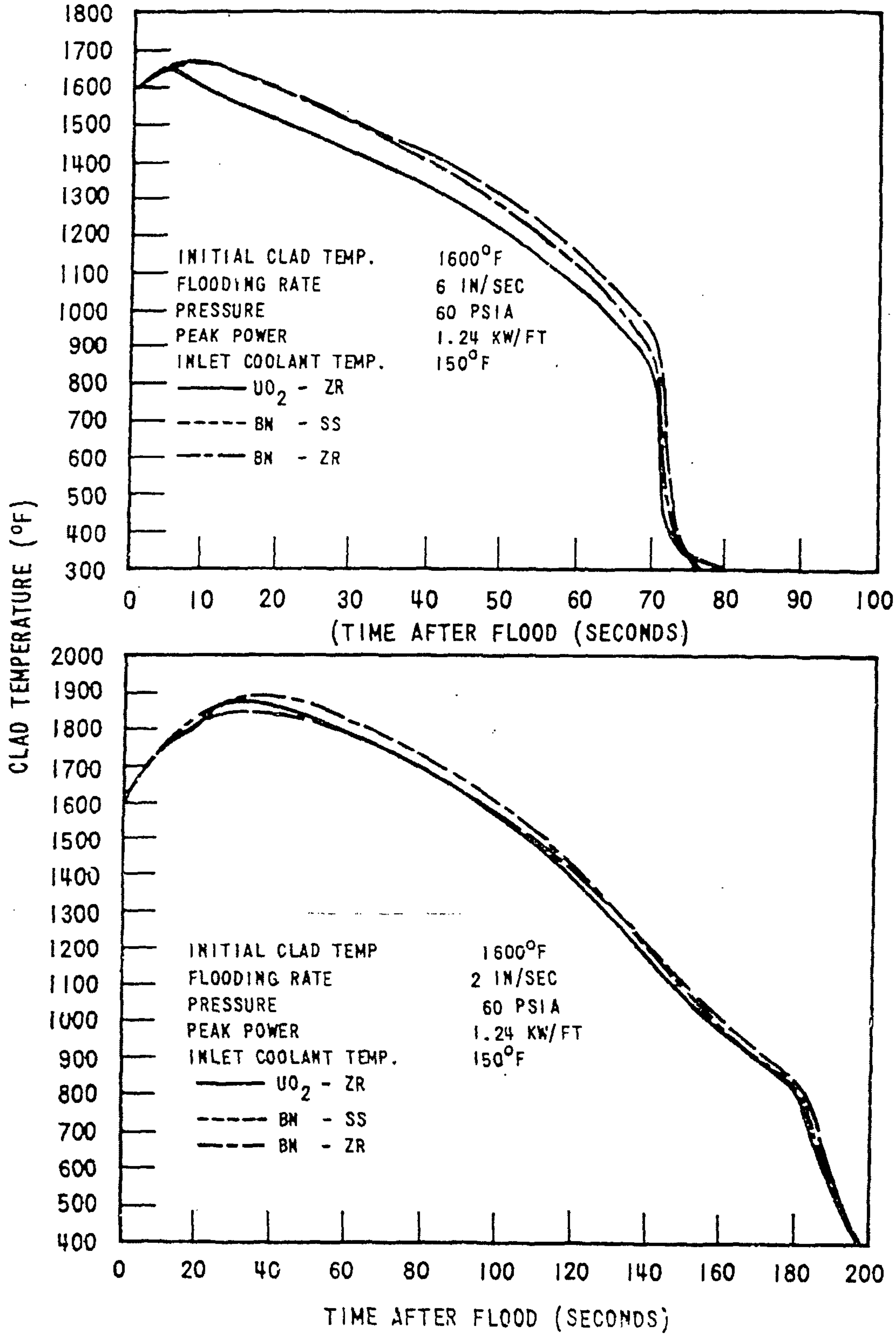

Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Saturday, July 30, 2011 6:45 PM
To: Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul
Subject: NRC-2009-05 54 (Fifth)
Attachments: Comment on PRM-50-93 and PRM-50-95 July 2011.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Edward Leyse's, Petitioner's, fifth response, dated July 30, 2011, 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 Edward Leyse