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Subject:	[External_Sender] Chairwoman Svinicki, Please Remedy YET MORE Problems with the NRC Technical Analysis of PRM-50-93/95
Date:	Thursday, April 19, 2018 3:01:16 AM

Dear Chairwoman Svinicki:

This letter is a follow-up to a letter I sent you on March 12, 2018, regarding **major problems** with the NRC technical analysis (dated March 18, 2016, according to ADAMS) of rulemaking petition, PRM-50-93. PRM-50-93 was submitted to the NRC on November 17, 2009. (This letter also concerns PRM-50-95, submitted on June 7, 2010.)

As I said, the technical analysis has numerous errors, misrepresentations, and omissions. In this e-mail, I address a few more of the technical analysis' **flaws.**

The staff cites results of tests conducted with stainless steel cladding as if they replicate the behavior of zirconium fuel-cladding in the event of a loss-of-coolant accident.

On page 10, the technical analysis of PRM-50-93 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 3-23, "Initial Clad Temperature Effect [on] Temperature Rise and Quench Time," in Lee et al. (1982), 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**. Linear extrapolation of initial cladding temperatures to predict final cladding temperature is inappropriate due to the increased radiative cooling at higher temperatures. Thus, contrary to the claim made by the petition, 'extrapolation' of data does not show "with high probability" that peak cladding temperatures will exceed 2,200 degrees F (1,204 degrees C)[1,478K]"[1] [emphasis not added].

First) The text of the technical analysis of PRM-50-93 does **NOT explain** that Figure 3-23 (of Lee *et al*., 1982) concerns three experimental programs that conducted tests with **stainless steel cladding**. The three experimental programs were PWR FLECHT SEASET, PWR FLECHT Cosine, and PWR FLECHT Skewed.[2]

Please ask the staff this question: Why does the staff fail to mention in the text of its technical analysis that it is discussing tests conducted with stainless steel cladding, **NOT** tests conducted with zirconium?

Second) Experimental results show that the staff's conclusions are **ERRONEOUS when**

applied to zirconium fuel-cladding—the type actually used in nuclear reactors.

At the higher temperatures at which the zirconium-steam reaction becomes significant, tests conducted with stainless steel cladding **do NOT** replicate the behavior of zirconium fuel-cladding in the event of a loss-of-coolant accident (LOCA).

As I told the staff in April 2014, in comments I submitted:

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 [peak cladding temperature] at the onset of reflood (in the entire 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.[3]

Yes, back in April 2014, I explained to the staff that it failed to consider that heat generated by the zirconium-steam reaction would cause zirconium cladding to increase higher than stainless steel cladding (in the temperature range in which the zirconium-steam reaction becomes significant).

Please ask the staff these questions: Why didn't the staff correct its mistakes after I pointed them out in 2014?

Why did the staff simply reiterate its **ERRONEOUS** claims?

THIRD) Experimental results show that the staff's conclusions are **ERRONEOUS**.

For nuclear reactors, **which use zirconium fuel cladding**, the staff's statement, "As the initial cladding temperature increases, the overall temperature rise **decreases**" [emphasis not added], is **ERRONEOUS** (in the temperature range in which the zirconium-steam reaction becomes significant).

As I told the staff in April 2014, in comments I submitted:

[I]n four Zircaloy tests—TH-1 test nos. 105, 107, 110, and 128—conducted with an average fuel rod power of 0.38 kW/ft; 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).

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. [4]

Yes, back in April 2014, I explained to the staff that **results from tests conducted with zirconium fuel-cladding** in a test reactor showed that (in the temperature range in which the zirconium-steam reaction becomes significant), as the initial cladding temperature increases, the overall temperature rise **increases**.

Thank you,

Mark Leyse

P.S. I have placed a portion of my April 2014 comments below.

[1] NRC, "Technical Safety Analysis of PRM-50-93/95 A Petition for Rulemaking to Amend 10 CFR 50.46 and Appendix K to 10 CFR Part 50," March 18, 2016, (ADAMS Accession No. ML16078A318), p. 10.

[2] 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, (ADAMS Accession No. ML070740214), p. 3.24.

[3] Mark Edward Leyse, "Comments on Nuclear Regulatory Commission's Draft Interim Reviews of Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554," April 12, 2014, (ADAMS Accession No: ML14104B253), pp. 32-33.

[4] Mark Edward Leyse, "Comments on Nuclear Regulatory Commission's Draft Interim Reviews of Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554," April 12, 2014, (ADAMS Accession No: ML14104B253), p. 33.

My April 2014 Comments:

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 Draft Interim Review 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*[1] [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 have also been a slight amount of actual decay heat[2]) and the peak measured cladding temperature was 2040°F.[3] 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,[4] PWR FLECHT Cosine,[5] and PWR FLECHT Skewed[6]—in its March 2013 Draft Interim Review 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*[7] [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 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;[8] 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).[9]

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.

(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*"[10] [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.

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[1] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

[2] 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.

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

[4] 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.

[5] G. P. Lilly, H. C. Yeh, L. E. Hochreiter, N. Yamaguchi, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report," WCAP-8838.

[6] 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.

[7] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

[8] 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.

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

[10] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.