CHAIRMAN Resource

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Sent: Monday, August 15, 2016 2:37 AM
To: CHAIRMAN Resource; CMRSVINICKI Resource; CMRBARAN Resource; RulemakingComments Resource; PDR Resource; CHAIRMAN Resource
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Subject: [External_Sender] Why does the NRC conduct unscientific reviews of the public's rulemaking petitions, NRC Chairman Burns? (NRC–2009–0554)

Dear Chairman Burns:

The NRC purports to encourage public participation. Your website states: “The NRC welcomes public participation in the rulemaking process.” That all sounds nice and friendly but I think it's only fair that your site also warn any prospective participants to expect a biased and unscientific review of any submitted material that's considered unfavorable to industry.

Better still, why not fix the NRC's rulemaking process? Please make it unbiased and scientific.

In this e-mail, I discuss the NRC's review of a petition for rulemaking that I submitted in 2009 (PRM-50-93). I allege that the Staff's review of PRM-50-93 has been biased and unscientific. Please note that I provide evidence below in Section I.

In PRM-50-93, among other things, I argued that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are inadequate for use in computer safety models like the NRC’s TRACE code. I asked that the computer safety models the NRC uses for loss-of-coolant accident evaluations be more realistic.

I alleged that current models under-predict the fuel-cladding temperatures that would occur in a loss-of-coolant accident. If what I claim is correct (and I think it is), it means that the power levels of reactors will likely need to be reduced. And that's certainly not a prospect that's agreeable to industry.

The NRC is congressionally mandated to protect the public, not serve the needs of industry. So, as a member of the public who took the time to research and write PRM-50-93, I expect an unbiased and scientific review of the material I submitted. **Please tell me, Chairman Burns, is that too much to ask for?**

I also expect an unbiased and scientific review of the material I submitted, because I believe public safety is compromised by unrealistic loss-of-coolant accident evaluations—ones that under-predict fuel-cladding temperatures.

Please Chairman Burns, tell the Staff to conduct an unbiased and scientific review of PRM-50-93. Please note that below in Sections II and III, I provide evidence that the NRC's TRACE code (using the Cathcart-Pawel and Baker-Just correlations) under-predicted cladding and steam temperatures in an experiment that Westinghouse conducted called FLECHT Run 9573.

Please note that below in Sections IV and V, I provide evidence that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are inadequate for use in computer safety models like the NRC's TRACE code.

Sincerely,

Mark Leyse

I. Evidence that the NRC has Conducted a Biased and Unscientific Review of PRM-50-93

A Freedom of Information Act request to the NRC revealed that in August 2015, the NRC had plans to deny my 2009 rulemaking petition (PRM-50-93). Their plan to deny was largely "supported" by their illegitimate computer simulation of a Westinghouse experiment called FLECHT Run 9573.[1]

As part of their technical analysis of PRM-50-93, the NRC did a computer simulation of what occurred when the FLECHT Run 9573 test bundle incurred runaway oxidation (or thermal runaway) (see Figure 1). They wanted to compare the results of their simulation to the data that Westinghouse reported. However, there was a big problem with the NRC's simulation. They did **not** simulate the section of the test bundle that incurred runaway oxidation.[2] (Or if they did simulate that section, they decided not to release their findings.)

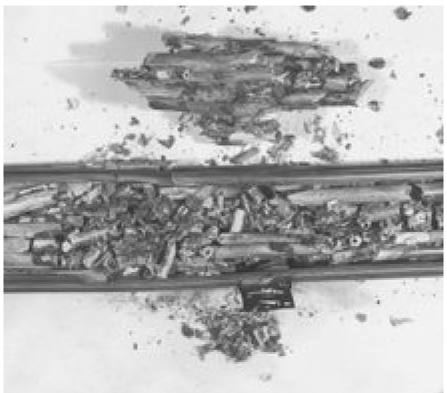

Figure 1. Section of the FLECHT Run 9573 Test Bundle that Incurred Runaway Oxidation

By way of an analogy: what the NRC did would be like simulating a forest fire and omitting trees reduced to ash and only simulating those that had been singed. After doing such a bogus simulation one might try to argue that trees actually do not burn down in forest fires. The NRC did something similar. They used the results of their simulation to argue that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are adequate for use in computer safety models like the NRC's TRACE code.[3]

On January 31, 2013, I gave a presentation to Chairwoman Allison M. Macfarlane and the four Commissioners. They invited me to present my views on a panel addressing public participation in the NRC's rulemaking process.[4] They apparently wanted my insights, because, in 2007, I raised a safety issue (in PRM-50-84) that they decided to incorporate into one of their regulations. (In 2012, the NRC Commissioners voted unanimously to approve a proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that is partly based on the safety issues I raised in PRM-50-84.[5])

In my presentation to the Commissioners, I discussed the NRC's computer simulation of FLECHT Run 9573. To quote myself, I said: "You cannot do legitimate computer simulations of an experiment that incurred runaway oxidation by not actually modeling the section of the test bundle that incurred runaway oxidation. So, the staff's…simulations were frankly a waste of money." I offered to meet with the staff members who were reviewing my 2009 rulemaking petition, to discuss it, "try to sort things out, expedite things."[6]

After everyone on the panel concluded their presentations, Chairwoman Macfarlane stated: "Let me first note that I think Mr. Leyse demonstrated and has been and is continuing to be in the process of demonstrating that the public actually has a lot of valuable input. The public actually knows things that people at government agencies don't know and may not be aware of, and actually, the social science literature is ripe with this information as well, confirming this is true."[7] Later on, Commissioner William Magwood assured me that he and the other NRC leaders would instruct their staff "to follow up on" my criticism of the NRC's computer simulation of FLECHT Run 9573.[8] He also asked the NRC's General Counsel Margaret Doane if it would be legal for me to directly discuss my 2009 rulemaking petition with NRC staff. And she affirmed that it would be legal as long as any discourse were held in an open forum. Commissioner Magwood said that they would also follow up on that issue and I thanked him.[9]

I thought I was making headway. The leadership of the NRC seemed receptive to my allegation that the computer simulation of FLECHT Run 9573 was inadequate. I hoped they'd order the NRC staff to conduct a legitimate simulation. But that didn't happen. Instead, a couple of months after the meeting on public participation, the NRC staff released yet more of its technical analysis of my 2009 rulemaking petition, including a statement that their simulation of FLECHT Run 9573 *over*-predicted the extent that zirconium burns in steam.[10] The NRC staff simply reiterated their claim that their simulation's results show that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are adequate for use in computer safety models like the NRC's TRACE code.

In November 2015, after I made a series of additional complaints, the NRC finally disclosed the results of a computer simulation of FLECHT Run 9573 that included the section of the test bundle that incurred runaway oxidation. And the simulation *under*-predicted temperatures Westinghouse had reported for that section.[11] This is powerful evidence that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are **inadequate** for use in computer safety models like the NRC's TRACE code.

Below in Sections II and III, I provide information on FLECHT Run 9573 as well as the NRC's computer simulation of FLECHT Run 9573 that included the section of the test bundle that incurred runaway oxidation.

II. "Low Temperature" Oxidation Rates Are Under-Predicted for FLECHT Run 9573

Westinghouse's "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" (hereinafter: "WCAP-7665") states that, "[t]he objective of the PWR FLECHT…test program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[12] Some of the FLECHT tests were conducted with bundles of heater rods sheathed in zirconium alloy (Zircaloy) cladding. Runaway oxidation was not expected to occur in any of the tests; however, the FLECHT Run 9573 test bundle incurred runaway oxidation.

The FLECHT Run 9573 test bundle incurred runaway oxidation around its seven foot elevation. WCAP-7665 states: "Post-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation. The heater rod failures were apparently caused by localized temperatures in excess of 2500°F." WCAP-7665 also states: "During the test, heater element failures started at 18.2 seconds... At the time of the initial failures, midplane [at the 6 foot elevation] clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure)."[13]

The NRC conducted TRACE code computer simulations of FLECHT Run 9573 and found that TRACE *under-predicted* temperatures that were reported by Westinghouse at the 7 ft elevation of the test bundle. On November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, sent me an e-mail regarding the NRC's TRACE computer simulation of FLECHT Run 9573. In his e-mail, Mr. Mohseni disclosed findings of "the completed simulation [for] the cladding and steam temperatures at the 7-ft elevation (at 18 seconds)."[14]

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

III. FLECHT Run 9573—a Comparison between Computer Safety Model Predictions and the Results Westinghouse Reported

According to Mr. Mohseni's e-mail, when the TRACE code used the Cathcart-Pawel and Baker-Just correlations, it predicted *cladding* temperatures of 1526 K (2287°F) and 1561 K (2350°F), respectively. And, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *steam* temperatures of 1370 K (2006°F) and 1397 K (2055°F), respectively. Those are predicted cladding and steam temperatures for the FLECHT Run 9573 test bundle at the 7-ft elevation, at 18 seconds.[15]

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

And Westinghouse reported that at 16.0 seconds, a steam probe at the 7 foot elevation recorded *steam* temperatures that exceeded 1644 K (2500°F). And a Westinghouse memorandum stated that after 12 seconds, the steam-probe thermocouple recorded "an extremely rapid rate of temperature rise (over 300°F/sec)."[16] (Who knows how high the steam temperatures actually were at 18 seconds; they were likely hundreds of degrees Fahrenheit higher than 1644 K (2500°F).)

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

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

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

IV. "Low Temperature" Oxidation Rates Are Under-Predicted for the CORA-16 Experiment

When Oak Ridge National Laboratory ("ORNL") investigators compared the results of the CORA-16 experiment—a BWR 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."[17]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16.[18] However, it is *unsubstantiated* that cladding strain actually increased reaction rates.

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, the NRC has pointed out that an NRC report, NUREG/CR-4412,[19] "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"[20] [emphasis not added].

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA [large-break loss-of-coolant accident] 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 1000°C [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[21]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[22] 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;"[23] the report provides a graph of the "rather sparse"[24] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[25]

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]."[26]

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."[27] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[28] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a brief period of time, tens of seconds, 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.[29]

Furthermore, ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F temperature range for CORA-16.

There is also one phenomenon the NRC did not consider in its 2011 analysis of CORA-16: "[t]he swelling of the [fuel] cladding…alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,”[30] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

CORA-16 is an example of an experiment that had zirconium-steam reaction rates that were under-predicted in the “low temperature” range from 1652°F to 2192°F by computer safety models. The CORA-16 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just correlation, are inadequate for use in computer safety models like the TRACE code.

V. Oxidation Models Are Unable to Predict the Fuel-Cladding Temperature Escalation that Commenced at “Low Temperatures” in the PHEBUS B9R-2 Test

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 uranium dioxide (UO_2) fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[31] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region (see Figure 1); the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[32]

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).
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> … *No mechanistic model is currently available to account for enhanced oxidation of pre-oxidized and cracked cladding*[33] [emphasis added].

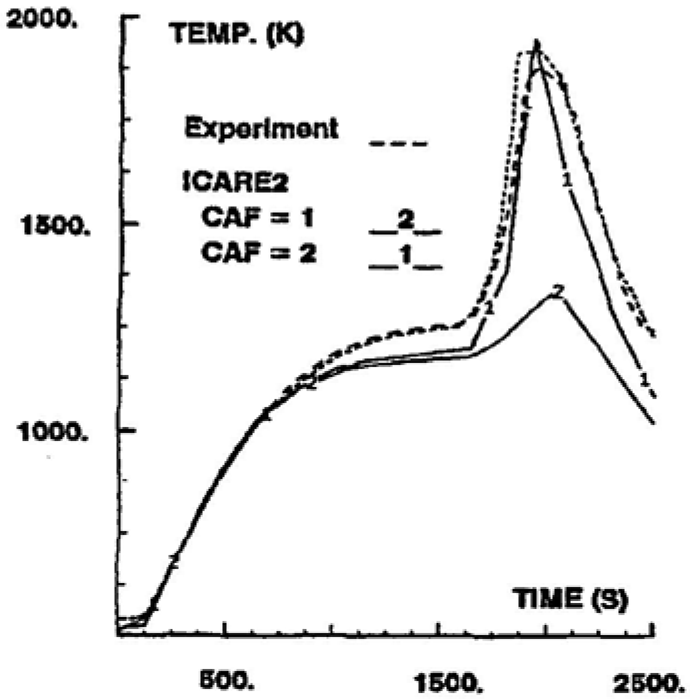

Figure 1. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[34]

Today, in 2016, oxidation models still cannot accurately predict the local fuel-cladding temperature escalation that commenced in PHEBUS B9R-2 in steam when local fuel-cladding temperatures were 1027°C (1880°F). The PHEBUS B9R-2 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just correlation, are inadequate for use in computer safety models like the TRACE code.

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to high burnup fuel. High burnup fuel rods would also be "pre-oxidized": when high burnup (and other) fuel rods are discharged from the reactor core and loaded into the spent fuel pool, the fuel cladding can have local zirconium dioxide (ZrO_2)

"oxide" layers that are up to 100 μm thick (or greater); there can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick.

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[1] Danial Doyle, Project Manager, e-mail (with attachment) regarding, "Division Director Alignment Meeting for PRM-50-93/95," August 5, 2015. See pages 1249, 1251, and 1252 of the PDF file of Interim response to a May 2016 Freedom of Information Act request by David Lochbaum of UCS for "All records not already publicly available in ADAMS related to the petition for rulemaking submitted by Mark Edward Leyse and designated as PRM-50-108 and NRC-2014-0171 by the staff," July 27, 2016, (ADAMS Accession No: ML16214A318).

[2] 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, (ADAMS Accession No. ML12265A277), pp. 7-9.

[3] 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, (ADAMS Accession No. ML12265A277), pp. 7-9.

[4] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf).

[5] Mark Leyse, "Petition for Rulemaking Addressing Crud Deposits on Fuel Cladding Surfaces and a Change in the Calculations for a Loss-of-Coolant Accident," PRM-50-84, March 15, 2007 (ADAMS Accession No. ML070871368); Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569; Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776; and NRC, Commission Voting Record, Decision Item: SECY-12-0034, Proposed Rulemaking—10 CFR 50.46(c): Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42), January 7, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/cvr/2012/2012-0034vtr.pdf).

[6] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf), pp. 55-56.

[7] *Id.*, pp. 65-66.

[8] *Id*., p. 83.

[9] *Id*., pp. 84-85.

[10] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," March 8, 2013, (ADAMS Accession No. ML13067A261), p. 4.

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

[12] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, (ADAMS Accession No: ML070780083), p. 1.1.

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

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

[15] *Id*.

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

[17] 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.

[18] 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.

[19] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, (ADAMS Accession No: ML083400371).

[20] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, (ADAMS Accession No: ML112211930), p. 3.

[21] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 27.

[22] *Id*., pp. 27, 30.

[23] *Id*., p. 30.

[24] *Id*.

[25] *Id*., p. 29.

[26] *Id*., p. 30.

[27] *Id*.

[28] 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.

[29] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 30.

[30] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, (ADAMS Accession No: ML041950059), p. 43.

[31] 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.

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

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

[34] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 312.