Rulemaking Comments **PRM-50-95 (75FR66007)** 6

From: Mark Leyse [markleyse@gmail.com]
Sent: Monday, December 27, 2010 7:51 PM
To: Rulemaking Comments; PDR Resource
Cc: Dave Lochbaum; necnp@necnp.org; Raymond Shadis; Powers, Dana A
Subject: NRC-2009-0554 (Third)
Attachments: Comment III Response to NEI.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Edward Leyse's, Petitioner's, third response, dated December 27, 2010, 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. In these comments on PRM-50-93 and PRM-50-95, Petitioner responds to the Nuclear Energy Institute's ("NEI")comments on PRM-50-93 and PRM-50-95, dated November 24, 2010.

Petitioner has responded to NEI's comments on PRM-50-93 and PRM-50-95 promptly: Petitioner notes that although NEI's comments are dated November 24, 2010, NEI's comments were placed into the docket folder for PRM-50-93 and PRM-50-95 on December 13, 2010.

Sincerely,

Mark Edward Leyse

DOCKETED
USNRC

December 28, 2010 (9:15am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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December 27, 2010

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

Attention: Rulemakings and Adjudications Staff

Subject: Response to the U.S. Nuclear Regulatory Commission's ("NRC") notice of solicitation of public comments on PRM-50-93 and PRM-50-95; NRC-2009-0554

Dear Ms. Vietti-Cook:

Enclosed is Mark Edward Leyse's, Petitioner's, third response 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. In these comments on PRM-50-93 and PRM-50-95, Petitioner responds to the Nuclear Energy Institute's ("NEI") comments on PRM-50-93 and PRM-50-95, dated November 24, 2010.

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Respectfully submitted,

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December 27, 2010

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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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

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 the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation

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[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 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.

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 a 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 of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a loss-of-coolant accident ("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.

In these comments on PRM-50-93 and PRM-50-95, Petitioner responds to Nuclear Energy Institute's ("NEI") comments on PRM-50-93 and PRM-50-95, dated November 24, 2010.

Petitioner has responded to NEI's comments on PRM-50-93 and PRM-50-95 promptly: Petitioner notes that although NEI's comments are dated November 24, 2010, NEI's comments were placed into the docket folder for PRM-50-93 and PRM-50-95 on December 13, 2010.

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[4] Data from multi-rod (assembly) severe fuel damage experiments 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.

II. Response to NEI's Comments on PRM-50-93 and PRM-50-95, Dated November 24, 2010

A. An Important Aspect of the CORA-16 Experiment that NEI Overlooked: Analyses that Used the Baker-Just and Cathcart-Pawel Correlations Under-Predicted Oxidation Kinetics in the CORA-16 Experiment

In NEI's comments on PRM-50-93 and PRM-50-95, NEI discusses the CORA-16 experiment. NEI discusses the temperature differences between the cruciform control blades and the fuel cladding in the CORA-16 experiment. And in NEI's comments, NEI states that the use of the Baker-Just and Cathcart-Pawel correlations is still appropriate.[7] Yet, unfortunately, NEI does not discuss or comment on the fact that Zircaloy oxidation in the CORA-16 experiment was under-predicted by analyses that used the Baker-Just and Cathcart-Pawel correlations—discussed in PRM-50-95 (pages 12-13, 26-27).

It is significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," presented in 1991, explicitly states "[c]ladding oxidation [in the CORA-16 experiment] was not accurately predicted by available correlations."[8] (In 1991, the Baker-Just and Cathcart-Pawel correlations was among the available correlations.)

And discussing "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[9] a boiling water reactor ("BWR") severe fuel damage experiment, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> The 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.*

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[7] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," November 24, 2010, Attachment, p. 2.

[8] 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.

[9] 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.

> ... Dr. Haste pointed out that he is chairing a committee (for the OECD) which is preparing a report on the state of the art with respect to Zircaloy oxidation kinetics. He will forward material addressing the low-temperature Zircaloy oxidation problems encountered in the CORA-16 analyses to ORNL [emphasis added].[10]

So, in the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations"[11] and "[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."[12] This indicates that the Baker-Just and Cathcart-Pawel correlations are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

In PRM-50-93, PRM-50-95, and in comments on PRM-50-93 and PRM-50-95, Petitioner has extensively discussed other data from multi-rod (assembly) severe fuel damage experiments that 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. Such experimental data, 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.

B. NEI's Misleading Characterization of the "Unrealistic" Balance of Heat Input to the Fuel Cladding and Heat Removal from the Fuel Cladding that Occurs in Severe Fuel Damage Experiments

In NEI's Comments on PRM-50-93 and PRM-50-95, NEI misleadingly characterizes some of the conditions of severe fuel damage ("SFD") experiments—in particular, the CORA experiments—as non-applicable to realistic LOCA conditions.

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[10] *Id.*

[11] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

[12] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.

Regarding heat input and heat removal from the fuel cladding in SFD experiments, NEI states:

> Severe accident tests are designed to result in the failure of the fuel, so that the melting behavior of the assembly can be studied. Under these scenarios steam is provided mainly to ensure the water-metal reaction occurs *and is not used to maintain a realistic balance of heat input and removal.* In the specific CORA tests referenced by the petition, *the combined cooling capability of both the steam and argon*[13] *is insufficient to arrest temperature increases* from the electrical heat input. Furthermore, in the CORA tests a sustained heat input is provided at a constant rate with *inadequate heat removal*, whereas, heat input under realistic LOCA conditions decreases exponentially with time while heat removal capability increases with time [emphasis added].[14]

NEI also states that "the differences in test conditions clearly invalidate the applicability of the CORA test to realistic LOCA conditions. The potential for excessive escalation of cladding temperature has to be determined through a balance of heat generation and removal as is presently accounted for in the LOCA licensing calculations."[15]

First, in a real LOCA there could be conditions similar to the conditions of some SFD experiments—including the CORA experiments. For example, in a LOCA where there would be reflood rates of less than one inch per second.

Regarding LOCAs with reflood rates of less than one inch per second, Appendix K to Part 50, I.D.5.b. states:

> During refill and during reflood when reflood rates are less than one inch per second, *heat transfer calculations shall be based on the assumption that cooling is only by steam*, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer [emphasis added].

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[13] Not all the CORA experiments discussed by Petitioner in PRM-50-93 and PRM-50-95 had argon, in addition to steam, in the transient and cooling phases of the experiments. However, in the CORA experiments argon was used in the gas preheat phase of the experiments; see S. Hagen, P. Hofmann, V. Noack, L. Sepold, G. Schanz, G. Schumacher, "Comparison of the Quench experiments CORA-12, CORA-13, CORA-17," Forschungszentrum Karlsruhe, FZKA 5679, 1996, pp. 3, 5.
[14] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554,"Attachment, p. 3.
[15] *Id.*, Attachment, p. 4.

So NEI's claim that the heat transfer that occurred in the CORA experiments is non-applicable to realistic LOCA conditions is clearly erroneous. Reflood rates of less than one inch per second are within the parameters of realistic LOCA conditions. And for reflood rates of less than one inch per second, Appendix K to Part 50, I.D.5.b. states that "heat transfer calculations shall be based on the assumption that cooling is only by steam."

It is unfortunate that NEI, representing the nuclear industry, does not realize that fuel-cladding that is only cooled by steam is "presently accounted for in the LOCA licensing calculations."[16]

Second, it is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" describes a method for assessing the conservatism of the 2200°F peak cladding temperature ("PCT") limit, as a boundary that would prevent autocatalytic oxidation from occurring. "Compendium of ECCS Research for Realistic LOCA Analysis" states that this can be accomplished by analyzing data from multi-rod SFD experiments—including data from the CORA program.

"Compendium of ECCS Research for Realistic LOCA Analysis" states:

> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. *This type of comparison implicitly includes...complex heat transfer mechanisms*...and the effects of fuel rod ballooning and rupture on coolability... Analysis of experiments performed in the Power Burst Facility, in the Annular Core Research Reactor, and in the NEILS-CORA (facilities in West Germany) program have shown that temperatures above 2200°F are required before the zircaloy-steam reaction becomes sufficiently rapid to produce an autocatalytic temperature excursion. Another group of relevant experimental data were produced from the MT-6B and FLHT-LOCA and Coolant Boilaway and Damage Progression tests conducted in the NRU Reactor in Canada. ...even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, *when design basis heat transfer and decay heat are considered*, some margin above 2200°F exists [emphasis added].[17]

[16] *Id.*

[17] 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, p. 8-2.

So according to "Compendium of ECCS Research for Realistic LOCA Analysis" experiments from the CORA program and other SFD research programs are useful for assessing the conservatism of the 2200°F PCT limit. "Compendium of ECCS Research for Realistic LOCA Analysis" also states that "[t]his type of comparison implicitly includes...complex heat transfer mechanisms"[18] and that "design basis heat transfer and decay heat are considered"[19] in such assessments of the conservatism of the 2200°F PCT limit.

Unfortunately, data from multi-rod (bundle) SFD experiments, actually, indicates that the 2200°F PCT limit is non-conservative. The conclusion of "Compendium of ECCS Research for Realistic LOCA Analysis" regarding data from multi-rod (bundle) experiments and the 2200°F PCT limit is erroneous.

For example, the paper, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," states:

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15 K/sec.[20]

A maximum heating rate of 15 K/sec. indicates that an autocatalytic oxidation reaction commenced: "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[21] So at the point

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[18] *Id.*

[19] *Id.*

[20] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230460, p. 83.

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

when peak cladding temperatures increased at a rate of greater than 10 K/sec. during the CORA experiments, autocatalytic oxidation reactions commenced at cladding temperatures between 2012°F and 2192°F.

Third, NEI oversimplifies the issue of the initial heatup rate that was used in many of the CORA experiments.

Regarding the constant heatup rate—used to simulate decay heat—in the CORA experiments, NEI states:

> Furthermore, in the CORA tests *a sustained heat input is provided at a constant rate* with inadequate heat removal, whereas, heat input under realistic LOCA conditions decreases exponentially with time while heat removal capability increases with time [emphasis added].[22]

The initial heatup rate of most of the CORA experiments discussed in PRM-50-93 and PRM-50-95 was 1 K/sec. It is significant that the LOFT LP-FP-2—conducted with actual decay heat—had an initial heatup rate of ~1 K/sec.[23] It is also significant that "heatup rates [of 1 K/s or greater] are typical of severe accidents initiated from full power."[24] And regarding the significance of the initial heatup rate in the LOFT LP-FP-2 experiment, "Review of Experimental Results on LWR Core Melt Progression" states:

> The higher initial heating rate [of ~1 K/sec.] in the LOFT [LP-]FP-2 experiment is related to the higher fraction of decay heat available following rapid blowdown of the coolant inventory in the reactor vessel. This higher heating rate leads to smaller oxide thickness on the cladding for a particular temperature and, therefore, more rapid oxidation. The increase in heating rate at the higher temperatures is the result of rapid oxidation of zircaloy and the strongly exothermic nature of the reaction (6.45 kJ/g Zr oxidized).[25]

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[22] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

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

[24] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2; this paper cites Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, as the source of this information.

[25] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," NUREG/CP-0114, Vol. 2, 1990, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042250131, p. 7.

So the initial heatup rate—1 K/sec—of most of the CORA experiments discussed in PRM-50-93 and PRM-50-95 was approximately the same initial heatup rate of the LOFT LP-FP-2—conducted with actual decay heat.

Fourth, NEI oversimplifies the issue of heat removal capability, in the event of a LOCA.

Regarding the issue of heat removal capability, in the event of a LOCA, NEI states:

> Furthermore, in the CORA tests a sustained heat input is provided at a constant rate with inadequate heat removal, whereas, heat input under realistic LOCA conditions decreases exponentially with time *while heat removal capability increases with time* [emphasis added].[26]

It is simply not true that in the event of a LOCA that heat removal capability would *always* increase with time. For example, in a pressurized water reactor ("PWR") LOCA in which there was steam binding, the heat removal capability would not necessarily increase with time.

C. Important Aspects of Reflood Heat Transfer Coefficients for PWR Fuel Rods and of Convective Heat Transfer Coefficients for BWR Fuel Rods Under Spray Cooling that NEI Overlooked

In NEI's comments on PRM-50-93 and PRM-50-95, NEI overlooks important aspects of the heat transfer coefficients that are used in ECCS evaluation calculations for Zircaloy fuel cladding. For example, the heat transfer coefficients used in Appendix K ECCS evaluation calculations for Zircaloy fuel assemblies in real reactor cores are based on data from thermal hydraulic experiments conducted with stainless steel heater-rod bundles. Trying to relate thermal hydraulic experiments conducted with stainless steel bundles to what would occur in a reactor core with Zircaloy bundles, in the event of a LOCA, simply does not work.

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[26] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

Regarding "[t]he effect of heat balance, expressed in terms of heat transfer coefficients, on accelerated oxidation," NEI states:

> The effect of heat balance, expressed in terms of heat transfer coefficients, on accelerated oxidation is illustrated in a case study shown in Figure 1. In this evaluation, double sided Cathcart-Pawel correlation was used for the metal-water reaction. Clearly with a heat transfer coefficient of ~20 W/m^2K the reaction is autocatalytic and cannot be stopped. This is comparable to what happens in the severe accidents tests, since the test objective is to melt the rods. However, with a heat transfer coefficient of ~50 W/m^2K, a rate significantly lower than what is calculated in realistic LOCA case, the reaction is not autocatalytic and temperatures above 2200°F (1204°C) can be reached without oxidation runaway. This demonstrates that the escalation of cladding temperature is a function of the balance between heat generation and removal.[27]

First, analyses that used the Cathcart-Pawel correlation under-predicted oxidation kinetics in the CORA-16 experiment (this is discussed in Section A of Petitioner's comments, pp. 6-7). So NEI's evaluation of NEI's case study would have had different results if the metal-water reaction rates modeled in NEI's analysis had been based on data from multi-rod experiments like the CORA-16 experiment.

It is not realistic to use correlations like the Cathcart-Pawel correlation—based on data from experiments conducted with single Zircaloy tube specimens, a few inches long or less—in ECCS evaluation calculations. Trying to relate experiments conducted with single Zircaloy tube specimens, a few inches long or less, to what would occur in a reactor core with Zircaloy fuel assemblies, in the event of a LOCA, simply does not work.

Second, NEI's evaluation of NEI's case study would have had different results if the heat transfer coefficients used in NEI's analysis had been based on data from thermal hydraulic experiments conducted with multi-rod Zircaloy bundles.

It is not realistic to use heat transfer coefficients based on data from experiments conducted with stainless steel and/or Inconel 600 bundles in ECCS evaluation calculations. Trying to relate thermal hydraulic experiments conducted with stainless steel and/or Inconel 600 bundles to what would occur in a reactor core with Zircaloy bundles, in the event of a LOCA, simply does not work.

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[27] *Id.*

It is significant that NRC states that "[h]eat transfer coefficients are not directly measurable quantities. They must be calculated from *measured temperatures,* known heat sources, and known thermal properties" [emphasis added].[28] Petitioner would add that heat transfer coefficients used for LOCA analyses of real reactor cores with Zircaloy fuel assemblies must *also* be calculated from thermal hydraulic experiments conducted with multi-rod Zircaloy bundles.

(It is noteworthy that NRC needs to conduct realistic thermal hydraulic experiments with multi-rod Zircaloy bundles in which the bundles would be heated up to at least 2200°F. Such experiments would also need to be conducted with varying reflood rates. And for BWRs such experiments would need to be conducted with varying amounts of coolant supplied to each fuel bundle by BWR core spray systems.)

Unfortunately, most thermal hydraulic experiments have been conducted with multi-rod stainless steel and Inconel 600 bundles. And it is significant that some of the thermal hydraulic experiments that have been conducted with multi-rod Zircaloy bundles have had results that do not conclusively demonstrate the effectiveness of ECCS in cooling the fuel cladding; *e.g.*, in cases in which there would be reflood rates of one inch or less per second.

The practice of using heat transfer correlations derived from stainless steel clad heater rods for ECCS evaluation calculations dates back to the Atomic Energy Commission ("AEC") rulemaking hearings: the AEC Commissioners concluded that "the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods.[29]

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[28] 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. 7.

[29] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1124. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

1. The Fallacy of the AEC Commissioners' Conclusion that "the Heat Transfer Mechanism is [Not] Different for Zircaloy and Stainless Steel": "that the Heat Transfer Correlations Derived from Stainless Steel Clad Heater Rods are Suitable for Use with Zircaloy Clad Fuel Rods"

To discuss the fallacy of the AEC Commissioners' conclusion that "the heat transfer mechanism is [not] different for zircaloy and stainless steel, and...that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods,"[30] Petitioner will discuss PWR FLECHT Run 9573. Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.

(Run 9573 was part of the PWR FLECHT test program; however, the exothermic zirconium-water reaction that occurred in the test is pertinent to both PWR and BWR Zircaloy fuel rods in LOCA environments. It is significant that a Zircaloy assembly used in the BWR FLECHT program—the Zr2K test assembly—also incurred autocatalytic oxidation.)

It is significant that "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> [T]he Commission sees no basis for concluding that the heat transfer mechanism is different for zircaloy and stainless steel, and believes that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods. It is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[31]

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final

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[30] *Id.*
[31] *Id.*

Report"] when compared to the equivalent stainless steel tests."[32] The NRC also stated, regarding the results of FLECHT run 9573, that the AEC Commissioners were not "concern[ed] about the zirconium-water reaction models."[33]

Discussing the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> [Another] reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating the [metal-water] reaction from the physical processes of cooling rods which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the assumption that the radiative and convective heat transfer processes for heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions.* The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.
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> *The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms* [emphasis added].[34]

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[32] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 16-17.

[33] *Id.*, p. 17.

[34] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-7.

And opining on the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The reasonable conclusion was reached that the effect of the difference between Zircaloy and stainless steel, if any, would be small. There is a difference, of course, in the rate of heat generation from steam oxidation, but this heat is deposited within the metal under the surface of the oxide film. *The presence of this heat source should not affect the heat transfer coefficients, which depend on conditions in the coolant outside the rod.*[35]

So the AEC Commissioners concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients, maintaining that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside the rod.

It is significant that within the first 18.2 seconds of FLECHT run 9573,[36] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[37] *i.e.*, more heat was transferred into the bundle midplane than was removed from that location. In petition for rulemaking 50-76 ("PRM-50-76"), Robert H. Leyse, the principal engineer in charge of directing the Zircaloy FLECHT tests and one of the authors of "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," states that "[t]he negative heat transfer coefficients [occurring within the first 18.2 seconds of run 9573] were calculated as a result of a heat transfer condition during which more heat was being transferred into the heater than was being removed from the heater[; used in the FLECHT tests to simulate fuel rods]. And the reason for that condition was that the heat generated from Zircaloy-water reactions at the surface of the

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[35] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1123-1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.
[36] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 3-97.
[37] *Id.*, p. 3-98.

heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[38]

So the heat generated from the exothermic oxidation reaction of the Zircaloy cladding (and Zircaloy spacer grids) was transferred from the cladding's reacting surface inward. Indeed, the Zircaloy-cladding heater rods were very hot internally, where the thermocouples were located; yet, nonetheless, the heater rods became a heat sink.[39]

Additionally, the exothermic oxidation reaction of the Zircaloy heated a mixture of steam and hydrogen, and entrained water droplets. Westinghouse agrees with this claim; in its comments regarding PRM-50-76, Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[40]

Regarding steam temperatures measured by the seven-foot steam probe, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states:

> At the time of the initial [heater element] failures, midplane 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).[41]

Therefore, it is reasonable to conclude that a superheated mixture of steam and hydrogen, and entrained water droplets, caused heating of Zircaloy cladding in the midplane location of the fuel rod. It is also reasonable to conclude that the "negative heat transfer coefficients [that] were observed at the bundle midplane for

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[38] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.
[39] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.
[40] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.
[41] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.

5...thermocouples"[42]—the occurrence of more heat being transferred into the bundle midplane than was removed from that location—within the first 18.2 seconds of FLECHT run 9573, were caused by an exothermic zirconium-water reaction. Additionally, it is reasonable to conclude that "the impression left from [FLECHT] run 9573" cannot be separated from concerns about zirconium-water reaction models.

Clearly, the exothermic zirconium-water reaction affects the coolant outside the cladding by heating a mixture of steam and hydrogen, and entrained water droplets; therefore, the zirconium-water reaction cannot legitimately be separated from cladding heat transfer mechanisms.

Furthermore, because, as Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam,"[43] the AEC Commissioners' conclusion that "the presence of...heat [generated from the exothermic zirconium-water reaction] should not affect...heat transfer coefficients, which depend on conditions in the coolant outside the rod"[44] is erroneous.

2. More on the Results of FLECHT run 9573 and the Fallacy of Using Heat Transfer Correlations Derived from Stainless Steel Clad Heater Rods in ECCS Evaluation Calculations

It is significant that FLECHT run 9573 incurred autocatalytic oxidation and had a lower initial cladding temperature than, and the same power level as, other FLECHT Zircaloy tests that did not incur autocatalytic oxidation. The primary difference between run 9573 and the other FLECHT Zircaloy tests was that run 9573 had the lowest flood rate. "Consolidated National Intervenors pointed out that most of [the Zircaloy] runs

[42] *Id.*, p. 3-98.
[43] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[44] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

were made at unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding rate was about one inch per second."[45]

It would be reasonable to postulate that if run 9573 were repeated—with the same or a lower coolant flood rate, yet with lower initial cladding temperatures (that in the event of a LOCA, would occur at the beginning of reflood at current and/or proposed PWRs) and a lower power level (within the operational range of current and/or proposed PWRs)—that the fuel assembly would still incur autocatalytic oxidation, because FLECHT run 9573 had the lowest flood rate of the four Zircaloy tests.

It is significant that for PWR FLECHT run 9573 the "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test"[46] and that "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples."[47] Yet the data from run 9573 is *not* considered valid. And "PWR FLECHT Final (Full Length Emergency Cooling Heat Transfer) Report" states:

> Properly used, PWR FLECHT test results can improve the accuracy of reactor LOCA analysis. The heat transfer correlations which were developed are *conservative* in that they do not take any credit for the effects of "fallback" or borated coolant and are *based on stainless steel clad data* [emphasis added].[48]

So Appendix K to Part 50—ECCS Evaluation Models I(D)(5)—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]"—is erroneously based on the assumption that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

Indeed, stainless steel cladding heat transfer coefficients are not a conservative representation of representation of Zircaloy cladding behavior, for some of the conditions that would occur in the event of a LOCA.

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[45] *Id.*
[46] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[47] *Id.*, p. 3-98.
[48] *Id.*, p. 5-4.

(It is noteworthy that the subsequent PWR FLECHT programs—like the FLECHT Low Flooding Rate Test Series—were conducted with stainless steel bundles. And the FLECHT-SEASET program was conducted with stainless steel bundles.

It is also noteworthy that the rig of safety assessment IV ("ROSA-IV") facility, which conducted PWR thermal hydraulic experiments, used Inconel 600 bundles.[49] And the Rod Bundle Heat Transfer ("RBHT") facility at Penn State University—currently investigating PWR-related problems—uses Inconel 600 fuel rod simulators.[50])

3. The Rate of Stainless Steel Oxidation is Small Relative to the Oxidation of Zircaloy at Temperatures Below 1400 K but the Rate of Reaction for Stainless Steel Exceeds that of Zircaloy above 1425 K; However, the Heat of Reaction is about One-Tenth that of Zircaloy, for a Given Mass Gain

Discussing one of Henry Kendall and Daniel Ford's, of Consolidated National Intervenors ("CNI"),[51] criticisms of the BWR-FLECHT tests (which would also apply to other thermal hydraulic experiments conducted with stainless steel bundles), "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The first complaint [regarding the BWR-FLECHT tests] was that although all BWR fuel rods are manufactured of a zirconium...alloy, Zircaloy, only 5 of the 143 FLECHT tests utilized [Zircaloy] rods. The remaining 138 tests were conducted with stainless steel...rods. *Since...[Zircaloy] reacts exothermically with water at elevated temperatures, contributing additional energy to that of the decaying fission products, the application of water to the core has the potential of increasing the heat input to the fuel rods rather than cooling them, as desired.* The small number of [Zircaloy] tests in comparison with the total test program was seriously faulted by the CNI [emphasis added].[52]

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[49] Yasuo Koizumi, Yoshinari Anoda, Hiroshige Kumamaru, Taisuke Yonomoto, Kanji Tasaka, "High-Pressure Reflooding Experiments of Multi-Rod Bundle at ROSA-IV TPTF," Nuclear Engineering and Design, Volume 120, Issues 2-3, June 1990, pp. 301-310.

[50] Donald R. Todd, Cesare Frepoli, Lawrence E. Hochreiter, "Development of a COBRA-TF Model for the Penn State University Rod Bundle Heat Transfer Program," 7th International Conference on Nuclear Engineering, Tokyo, Japan, April 19-23, 1999, ICONE-7827, p. 3.

[51] Henry Kendall and Daniel Ford of Union of Concerned Scientists were the principal technical spokesmen of Consolidated National Intervenors, in the AEC ECCS rulemaking hearing.

[52] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," EQL Report No. 9, pp. A8-2, A8-6.

And discussing the use of stainless steel heater-rod assemblies in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The [stainless steel] rods were apparently chosen primarily for their durability. They could be used repeatedly in testing (for 30 or 40 individual tests) without substantial changes in response over the series. . . .
>
> On the other hand, *as a result of metal-water reactions, [Zircaloy] rods could be used only once* and then had to be subjected to a destructive post-mortem examination after the test [emphasis added].[53]

It is significant that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400 K. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[54] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425 K. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain*" [emphasis added].[55]

4. The Results of PWR Thermal Hydraulic Experiments Conducted with Zircaloy Bundles that Demonstrate that Low Reflood Rates do Not Prevent Zircaloy Cladding Temperatures from having Substantial Increases

National Research Universal's ("NRU") thermal-hydraulic experiments were conducted in the early '80s. NRU's thermal-hydraulic experiments were conducted with single bundles of full-length Zircaloy cladding, driven by low-level fission heat: an amount to simulate decay heat. In NRU Thermal-Hydraulic Experiment 1 ("TH-1"), a total of 28 tests were conducted. The tests were intended to simulate LB LOCAs. The

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[53] *Id.*, p. A8-6.
[54] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2.
[55] *Id.*, p. 4.4.

TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[56]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[57] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[58] "low enough that superheated steam conditions [would] exist at the loop inlet instrument location. The superheat requirement [was] imposed so that meaningful steam temperatures [could] be measured."[59])

As discussed in PRM-50-93 (page 18), the TH-1 tests[60] demonstrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126 (reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.7 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F).

Compare this to some of the TH-1 tests that had reflood rates of 5.9 in./sec. or greater: test no. 120 (reflood rate of 5.9 in./sec.) had a PCT at the start of reflood of 1460°F and an overall PCT of 1611°F (an increase of 151°F), test no. 113 (reflood rate of 7.6 in./sec.) had a PCT at the start of reflood of 1408°F and an overall PCT of 1526°F (an increase of 118°F), test no. 115 (reflood rate of 9.5 in./sec:) had a PCT at the start of reflood of 1666°F and an overall PCT of 1758°F (an increase of 92°F).

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[56] 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.
[57] *Id.*, p. 10.
[58] 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, p. 6-5.
[59] *Id.*
[60] For all of the values of reflood rates and PCTs in the TH-1 tests see 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, p. 13. This information is also available in PRM-50-93: Appendix D Table 1. Experimental Heat Cladding Temperatures (The 28 Tests from Thermal-Hydraulic Experiment 1).

It seems obvious that if the three TH-1 tests with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 30 seconds or higher, or had PCTs at the start of reflood that were 1200°F or higher, that the fuel assemblies, with high probability, would have incurred autocatalytic (runaway) oxidation, clad shattering, and failure—like FLECHT run 9573. It certainly seems obvious that if the parameters were the same for test no. 115 (PCT at the start of reflood of 1666°F), except it had a reflood rate of 1.2 in./sec. or lower, that its overall PCT would have increased above 2200°F and the fuel assembly, with high probability, would have incurred autocatalytic oxidation, clad shattering, and failure—like FLECHT run 9573.

It is significant that in NEI's comments, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," dated April 12, 2010, NEI states:

> Depending on the plant design, core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F...[61]

If indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F..."[62] it is highly problematic, because it means that, with high probability, reflood rates of 1 in./sec. or lower would not be sufficient to quench the core.

a. TH-1 Test No. 130

In TH-1 test no. 130, there was a reflood rate of 0.7 in./sec. At the start of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[63]

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

[61] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," April 12, 2010, Attachment, p. 3.
[62] *Id.*
[63] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.

cladding temperature was 2040°F.[64] 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 autocatalytic oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

(It is significant that 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 would have been 0.37 kW/ft,[65] 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.

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.[66]

(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.)

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[64] *Id.*
[65] *Id.*, p. 10.
[66] 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.

5. Criticizing the BWR Thermal Hydraulic Experiments, J. W. McConnell of Aerojet Concluded that "the Ability to Predict Accurately the Heat Transfer Coefficient and Metal-Water Reactions May Not be Proven"

It is significant that, regarding Aerojet internal memoranda that criticize the BWR-FLECHT program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" reports:

> J. W. McConnell (who will be co-author, with Dr. Griebe, of the as-yet-unpublished BWR-FLECHT final report from [Aerojet]) wrote:
>
> "There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the GE determination to prove out their ECC systems. Their role in this program can only be described as a conflict of interest as is the Westinghouse portion of PWR-FLECHT. Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ... A combination of poor data acquisition and transmission, faulty test approaches (probably caused by crude test facilities) and the marginal nature of these tests has produced a large amount of questionable data. It appears probable that the results of these tests can be interpreted. *But the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven. From a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective* [emphasis added]."[67]

So J. W. McConnell concluded that "the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven."[68] It is also significant that J. W. McConnell concluded that "from a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective"[69] in the BWR-FLECHT program.

(It is noteworthy that regarding the prospect of planning and conducting a new BWR-FLECHT program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> No recovery from the defects in the BWR-FLECHT Program are possible without a new program of greater scope being planned and carried out, like a new PWR-FLECHT Program, carried out in a way essentially free

[67] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 5.11.
[68] *Id.*
[69] *Id.*

of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[70]

Petitioner would add that such a new BWR-FLECHT program would have to be conducted with Zircaloy fuel assemblies. It would also be necessary that the PCTs of such tests exceeded those of the PWR TH-1 tests, conducted at Chalk River in the early '80s, where the test planners—"for safety purposes"—did not want the maximum PCTs of the TH-1 tests to exceed 1900°F[71]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.)

6. The primary BWR Heat Transfer Experiments Conducted since the BWR-FLECHT Tests, were Conducted with Inconel 600 Bundles

Unfortunately, it seems that none of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Zircaloy fuel assemblies.

Perhaps all of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Inconel 600 fuel rod simulators. For example, the Two-Loop Test Apparatus ("TLTA") facility had electrically heated Inconel 600 fuel rod simulators,[72] the Rig of Safety Assessment ("ROSA") III facility had electrically heated Inconel 600 fuel rod simulators,[73] and the Full Integral Simulation Test ("FIST") facility had electrically heated Inconel 600 fuel rod simulators.[74]

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[70] *Id.*, p. 5.41.

[71] 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, p. 3-3.

[72] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, August 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072480029, p. 5-27.

[73] Y. Koizumi, M. Iriko, T. Yonomoto, K. Tasaka, "Experimental Analysis of the Power Curve Sensitivity Test Series at ROSA-III," Nuclear Engineering and Design, 86, 1985, pp. 268, 270.

[74] General Electric, "BWR Full Integral Simulation Test (FIST) Program Facility Description Report" NUREG/CR-2576, EPRI NP-2314, GEAP-22054, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML073461126, pp. 2-32, 2-37; and Siemens, "EXEM BWR-2000 ECCS Evaluation Model," EMF-2361 (NP), October 2000, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML003772936, p. 5-2.

Additionally, the BWR FIX-II test facility had electrically heated Inconel 600 fuel rod simulators[75] and the NUPEC BWR Full-Size Fine-Mesh Bundle Test ("BFBT") facility had electrically heated Inconel 600 fuel rod simulators.[76]

Petitioner has not been able to locate information identifying the cladding material that was used in the fuel rod simulators in the 30° Steam Sector Test Facility ("SSTF"); in the SSTF, it is doubtful that Zircaloy was used as the fuel rod simulator cladding material. The SSTF experiments used steam injection to simulate core heat[77] and the maximum temperature of the steam was 800 F.[78]

It is significant that Inconel 600 does not oxidize nearly as much as Zircaloy in the design-basis accident temperature range.

Discussing Inconel 600's resistance to oxidation, "INCONEL alloy 600," states:

> INCONEL alloy 600 is widely used in the furnace and heat-treating fields for retorts, boxes, muffles, wire belts, roller hearths, and similar parts which require resistance to oxidation and to furnace atmospheres. ... The alloy's resistance to oxidation and scaling at 1800°F (980°C) is shown in Figure 11.[79]

Figure 11 of "INCONEL alloy 600," depicts a graph of the results of cyclic oxidation tests at 1800°F (980°C), in which there were alternating intervals of 15 minutes of heating and 5 minutes of cooling in air: Inconel 600 oxidized less than stainless steel (type 304), stainless steel (type 309), and Inconel 800HT. Inconel 600 oxidized very little over a period of 1000 hours of cyclic exposure time.[80]

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[75] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, pp. 5-119, 5-129.

[76] B. Neykov, F. Aydogan, L. Hochreiter, K. Ivanov, H. Utsuno, K. Fumio, E. Sartori, "NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT) Benchmark," Volume I: Specifications, NEA/NSC/DOC(2005)5, June 2005, pp. 15, 34.

[77] 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, p. 6.5-11.

[78] NRC, (Appendix A) "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053620415, Appendix A, p. A-208.

[79] Special Metals Corporation, "INCONEL alloy 600," www.specialmetals.com, SMC-027, 2008, p. 11.

[80] *Id.*

Additionally, in an Advisory Committee on Reactor Safeguards, subcommittee meeting on thermal hydraulic phenomena, on July 7, 2008, a participant, Mr. Kelly, discussing LOCA phenomena, stated that Inconel has "almost no oxidation."[81]

So Henry Kendall and Daniel Ford's criticisms of the BWR FLECHT tests conducted with stainless steel fuel rod simulators would also apply to BWR thermal hydraulic experiments conducted since the early 1970s with Inconel 600 fuel rod simulators.

It is significant that interpretations of the results of experiments conducted with Inconel 600 fuel rod simulators would most likely lead the interpreters to false conclusions. For example, a multi-rod Inconel 600 bundle heated up to peak cladding temperatures between 1832°F and 2200°F would not incur autocatalytic oxidation; however, a multi-rod Zircaloy bundle heated up to peak cladding temperatures between 1832°F and 2200°F would (with high probability) incur autocatalytic oxidation.

D. NEI's Claims Regarding the QUENCH-06 Experiment are Unsubstantiated

After NEI discusses the results of NEI's case study—shown in Figure 1 of NEI's comments—NEI claims that the results of NEI's case study are reinforced from calculations conducted in support of the QUENCH-06 experiment.

Regarding the QUENCH-06 experiment, NEI states:

> This [the results of NEI's case study, shown in Figure 1 of NEI's comments] is reinforced from calculations conducted in support of the Quench-06 test.[82] The maximum calculated bundle temperatures calculated in the simulated Quench-06 experiment are presented in Figure 2. *This experiment showed that with the proper heat balance it is possible for the cladding to attain high temperatures without approaching runaway oxidation* (until the power transient was initiated after 6000 seconds) [emphasis added].[83]

What NEI overlooks regarding the QUENCH-06 experiment is that the QUENCH-06 experiment had a *low* heatup rate—0.32 K/s between 1450 K (2150°F) and

[81] Mr. Kelly, NRC, Advisory Committee on Reactor Safeguards, Transcript of Subcommittee Meeting on Thermal Hydraulic Phenomena, July 7, 2008, p. 168.
[82] W. Hering, *et al.*, "Comparison and Interpretation Report of the OECD International Standard Problem No. 45 Exercise (QUENCH-06)," Forschungszentrum Karlsruhe, FZKA 6722, 2002.
[83] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

1750 K (2690°F)[84]—and that the initial heatup rate of a SFD experiment can determine whether the experiment will incur runaway oxidation or not.

It is significant that Figure 13 of "Current Knowledge on Core Degradation Phenomena, a Review" shows that with an initial core heat-up rate of ≤0.2 K/sec there would be *no temperature escalation.* Figure 13 also shows that with an initial core heat-up rate of ≥1 K/sec there would be a temperature escalation when cladding temperatures reached approximately 1200°C: when cladding temperatures reached 1200°C, the heatup rate would become ≥10 K/sec.[85]

So NEI's claim that the QUENCH-06 experiment "showed that with the proper heat balance it is possible for the cladding to attain high temperatures without approaching runaway oxidation,"[86] is unsubstantiated, because the low heatup rate of the QUENCH-06 experiment—0.32 K/s between 2150°F and 2690°F[87]—would have affected the QUENCH-06 experiment's results.

(It is noteworthy that, regarding the influence of heat-up rates on liquefaction of materials, "Current Knowledge on Core Degradation Phenomena, a Review" states:

> In addition to the temperature of the core, the local heat-up rates also have an important influence on the in-vessel core melt progression. These local heat-up rates can be largely controlled by local steam availability because of the importance of the exothermic Zircaloy/steam reaction. *At initial low heat-up rates <0.5 K/s, the fuel cladding is completely oxidized to ZrO_2 under steam-rich conditions before reaching the melting point of metallic Zircaloy.* As a result, fuel rod melting will not occur until 2600°C. *At initial heat-up rates above 1 K/s, temperatures are reached that permit the Zircaloy metal to melt and dissolve UO_2 before all the Zircaloy becomes oxidized* [emphasis added].[88])

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[84] L. Sepold, W. Hering, C. Homann, A. Miassoedov, G. Schanz, U. Stegmaier, M. Steinbrück, H. Steiner, J. Stuckert, "Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45)," Forschungszentrum Karlsruhe, FZKA 6664, 2004, p. iii, Abstract.

[85] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 205. It is significant that Figure 1 of "Current Knowledge on Core Degradation Phenomena, a Review" states that "[s]tart of rapid Zircaloy oxidation by [steam leads to] uncontrolled temperature escalation;" the temperature escalation commences at approximately 1200°C (p. 196).

[86] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

[87] L. Sepold, W. Hering, C. Homann, A. Miassoedov, G. Schanz, U. Stegmaier, M. Steinbrück, H. Steiner, J. Stuckert, "Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45)," Forschungszentrum Karlsruhe, FZKA 6664, p. iii, Abstract.

[88] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 204.

(It is also noteworthy that discussing the QUENCH-06 experiment, "Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45)" states:

> [In the QUENCH-06 experiment, at] the end of the stabilization period the bundle was ramped by stepwise increases in power up to about 11 kW to reach ~1473 K, the target temperature for pre-oxidation. This temperature was maintained for about 4600 s by control of the electrical power to reach a desired oxide layer thickness of about 200 μm.[89])

E. Response to NEI's Comments on the FLHT-1 Test: A Test in which Test Conductors were Not Able to Prevent Runaway Oxidation by Increasing the Coolant Flow Rate when Peak Cladding Temperatures Reached Approximately 2200°F

Regarding the fact that runaway oxidation does not commence at a specific temperature in SFD experiments, NEI states:

> The petitioner states that Zircaloy fuel assemblies would incur an autocatalytic oxidation, if they reach local cladding temperatures between approximately 1832°F (1000°C) and 2192°F (1200°C) (page 64 of PRM 50-95). An autocatalytic reaction does not occur at a specific temperature, but it occurs when the heat generation from the cladding metal-water reaction exceeds the cladding cooling by convection and radiation. This accounts for the lack of a fixed temperature for the accelerated reaction observed in the severe accidents mentioned by the petitioner.[90]

(Actually, in PRM 50-95, Petitioner states that, in the event of a LOCA, Zircaloy fuel assemblies would, *with high probability*, incur autocatalytic oxidation, if they reached temperatures between approximately 1832°F (1000°C) and 2192°F (1200°C). Petitioner does not state that autocatalytic oxidation would *always* commence at temperatures between approximately 1832°F and 2192°F.)

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[89] L. Sepold, W. Hering, C. Homann, A. Miassoedov, G. Schanz, U. Stegmaier, M. Steinbrück, H. Steiner, J. Stuckert, "Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45)," Forschungszentrum Karlsruhe, FZKA 6664, p. iii, Abstract.

[90] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

And regarding the first full-length high-temperature severe fuel damage ("FLHT-1") test, NEI states:

> A range [a temperature range for runaway oxidation commencing] between 2012°F (CORA 2-3 tests) and 2200°F (1204°C) (FLHT-1 test) is indicated in the petition. The reaction initiating temperature is dependent upon each experiment's cladding cooling condition. *If enough cooling is provided, the reaction can be terminated as occurred in the FLHT-1 test at 2150°F* [emphasis added].[91]

(Actually, in PRM 50-95, Petitioner reports that runaway oxidation commenced at 1832°F in the CORA-2 and CORA-3 experiments.[92] And in PRM-50-93 and PRM-50-93, Petitioner, argues that runaway oxidation commenced at approximately 2275°F or lower in the FLHT-1 test. Petitioner bases this argument on the fact that "Full-Length High-Temperature Severe Fuel Damage Test 1" reports that the test conductors could not control the Zircaloy oxidation rate and terminate the cladding-temperature increase by increasing the coolant flow rate, after peak cladding temperatures reached approximately 2200°F.[93])

It is significant that NEI points out that in the FLHT-1 test, the test conductors were able to prevent runaway oxidation by increasing the coolant flow rate when peak cladding temperatures reached approximately 2150°F (reported on in "Full-Length High-Temperature Severe Fuel Damage Test 1"[94]). It is also significant that in the FLHT-1 test that the test conductors were *not* able to prevent runaway oxidation by increasing the coolant flow rate when peak cladding temperatures reached approximately 2200°F.[95]

Clearly, the fact that in the FLHT-1 test, the test conductors were not able to prevent runaway oxidation by increasing the coolant flow rate when peak cladding temperatures reached approximately 2200°F, is another piece of evidence that indicates that the 10 C.F.R. § 50.46(b)(1) 2200°F PCT limit is non-conservative.

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[91] *Id.*

[92] 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, Abstract and p. 41.

[93] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, p. 4.6.

[94] *Id.*

[95] *Id.*

NEI's statement that "[i]f enough cooling is provided, the [runaway oxidation] reaction can be terminated as occurred in the FLHT-1 test at 2150°F,"[96] is not a valid argument that the 2200°F PCT limit is conservative, given the fact that in the FLHT-1 test, test conductors were *not* able to prevent runaway oxidation by increasing the coolant flow rate when peak cladding temperatures reached approximately 2200°F.

In fact, the FLHT-1 "test plan called for a gradual temperature increase to approximately 2150 K (3400°F),"[97] but "the planned [test] operations and predicted test behavior"[98] obviously did not work.

Discussing the FLHT-1 test plan in more detail, "Full-Length High-Temperature Severe Fuel Damage Test 1" states:

> Once the power is set, the test will be started through its transient operation. *The term transient is somewhat of a misnomer*; operation will consist of a series of preplanned, discrete flow-reduction steps. The size and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate. . . .
>
> The bundle [coolant] flow rate will then be decreased in a series of precalculated flow steps... The duration of the time between steps is dictated by the time needed to reach near steady state and also by *the requirement that the Zircaloy-steam reaction be limited.* About 14 steps, each of about 1/2 hr. duration, are expected. *The last flow reduction step will be calculated to give a peak cladding temperature of about 2150 K (3400°F). ...*
>
> The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150 K (3400°F) [emphasis added].[99]

Indeed, the test conductors must have been taken by surprise when they could not control the zircaloy oxidation rate by increasing the coolant flow rate. They realized that there was no way to terminate the cladding-temperature increase—after peak cladding temperatures reached approximately 1475 K (2200°F)—short of reducing the reactor

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[96] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.
[97] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. v.
[98] *Id.*
[99] *Id.*, pp. 4.3-4.5.

power to zero power, as they did "85 seconds after the start of the [cladding temperature] excursion."[100]

It is important to remember that the events described above occurred within a period of approximately 85 seconds: peak cladding temperatures increased from approximately 1520 K (~2275°F) or lower to approximately 2275 K (3635°F), within approximately 85 seconds.

The description of the procedure of the FLHT-1 test in "Full-Length High-Temperature Severe Fuel Damage Test 1," also indicates that the rapid temperature increase began at a temperature of approximately 1520 K (~2275°F) or lower. "Full-Length High-Temperature Severe Fuel Damage Test 1" states:

> Typical cladding temperature behavior at one position in the assembly during the test is shown in Figure 4.1. At about 60 to 70 min. along the abscissa, a temperature increase [commenced] when the [bundle coolant] flow rate was about 9 kg/hr. (20 lb/hr.). The [cladding] temperature increased until about 95 min. and [reached] 1450 K (2150°F), at which time the bundle coolant [flow] rate was increased to 18 kg/hr. (40 lb/hr.) to stabilize the temperature. However, the [cladding] temperature rapidly dropped to about 1060 K (1450°F). The bundle coolant flow rate was then decreased through a series of steps to a minimum of 9 kg/hr. (20 lb/hr.). This action stopped the temperature decrease and started another temperature rise. *When the temperature reached about 1475 K (2200°F), the bundle coolant flow [rate] was again increased to stop the temperature ramp.* This led to a stabilized condition. *The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly* until the test director requested that the reactor power be reduced to zero power [emphasis added].[101]

First, it is obvious from the above description (and from Figures 4.1 and 5.4 of "Full-Length High-Temperature Severe Fuel Damage Test 1") that when cladding temperatures reached approximately 1475 K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition"[102] was not achieved. Cladding temperatures continued to rise. This is clearly stated: "The flow was increased in steps and reached a

[100] *Id.*, p. 4.6.
[101] *Id.*
[102] *Id.*

maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly..."[103]

Second, it is obvious that the rapid metal-water reaction began at cladding temperatures far lower than 1700 K (2600°F), as reported in "Full-Length High-Temperature Severe Fuel Damage Test 1."[104] It makes no sense that the autocatalytic oxidation reaction would have begun at 1700 K (2600°F). How can it be explained that after the coolant flow rate was increased—when cladding temperatures reached approximately 1475 K (2200°F)—that the cladding temperatures were able to increase by 225 K (400°F)? Why would the test conductors have not been able to terminate the cladding-temperature rise, as they did earlier in the test when cladding temperatures reached 1450 K (2150°F)? And how can it be explained that the test conductors did not have enough time to increase the coolant flow rate back up to 18 kg/hr. (40 lb/hr.), as they did when cladding temperatures reached 1450 K (2150°F), earlier in the test?

So peak cladding temperatures reached approximately 1475 K (2200°F) and the test conductors could not terminate the temperature rise by increasing the coolant flow rate; they increased the flow rate up to approximately 15 kg/hr. (34 lb/hr.) yet still could not prevent the autocatalytic oxidation reaction. The onset of the autocatalytic oxidation reaction must have taken them by surprise.

The FLHT-1 test is highly significant precisely because, once cladding temperatures reached as high as approximately 1475 K (2200°F), the test conductors could not prevent the cladding-temperature rise by increasing the coolant flow rate.

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[103] *Id.*

[104] It is noteworthy that "Full-Length High-Temperature Severe Fuel Damage Test 1" states that at approximately 1700 K (2600°F) the Zircaloy cladding in the FLHT-1 test began to rapidly oxidize, causing a rapid local bundle temperature excursion (p. 4.11); however, it is far more likely that the Zircaloy cladding actually began to rapidly oxidize at a temperature of approximately 1520 K (~2275°F) or lower. "Full-Length High-Temperature Severe Fuel Damage Test 1" has inconsistent statements regarding the time that the Zircaloy cladding temperature excursion began—the autocatalytic (runaway) oxidation reaction.

"Full-Length High-Temperature Severe Fuel Damage Test 1" states that "[t]he reactor power was decreased at approximately 17:11:07, 85 seconds after the start of the [cladding temperature] excursion" (p. 4.6); *i.e.*, the cladding temperature excursion began at 17:09:42. However, "Full-Length High-Temperature Severe Fuel Damage Test 1" also states that the cladding temperature excursion began 18 seconds latter at 17:10:00—when the cladding temperature was 1700 K (p. 4.11). The difference of 18 seconds is highly significant, because it means that the cladding temperatures were much lower than 1700 K when the temperature excursion actually began.

Increasing the coolant flow rate did not prevent the onset of an autocatalytic oxidation reaction—which occurred at cladding temperatures of approximately 1520 K (~2275°F) or lower.

F. Some of NEI's Statements Could be Used to Support Making Regulations Stipulating Minimum Reflood Rates and Minimum Allowable Amounts of Coolant to be Supplied to Each Fuel Assembly by BWR Core Spray Systems

Some of NEI's statements regarding PRM-50-93 and PRM-50-95 could be used to support making regulations stipulating minimum reflood rates [105] and minimum allowable amounts of coolant to be supplied to each fuel assembly by BWR core spray systems.[106] For example, NEI states that "[e]vidence shows that with sufficient cooling to account for the heat generation from [the] metal-water reaction the threat of clad melting is abated."[107]

In NEI's comments, NEI also states:

> *At any temperature approaching the 10 CFR 50.46 limit, a significant decrease in cooling could lead to a rapid increase in heating rate.* Such a situation would have to be analyzed on a case-by-case basis, since so many variables exist. A balance between heat addition and removal must be understood in order to make conclusions about any phenomena impacting the system while experiencing such a self-sustaining reaction [emphasis added].[108]

Indeed, in the event of a LOCA, "[a]t any temperature approaching the 10 CFR 50.46 [2200°F PCT] limit, a significant decrease in cooling could lead to a rapid increase

[105] 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.

[106] "Resolution of Generic Safety Issues: Item A-16: Steam Effects on BWR Core Spray Distribution" states that "to ensure the health and safety of the public, [BWR] core spray systems must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores."

[107] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 3.

[108] *Id.*

in [the] heating rate."[109] For this reason, it makes sense for NRC to make new regulations stipulating minimum reflood rates and minimum allowable amounts of coolant to be supplied to each fuel assembly by BWR core spray systems.

(It is noteworthy that neither PRM-50-93 nor PRM-50-95 requested a regulation stipulating that BWR core spray systems must supply minimum allowable amounts of coolant to each fuel bundle in the BWR core, in the event of a LOCA. In the event of a LOCA at a BWR, it would be important to supply each fuel assembly with a minimum amount of coolant to help ensure that the fuel cladding would be cooled.)

G. Temperature Differences of the BWR Cruciform Control Blades and the Fuel Cladding in the Event of a LOCA

In NEI's comments on PRM-50-93 and PRM-50-95, NEI discusses the temperature differences between the BWR cruciform control blades and the fuel cladding in the CORA-16 experiment.

Regarding this issue, NEI states:

> The petitioner also states that current BWR components (control blades) would be damaged if the cladding reaches a temperature between 1832°F (1000°C) and 2192°F (1200°C) (page 65 of PRM 50-95). The petitioner's basis for this statement is based upon the melting reaction between B_4C and stainless steel beginning at approximately 1832°F (1000°C) and accelerating above 2192°F (1200°C). LOCA licensing calculations indicate that when the 1832°F (1000°C) cladding temperature is reached, the temperatures in the control blades are at least 392°F (200°C) lower. This is corroborated by the CORA-16 temperature measurements (Figures 16 and 17 of FZKA 7447 report January 2009). Thus, a 2200°F (1204°C) limit in the cladding temperature is enough to ensure not reaching 1832°F (1000°C) in the control blade. The cladding temperature proposed limit of 1832°F (1000°C) to prevent the initiation of control blade melting at 1832°F (1000°C) is not justified.[110]

First, NEI is correct that the temperature of the BWR cruciform control blades would be significantly below that of peak fuel cladding temperatures, in the event of a LOCA, as demonstrated by the CORA-16 experiment. However, if the fuel cladding were to incur runaway oxidation between 1832°F and 2192°F, peak cladding

[109] *Id.*
[110] *Id.*, Attachment, p. 4.

temperatures would begin rapidly increasing at tens of degrees Fahrenheit per second. And within tens of seconds peak cladding temperatures would increase to over approximately 2600°F and temperatures of the cruciform control blades would also increase to temperatures over approximately 2192°F (1200°C). This is seen in the figures NEI cites in NEI's comments on the CORA-16 experiment: figures 16 and 17 of FZKA 7447.[111]

Clearly, the fact that there would be complete liquefaction of the stainless steel of the BWR control blade at approximately 1250°C (2282°F), instead of at temperatures between 1375 and 1425°C (2507 and 2597°F),[112] is a significant nuclear power safety issue. And, clearly, data from the CORA-16 experiment—i.e., the B_4C-stainless steel reaction beginning at approximately 1000°C (1832°F) and the stainless steel cladding of the B_4C absorber material liquefying very quickly above 1200°C (2192°F)[113]—is further evidence that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F should be lowered. Lowering the 2200°F PCT limit would provide a necessary margin of safety and help prevent a partial or complete meltdown, in the, event of a LOCA.

It is significant that in Dr. Robert E. Henry's (of Fauske & Associates) presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007,[114] one of the presentation slides states that "the core damage was generally caused by the cladding oxidation."[115] And another one of Dr. Robert E. Henry's presentation slides states that "[t]he chemical energy release [from the oxidation of the

[111] See Appendix A Figure 16. CORA-16; Temperatures of Unheated Rods and Figure 17. CORA-16; Temperatures of the Absorber Blade.
[112] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," Oak Ridge National Laboratory, 1997, pp. 4-5.
[113] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, p. 11.
[114] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 ANS/ENS International Meeting, November 11, 2007, seven of these presentation slides are in attachment 2 of the transcript from "10 C.F.R. 2.206 Petition Review Board Re: Vermont Yankee Nuclear Power Station", July 26, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML102140405, Attachment 2.
[115] *Id.*

Zircaloy fuel cladding by steam] caused the core to overheat faster and eventually melt or liquefy the individual constituents."[116]

(It is noteworthy that, in 1981, Fauske & Associates developed the Modular Accident Analysis Program ("MAAP") code in response to the TMI-2 accident—under sponsorship from Electric Power Research Institute ("EPRI") and MAAP Users Group.)

Second, not mentioned in PRM-50-95, is the fact that, in the event of a LOCA, there could be chemical interactions between Zircaloy and stainless steel and between Zircaloy and Inconel at "low temperatures."

It is significant that "[t]he chemical reaction between Inconel and Zircaloy influences the meltdown of the reactor core in the vicinity of Inconel grid spacers."[117]

Regarding the relatively low temperatures at which chemical interactions between Inconel and Zircaloy could occur, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core" states:

> *Grid spacers can have a significant impact on the progression of damage in a reactor core during a severe accident.* ... The impact of grid spacers on damage progression has been revealed by out-of-pile experiments in Germany[118] and Japan,[119] in-pile experiments at the PBF facility in Idaho,[120] and by examinations of the damaged Three Mile Island (TMI-2) core.[121] *The experiments in Germany and Japan have revealed the existence of chemical interactions between Inconel and Zircaloy that take place at temperatures as low as 1273 K [(1832°F)]*, more than 200 K lower than the melting temperature of Inconel. *Thus in a reactor core*

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[116] *Id.*

[117] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," Nuclear Engineering and Design 146, 1994, Abstract, p. 427.

[118] E.A. Garcia, P. Hofmann, and A. Denis, "Chemical Interaction between Inconel Spacer Grids and Zircaloy Cladding; Formation of Liquid Phases due to Chemical Interaction and Its Modeling," Kernforschungszentrum Karlsruhe, KfK 4921; S. Hagen, *et al.*, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C," Kernforschungszentrum Karlsruhe, KfK 4378, September 1990; and P. Hofmann, *et al.*, "Low-Temperature Liquefaction of LWR Core Components," Severe Accident Research Program Partners Review Meeting, Brookhaven National Laboratory, Upton, New York, April 30 to May 4, 1990.

[119] F. Nagase, *et al.*, "Interaction between Zircaloy Tube and Inconel Spacer Grid at High Temperature," JAERI-M 90-165, Japan Atomic Energy Research Institute, August 1990.

[120] D.A. Petti, *et al.*, "PBF Severe Fuel Damage Test 1-4 Test Results Report," NUREG/CR-5163, EGG-2542, EG&G Idaho Inc., December 1986.

[121] E.L. Tolman, *et al.*, "TMI-2 Accident Scenario Update," EGG-TMI-7489, EG&G Idaho, Inc., December 1986.

with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers [emphasis added].[122]

It is significant that grid spacers would effect the progression of damage in a reactor core during a LOCA if temperatures were to reach approximately 2012°F;[123] and significant that experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.

And discussing chemical interactions between Zircaloy and stainless steel and between Zircaloy and Inconel, in more detail, "Current Knowledge on Core Degradation Phenomena, a Review" states:

> The Zircaloy/stainless steel (1.4919; corresponds to [stainless steel] Type 316 with 18 wt% Ni and 8 wt% Cr) interactions are important with respect to the contact between the absorber rod cladding and the Zircaloy guide tube and between the Inconel spacer grid and the Zircaloy fuel rod cladding. In both cases, the iron-zirconium and the nickel-zirconium phase diagrams show that due to eutectic interactions, early melt formation has to be expected, which initiates the melt progression within the fuel assembly at low temperatures. *Liquid phases form at temperatures <1000°C; however, the reaction kinetics become significant only above 1100°C.* This was seen in the CORA tests, where fuel rod bundles were heated up to complete meltdown.' In all cases, the damage of the bundle was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions. *Localized liquefaction of these components started around 1200°C.*[124]
>
> The reaction kinetics between Zircaloy and stainless steel can be divided into a reaction zone growth rate in Zircaloy and one in stainless steel, as shown in Fig. 11. One can see that the Zircaloy is attacked more strongly than the stainless steel. Oxide layers on the Zircaloy cladding outside diameter delay the chemical interactions between Zircaloy and steel, but they cannot prevent them. *The influence of oxide layers becomes less important at temperatures >1100°C, since the dissolution of the protecting ZrO_2 layers occurs rather fast and the stainless steel is then in contact with metallic Zircaloy or oxygen-stabilized α-Zr(O).*[125]

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[122] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.
[123] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[124] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[125] P. Hofmann, M. Markiewicz,, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Stainless Steel at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 5106, 1994.

> In a first approach, the reaction behavior of Zircaloy with Inconel 718 is comparable to that with Type 316 stainless steel.[126] *At temperatures <1100°C, Inconel attacks the Zircaloy faster than stainless steel; above 1100°C, the situation is the reverse.* In both cases, the melting of a relatively large quantity of Zircaloy with limited melting of the adjacent stainless steel or Inconel takes place. *During heat-up of the stainless steel/Zircaloy and Inconel/Zircaloy reaction systems, a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C. This may be the reason that melt progression in a fuel rod bundle initiates at absorber rod cladding (stainless steel)/Zircaloy guide tube contact areas and Inconel spacer grid/Zircaloy fuel rod contact locations*[127] [emphasis added].[128]

It is significant that in the CORA tests, in which fuel rod bundles were heated up to complete meltdowns, that "the damage of the [bundles] was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions"[129] and that "[l]ocalized liquefaction of these components started around 1200°C [(2192°F)]."[130] It was also observed in the CORA tests that "[l]iquid phases form at temperatures <1000°C [(1832°F)]"[131] and that "the reaction kinetics become significant only above 1100°C [(2012°F)]."[132]

It is significant that in Dr. Robert E. Henry's (of Fauske & Associates) presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007,[133] one of the presentation slides states that "the core damage was generally caused by the cladding oxidation."[134] And another one of Dr. Robert E. Henry's

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[126] P. Hofmann, M. Markiewicz, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Inconel 718 at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 4729, 1994.
[127] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[128] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[129] *Id.*
[130] *Id.*
[131] *Id.*
[132] *Id.*
[133] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 ANS/ENS International Meeting, November 11, 2007, seven of these presentation slides are in attachment 2 of the transcript from "10 C.F.R. 2.206 Petition Review Board Re: Vermont Yankee Nuclear Power Station", July 26, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML102140405, Attachment 2.
[134] *Id.*

presentation slides states that "[t]he chemical energy release [from the oxidation of the Zircaloy fuel cladding by steam] caused the core to overheat faster and eventually melt or liquefy the individual constituents."[135]

(It is noteworthy that, in 1981, Fauske & Associates developed the MAAP code in response to the TMI-2 accident—under sponsorship from EPRI and MAAP Users Group.)

H. Response to NEI's Comments on the Hobson/Rittenhouse Furnace Experiments

Discussing the Hobson/Rittenhouse furnace experiments, NEI states:

> Although not well addressed at the time of the 1973 Hearings, the accuracy of Hobson's oxidation temperatures of 2200°F (1204°C) and 2400°F (1315°C) has been challenged by the subsequent investigators. The temperature reported in Reference 1[136] was the furnace temperature rather than actual specimen temperature that is more accurately measured with a directly spot-welded thermocouple as has been done by investigators such as Cathcart-Pawel and more recently at ANL. Considering the high oxidation heat, actual specimen temperature reported as 2200°F (1204°C) in the Hobson experiments was probably close to ~2300°F (~1260°C).[137]

On the same point that NEI makes, regarding the significant exothermic heat of oxidation of Zircaloy that was not well recognized in the Hobson/Rittenhouse furnace experiments, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report" states:

> It is important to realize that in the early experiments of oxidation of Zircaloys at high temperatures,[138] specimen temperatures were not measured directly; *e.g.*, by using spot-welded thermocouples. Likewise,

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[135] *Id.*

[136] Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," ANS Topical Meeting on Water Reactor Safety, 1973, Salt Lake City, pp. 274-288.

[137] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-95); NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC-2009-0554," Attachment, p. 2.

[138] Hesson, J. C., *et al.*, "Laboratory Simulations of Cladding-Steam Reactions Following Loss-of-Coolant Accidents in Water-Cooled Power Reactors," Argonne National Laboratory, ANL-7609, January 1970; Hobson, D. O., Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972; and Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," pp. 274-288.

specimen temperatures in the experiment of Baker-Just[139] were determined indirectly. *Before [the] mid-1970s, it appears that the effect of the large exothermic heat of oxidation of [Zircaloy] was not well recognized by the investigators.* In Hobson's experiments, [140] the temperature of [the] Zircaloy tube being oxidized was assumed to be the same as the temperature of the uniform central zone of the high-temperature furnace. This assumption would be reasonable for low temperatures; *e.g.*, <800°C. *However, at higher temperatures—e.g., >1100°C—high rate of self-heat generation from oxidation causes actual specimen temperature significantly higher than that of the furnace temperature. In this respect, actual oxidation temperature of a Zircaloy tube reported in Hobson's experiment is believed to be significantly higher, e.g., 1200°C vs. 1260°C* [emphasis added].[141]

It is significant that, according to "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," in the Hobson/Rittenhouse furnace experiments, the temperature of a Zircaloy tube would have been approximately 1260°C when the furnace temperature was 1200°C. So in the Hobson/Rittenhouse furnace experiments, the radiative heat losses of the Zircaloy tube specimens to the furnace environment—that apparently at 1200°C was approximately 60°C lower than the specimen temperature—would have affected the specimens' oxidation kinetics in the experiments.

The hot spot (at 1260°C) of fuel rods in a reactor core, in a LOCA environment, would have a greater oxidation rate than a Zircaloy tube specimen (at 1260°C) in a furnace environment in which the furnace temperature was 1200°C.

(It is noteworthy that "[b]efore [the] mid-1970s, it appears that the effect of the large exothermic heat of oxidation of [Zircaloy] was not well recognized by the investigators,"[142] because the Baker-Just equation—required by Appendix K to Part 50 I(A)(5)—which calculates the rate of energy release from the metal-water reaction, dates back to 1962.)

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[139] 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.
[140] Hobson, D. O., Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," ORNL-4758 and Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," pp. 274-288.
[141] Nuclear Energy Agency, OECD, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," NEA No. 6846, 2009, p. 38.
[142] *Id.*

III. CONCLUSION

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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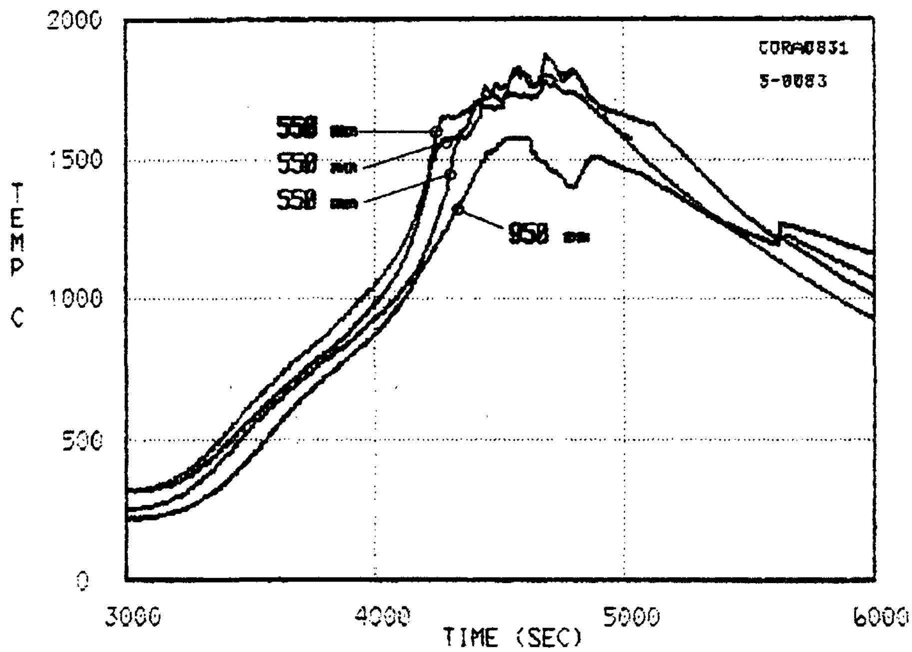
Appendix A Figure 16. CORA-16; Temperatures of Unheated Rods and Figure 17. CORA-16; Temperatures of the Absorber Blade[1]

[1] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, pp. 62-63.



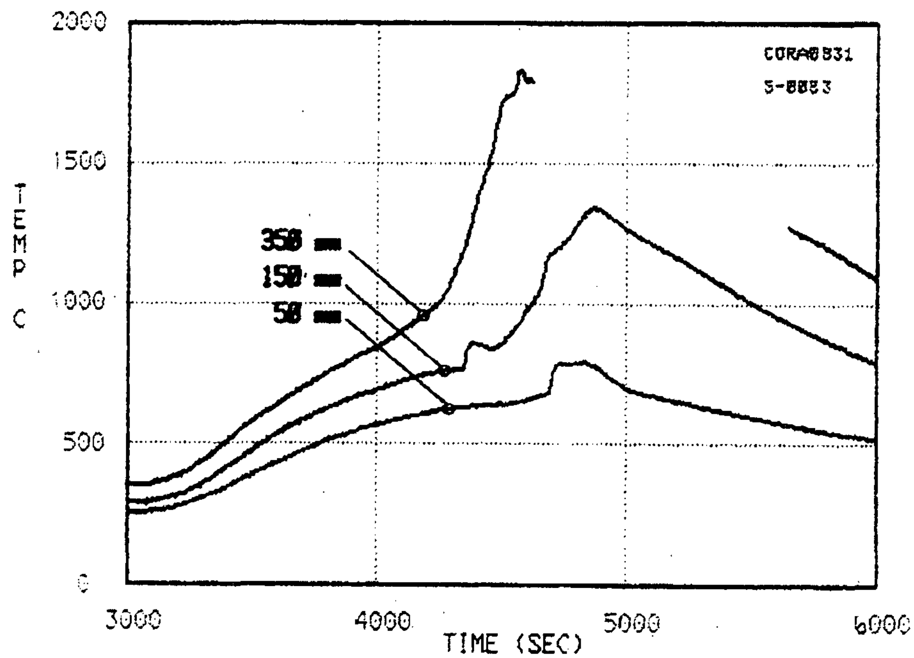


Fig. 16: CORA-16; Temperatures of unhheated rods

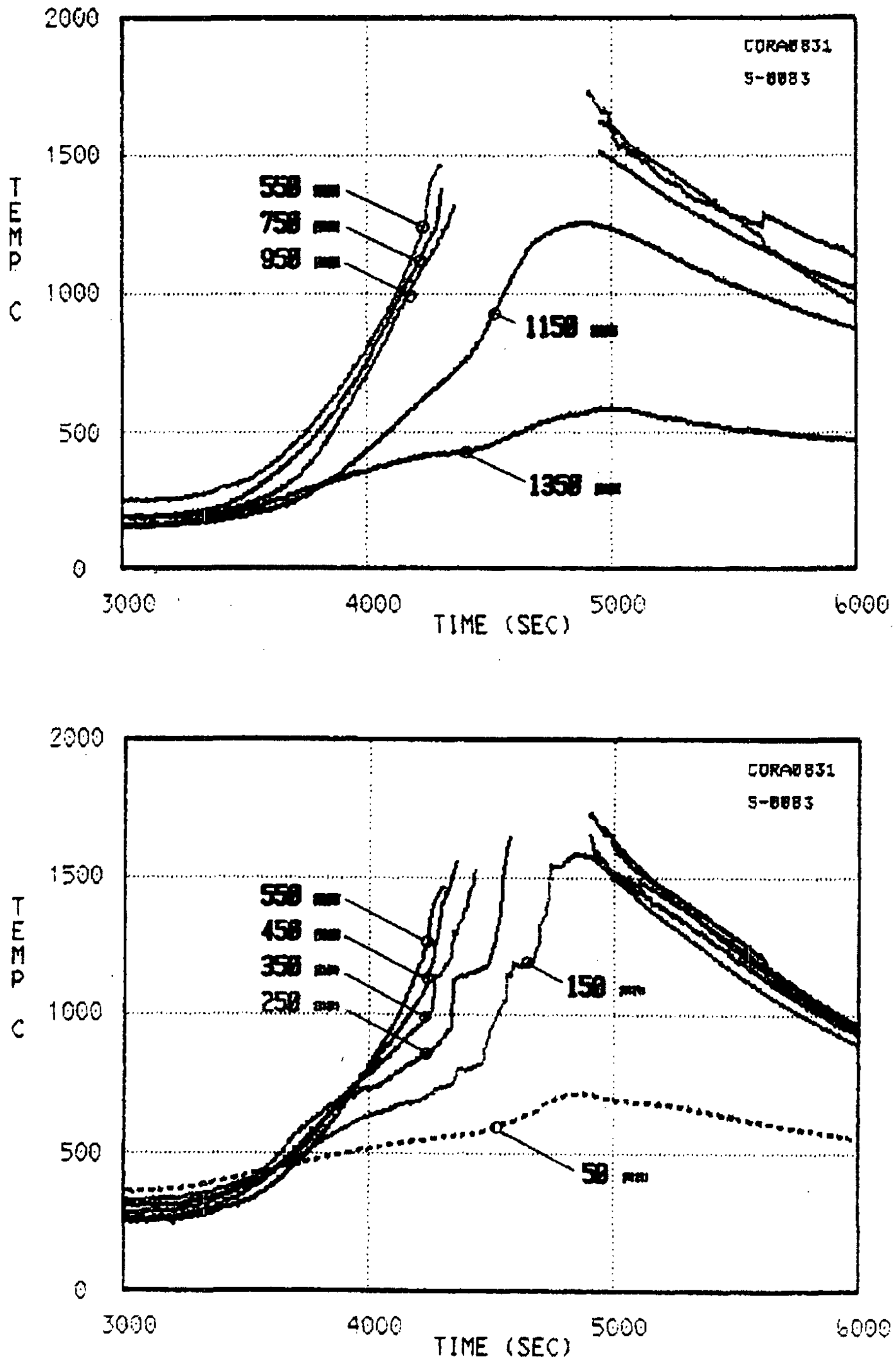


Fig. 17: CORA-16; Temperatures of the absorber blade

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