From:	Mark Leyse
To:	Boska, John
Cc:	Deborah Brancato; Phillip Musegaas; Dudley, Richard
Subject:	Supplementary Information to the PRB Meeting (July 11, 2011)
Date:	Thursday, July 21, 2011 2:25:23 AM
Attachments:	Supplement to PRB Teleconference July 11, 2011.pdf

Dear Mr. Boska:

Attached is supplementary Information to the PRB meeting we had on July 11, 2011, regarding the Riverkeeper 2.206 petition on Indian Point. This information covers material I discussed, with references for some of the reports I referred to in the meeting. I think it might be helpful to the PRB to have this information; I had said I would send it.

Would you please attach this information to the transcript for the meeting?

Thank you,

Mark Leyse

Supplementary Information for the Transcript of the Petition Review Board Meeting (July 11, 2011), Regarding the Riverkeeper 2.206 Petition on Indian Point

In 1971, in the Indian Point Unit 2 licensing hearing, intervenors argued that data from the First Transient Experiment of a Zircaloy Fuel Rod Cluster (FRF-1 experiment) indicates that ECCS evaluation models under-predict the amount of hydrogen produced in that experiment. This in turn meant that ECCS evaluation models would under-predict the amount of hydrogen produced in the event of a LOCA. The FRF-1 experiment was "performed with a seven-rod bundle of 27 [inch] long Zircaloy-clad UO_2 fuel rods in [a] flowing steam atmosphere,"[1] in the TREAT facility.

It is reported that, in the FRF-1 experiment, at cladding temperatures of approximately 1800°F, the Zircaloy-steam reaction generated 1.2 ± 0.6 liters of hydrogen.[2] Intervenors argued that data from FRF-1 indicates that ECCS evaluation models using the Baker-Just correlation under-predict Zircaloy-steam reaction rates at 1800°F. The AEC had stated that at 1800°F, the Zircaloy-steam reaction is predicted to be "negligible"[3] and, in the IP-2 licensing hearing, Westinghouse testified that no Zircaloy-steam reaction would be predicted at 1800°F.[4]

However, Westinghouse also argued that there had been problems with temperature measurements in the FRF-1 experiment, that there had been "an uncertainty in the temperatures of the fuel [cladding] during the experiment"[5] and that "one cannot make a direct inference on reported temperatures and lead yourself to the conclusion that

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[1] 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, Abstract.

[2] *Id*., p. 16.

[3] 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.

[4] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, p. 2152.

[5] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350642, p. 2298.

the extent of zirc-water reaction was higher or much higher than would have been predicted by Baker-Just."[6]

Instead of conducting a series of more tests in the TREAT facility (perhaps they conducted one additional test), the transient test program in the TREAT facility, for Zircaloy-clad fuel rods with UO_2 fuel, was terminated due to a lack of funding[7] and "[s]upport of [Oak Ridge] work on fuel rod failure [was] terminated at the end of FY-71."[8]

In the IP-2 licensing hearing, Union of Concerned Scientists pointed out that "[t]he authors of that Oak Ridge report, ORNL-4635,[1] contend[ed] that [the FRF-1 experiment] is the most realistic simulation of loss-of-coolant accident conditions to date,"[9] up to 1971.

Westinghouse disagreed with the authors of ORNL-4635, opining that the four Zircaloy tests conducted in the PWR FLECHT program provided a more realistic representation of the Zircaloy-steam reaction in a LOCA environment, than the FRF-1 experiment; and that the PWR FLECHT results were in "very good agreement with the Baker-Just equation."[10]

In the last PRB meeting, I criticized Westinghouse's examinations of the oxide samples that were taken from rods from the four Zircaloy PWR FLECHT tests. To repeat, Westinghouse did not obtain samples from the locations of the rods from FLECHT runs 8874 and 9573 that incurred runaway oxidation. And it is likely that the sections of the bundles that Westinghouse did examine from runs 8874 and 9573 were steam starved.

In the last PRB meeting, I did not include FLECHT run 8874; I only mentioned run 9573. In the PWR FLECHT program, there were four runs conducted with Zircaloy multi-rod bundles and two of these bundles incurred runaway oxidation.

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[6] *Id.*, p. 2299.
[7] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.
[8] *Id.*, p. ix.
[9] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43. See also 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, p. 2300.
[10] 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, p. 2299.

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[11]—during PWR FLECHT runs 8874 and 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions occurred, which cannot be detected in the post-test investigation.

So Westinghouse's application of the Baker-Just correlation to the oxide layers on the bundles from FLECHT runs 8874 and 9573 were to locations that were most likely steam starved. That is not a legitimate verification of the adequacy of the Baker-Just correlation for use in ECCS evaluation models.

And in recent years the NRC used this same data from the four PWR FLECHT Zircaloy runs in its safety analysis of PRM-50-76, which was submitted in 2002. And the NRC basically made the same arguments that Westinghouse made (but included the Cathcart-Pawel correlation), not realizing that they were basing their claims on samples that were taken from locations that would have had local steam starvation conditions, which cannot be detected in the post-test investigation. That's for two of the bundles. Again, that is not a legitimate verification of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in ECCS evaluation models.

In the early 1980s, the NRC contracted with National Research Universal at Chalk River, Ontario, Canada to run a series of tests, including the Thermal-Hydraulic Experiment 1, to evaluate the 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] in the NRU reactor. The TH-1 experiment was conducted with low-level fission heat to simulate decay heat:[13] the average fuel rod power for the tests was 0.37 kW/ft[14] and the peak power was 0.55 kW/ft.[15]

[11] 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.

[12] 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.

[13] 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

In a comparison between the data from TH-1 and 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 Westinghouse's "PWR FLECHT Final Report,"[16] 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,[17] 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).[18] 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).[19]

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.

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 also under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128.

Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119, p. 1 (hereinafter "Prototypic Thermal-Hydraulic Experiment").

[14] *Id.*, p. 10.

[15] 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").

[16] 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").

[17] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.

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

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

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, which I discussed in the last PRB meeting.

In TH-1 test no. 130, the reactor shutdown when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the Zircaloy-steam reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[20] So the peak cladding temperature increased by 190°F after the reactor shutdown, because of the heat generated from the Zircaloy-steam 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.

So data from thermal hydraulic experiments 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 loss-of-coolant accidents.

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