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ADJUDICATIONS STAFF

April 28, 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; 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, published in the Federal Register, January 25, 2010. In these comments on PRM-50-93, Petitioner responds to the Nuclear Energy Institute's comments on PRM-50-93, dated April 12, 2010.

Respectfully submitted,

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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; NRC-2009-0554

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

I. Statement of Commentator's ("Petitioner") Interest

On November 17, 2009, Mark Edward Leyse, Commentator ("Petitioner") submitted a petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the U.S. 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 calculations be based on data from multi-rod (assembly) severe fuel damage

[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) peak cladding temperature ("PCT") limit of 2200°F is non-conservative.

[2] It can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (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 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.

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 March 15, 2007, Petitioner submitted a petition for rulemaking, PRM-50-84 (ADAMS Accession No. ML070871368). In 2008, the NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process. PRM-50-84 requested new regulations: 1) to require licensees to operate LWRs under conditions that effectively limit the thickness of crud (corrosion products) and/or oxide layers on fuel cladding, in order to help ensure compliance with 10 C.F.R. § 50.46(b) ECCS acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.

Additionally, PRM-50-84 requested that the NRC amend Appendix K to Part 50—ECCS Evaluation Models I(A)(1), *The Initial Stored Energy in the Fuel*, to require that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of crud and/or oxide layers on cladding plays in increasing the stored energy in the fuel. PRM-50-84 also requested that these same requirements apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.

PRM-50-84 was summarized briefly in the American Nuclear Society's *Nuclear News*'s June 2007 issue[6] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[7] by the Union of Concerned Scientists.

[4] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating 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] American Nuclear Society, *Nuclear News*, June 2007, p. 64.

[7] David Lochbaum, Union of Concerned Scientists, "Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84)," July 31, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072130342, p. 3.

Petitioner also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis," which was presented at the American Nuclear Society's 2009 Winter Meeting, November 15-19, 2009, Washington, D.C.

In these comments on PRM-50-93, Petitioner responds to the Nuclear Energy Institute's ("NEI") comments on PRM-50-93, dated April 12, 2010.

II. Response to the Nuclear Energy Institute's Comments on PRM-50-93

A. NEI's Misrepresentations of Petitioner's Arguments in PRM-50-93

In Petitioner's response to NEI comments on PRM-50-93, Petitioner will begin by addressing NEI's misrepresentations of Petitioner's argument in PRM-50-93.

First, in NEI's comments on PRM-50-93, NEI erroneously states:

> The petitioner claims that [FLECHT Run 9573] demonstrates that the zirconium-water autocatalytic reaction was reached at temperatures below 2200°F.[8]

In no section of PRM-50-93, and in no section of Petitioner's comments on PRM-50-93, does Petitioner state that a zirconium-water autocatalytic reaction was reached at temperatures below 2200°F in FLECHT Run 9573.

In PRM-50-93 (on page 49), Petitioner quotes Westinghouse's comments on PRM-50-76. As quoted in PRM-50-93, Westinghouse stated, "[d]espite the severity of the conditions [of FLECHT Run 9573] and the observed extensive zirconium-water reaction, the oxidation was within the expected range and runaway oxidation [occurred] beyond 2300°F."[9]

Then in PRM-50-93 (on page 49), Petitioner states that "an occurrence of runaway (autocatalytic) oxidation at a temperature greater than 2300°F (assuming that means at a temperature below 2400°F) is not within 'the expected range' of what the Baker-Just correlation would predict: the Baker-Just correlation predicts that

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[8] 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. 2.

[9] 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.

autocatalytic oxidation of Zircaloy occurs at cladding temperatures of approximately 2600°F."[10, 11]

So, in PRM-50-93, Petitioner pointed out that Westinghouse stated that "runaway oxidation [occurred] beyond 2300°F"[12] in FLECHT Run 9573; Petitioner did not claim that runaway oxidation occurred below 2200°F in FLECHT Run 9573.

(It is noteworthy that in its comments on PRM-50-93, NEI erroneously classifies FLECHT Run 9573 as a "multirod severe fuel test."[13] NEI does not seem to understand what kind of experiments the PWR Full Length Emergency Cooling Heat Transfer ("FLECHT") experiments were. The FLECHT experiments were thermal hydraulic experiments, not severe damage fuel experiments. In PRM-50-93 (on page 48), Petitioner states that "FLECHT run 9573 was a thermal hydraulic test; however, in some respects it resembled a severe fuel damage test."[14])

Second, in NEI's comments on PRM-50-93, NEI erroneously states:

> The petitioner bases the claim for a fixed minimum reflood rate on FLECHT Run 9573.[15]

In PRM-50-93, Petitioner argues for a new regulation stipulating minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA"), *primarily* by citing experimental data from the National Research Universal ("NRU") Thermal-Hydraulic Experiment 1 (a total of 28 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), NRU Thermal-Hydraulic Experiment 2 (a total of 14 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), and NRU Thermal-Hydraulic Experiment 3 (a total of three thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven

[10] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."
[11] Mark Edward Leyse, PRM-50-93, November 17, 2009, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML093290250, p. 49.
[12] H. A. Sepp, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[13] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 2.
[14] Mark Edward Leyse, PRM-50-93, p. 48.
[15] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

by low-level fission heat). (In PRM-50-93, Petitioner discusses the NRU reactor thermal-hydraulic experiments on pages 14-20, 24, 73-74, 75, and Appendix D lists data from the 28 tests conducted in Thermal-Hydraulic Experiment 1.)

Third, in the cover letter of NEI's comments on PRM-50-93, NEI misleadingly states:

> In support of this request, the petitioner cites results from two out of many tests performed over 25 years ago.[16]

In the passage above from NEI's cover letter, NEI does not identify the two experiments it is referring to; however, in the attachment, "NEI Comments on Petition for Rulemaking (PRM-50-93)," NEI comments on two experiments discussed in PRM-50-93: FLECHT Run 9573 and the LOFT LP-FP-2 experiment.

In PRM-50-93, and in Petitioner's comments on PRM-50-93, Petitioner discusses data from over 60 experiments (tests) to argue for the regulations PRM-50-93 proposes.

Regarding reflood rates, Petitioner *primarily* discusses data from the following experiments: NRU Thermal-Hydraulic Experiment 1 (a total of 28 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), NRU Thermal-Hydraulic Experiment 2 (a total of 14 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), NRU Thermal-Hydraulic Experiment 3 (a total of three thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat).

Regarding reflood rates, Petitioner also discusses data from the following experiments: FLECHT Run 9573 (a thermal hydraulic test conducted with full-length Zircaloy fuel rod simulators), FLECHT-SEASET test 31504 (a thermal hydraulic test conducted with full-length stainless steel fuel rod simulators), FLECHT Runs 6553 and 9278 (thermal hydraulic tests conducted with full-length stainless steel fuel rod simulators). (Regarding reflood rates, FLECHT-SEASET test 31504 and FLECHT Runs 6553 and 9278 are discussed in Petitioner's comment on PRM-50-93, dated March 15, 2010.)

Regarding the metal-water reaction rate and/or experimental data that indicates the Baker-Just and Cathcart-Pawel equations are non-conservative, Petitioner discusses

[16] *Id.*, Cover Letter, p. 1.

data from the following multi-rod experiments: the Power Burst Facility ("PBF") Severe Fuel Damage ("SFD") 1-1 test, PBF SFD 1-3 test, PBF SFD 1-4 test, NRU Materials Test 6B, NRU Reactor Full-Length High-Temperature 1 Test, the LOFT LP-FP-2 experiment, the CORA Experiments as a whole, the CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments, the PHEBUS B9R test, the QUENCH-04 test, PWR FLECHT Run 9573, and the BWR FLECHT Zr2K test. (The CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments, and the BWR FLECHT Zr2K test are discussed in Petitioner's comment on PRM-50-93, dated March 15, 2010.)

(PWR FLECHT Run 9573 and the BWR FLECHT Zr2K test were thermal hydraulic tests; however, in some respects they resembled severe fuel damage tests.)

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner *primarily* discusses data from the following multi-rod experiments: the LOFT LP-FP-2 experiment, the CORA Experiments as a whole, and the CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments.

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner also discusses data from the BWR FLECHT Zr2K test: Petitioner points out that graphs of thermocouple measurements taken during the Zr2K test depict temperature excursions that began when cladding temperatures reached between approximately 2100 and 2200°F.

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner also discusses data from experiments, where the onset of autocatalytic oxidation occurred above 2200°F. It can be concluded that 2200°F peak cladding temperature ("PCT") limit does not provide a necessary margin of safety from the following experiments: NRU Reactor Full-Length High-Temperature 1 Test, the PHEBUS B9R test, and the QUENCH-04 test.

B. NEI's Misinterpretations of FLECHT Run 9573 and Misrepresentations of Petitioner's Discussion of FLECHT Run 9573 in PRM-50-93

First, as stated above, NEI erroneously classifies FLECHT Run 9573 as a "multirod severe fuel test."[17] NEI does not seem to understand what kind of experiments the PWR Full Length Emergency Cooling Heat Transfer experiments were. The FLECHT experiments were thermal hydraulic experiments, not severe damage fuel experiments. (In PRM-50-93 (on page 48), Petitioner states that "FLECHT run 9573 was a thermal hydraulic test; however, in some respects it resembled a severe fuel damage test."[18])

Second, in NEI's comments on "multirod severe fuel tests," NEI states:

> The petitioner claims that [FLECHT Run 9573] demonstrates that the zirconium-water autocatalytic reaction was reached at temperatures below 2200°F. The petitioner's use of autocatalytic is wrong. What occurred is that the oxidation became significantly out of balance with the cooling taking place.[19]

As mentioned above, in no section of PRM-50-93, and in no section of Petitioner's comments on PRM-50-93, does Petitioner state that a zirconium-water autocatalytic reaction was reached at temperatures below 2200°F in FLECHT Run 9573.

In PRM-50-93 (on page 49), Petitioner quotes Westinghouse's comments on PRM-50-76. As quoted in PRM-50-93, Westinghouse stated, "[d]espite the severity of the conditions [of FLECHT Run 9573] and the observed extensive zirconium-water reaction, the oxidation was within the expected range and runaway oxidation [occurred] beyond 2300°F."[20] So, in 2002, Westinghouse stated that runaway oxidation (or autocatalytic oxidation) occurred in FLECHT Run 9573, seven years before Petitioner stated that runaway oxidation (or autocatalytic oxidation) occurred in FLECHT Run 9573, in PRM-50-93. Evidently, NEI believes Westinghouse's description of runaway oxidation occurring in FLECHT Run 9573 is erroneous.

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[17] *Id.*, Attachment, p. 2.
[18] Mark Edward Leyse, PRM-50-93, p. 48.
[19] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 2.
[20] H. A. Sepp, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.

Third, as discussed above, NEI erroneously states:

> The petitioner bases the claim for a fixed minimum reflood rate on FLECHT Run 9573.[21]

NEI's statement is erroneous. In PRM-50-93, Petitioner argues for a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA, *primarily* by citing experimental data from the NRU Thermal-Hydraulic Experiment 1 (a total of 28 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), NRU Thermal-Hydraulic Experiment 2 (a total of 14 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), and NRU Thermal-Hydraulic Experiment 3 (a total of three thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat). (In PRM-50-93, Petitioner discusses the NRU reactor thermal-hydraulic experiments on pages 14-20, 24, 73-74, 75, and Appendix D lists data from the 28 tests conducted in Thermal-Hydraulic Experiment 1.)

C. NEI's Misrepresentations and Misinterpretations of the LOFT LP-FP-2 Experiment

First, in the cover letter of NEI's comments on PRM-50-93, NEI misleadingly states:

> Results from the second test were discounted by the original experimenters because of instrumentation problems.[22]

In the passage above from NEI's cover letter, NEI does not identify the second experiment it is referring to; however, in the attachment, "NEI Comments on Petition for Rulemaking (PRM-50-93)," NEI comments on two experiments discussed in PRM-50-93: FLECHT Run 9573 and the LOFT LP-FP-2 experiment. In the attachment, NEI comments on the thermocouples used in the LOFT LP-FP-2 experiment and states that "according to NUREG/IA-0049, the cause of the rapid temperature rise [in the LOFT LP-

[21] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.
[22] *Id.*, Cover Letter, p. 1.

FP-2 experiment] resulted from shunting of the thermocouple leads through a region of high temperature."[23]

NEI's statement that "[r]esults from the second test were discounted by the original experimenters because of instrumentation problems,"[24] is misleading.

Indeed, there were some thermocouple readings from the LOFT LP-FP-2 experiment that were considered erroneous. This is discussed in Petitioner's comment on PRM-50-93, dated March 15, 2010 (pages 20-23).

In Petitioner's comment on PRM-50-93, dated March 15, 2010 (page 21), regarding core temperature measurements in the LOFT-LP-FP-2 experiment, Petitioner quotes "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2;" it states:

> From the analyses of core temperature measurements in [the LOFT] LP-FP-2 [experiment], the rapid increase in temperature shown in fig 14.[25] was a result of the oxidation of zircaloy which became rapid at temperatures in excess of 1400 K. Further examination of such high temperatures measured by thermocouples gave rise to the detection of a cable shunting effect which is defined in "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2,"[26] as the formation of a new thermocouple junction on the thermocouple cable due to exposure of the cable to high temperature. Experiments were designed and conducted by EG&G Idaho to examine the cable shunting effect. The results of these experiments indicate that the cladding temperature data in LP-FP-2 contain deviations from true temperature due to cable shunting after 1644 K is reached. This temperature is within the range when rapid metal-water reaction occurs. An example of such temperature deviation due to cable shunting is shown in fig. 15.[27, 28]

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[23] *Id.*, Attachment, p. 3.
[24] *Id.*, Cover Letter, p. 1.
[25] See Appendix A of PRM-50-93 Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation.
[26] M. L. Carboneau, V. T. Berta, and S. M. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989.
[27] See Appendix A of Petitioner's comment on PRM-50-93, dated March 15, 2010 Fig. 15 Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM.
[28] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," GRS-Garching, Proceedings of the OECD (NEA) CSNI Specialist Meeting on Instrumentation to Manage Severe Accidents, Held at Cologne, F.R.G. March 16-17, 1992, p. 135.

As a whole the data from the LOFT-LP-FP-2 experiment is considered valid. It seems that NEI does not realize that the data from the LOFT-LP-FP-2 experiment is highly regarded. Indeed, NEI seems to fail to grasp that the paper they cite, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, was written precisely because the data from the LOFT-LP-FP-2 experiment is considered valid: "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" discusses analyses of data from the LOFT LP-FP-2 experiment with the RELAP5/MOD2 and SCDAP/MOD1 codes.

It is significant that the abstract of "Design Report: SCDAP/RELAP5 Reflood Oxidation Model" states:

> Current SCDAP/RELAP5 oxidation models have proven to underpredict oxidation, and therefore hydrogen production, when modeling reflood during in-pile tests. As an example, while OECD LOFT Experiment LP-FP-2 shows significant increases in temperature and pressure during reflood due to increased oxidation, only minimal additional oxidation is currently predicted with SCDAP/RELAP5.[29]

It is also significant that "Design Report: SCDAP/RELAP5 Reflood Oxidation Model" states:

> Based upon the body of work documented in this report, the authors believe they can make several pertinent recommendations. The first regards the validation of the reflood oxidation models incorporated into SCDAP/RELAP5 with this report. ...
>
> The reflood of OECD LOFT Experiment LP-FP-2 also seems to provide a unique opportunity for code validation and assessment, which would provide the user community [with] an understanding of the uses and limitations of the new code models.[30]

Furthermore, data from the LOFT-LP-FP-2 experiment is still being used (in 2010) to benchmark several severe accident codes. In Petitioner's comment on PRM-50-93, dated April 12, 2010 (pages 32-36), Petitioner discusses the fact that developers have used data from the LOFT-LP-FP-2 experiment to help validate the ICARE/CATHARE and ASTEC codes.

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[29] E. W. Coryell, S. A. Chavez, K. L. Davis, M. H. Mortensen, "Design Report: SCDAP/RELAP5 Reflood Oxidation Model," October 1992, EG&G Idaho, Inc., Idaho National Engineering Laboratory, EGG-RAAM-10307, Abstract, p. i.

[30] *Id.*, p. 41.

Additionally, data from the LOFT LP-FP-2 experiment has been used to benchmark the Modular Accident Analysis Program ("MAAP") code. And, as it turns out, the nuclear industry thinks rather highly of the MAAP code. A report Electric Power Research Institute ("EPRI") wrote on behalf of NEI, in 2006, "Program on Technology Innovation: Continued Technical Support to NEI on Risk-Informed Regulations," states:

> On several occasions the Nuclear Regulatory Commission (NRC) has requested the use of an alternative code (specifically RELAP) to justify risk-informed submittals that initially used the MAAP code. *It has long been the industry position that MAAP is the thermal hydraulic code of choice for risk-informed submittals*. The purpose of the plan is to develop a strategy to enhance the acceptance of the MAAP code by the NRC for risk-informed submittals.
>
> It should be recognized that the MAAP code was indeed developed for the investigation of severe accident phenomena as opposed to detailed thermal hydraulic analysis. However, modifications to the code as well as *various benchmarks with experiments*, actual plant events, and other thermal hydraulic codes have shown MAAP to be very robust when addressing various thermal hydraulic issues [emphasis added].[31]

So the industry's position is that "MAAP is the thermal hydraulic code of choice for risk-informed submittals"[32] and in the report EPRI wrote on behalf of NEI, the paper, "Simulation of LOFT Experiment LP-FP-2 Using Modular Accident Analysis Program (MAAP) Version 3.0,"[33] is listed in both appendixes EE and FF.

As quoted in PRM-50-93 (page 39), regarding the value of the data from the LOFT LP-FP-2 experiment, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

> Data from [the LOFT LP-FP-2] experiment provide a wealth of information on severe accident phenomenology. The results provide important data on early phase in-vessel behavior relevant to core melt progression, hydrogen generation, fission product behavior, the composition of melts that might participate in core-concrete interactions, and the effects of reflood on a severely damaged core. The experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.

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[31] K. Canavan, *et al.*, EPRI, "Program on Technology Innovation: Continued Technical Support to NEI on Risk-Informed Regulations," 1013580, Technical Update, December 2006, p. 1-23.
[32] *Id.*
[33] Fauske & Associates, "Simulation of LOFT Experiment LP-FP-2 Using Modular Accident Analysis Program MAAP Version 3.0."

> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[34]

Second, regarding rapid cladding temperature increases in the LOFT LP-FP-2 experiment, NEI misleadingly states:

> [A]ccording to NUREG/IA-0049, the cause of the rapid temperature rise resulted from shunting of the thermocouple leads through a region of high temperature. Thus, there is some uncertainty in the results of [the LOFT LP-FP-2 experiment]. The reported temperature at the initiation of rapid oxidation is not an accurate depiction of the cladding temperature without some form of interpretation.[35]

Regarding, the shunting of the thermocouple leads through high temperature regions, NUREG/IA-0049, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states:

> During the transient, the temperatures on the outside of the shroud increased steadily from 740 to about 1700 sec. This is illustrated in Figure 3.8, which compares the temperatures on the south side of the shroud. At approximately 1700 sec., the heatup rate increases. At about the same time, the thermocouples near the outside of the shroud also start to heat up more rapidly. Figure 3.9 illustrates this by comparing the temperatures at various elevations in the 2nd fuel module, just adjacent to the shroud south wall. By the time the reflood turns the temperatures around (1785 sec.), all of these temperatures exceed the shroud temperatures at the same elevation. The cause of this rapid heatup is not presently known, but it may be an effect caused by the thermocouple leads passing through a hot area as they exit from the top of the core (shunting) rather than by a true local effect.[36]

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[34] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 3. 23.

[35] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

[36] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML062840091, p. 33.

Regarding, the shunting of the thermocouple leads through high temperature regions, NUREG/IA-0049, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" also states:

> Figure 5.17 shows an excellent agreement between the calculated and measured peripheral clad temperatures at the 10-inch elevation until about 1700 sec. At 1700 sec., the thermocouples near the outside of the shroud, particularly at lower elevations, began an extraordinary temperature excursion. The cause of the rapid peripheral temperature rise is somewhat uncertain. The exothermic reaction between zircaloy and water is not considered a possibility because the initiation temperatures were too low; nor is radiation from the shroud wall likely because the wall temperature is lesser than that reached by the fuel rod thermocouples at this elevation. It is judged that the rapid temperature rise was caused by shunting of the thermocouple leads, where they passed through an area of high temperature[37] (near the top of the core). Therefore, the differences with the calculated results are meaningless.[38]

NEI misrepresents the data collected from the LOFT LP-FP-2 experiment when NEI states that "according to NUREG/IA-0049, the cause of the rapid temperature rise resulted from shunting of the thermocouple leads through a region of high temperature. Thus there is some uncertainty in the results of [the LOFT LP-FP-2 experiment."[39] It is clear from the two passages above that NUREG/IA-0049 discusses a rapid temperature rise that was caused by shunting of the thermocouple leads, where they passed through a hot temperature area, *at 1700 sec*. It is also pertinent that NUREG/IA-0049, states that the rapid temperature rise caused by shunting of the thermocouple leads occurred near the outside of the shroud and at peripheral clad locations at the 10-inch elevation.

Clearly, the shunting of the thermocouple leads is not pertinent to the "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water [that] occurred at about *1430 sec.* and 1400 K on a guide tube at the 0.69-m (27-in.) elevation"[40] in the LOFT LP-FP-2 experiment [emphasis added].

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[37] M. L. Carboneau, *et al.*, "OECD LOFT Fission Product Experiment LP-FP-2 Data Report," OECD LOFT-T-3805, OECD, May 1987.

[38] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, p. 79.

[39] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

[40] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, p. 30.

In more detail, as quoted in PRM-50-93 (on pages 39-40), discussing the metal-water reaction measured-temperature data of the LOFT LP-FP-2 experiment, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states:

> The first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation. This temperature is shown in Figure 3.7. A cladding thermocouple at the same elevation (see Figure 3.7) reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase. Note that, due to the limited number of measured cladding temperature locations, the precise location of the initiation of [the] metal-water reaction on any given fuel rod or guide tube is not likely to coincide with the location of a thermocouple. Thus, the temperature rises are probably associated with precursory heating as the metal-water reaction propagates away from the initiation point. Care must be taken in determining the temperature at which the metal-water reaction initiates, since the precursory heating can occur at a much lower temperature. It can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400 K (2060°F).[41, 42]

It is significant that "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states "[t]he first recorded and *qualified* rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K" [emphasis added]. So, in the LOFT LP-FP-2 experiment, the rapid temperature rise associated with the rapid reaction between Zircaloy and water that commenced at approximately 1400°K was *qualified.*

Furthermore, just because, for example, "a cladding thermocouple at the [at the 0.69-m (27-in.) elevation] reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase,"[43] it does not mean that other temperature measurements in the LOFT-LP-FP-2 experiment were not valid.

And as discussed above, EG&G Idaho examined the cable shunting effect that occurred in the LOFT-LP-FP-2 experiment, at locations other than those discussed in

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[41] *Id.*, pp. 30, 33.

[42] See Appendix F of PRM-50-93 Figure 3.7. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 and Figure 3.10. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 with Saturation Temperature.

[43] *Id.*, pp. 30, 33.

NUREG/IA-0049. And EG&G Idaho determined that "the cladding temperature data in LP-FP-2 contain deviations from true temperature due to cable shunting after 1644 K is reached."[44] Furthermore, EG&G Idaho did not disqualify the rapid increase in cladding temperatures that commenced at approximately 1400 K, as a result of the Zircaloy-water reaction.

And regarding the expertise of the test design of the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> The last experiment of the OECD LOFT Project LP-FP-2, conducted on [July] 9, 1985, was a severe core damage experiment. It simulated a LOCA caused by a pipe break in the Low Pressure Injection System (LPIS) of a four-loop PWR as described in "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2."[45] *The central fuel assembly of the LOFT core was specially designed and fabricated for this experiment and included more than 60 thermocouples for temperature measurements.* ...
>
> Experience available in EG&G Idaho from TMI-2 analyses and from the PBF severe fuel damage scoping test conducted in October 1982 were utilized in the design, conduction and analyses of this experiment. LP-FP-2 costs [were] $25 million out of [the] $100 million [spent] for the whole OECD LOFT project [emphasis added].[46]

So the LOFT core had more than 60 thermocouples for temperature measurements.

D. Response to NEI's Claims in NEI's "Background" Section

In NEI's "Background" section, NEI states:

> [T]he petitioner questions the adequacy of the [Baker Just and Cathcart-Pawel] correlations used [for] calculating the metal-water reaction rates. These issues are very similar to those the petitioner raised in Docket number PRM-50-76 (Federal Register of August 9, 2002, Volume 67, Number 154). At the time, the NRC concluded that Appendix K of 10

[44] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 135.
[45] M. L. Carboneau, V. T. Berta, and S. M. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989.
[46] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 133.

> CFR Part 50 and the existing guidance on best-estimate Emergency Core Cooling Systems (ECCS) evaluation models are adequate for assessing ECCS performance for US Light Water Reactors (LWRs) using Zircaloy-clad UO_2 at burnup levels authorized in plant licensing bases. It is the industry's position that the NRC's previous conclusions remain valid.[47]

(It is noteworthy that PRM-50-76 and PRM-50-93 were submitted by different petitioners: Robert H. Leyse and Mark Edward Leyse, respectively.)

First, it is significant that regarding the high burnup single rod furnace tests conducted at Argonne National Laboratory ("ANL")—at the NRC's Advisory Committee on Reactor Safeguards ("ACRS"), Reactor Fuels Committee meeting on April 4, 2001—Dr. Ralph Meyer stated:

> The work started with real specimens last summer when we received the BWR rods from the Limerick plant, and it's slow going. We have done a number of the oxidation kinetics measurements, and I can just give you a qualitative result of that.
>
> Oxidation kinetics seem somewhat faster for high burnup fuel than for fresh fuel. *So we get oxidation rates that are higher than [the] Cathcart-Pawel correlation*, for example, whereas when we measure for fresh tubing, we can reproduce the Cathcart-Pawel correlation [emphasis added].[48]

So Dr. Ralph Meyer stated, "we get oxidation rates that are higher than [the] Cathcart-Pawel correlation,"[49] for high burnup fuel, in an ACRS, Reactor Fuels Committee meeting, more than a year before PRM-50-76—which argued that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA—was submitted. Yet in the NRC's technical safety analysis[50] and report on its denial of PRM-50-76, the NRC did not include any information regarding the oxidation rates of high burnup fuel that had been measured in single rod furnace tests conducted at ANL.

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[47] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 1.

[48] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Committee, Meeting, April 4, 2001.

[49] *Id.*

[50] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," April 29, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041210109.

Second, it is significant that in 2005, in the NRC's report on its denial of PRM-50-76, the NRC stated:

> No data or evidence was...found in NRC records to suggest that the research, calculation methods, or data used to support ECCS performance evaluations were sufficiently flawed so as to create significant safety problems. NRC's technical safety analysis demonstrates that current procedures for evaluating performance of ECCS are based on sound science and that no amendments to the NRC's regulations and guidance documents are necessary. ...the NRC [has not] found, the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material.[51]

So the NRC was unable to locate data in NRC records from multi-rod (assembly) severe fuel damage experiments that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. And the NRC was unable to perceive "the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material."[52] For example, the NRC was unable to locate data in NRC records from the LOFT LP-FP-2 experiment that indicates that an autocatalytic oxidation reaction of Zircaloy cladding occurred at a temperature hundreds of degrees Fahrenheit below what either the Baker-Just or Cathcart-Pawel equations would predict.

Clearly, the NRC's conclusions regarding the Baker Just and Cathcart-Pawel correlations, in its denial of PRM-50-76, were not based on a review of pertinent experimental data.

E. Response to NEI's Claims in NEI's "Zirconium-Water Reaction" Section

It is significant that in NEI's "Multirod Severe Fuel Tests" section, NEI states:

> Rapid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F).[53]

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[51] 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, p. 23.

[52] *Id.*

[53] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

(According to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F). Also, according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" the peak measured cladding temperature reached 2100°K (3320°F) within approximately 75 seconds[54] (the melting point of Zircaloy is approximately 3308°F[55]). And according to another report, once the Zircaloy cladding began rapidly oxidizing, cladding temperatures increased at a rate of approximately 18°F/sec. to 36°F/sec.[56])

Of course, 1430 K (2114°F) is below the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F, so in the interest of public and plant-worker safety and conservatism, the NRC should regard NEI's statement that "[r]apid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F)"[57] in the LOFT-LP-FP-2 experiment, as another piece of evidence that indicates the 2200°F PCT limit is non-conservative.

NEI's statement should also be regarded as another piece of evidence that indicates the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. Which, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

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[54] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, pp. 23, 30.

[55] 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.

[56] 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; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[57] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

It is significant that in NEI's "Conclusions" section, NEI states:

> [T]he petitioner's claim that the autocatalytic runaway regime begins below 2200°F and that the current [metal-water reaction rate] correlations are non-conservative is not substantiated for conditions where core cooling within the capability of current design exists (*i.e.*, realistic balance of heat addition and removal).[58]

It is NEI's statement above that is unsubstantiated; furthermore, NEI is overly optimistic about what the "realistic balance of heat addition and removal" in the event of a LOCA would actually be.

It is significant that in the ACRS, Reactor Fuels Subcommittee Meeting, on September 29, 2003, Dr. Dana A. Powers stated:

> ...I have seen some calculations...dealing with heat transfer of single rods versus bundles which says, well, on heat transfer effects, I just don't learn anything from single rod tests. So I really have to go to bundles, and even multi-bundles to understand the heat transfer. The question we're struggling with now is a modified question. Is there more we need to do to understand what goes on in the reactor accident?[59]

And regarding how heat transfer affects the temperature at which the autocatalytic oxidation of Zircaloy cladding occurs—at the NRC's ACRS, Reactor Fuels Committee meeting on April 4, 2001—Dr. Ralph Meyer stated:

> There doesn't seem to be any magic temperature at which you get some autocatalytic reaction that runs away. *It's simply a matter of heat balances: how much heat from the chemical process and how much can you pull away* [emphasis added].[60]

In PRM-50-93, and in Petitioner's comments on PRM-50-93, Petitioner also argues that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA, because they were not developed to consider how heat transfer would affect zirconium-water reaction kinetics. (Petitioner quotes many reports stating that heat transfer affects zirconium-water reaction kinetics.)

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[58] *Id.*, p. 4.

[59] Dr. Dana A. Powers, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Transcript, September 29, 2003, pp. 211-212.

[60] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Committee, Meeting, April 4, 2001. In the transcript the second sentence was transcribed as a question; however, the second sentence was clearly not phrased as a question.

In PRM-50-93, and in Petitioner's comments on PRM-50-93, Petitioner discusses data from many multi-rod (assembly) severe fuel damage experiments that indicates the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. Petitioner also discusses data from two multi-rod (assembly) thermal hydraulic experiments indicating the same.

Discussing single rod furnace tests that were conducted at ANL, NEI states:

> Recent tests conducted at Argonne National Laboratory (ANL) and documented in NUREG/CR-6967, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents" July 31, 2008 (ML082130389) have demonstrated that the [Baker-Just] correlation over-predicts the zirconium-water reaction by as much as 30% at the limiting temperature (2200°F)[61]

(It is noteworthy that regarding the high burnup single rod furnace tests conducted at ANL—at the NRC's ACRS, Reactor Fuels Committee meeting on April 4, 2001—Dr. Ralph Meyer stated:

> The work started with real specimens last summer when we received the BWR rods from the Limerick plant, and it's slow going. We have done a number of the oxidation kinetics measurements, and I can just give you a qualitative result of that.
>
> Oxidation kinetics seem somewhat faster for high burnup fuel than for fresh fuel. *So we get oxidation rates that are higher than [the] Cathcart-Pawel correlation*, for example, whereas when we measure for fresh tubing, we can reproduce the Cathcart-Pawel correlation [emphasis added].[62])

It is significant that when Dr. Dana A. Powers stated "I have seen some calculations...dealing with heat transfer of single rods versus bundles which says, well, on heat transfer effects, I just don't learn anything from single rod tests. So I really have to go to bundles, and even multi-bundles to understand the heat transfer,"[63] he was discussing the ANL single rod tests with Mike Billone—the lead author of "Cladding

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[61] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 2.

[62] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Committee, Meeting, April 4, 2001.

[63] Dr. Dana A. Powers, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Transcript, September 29, 2003, pp. 211-212.

Embrittlement During Postulated Loss-of-Coolant Accidents"[64]—and others in an ACRS meeting.

It is also significant that "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents" states:

> Because the sample has such low thermal mass per unit length, it is important to ramp to the hold temperature at a relatively fast rate for these tests without *temperature overshoot due to the initially rapid heat generation rate from cladding oxidation.* In setting the controller parameters, the requirements are that the temperature overshoot during the ramp be <20°C relative to the target hold temperature for a short period of time (few seconds), and that the average hold temperature be within 10°C of the target temperature. ... Temperature overshoot is not much of an issue for long-time oxidation temperatures ≤1100°C, but it can have a significant embrittlement effect for higher oxidation temperatures. For tests conducted at 1200°C, temperature overshoot was minimized by slowing down the heating rate at ramp temperatures within 50-100°C of the target temperature [emphasis added].[65]

So in the ANL single rod tests "temperature overshoot due to the initially rapid heat generation rate from cladding oxidation"[66] was a phenomenon that had to be controlled by various test procedures.

But clearly, it would not be possible to investigate the oxidation kinetics of Zircaloy fuel-cladding bundles under isothermal conditions at temperatures between 1000°C and 1200°C. If such an attempt were made, it would not be possible to meet the experimental protocol of isothermal conditions, because the energy from the exothermic Zircaloy-steam oxidation would cause a temperature excursion.

It is significant that regarding the uncontrollable Zircaloy-steam reaction that would occur in the event of a LOCA, "Current Knowledge on Core Degradation Phenomena, a Review" sates:

> Oxidation of Zircaloy cladding materials by steam becomes a significant heat source which increases with temperature; *if the heat removal*

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[64] M. Billone, *et al.*, "Cladding Embrittlement During Postulated Loss-of Coolant Accidents" NUREG/CR-6967, July 2008, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML082130389.
[65] *Id.*, p. 17.
[66] *Id.*

capability is lost, it determines a feedback between temperature increase and cladding oxidation [emphasis added].[67]

Furthermore, Figure 1[68] of the same paper depicts that the "start of rapid [Zircaloy] oxidation by H_2O [causes an] uncontrolled temperature escalation," at 1200°C (2192°F),[69] and Figure 13[70] of the same paper depicts that if the initial heat up rate is 1 K/sec. or greater, a cladding temperature excursion would commence at 1200°C (2192°F), in which the rate of increase would be 10 K/sec. or greater.[71]

It is significant that "if the heat removal capability is lost [from the oxidation of Zircaloy cladding materials by steam], it determines a feedback between temperature increase and cladding oxidation;"[72] and that "any failure to remove the heat of the Zircaloy-steam reaction from the fuel cladding can result in an increase in the temperature of the cladding."[73]

And this is what occurred in the LOFT LP-FP-2 experiment where "[r]apid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F)"[74] or 1400 K (2060°F).[75]

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[67] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 195.

[68] See Appendix B of Petitioner's comment on PRM-50-93, dated April 12, 2010 Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.

[69] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 196.

[70] See Appendix B of Petitioner's comment on PRM-50-93, dated April 12, 2010 Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core.

[71] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 205.

[72] *Id.*, p. 195.

[73] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079, p. 119.

[74] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

[75] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.

F. Response to NEI's Claims in NEI's "Multirod Severe Fuel Tests" Section

In NEI's comments NEI, states:

> The petitioner relies heavily on the results of two assembly tests with fuel damage, FLECHT Run 9573 and LOFT LP-FP-2.[76]

It is important to clarify that Petitioner cites data from many multi-rod severe fuel damage experiments in PRM-50-93. (PWR FLECHT Run 9573 and the BWR FLECHT Zr2K test were thermal hydraulic tests; however, in some respects they resembled severe fuel damage tests.)

Regarding the metal-water reaction rate and/or experimental data that indicates the Baker-Just and Cathcart-Pawel equations are non-conservative, Petitioner discusses data from the following multi-rod experiments: the Power Burst Facility ("PBF") Severe Fuel Damage ("SFD") 1-1 test, PBF SFD 1-3 test, PBF SFD 1-4 test, NRU Materials Test 6B, NRU Reactor Full-Length High-Temperature 1 Test, the LOFT LP-FP-2 experiment, the CORA Experiments as a whole, the CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments, the PHEBUS B9R test, the QUENCH-04 test, PWR FLECHT Run 9573, and the BWR FLECHT Zr2K test. (The CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments, and the BWR FLECHT Zr2K test are discussed in Petitioner's comment on PRM-50-93, dated March 15, 2010.)

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner *primarily* discusses data from the following multi-rod experiments: the LOFT LP-FP-2 experiment, the CORA Experiments as a whole, and the CORA-2, CORA-3, CORA-7, CORA-9, CORA-12, CORA-13, CORA-15, and CORA-16 experiments.

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner also discusses data from the BWR FLECHT Zr2K test: Petitioner points out that graphs of thermocouple measurements taken during the Zr2K test depict temperature excursions that began when cladding temperatures reached between approximately 2100 and 2200°F.

[76] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 2.

Regarding the calculated maximum fuel element cladding temperature limit, Petitioner also discusses data from experiments, where the onset of autocatalytic oxidation occurred above 2200°F. It can be concluded that 2200°F peak cladding temperature ("PCT") limit does not provide a necessary margin of safety from the following experiments: NRU Reactor Full-Length High-Temperature 1 Test, the PHEBUS B9R test, and the QUENCH-04 test.

1. FLECHT Run 9573

For information on NEI's account of FLECHT Run 9573, see the text in Section B above: "NEI's Misinterpretations of FLECHT Run 9573 and Misrepresentations of Petitioner's Discussion of FLECHT Run 9573 in PRM-50-93."

In addition to the text in the section above, it is noteworthy, that Petitioner's *primary* conclusions from the experimental data of FLECHT Run 9573, stated in PRM-50-93 (page 71), are:

> FLECHT run 9573 demonstrates that the metal-water reaction becomes autocatalytic at temperatures lower than what the Baker-Just and Cathcart-Pawel equations predict. Westinghouse stated that run 9573 incurred autocatalytic oxidation at a temperature greater than 2300°F[77] (most likely, meaning at a temperature below 2400°F); the Baker-Just and Cathcart-Pawel equations predict that autocatalytic oxidation of Zircaloy cladding occurs at approximately 2600°F and 2700°F, respectively.[78]
>
> The results from FLECHT run 9573 also demonstrate that stainless steel cladding heat transfer coefficients are not always a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.[79]

2. The LOFT LP-FP-2 Experiment

In NEI's comments, NEI has misrepresented and misinterpreted the LOFT LP-FP-2 experiment: NEI states that "[r]esults from the second test were discounted by the

[77] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[78] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."
[79] Mark Edward Leyse, PRM-50-93, November 17, 2009, p. 71.

original experimenters because of instrumentation problems,"[80] and that "according to NUREG/IA-0049, the cause of the rapid temperature rise resulted from shunting of the thermocouple leads through a region of high temperature"[81]

NUREG/IA-0049, explicitly sates:

> The first recorded and *qualified* rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation [emphasis added].[82]

Data from the LOFT-LP-FP-2 experiment is still being used (in 2010) to benchmark several severe accident codes. It is also significant that "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states: "[t]he LOFT LP-FP-2 experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[83]

Additionally, it is significant that in NEI's comments, NEI states:

> Rapid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F). The LOFT thermocouples had a reported uncertainty of 5% under ambient conditions but this uncertainty increased during the later stages of the transient because of thermocouple drift and as a result of cladding oxidation and ballooning.[84]

First, NEI provides no data to support NEI's claim that "The LOFT thermocouples had a reported uncertainty of 5% under ambient conditions but this uncertainty increased during the later stages of the transient because of thermocouples drift and as a result of cladding oxidation and ballooning."[85]

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[80] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Cover Letter, p. 1.
[81] *Id.*, Attachment, p. 3.
[82] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.
[83] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3. 23.
[84] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.
[85] *Id.*

(It would be helpful to have notes for such statements, complete with report titles and page numbers.)

It is significant that "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 sec. and 1400 K *on a guide tube* at the 0.69-m (27-in.) elevation,"[86] *not on a fuel rod*, in the LOFT LP-FP-2 experiment [emphasis added].

Second, NEI states that "Rapid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F)."[87]

(According to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F). Also, according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" the peak measured cladding temperature reached 2100°K (3320°F) within approximately 75 seconds[88] (the melting point of Zircaloy is approximately 3308°F[89]). And according to another report, once the Zircaloy cladding began rapidly oxidizing, cladding temperatures increased at a rate of approximately 18°F/sec. to 36°F/sec.[90])

Of course, 1430 K (2114°F) is below the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F, so in the interest of public and plant-worker safety and conservatism, the NRC should regard NEI's statement that "[r]apid cladding oxidation was observed when cladding thermocouples reported a temperature of

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[86] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, p. 30.

[87] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

[88] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, pp. 23, 30.

[89] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," p. 3-1.

[90] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, "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, p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, as the source of this information.

approximately 1430 K (2114°F)"[91] in the LOFT-LP-FP-2 experiment, as another piece of evidence that indicates the 2200°F PCT limit is non-conservative.

NEI's statement should also be regarded as another piece of evidence that indicates the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. Which, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

For additional information on NEI's account of the LOFT-LP-FP-2 experiment, see the text in Section C above: "NEI's Misrepresentations and Misinterpretations of the LOFT LP-FP-2 Experiment."

G. Response to NEI's Claims in NEI's "Reflood Rates" Section

First, in NEI's comments on PRM-50-93, NEI erroneously states:

> The petitioner bases the claim for a fixed minimum reflood rate on FLECHT Run 9573.[92]

In PRM-50-93, Petitioner argues for a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA, *primarily* by citing experimental data from the NRU Thermal-Hydraulic Experiment 1 (a total of 28 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), NRU Thermal-Hydraulic Experiment 2 (a total of 14 thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat), and NRU Thermal-Hydraulic Experiment 3 (a total of three thermal hydraulic tests conducted with full-length Zircaloy fuel rods, driven by low-level fission heat).

It is noteworthy that in NEI's comments on PRM-50-93, NEI does not comment on NRU Thermal-Hydraulic Experiment 1, NRU Thermal-Hydraulic Experiment 2, and NRU Thermal-Hydraulic Experiment 3. In the early 1980s, the NRC contracted with NRU at Chalk River, Ontario, Canada to run a series of LOCA tests in the NRU reactor. 45 tests were conducted to evaluate the thermal-hydraulic behavior of a full-length 32-rod

[91] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.
[92] *Id.*

Zircaloy assembly during the heatup, reflood, and quench phases of a large-break LOCA. In PRM-50-93, Petitioner discusses the NRU reactor thermal-hydraulic experiments on several pages (pages 14-20, 24, 73-74, 75) and Appendix D lists data from the 28 tests conducted in Thermal-Hydraulic Experiment 1, yet NEI has not commented on the NRU reactor thermal-hydraulic experiments in NEI's comments.

Second, it is significant that in NEI's "Multirod Severe Fuel Tests" section, NEI states:

> Depending on the plant design, core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F; these are significantly lower than in FLECHT Run 9573 and at flooding rates substantially above the 1.1 inch/second of this test. Flooding rates as low as [1.1 inch/second] are possible only after significant cooling is established within the core.[93]

NEI makes the above claim, yet NEI provides no experimental data to substantiate the above claim. NEI does not provide any experimental data that indicates what initial reflood rates would be or what the time duration of the initial reflood rates would be before the effects of steam binding set in. NEI also does not provide any experimental data from tests conducted with full-length Zircaloy cladding that indicates that there would in fact be significant cooling in the core when reflood rates dropped to 1 in./sec. or lower.

(It would be helpful to have notes for such claims, complete with report titles and page numbers.)

And, as pointed out above, in NEI's comments on PRM-50-93, NEI does not comment on NRU's thermal-hydraulic experiments conducted in the early '80s. One of the primary reasons that Petitioner discusses NRU's thermal-hydraulic experiments, is that they were conducted with full-length Zircaloy cladding, driven by low-level fission heat.

If indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F,"[94] this 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.

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[93] *Id.*
[94] *Id.*

It is significant that "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction" states:

> Bottom reflood progresses very quickly during the onset of reflood. However, the intense steam generation soon retards the overall progression of the quench front to a relatively uniform progression. Nevertheless, good core quenching rates are achieved even for flooding rates of one inch per second.
>
> ... During reflood, the flow regime, cladding temperature rise and quench behavior is strongly dependant on the flooding rate.[95]

It is important to note that when "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction," states that "good core quenching rates are achieved even for flooding rates of one inch per second," this claim is based on the results of tests conducted with stainless steel cladding, *not* driven by low-level fission heat.

(In the event of a 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.)

Regarding Thermal-Hydraulic Experiment 1 ("TH-1"), PRM-50-93 (page 18) states:

> The TH-1 tests illustrate 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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[95] "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction," Attachment 3 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 2; Attachment 3 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720713; the letter's Accession Number: ML021720690.

> 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.[96]

So, clearly, if indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F,"[97] it is highly problematic, and additional evidence that indicates that the NRC should make a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA.

III. Conclusion

In NEI's comments, NEI only commented on two experiments (FLECHT Run 9573 and the LOFT LP-FP-2 experiment) out of the more than 60 experiments discussed in PRM-50-93.

It is noteworthy that in PRM-50-93, Petitioner discusses a number of severe fuel damage experiments that Electric Power Research Institute ("EPRI") lists in "Program on Technology Innovation: Continued Technical Support to NEI on Risk-Informed Regulations"[98]: a report EPRI wrote on behalf of NEI. In the report, EPRI has two appendixes—Appendix EE Compendium of Source Term Report and Appendix FF Listing of Reports Related to Severe Accidents—that list at least four papers on different CORA experiments and at least one paper on the LOFT LP-FP-2 experiment (mentioned above).)

[96] Mark Edward Leyse, PRM-50-93, November 17, 2009, p. 18.
[97] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.
[98] K. Canavan, *et al.*, EPRI, "Program on Technology Innovation: Continued Technical Support to NEI on Risk-Informed Regulations," 1013580, Technical Update, December 2006.

In the report EPRI wrote on behalf of NEI, the paper "First Results of CORA Post Test Examinations (CORA Bundle Test B),"[99] is listed in both appendixes EE and FF. Petitioner has not read this paper; however, CORA Bundle Test B is mentioned in a paper discussed in two of Petitioner comments on PRM-50-93, dated March 15, 2010 and April 12, 2010.

Discussing the exothermic Zircaloy-steam reaction that occurred in the CORA-2 and CORA-3 experiments, "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)" states:

> As already observed in previous tests [(*CORA Test B* and CORA Test C)],[100] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation; the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles [emphasis added].[101]

As discussed in PRM-50-93, on pages 26-27, 38-45, 51-55, "[t]he 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,"[102] and this occurred in CORA Bundle Test B, commencing at a temperature below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

It is unfortunate that NEI did not comment on the CORA experiments that were discussed at length in PRM-50-93 and in Petitioner's comments on PRM-50-93.

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[99] Peter Hofmann, "First Results of CORA Post Test Examinations (CORA Bundle Test B)," SFD Meeting, May 1987.

[100] S. Hagen *et al.*, "Interactions between Aluminium Oxide Pellets and Zircaloy Tubes in Steam Atmosphere at Temperatures above 1200°C (Posttest Results from the CORA Tests B and C)," KfK-4313, 1988.

[101] 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)," KfK 4378, p. 41.

[102] 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.

Additionally, it is noteworthy that many of the papers listed in Appendix EE and Appendix FF report on experiments that were conducted more than 30 years ago.

In NEI's comments, NEI has misrepresented Petitioner's arguments regarding FLECHT Run 9573: 1) in no section of PRM-50-93, and in no section of Petitioner's comments on PRM-50-93, does Petitioner state that a zirconium-water autocatalytic reaction was reached at temperatures below 2200°F in FLECHT Run 9573; and 2) in PRM-50-93 and in Petitioner's comments on PRM-50-93, Petitioner does not "[base] the claim for a fixed minimum reflood rate on FLECHT Run 9573."[103]

In NEI's comments, NEI has misrepresented and misinterpreted the LOFT LP-FP-2 experiment: NEI states that "[r]esults from the second test were discounted by the original experimenters because of instrumentation problems,"[104] and that "according to NUREG/IA-0049, the cause of the rapid temperature rise resulted from shunting of the thermocouple leads through a region of high temperature"[105]

NUREG/IA-0049, explicitly sates:

> The first recorded and *qualified* rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation [emphasis added].[106]

Data from the LOFT-LP-FP-2 experiment is still being used (in 2010) to benchmark several severe accident codes. It is also significant that "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states: "[t]he LOFT LP-FP-2 experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[107]

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103 NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.
104 *Id.*, Cover Letter, p. 1.
105 *Id.*, Attachment, p. 3.
106 J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.
107 S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3. 23.

In NEI's comments, NEI makes two statements that provide additional evidence that the NRC should make the regulations proposed in PRM-50-93 into legally binding regulations.

First, discussing the LOFT LP-FP-2 experiment , NEI states:

> Rapid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F).[108]

(According to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F). Also, according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" the peak measured cladding temperature reached 2100°K (3320°F) within approximately 75 seconds[109] (the melting point of Zircaloy is approximately 3308°F[110]).)

Of course, 1430 K (2114°F) is below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F, so in the interest of public and plant-worker safety and conservatism, the NRC should regard NEI's statement that "[r]apid cladding oxidation was observed when cladding thermocouples reported a temperature of approximately 1430 K (2114°F)"[111] in the LOFT-LP-FP-2 experiment, as another piece of evidence that indicates the 2200°F PCT limit is non-conservative.

NEI's statement should also be regarded as another piece of evidence that indicates the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. Which, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

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108 NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

109 J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," NUREG/IA-0049, pp. 23, 30.

110 NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," p. 3-1.

111 NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," Attachment, p. 3.

Second, it is significant that NEI states:

> Depending on the plant design, core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F...[112]

So, clearly, if indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F,"[113] it is highly problematic, and additional evidence that indicates that the NRC should make a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA.

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

Respectfully submitted,

Mark Edward Leyse
P.O. Box 1314
New York, NY 10025
markleyse@gmail.com

Dated: April 28, 2010

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[112] *Id.*
[113] *Id.*

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Thursday, April 29, 2010 4:01 PM
To: Rulemaking Comments
Subject: NRC-2009-0554
Attachments: Response to NEI Comments on PRM-50-93.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is a cover letter and my third response, dated April 28, 2010, to the NRC's notice of solicitation of public comments on PRM-50-93, NRC-2009-0554, published in the Federal Register, January 25, 2010. In these comments, I respond to the Nuclear Energy Institute's comments on PRM-50-93, dated April 12, 2010.

Sincerely,

Mark Leyse

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